Study on the Applicability of the Regulatory Framework for Nuclear Facilities to Fusion Facilities.

Towards a specific regulatory framework for fusion facilities

Final Report

GRS and KIT
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EUROPEAN COMMISSION

Directorate-General for Energy
Directorate D — Nuclear Energy, Safety and ITER
Unit D.4 — ITER

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CONTENT

ABBREVIATIONS

AC	Alternating Current
ACB	Advanced Ceramic Breeder, Auxiliary Cold Boxes
ACP	Activated Corrosion Products
AFE	Active Front End
ALARA	As Low As Reasonably Achievable
ALARP	As Low As Reasonably Practicable
AMF	Active Management Facility
ASME	American Society of Mechanical Engineers
ASN	Autorité de Sûreté Nucléaire
BB	Breeding Blanket
BDBA	Beyond Design Base Accident
Be	Beryllium
BoP	Balance of Plant
BPV	Boiler and Presser Vessel (Committee on Construction of Nuclear Facility Components)
BSS	Back Supporting Structure
BWR	Boiling Water Reactor
BZ	Breeding Zone
C	Celsius
CB	Cassette Body (divertor component)
CCWS	Component Cooling Water System
CFETR	China Fusion Engineering Test Reactor
CFR	Code of Federal Regulations
CHWS	Chilled Water System
CLAM	China Low Activation Martensitic (steel)
CMM	Cassette Multifunctional Mover
CMSB	Cryogenic Molecular Sieve Bed
CN	China
CNSC	Canadian Nuclear Safety Commission
CODAC	Control, Data Access and Communication
COPD	Chronic Obstructive Pulmonary Disease
CPS	Coolant Purification System
CS	Central Solenoid
CTB	Coil Termination Box
CTCB	Cryoplant Termination Cold Box
CTM	Cassette Toroidal Mover
CVB	Cold Valve Box
CVCS	Chemical and Volume Control System
D	Deuterium
D&C	Diagnostic and Control
DBA	Design Basis Accident
DCD	Direct Coupling Design
DCLL	Dual Coolant Lead-Lithium
DD	Deuterium-Deuterium
DEC	Design Extension Conditions
DEMO	Demonstration fusion power plant
DiD	Defence-in-Depth
DIR	Direct Internal Recycling
DHRS	Decay Heat Removal System
DIR	Direct Internal Recycling
DIV	Divertor
DN	Double Null (divertor configuration)
DOE	Department of Energy
DONES	DEMO-Oriented Neutron Source
dpa	Displacement rate per atom
DT	Deuterium-Tritium
DSP	Digital Signal Processor
EA	Environmental Agency (UK)
EC	European Commission, Electron Cyclotron
ELM	Edge Localized Mode
EM	Electromagnetic

ENEA	Agenzia Nazionale per le Nuove Tecnologie, l'Energia e lo Sviluppo economico sostenibile
ENSREG	European Nuclear Safety Regulators Group
EOWG	Open-Ended Waveguide (antenna)
EPR	Environmental Permitting Regulations
EP&R	Emergency Preparedness and Response
ESS	Energy Storage System
EU	European Union
EURATOM	European Atomic Energy Community
EUROFER	European Steel Association
EV	Expansion Volume
FDU	Fast Discharge Unit
FFMEA	Functional Failure Mode and Effect Analysis
FGM	Functional Graded Material
FNSF	Fusion Nuclear Science Facility
FPGA	Field Programmable Gate Arrays
FPP	Fusion Power Plant
fpy	Full power year
FW	First Wall
GLC	Gas-Liquid Contactor
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit (German TSO)
GSSR	Generic Site Safety Report
GW	Gigawatt
H	Hydrogen
HCD	Helicon Current Drive (system)
H&CD	Heating and Current Drive (system)
HBRB	High-Breeding Ratio Blanket
HCCB	Helium Cooled Ceramic Breeder (blanket)
HCPB	Helium-cooled Pebble Bed (blanket)
H	Hydrogen, High (confinement mode) is→
Ha	Hartmann number
He	Helium
HLW	High Level Waste
HSE	Health and Safety Executive (UK)
HTS	High Temperature Superconductor
HV	High Voltage
HVAC	Heating, Ventilation Air Condition
HX	Heat Exchanger
IA	Inherent Availability
IAEA	International Atomic Energy Agency
IB	Inboard (blanket)
IC	Ion Cyclotron
I&C	Instrumentation and Control
ICD	Indirect Coupling Design
IFMIF	International Fusion Materials Irradiation Facility
IHTS	Intermediate Heat Transfer System
IHX	Intermediate Heat Exchanger
ILW	Intermediate Level Waste
IML	Inboard Mid-plane Limiter
INB	Installation Nucléaire de Base
INSAG	International Nuclear Safety Advisory Group
IR	Isotope Rebalancing
IRR17	Ionising Radiations Regulations 2017
IRSN	Institut de Radioprotection et de Sûreté Nucléaire (French TSO)
ISS	Isotopic Separation System
ITER	International Thermonuclear Experimental Reactor
IVC	In-Vessel Component
IVT	Inner Vertical Target (divertor component)
JET	Joint European Torus
KDII	Key Design Integration Issue
KIT	Karlsruhe Institute of Technology
kPa	Kilopascal
KSTAR	Korea Superconducting Tokamak Advanced Research

KTA	Kerntechnischer Ausschuss (German Nuclear Standardization Organization)
L	Low (confinement mode)
LC	Load Coil
LHCD	Low Hybrid Wave Current Drive
LHe	Liquid Helium
Li	Lithium
LLW	Low Level Waste
LN2	Liquid Nitrogen
LOCA	Loss of Coolant Accident
LOFA	Loss of Flow Accident
LOVA	Loss of Vacuum Accident
LOOP	Loss of Offsite Power
LTS	Low Temperature Superconductor
LVC	Liquid Vacuum Contactor
MBTL	Multi-Beam Transmission Line
MCP	Main Coolant Pump
MDT	Mean Downtime
MEST	Magnetic Energy Storage and Transfer (system)
MIMO	Multiple-Input-Multiple-Output
MLI	Multi-Layer Insulation
MMC	Modular Multilevel Converters
MPa	Megapascal
MSA	Mid-Steering Antenna
MTBF	Mean Time Between Failure
MTTR	Mean Time To Repair
MW	Megawatt
Nb	Niobium
NB	Neutral Beam
NBI	Neutral Beam Injection (system)
NEA	Nuclear Energy Agency (OECD)
NEG	Non-Evaporable Getter
NEIMA	Nuclear Energy Innovation and Modernization Act
NIF	National Ignition Facility (USA)
NMM	Neutron Multiplier Material
NNBI	Negative ion-based Neutral Beam Injector
NNSA	National Nuclear Safety Administration (China)
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission (USA)
NTM	Neoclassical Tearing Mode
NTM	Neo Classical Tearing Mode
OA	Operational Availability
OB	Outboard (blanket)
OECD	Organisation for Economic Co-operation and Development
OFC	Oxygen-Free Copper
OFHC	Oxygen-free High Thermal Conductivity
OML	Outboard Mid-plane Limiters
ONR	Office for Nuclear Regulation (UK)
OLL	Outboard Lower Limiter
ORE	Occupational Radiation Exposure
OTSG	Once Through Steam Generator
OVT	Outer Vertical Target (divertor component)
PAR	Passive Autocatalytic Recombiners
PAT	Power Ascension Tests
PAV	Permeator Against Vacuum
PbLi	Lithium-lead
PBS	Plant Breakdown Structure
PC	Plasma Chamber
PCS	Power Conversion System
PES	Plant Electrical Supply
PF	Poloidal Field (Coil)
PFC	Plasma-Facing Component
PFU	Plasma-Facing Unit
PG	Purge Gas

PHTS	Primary Heat Transfer System
PIE	Postulated Initiation Events
PR	Protium Removal
PRF	Permeation Reduction Factor
PRZ	Pressurizer
PSD	Plasma Shutdown (system)
PTG	Power Transmission Grid
RAFM	Reduced Activation Ferritic Martensitic (steel)
RAMI	Reliability Availability Maintainability Inspectability
RBD	Reliability Block Diagram
RCC-MRx	Règles de Conception et de Construction des matériels mécaniques des installations nucléaires hautes températures, expérimentales et de fusion (AFCEN code)
RD	Rupture Disc
RF	Radio Frequency
RH	Remote Handling
RID	Residual Ion Deflection (unit)
RM	Remote Maintenance
RMP	Resonant Magnetic Perturbation
RMSB	Reactive Molecular Sieve Bed
RPrS	Rapport Préliminaire de Sûreté, Preliminary Safety Report
SAP	Safety Assessment Principles
SC	Sink Coil
SDDR	Shutdown dose rate
SF	Safety Fundamentals (IAEA Safety Standard Series)
SG	Steam Generator
Si	Silicon
SIC	Safety Important Classification (or Component)
SMS	Single-Module Segment
SN	Single Null (Divertor configuration)
SOL	Scrape-Off Layer
SSC	Structures, Systems and Components
SSR	Specific Safety Requirements (IAEA Safety Standard Series)
ST	Suppression Tank
SWCB	Super-critical Water-Cooled solid Breeder
T	Tritium / Temperature / Tesla
TBR	Tritium Breeding Rate
TCVB	Thermal Shield Cold Valve Box
TECDOC	IAEA Technical Documents Series
TEP	Tokamak Exhaust Processing
TER	Tritium Extraction and Removal (system)
TEU	Tritium Extraction Unit
TF	Toroidal Field (coil)
TG	Turbine Generator
TPB	Tritium Permeation Barriers
TWA	Travelling Wave Antenna
Ti	Titanium
TLK	Tritium Labor Karlsruhe (part of KIT)
TS	Transfer System
TSO	Technical Safety Organization
UK	United Kingdom
UKAEA	United Kingdom Atomic Energy Authority
UL	Upper Limiter
USA	United States of America
VDE	Vertical Displacement Events
VLLW	Very Low Level Waste
VSC	Voltage Source Converters
VV	Vacuum Vessel
VVPSS	Vacuum Vessel Pressure Suppression System
W	Tungsten
WBS	Work Breakdown Structure
WCLL	Water-cooled Lithium Lead (blanket)
WCSB	Water-cooled Solid Breeder (blanket)

WDS	Water Detritiation System
WENRA	Western European Regulators Association
WOOD	British TSO
WP	Winding Pack

1 EXECUTIVE SUMMARY

This report describes the results obtain in the study "On the Applicability of the Regulatory Framework for Nuclear Facilities to Fusion Facilities. Towards a specific regulatory framework for fusion facilities" carried out by GRS and KIT under the contract 356842-ENER-D4-2020-64. The study was split into five different tasks that yielded in five deliverables.

In Task 1 an overview of international approaches to fusion regulation and existing safety requirements for fusion facilities was compiled. As expected, the review of the different approaches confirmed that presently no country has a dedicated and specific regulatory framework for fusion facilities. Nevertheless, it was observed that the competent regulatory authorities in more and more countries pay attention to this issue. Based on the insights gained in Task 1 it has been concluded that the development of a future fusion specific regulatory framework needs

- to consider the different sources and zones in a fusion facility with radiation exposure risks;
- to develop a graded approach for fusion facilities;
- to establish a safety concept for future fusion facilities;
- to consider internal and external hazards, including non-radiological hazards specific for fusion facilities;
- to consider the approach of design extension conditions to enhance defence in depth;
- to develop a regulatory approach for licensing and oversight of a first-of-a-kind facility;
- to enhance regulatory readiness by capacity building of competent authorities;
- to include international and national operational experience feedback;
- to consider the need for interim storage and potentially final disposal.

In Task 2 the main differences between fusion and fission facilities have been identified. Detailed technical descriptions of 17 fusion specific systems have been completed and detailed using the example of the DEMO configuration. In addition, since fusion technology is in a development stage, the established descriptions emphasize various technical solutions to realize such systems. In addition, a variety of regulatory documents were screened including Council Directives / Directives of the European Parliament and IAEA Safety Standard Series. The IAEA Safety Standard Series provides a hierarchically structured set of safety requirements and substantiating safety guides. Whereas the high-level requirements are directly applicable to fusion power plants, specific safety requirements and guidance addressing typical systems of fusion power plants are not yet addressed. These results constitute the technical deliverable D2: "Main differences between fusion and fission facilities and specific safety requirements relevant for fusion facilities".

Within Task 3 safety requirements for fusion specific systems have been derived. As IAEA has already performed an analysis of the applicability of IAEA Safety Standard SSR-4 on fusion facilities and a first adaption of this safety requirement to fusion facilities, the work performed within Task 3 discusses some basic safety principles and specific safety aspects of fusion facilities not yet addressed in the IAEA work. Safety issues of those fusion specific systems described in Task 2 are considered and proposals for safety requirements have been derived. Insights from various on-going DEMO projects around the world as well as from the ITER project reflect a broad spectrum of different design solutions. A structure for safety requirements for fusion facilities was drafted and first safety requirements were derived and formulated. In addition, the interface between safety and security has been addressed. Based on these finding a safety concept for fusion facilities has been derived based on the defence-in-depth concept, fundamental safety functions, the consideration of internal and external hazards, and all phases of facility lifetime. Recommendations for a legal framework as well as a framework for safety requirements and guides have been developed. These results constitute the technical deliverable D3: "Recommendations for the implementation of fusion specific safety requirements".

In Task 4 an action plan for the implementation of the recommendations defined in Task 3 has been established. The action plan includes 13 steps including to establish a common European legal framework, to agree on fusion specific implementation of the defence in depth concept and a graded approach, to develop safety requirements for fusion facilities using D-T plasmas and magnetic confinement, to establish a system for operating experience feedback, to develop safety guides for different aspects of fusion specific technical aspects, and to develop fusion specific codes and standards. For each step of the action plan, the involved stakeholders have been identified. These results constitute the technical deliverable D4: "Action plan for the development of a targeted and proportionate regulatory framework".

In Task 5 a workshop with international fusion experts has been carried out. The workshop "On the applicability of the regulatory framework for nuclear facilities to fusion facilities. Towards a specific regulatory framework for fusion facilities" took place from 22nd – 24th November 2021 in an online format. Altogether 43 participants from licensees, research organisations, technical safety organizations, international organisations and national regulators took part at the workshop. Proceedings of the workshop have been distributed.

2 PROJECT DESCRIPTION

Existing nuclear regulatory frameworks are traditionally based on fission reactors and address the specific risks of such facilities. Fusion reactors are based on a completely different nuclear process and pose specific risks different from those risks imposed by fission processes. The following differences can be emphasized:

- Use of strong magnetic fields;
- Use of very high electric currents and voltages;
- Plasma operation and plasma transient (disruption);
- Handling and monitoring of large quantities of tritium;
- Risk of mobilizable activation products, dust from the plasma facing components, tritium, and sputtering products;
- Use of cryogenic systems;
- Presence of potential toxic or hazardous materials such as beryllium.

The objective of this study is to consider the specific risks of fusion facilities and to develop a skeletal structure of a regulatory framework specific for fusion facilities. It will be analysed to identify which parts of the existing framework for fission facilities is considered important for fusion facilities. Some of the requirements could be directly adopted to fusion facilities, need to be adapted to the specifics of fusion facilities, or even requires development of new requirements. To form the basis for a fusion specific regulatory framework, this study performs a survey of regulatory approaches in selected countries working on fusion facilities like DEMO and ITER to show how regulation of fusion facilities is envisaged.

The goal of this study is to achieve the following results:

- Overview of international approaches to fusion regulation and existing safety requirements for fusion facilities;
- Screening of existing international safety requirements and analysis of their applicability to fusion facilities;
- Identification of fusion specific safety issues;
- Proposal of a safety concept for fusion facilities and overarching safety requirements;
- Proposal of an action plan to establish a fusion specific regulatory framework.

This report focuses on fusion facilities with magnetic confinement using tritium, such as the current design proposals of the DEMO facility concept.

3 DESCRIPTION OF INTERNATIONAL APPROACHES TO FUSION REGULATION, LIST OF EXISTING GUIDANCE SYSTEMS, AND ASSESSMENT OF THEIR RELEVANCE FOR FUTURE FUSION FACILITIES

At present, no country has a dedicated comprehensive fusion-specific regulatory framework for the whole lifecycle from siting to decommissioning of fusion facilities. Safety requirements applied to fusion facilities are based primarily on experience with activities related to fission facilities. Therefore, in a first step approaches in different countries were screened and collected. In a second step the relevance of these approaches for future fusion facilities was assessed and the gaps in the transferability of the existing approaches to future fusion facilities is pointed out.

3.1 Screening and collection of information on international approaches for fusion regulation

A screening of the open literature has been conducted to identify countries, which either already operate fusion facilities, are constructing such facilities or have been done assessments of their current regulation status and its applicability to fusion facilities. The focus has been set on facilities which include tritium as fuel for the fusion process.

In the UK there exists already a fusion facility (JET) which used and will use again tritium. In France the ITER project has been licensed for construction which will also use tritium. USA, China, and Korea have plans for future fusion facilities and work has been published on how to apply the current nuclear regulation on a fusion facility as well as the gaps of the regulation. In Germany no plans for such a fusion facility exists. Nevertheless, the application of the current regulation for an advanced fusion experiment is summarized here as well as the main results of a published review of the safety concept for fusion reactor concepts and transferability of the German nuclear fission regulation to potential fusion power plants.

For other countries no detailed publication has been found. In the following, the regulatory approaches in selected countries in Europe (France, UK, Germany, Russia), America (USA) and Asia (China, Korea) are described.

3.1.1 France

Information about the current regulatory approach to fusion facilities in France is based on the Decree to license “The nuclear facility called ITER” /MINI 12/, the presentations by M. Gandolin at the 1st Joint IAEA ITER Technical Meeting on Safety and Radiation Protection for Fusion Reactors /GAND 20/and by J. Elbez Uzan at the Virtual Public Forum A Regulatory Framework for Fusion /ELBE 20/.

Until now, France has no fusion-specific safety and licensing regulations. National laws and European directives are of the highest legal effect. Decrees and orders of the government are also of legal effect. Technical regulatory decisions by the nuclear regulator ASN (Autorité de Sûreté Nucléaire) and by government certifications and individual resolutions by the ASN are also legally binding. Non legally binding are the ASN guides and ASN Fundamental Safety Rule.

Laws

Applicable laws to fusion facilities include the Labour Code, the Public Health Code and the Environmental Code for the most significant ones. These laws constitute an “integrated” legal system /GAND 20/ preventing or managing all risks, covering major risks and detrimental effects, are applicable to radioactive as well as non-radioactive risks, and cover the public, environmental protection and worker safety.

Basic principles of the Environmental Code are: a) prevention, b) precaution, c) follow the polluter pays principle, and d) require the participation of the public. The Public Health Code defines the key principles for radiation protection, namely justification, optimization, and limitation to the exposure to radiation.

In /GAND 20/ the different steps required by the laws to license a civil land-based nuclear installation such as the ITER facility (“INB”, Installation Nucléaire de Base) are described. These steps were an assessment of the preliminary safety design options, involving the regulator ASN and requiring a preliminary version of the Preliminary Safety Reports (RPrS, Rapport Préliminaire de Sûreté), an application for the authorization to create an INB sent to the responsible minister, the review and assessment of the application by the ministries, the regulator and TSO, an public enquiry performed by the local authority, public participation by e.g. a local committee and other European countries, the issuing of the license by a decree of the Prime Minister and Minister in charge.

In France INB are defined as nuclear reactors, fuel cycle facilities, storage facilities, most powerful electron and ion accelerators and all facilities using significant amount of nuclides according to the article L.593-2 of the Environmental Code. Therefore, INB regime is mainly based on their nuclides inventory and activity and is not limited to facilities with fissile materials. According to the Decree 2007-830 of 11 May 2007, a facility authorized to contain more than 27 g of tritium needs an "INB" license. With a specific activity of 0,36 PBq/g, this amount of tritium corresponds to an activity of about 10 PBq, which is a factor of 10 million of the exemption value as defined in 2013/59/Euratom. Compared to this, the IAEA dangerous quantity of radioactive material (D-value) for tritium is set to 2 PBq /IAEA 06b/. The D-value is, according to the IAEA, that quantity of radioactive material, which, if uncontrolled, could result in the death of an exposed individual or a permanent injury that decreases that person's quality of life.

Decrees and technical regulatory resolutions

The fusion facility ITER, which is currently being constructed in the south of France, is licensed according to the Environment Code /MINI 12/ as INB, because it is authorized to contain up to 4 kg of tritium on the whole facility including about 1 kg in the VV /MIN 12/. ITER has been licensed as INB number 174 by the decree /MINI 12/ in 2012.

The decree /MINI 12/ contains two Articles. Article 1 specifies the plant boundaries of the INB and a list of its buildings. Article 2 rules the characteristics of the facility. These include limits for the radioactive inventory, the main fusion specific parameters like fusion power and plasma current, the fuel (deuterium and tritium), the successive operational phases, accident prevention including postulate initiating events and basic safety functions, i.e. management of confinement and limiting exposure to ionising radiation, protecting the facility against internal and external hazards, and the operation of the facility. The latter specifies different hold points before a certain operational phase like commissioning, operation and dismantling of the facility. The decree includes also rules for the handling of beryllium.

The decree /MINI 12/ is implemented by several decisions by the regulator (ASN Decision no 2013-DC-0379 of 12 November 2013, modified by ASN Decision 2015-DC-0529 of 22 October 2015, modified by ASN Decision 2017-DC-0601 of 24 August 2017). According to /GAND 20/ these decisions include over sixty additional requirements. These cover the design of the facility, several so-called hold points during construction and before the commissioning of different operational phases. Also included are requirements for regular information for the regulator and the public, an emergency preparedness and response plan, and radiation protection measures.

Outlook

The technical safety organisation IRSN (Institut de Radioprotection et de Sûreté Nucléaire), who has been and is supporting the French regulator ASN during the licensing process of ITER published a report about "Safety and radiation protection considerations for demonstration reactor that follow the ITER facility" /PERR 17/. The following points were identified, which should be examined in the design phase of a fusion power demonstration plant:

- The residual heat removal during operation of the plasma, and during transfer and storage on site;
- The radiation exposure risks for the operators/during maintenance;
- The types of accidents to be considered for safety analyses;
- The radioactive releases into the environment under normal operation;
- How to limit the amount of tritium on site;
- The waste produced.

So very likely, these points would be considered in future licensing processes for post-ITER fusion facilities in France.

3.1.2 United Kingdom

The UK does not have a specific regulatory framework for fusion facilities. The existing experimental fusion facilities at Culham are not licensed as a nuclear site, but similar to other non-nuclear licensed sites e.g. major industrial users, universities, hospitals, research facilities. Therefore, the competent regulatory authority is not the Office for Nuclear Regulation (ONR), but the Health & Safety Executive (HSE) and Environment Agency (EA).

According to /HSE 20/, the operator of the fusion facilities (e.g. JET) at Culham, UKAEA (United Kingdom Atomic Energy Authority), has chosen to apply a nuclear style staged safety case approach to assessment of hazards to provide justification that risk is ALARA / ALARP (As Low As Reasonably Achievable / As Low As Reasonably Practicable). The relevant safety standards of the UKAEA which are applied are equivalent to those which would apply to a site licensed under the Nuclear Installations Act 1965. According to /BEL 99/the Safety Assessment Principles (SAPs) for nuclear licensed site published by the Health and Safety Executive in 1992 were applied. The SAPs include both deterministic and probabilistic criteria. These were supplemented by internal UKAEA standards where appropriate.

Amongst other health and safety legislation, JET is also subject to the Ionising Radiations Regulations 1985 and the Radioactive Substances Act 1993 (RSA93) through which official approval for radioactive holdings and waste disposal is granted by the Environment Agency /BEL 99/. Additional regulations of relevance for fusion facilities are:

The Environmental Permitting (England and Wales) Regulations 2016

The Environmental Permitting Regulations (EPR) specify which activities require an environmental permit. These are collectively described as “regulated facilities”. There are currently twelve different kinds, or “classes” of regulated facility.

Schedule 23 of the EPR deals with radioactive substances activities involving the use of radioactive material, accumulation and disposal of radioactive waste, and security of radioactive material. According to schedule 23, radioactive substances activities, where an operator keeps or uses radioactive materials or receives, accumulates and disposes of radioactive waste, requires an environmental permit, unless it is outside the scope of regulation or exempted under the regulations.

Schedule 23 of the EPR replaces and repeals all of the RSA93, in England and Wales, which previously controlled the radiation exposure resulting from radioactive wastes entering the environment.

The Ionising Radiations Regulations 2017

The Ionising Radiations Regulations 2017 (IRR17) form the regulatory framework for the radiation protection. The IRR17 requires the appointment of Radiation Protection Supervisors and Advisers, control and restriction of exposure to ionising radiation (including dose limits), and a requirement for local rules such as controlled areas. It includes legal requirements for prior authorisation of the use of particle accelerators and x-ray machines. In the last revision of 2017, a graded approach for regulation has been implemented. The IRR17 also implement the Basic Safety Standards Directive 2013/59/Euratom /HSE 18/.

Towards Fusion Energy - The UK Government’s proposals for a regulatory framework for fusion energy of October 2021

In October 2021 the U.K. government publish the so call “Green Paper” about its proposal for a regulatory framework for fusion energy /SSBE 21/. It is based on the technical analysis by the United Kingdom Atomic Energy Authority /UKAE 21/. The UK governments proposal started a public consultancy process which should result in a regulatory framework. The proposal has the following five main objectives:

1. To maintain the existing regulatory approach to operational permitting of fusion facilities, given that the radiological hazard of a fusion power plant will be increased but not fundamentally different in type of hazard from current fusion research facilities

2. To clarify fusion’s status with regards to existing nuclear regulations and introduce new provisions necessary for the efficient, effective and proportionate regulation of fusion power plants

3. To work with the regulators to consider whether and how enhanced engagement and new guidance for fusion developers could help support the safe and rapid deployment and commercialisation of fusion energy technology

4. To keep related policy under review areas as fusion energy technology develops

5. To review the overall regulatory approach to fusion no less frequently than every 10 years, on the basis of the remaining uncertainties around the technologies involved in a fusion power plant

As under the existing regulation applied especially to JET, the proposal is to also regulate future fusion facilities of all sizes not as nuclear sites but as a radioactive substance activity. The current regulatory framework should be retained and developed further. This would mean that in England, the regulation of fusion facilities will continue to be led by the Environment Agency (EA) and the Health and Safety Executive (HSE). Fusion operators would not be required to obtain a nuclear site licence and so fusion facilities would not be required to be on a nuclear site regulated by the Office for Nuclear Regulation (ONR) /SSBE 21/. However, considering the uncertainties involved in fusion power plants, the U.K. government keeps the option to change the regulatory framework for fusion facilities if fusion design choices in the future involve a considerably higher degree of radiological hazard than is currently expected and will review its decision regularly.

The proposal also contains suggestions how the current regulation of fusion facilities should be extended and enhanced with, especially with more public engagement and guidance for developers.

3.1.3 Germany

In Germany, no special regulation for fusion facilities exists. The German Atomic Energy Act is not applicable, because it is limited to facilities using fissile material (e.g. uranium, plutonium). The largest German fusion facility, Wendelstein 7-X, is regulated using the German Radiation Protection Law, superseding the former German Radiation Protection Ordinance, applied at the time of the construction of it.

German Radiation Protection Law

According to § 12 and § 17 of the Radiation Protection Law, anyone who intends to operate a plasma equipment/facility which does exceed a local dose rate of 10 micro-sievert per hour at a distance of 10 cm from the walls of the area which is inaccessible for electrotechnical reasons during operation requires an operating license. An accelerator or plasma facility generating more than 10^{12} neutrons per second requires an additional construction license, because it is expected that this kind of facilities may result in radiation exposure of the surrounding population through direct or scattered radiation or through discharges of radioactive material. § 11 specifies the requirements to grant a construction license like trustworthiness of the license holder, the appointment of an adequate number of radiation protection officers, the limitation of the radiation exposure of the personal of the facility, the environment and the public, both during normal operation and design basis accidents.

Also covered by this law are the required documentation, the termination of the operation, and supervisory examinations.

3.1.4 Russia

If research of fusion reactions in high-temperature plasma based on non-radioactive isotopes of hydrogen with the use of charged particle accelerators is conducted, such a facility is licensed as "generating source of ionizing radiation". Issuance of licenses is carried out by the Russian Federal Service for Surveillance on Consumer Rights Protection and Human Wellbeing (Rospotrebnadzor) in compliance with the Federal law from May 4, 2011 No. 99-FZ "On Licensing of Certain Types of Activities" and the Decree of the Government of the Russian Federation from April 2, 2021 No. 278 "On Licensing of Activities in the field of Ionizing Radiation Sources Use (Generating) (Excluding Use of Such Sources in Medical Activities)".

If research of fusion reactions with tritium (for instance, D-T reactions) is conducted at a facility, a license authorizing to conduct activities with the use of radioactive substances in R&D and/or a license authorizing management of radioactive substances is issued. A facility will fall in this category if its tritium inventory is more than 0.2 g. Such licenses are issued by the Federal

Environmental, Industrial and Nuclear Supervision Service of Russia (Rostechnadzor) in accordance with the Federal law from November 21, 1995 No. 170-FZ "On Atomic Energy Use" and the Decree of the Government of the Russian Federation from March 29, 2013 No. 280 "On Licensing of Activities in the field of Atomic Energy Use".

3.1.5 USA

The USA has no specific regulatory framework for fusion facilities. However, in 1996, the Department of Energy (DOE) publishes the DOE Standard DOE-STD-6002-96 "Safety of Magnetic Fusion Facilities: Requirements" /DOE 96/ which provide requirements for developing design and operations envelopes to ensure safety of magnetic fusion facilities. This Standard was accompanied by DOE-STD-6003-96 "Safety of Magnetic Fusion Facilities: Guidance" /DOE 96b/ providing guidance for the implementation of the requirements identified in DOE-STD-6002-96.

The general U.S. legislation and regulations system on nuclear and radiation safety is structured as a three-level framework: the nuclear safety related Acts, the implementing Federal Regulations, and the supporting Regulatory Guides. Industry consensus standards related to the nuclear and radiation safety can also be made as part of the regulations by endorsing them in the Federal Regulations or Regulatory Guides or by including them into the licensing basis of a facility. /IAEA 11/

In 2009, the Nuclear Regulatory Commission (NRC) staff raised the question of the regulation of fusion-based power generator devices with the commissioners of the NRC /NRC 09/. The NRC staff focused on the definition of "utilization facility" in the Atomic Energy Act in their analysis and believed that this could incorporate fusion devices. The NRC staff recommended that the NRC establish jurisdiction over fusion facilities /ROMA 20/.

In response to the request of the staff, the Commission agreed to assert regulatory jurisdiction over fusion, but punted instructing the staff to leave the question on the issue of how to regulate fusion facilities alone until the technology developed further /ROMA 20/.

In 2018, the US Congress passed the Nuclear Energy Innovation and Modernization Act (NEIMA). This act (Sec. 103) requires the following: "For commercial advanced nuclear reactors, the NRC must

(1) establish stages within the licensing process;

(2) increase the use of risk-informed, performance-based licensing evaluation techniques and guidance; and

(3) establish by the end of 2027 a technology-inclusive regulatory framework that encourages greater technological innovation."

The act (sec. 3 para 1) defines the term 'advanced nuclear reactor' as a nuclear fission or fusion reactor, including a prototype plant (as defined in Sections 10 CFR 50.2 and 10 CFR 52.1), with significant improvements compared to commercial nuclear reactors under construction.

In 2021 NRC started a series of workshops to discuss the options for a regulatory framework for fusion energy systems. These discussions have been initiated due the emergence of several initiatives to develop commercial fusion facilities in the USA based on various technology options. The first workshop took place in January 2021, followed by a second workshop in March. A third workshop was scheduled for June 2021. A white paper /NRC 21/ was submitted to the Advisory Committee on Reactor Safeguards in May 2021. This paper discusses three options to regulate commercial fusion power plants in the USA in the future:

(1) Utilization Facility Approach

(2) Byproduct Material Approach

(3) Hybrid or New Approach

Developers and organizations affiliated with the fusion industry consider option (1) as overly restrictive as utilization facilities are usually fission-based facilities having a higher risk to the public compared with fusion facilities. Concerning option (2) NRC staff documents some concerns on the definition of the term "byproduct" in the USA regulations and if this definition covers all situations of radioactive materials within a fusion facility. Some concerns are expressed, that the risk of commercial fusion facilities may be higher compared with typical facilities licenced under the byproduct approach. Option (3) proposes a hybrid approach. This approach takes into account the risk of a proposed facility which governs the selection of the applicable regulations. The so-called fragmented approach selects the applicable framework -either the utilization facility model under 10 CFR Part 53 or the byproduct material model under 10 CFR Part 30- based on decision criteria to be defined. The consolidated approach requires the rulemaking of fusion specific regulations integrating a graded approach to allow the appropriate application of the stringency of requirements related to the hazard potential of the proposed fusion facility. Industry expressed a favour for option (3). Yet, no final decision has been made. NRC continues regularly exchange and discussions with the stakeholders.

Atomic Energy Act

In the US, the Atomic Energy Act of 1954, provides the legal framework for all regulation of civilian nuclear installations. This act lays down the general principles and concepts of the regulation of all civilian uses of radioactive materials and nuclear energy. It leaves the details to the nuclear regulator. Since 1974, the competent nuclear regulator is the NRC. The Atomic Energy Act empowers the NRC to establish by rule or order, and to enforce, such standards to govern these uses as "the Commission may deem necessary or desirable in order to protect health and safety and minimize danger to life or property". The NRC implements the Atomic Energy Act through regulations that are issued in accordance with the Administrative Procedure Act, a law that provides general rules and procedures for all Federal agencies, including the NRC /NRC 19/.

Under Section 274 of the Atomic Energy Act, the NRC may enter into an agreement with a State ("Agreement State") for discontinuance of the NRC's regulatory authority over some materials licensees within the State. The State must first show that its regulatory program is compatible with the NRC's and adequate to protect public health and safety. The NRC retains authority over, among other things, nuclear power plants within the State and exports from the State /NRC 20/. According to NRC, more than 20,000 active source, byproduct, and special nuclear materials licenses are in place in the United States. About a quarter are administered by the NRC, while the rest are administered by 37 Agreement States /NRC 20/.

The Atomic Energy Act controls both the civilian use of radioactive materials and facilities, including

- "byproduct materials": Examples of byproduct material are tritium (hydrogen-3), carbon-14, flourine-18, krypton-87, cobalt-57, and discrete sources of radium-226 /NRC 20/;
- "source material": uranium and thorium;
- "special nuclear material": enriched uranium and plutonium;
- facilities that use, produce, or incorporate radioactive materials;
- facilities that use atomic energy in such quantity as to be of significance to the national interest and public health.

In this context the term "atomic energy" means all forms of energy released in the course of nuclear fission or nuclear transformation.

The Atomic Energy Act, Chapter 10, Section 101, prohibits possession and operation of a production and utilization facility without a valid license issued by the NRC. Section 103, which applies to facilities for industrial or commercial purposes, also states that such licenses are subject to conditions that the NRC may establish by rule or regulation to carry out the purposes and provisions of the Atomic Energy Act /NRC 19/.

NRC Regulatory Framework

Regulation of the NRC fall under Title 10 of the Code of Federal Regulations (10 CFR). This title is split into various parts, addressing different aspects of the nuclear regulation framework. In the following some of the relevant parts for licensing and regulating nuclear material and facilities are addressed. In general, 10 CFR Part 50 addresses nuclear facilities, however it was developed for nuclear power reactors. 10 CFR 30 regulates radioactive material such as tritium, but not the facility itself.

10 CFR Part 20: Standards for Protection Against Radiation

The NRC regulations in 10 CFR Part 20 establish requirements for radiation protection for all NRC licensees. Each subpart of 10 CFR Part 20 addresses a specific area of radiation protection, such as occupational and public dose limits, posting, surveys, monitoring, waste disposal, and reporting requirements. /NRC 19/

10 CFR Part 30 – 37: Rules regulating byproduct material

Parts 30 to 37 of title 10 of the CFR cover the regulation of radioactive materials known as "byproduct materials". This includes basically all radioactive materials that are not fertile or fissile such as uranium, thorium and plutonium. Waste and tailings produced in the extraction or concentration of uranium and thorium are also not covered by 10 CFR Part 30 – 37, but in 10 CFR Part 40.

The framework of 10 CFR Part 30 – 37 addresses the regulation of the radioactive material, rather than the regulation of a facility. Therefore, fusion facilities are not explicitly addressed. However, according to reference /ROMA 20/ fusion experiments and other installations generating irradiation such as cyclotrons are already being licensed by NRC using in the framework of 10 CFR Part 30 – 37.

The regulation with respect to tritium is addressed at several sections of 10 CFR Part 30 – 37. § 30.55 for example addressed the so-called tritium reports.

According to reference /ELLI 20/, the framework of 10 CFR Part 20 and 10 CFT Part 30 has already been used to regulate the Phoenix fusion device located in Wisconsin under Wisconsin state authority.

10 CFR Parts 50 – 55: Rules regulating production and utilization facilities

Parts 50 to 55 of 10 CFR deal with the regulation of production and utilization facilities and are mainly addressing nuclear power reactors. However, the Atomic Energy Act defines "utilization facility" to include in general those facilities that use special nuclear material or atomic energy in such quantity as to be of significance to the common defence and security, or in such manner as to affect the health and safety of the public.

Most regulations of the framework of 10 CFR Part 50 were intended for nuclear power plants. Given the safety concerns with nuclear reactors and the high amount of nuclear materials stored in irradiated nuclear fuel that could impact the public if released these regulations are quite detailed and prescriptive. The regulations cover the different phase of a lifecycle of a nuclear power plant, i.e. licensing, operation, license renewal, decommissioning and termination of license, as well as technical aspects such as which standards are to be used.

In general, fusion facilities generate atomic energy as defined by the Atomic Energy Act. There are ongoing discussions between the different stakeholders in der U.S. if these facilities fall under fall under 10 CFR Parts 50 – 55, i. e. the rules regulating production and utilization facilities or if they should be regulated under 10 CFR Parts 30 – 37, i.e. the rules regulating byproduct material. The decision will depend on the evaluation if these facilities are of significance to the common defence and security, or in such manner as to affect the health and safety of the public.

Industry Standards

There is an ongoing effort within The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Committee on Construction of Nuclear Facility Components (Section III) to develop rules for the construction of fusion energy devices. The Standards Committee of Section III, Division 4 and its Subgroup on Fusion Energy Devices are developing these new fusion Code rules. A draft standard for trial use (ASME FE.1-2018) has been published in 2018 /ASME 18/ and is currently under revision. This code will probably change significantly in future updates.

These rules cover fusion-energy-related components such as vacuum vessels, cryostats, and superconducting magnet structures and the interactions of these components. Related support structures, including metallic and non-metallic materials, containment or confinement structures, and in-vessel components such as fusion-system piping, vessels, valves, pumps, and supports, are also covered. The rules contain requirements for materials, design, fabrication, testing, examination, inspection, certification, and stamping /ASME 18/.

3.1.6 China

Information about the current regulatory approach to fusion facilities is based on the publication by Shen et al. /SHEN 19/ and the presentation by Wang at the 1st Joint IAEA ITER Technical Meeting on Safety and Radiation Protection for Fusion Reactors /WANG 20/.

Until now, China has no fusion-specific safety and licensing regulations. There are two nuclear safety related laws (another one in the process of enactment), seven administrative regulations, 27 department rules, about 90 safety guides, and more than 180 technical documents.

National laws are of the highest legal effect. Administrative regulations are also of legal effect. Department rules and safety guides are approved and issued by various administrative departments under the state council. The technical documents are the lowest level, and they are approved by competent authorities, supporting the department rules and safety guides.

Laws

In China two laws are approved in the nuclear field: Law of the People’s Republic of China on Prevention and Control of Radioactive Pollution in 2003, and Nuclear Safety Act in 2017.

The laws are technology-neutral and the highest-level laws for nuclear facilities. According to the definition in the Nuclear Safety Act, the nuclear facilities according to article 2 include

- nuclear power plants,
- research reactors,
- nuclear fuel cycle facility,
- radioactive waste storage and treatment facilities,
- nuclear materials including U-235, U-233, Pu-239 and
- other nuclear materials that needed to be controlled by other administrative regulations.

Fusion facilities are not explicitly mentioned, and tritium is not clearly defined in the nuclear material list of the Nuclear Safety Act. But according to an administrative regulation on nuclear material control (HAF501) it belongs to nuclear material in China.

Licenses for construction of the facility and for the operation with tritium (having an inventory larger than the allowable limit of ~0.1 g) shall be obtained according to article 22 and article 38 of the Nuclear Safety Act. Proper security measures shall be adopted to ensure the security of the facility and the tritium inventory.

According to article 21 of the Law of the People’s Republic of China on Prevention and Control of Radioactive Pollution the design, construction, and operation of the facilities for the prevention and control of radioactive pollution shall be performed simultaneously. Therefore, the tritium plant and hot cell shall also be considered at the conceptual design phase.

According to article 8 of the Nuclear Safety Act, the enforced safety criteria shall be developed considering the future development of science and technology.

Both laws require that a restricted area shall be set, and on-site emergency response plan shall be developed in case of accidents.

According to article 18 and 19 of the Nuclear Safety Act, the routine release of liquid and gaseous effluents shall be controlled and monitored, the occupational radiation exposure and the radwaste production shall be well controlled and kept as low as reasonably achievable.

Administrative regulations

Nuclear safety management, administrative organization and its responsibilities and surveillance procedures, are stated in existing administrative regulations, which include civilian nuclear installations, nuclear materials, radioactive waste, civil nuclear safety equipment, radioactive materials transportation, nuclear accidents, etc. In /SHEN 19/ four professional administrative nuclear safety regulations have been identified as related to fusion facilities.

According to requirements of the Regulations on the Civil Nuclear Safety Supervision and Administration of Civil Nuclear Facilities (HAF001), a nuclear safety licensing system must be implemented in China, which is established and approved by National Nuclear Safety Administration. Correspondingly, a safety licensing system needs also to be adapted for fusion facilities from current regulations. For examples, a fusion facility can only be constructed after it is reviewed and the nuclear facility construction license has been issued. The commissioning can be started after approval to fuel/store tritium at the plant. The facility cannot be formally operated till the nuclear facilities operation license is obtained.

In HAF001, the "nuclear facilities" according to article 2 include:

- Nuclear Power Plant;
- Research Reactor, Test Reactor, Critical Facility;
- Nuclear Fuel Cycle Facility;
- Radwaste Treatment & Disposal Facility;
- Other Nuclear Facility Requiring Supervision.

According to this definition, fusion facilities may be treated as a Research Reactor or Other Nuclear Facility Requiring Supervision.

Activities in a fusion facility including possessing, using, producing, storing, transporting and disposing of tritium shall comply with the Regulations on Nuclear Materials Control (HAF501), Compiling Requirements for the Nuclear Material Control and Accounting Documents in Nuclear Material License Application (EJ/T 1095-1999). In general, the licensing of tritium is approved by National Nuclear Safety Administration and issued by State Administration of Science, Technology and Industry for National Defence of P. R. China.

According to the Regulation on the Safety Management of Radioactive Waste, waste of a fusion facility containing radionuclides or contaminated with radionuclides, with the density or specific activities of radionuclides higher than the clearance level determined by the state, which is expected to be no longer used, is treated as radioactive waste. This is consistent with the definition of radioactive waste in the IAEA Safety Glossary. Besides, for a fusion facility, various necessary measures should be taken to reduce the mass and volume of fusion radwaste as much as possible.

For the supervision and management of safety equipment of a fusion facility, it shall comply with the Regulations on the Supervision and Management of Civilian Nuclear Safety Equipment, especially for the vacuum vessel, which is the most important component of the first confinement barrier. In order to prevent the risk of nuclear proliferation and terrorism, equipment and technology export activities in a fusion facility, must obey the Regulations on Nuclear Export Control of P.R. China and the Regulations on Export Control of Nuclear Dual-Use Goods and Related Technology. Furthermore, operation and management of a fusion facility may also need adequate safeguard.

Department rules

Department rules regarding nuclear and radiation safety are approved and issued by the National Nuclear Safety Administration (CN NNSA). There exist one general rule series and nine specific rule series. The specific series includes rules about: Nuclear Power Plant, Research Reactor, Non-reactor Nuclear Fuel Cycle Facilities, Radioactive Waste Management, Nuclear Materials, Civil Nuclear Safety Equipment Supervision and Management, Transport of Radioactive Materials, Radioisotopes and Ray devices, and Radiation Environment. According to /SHEN 19/ none of these documents specifically address fusion.

According to /SHEN 19/, a fusion facility with a thermal power of about 1 GW is proposed to belong to the class III of research reactors (with the criteria of thermal power greater than 10 MW) following the implementation rules of the regulations on the safety supervision and administration of civil nuclear facilities. The department rules about research reactors can be divided into three categories: nuclear safety licensing systems and processes; design safety regulations; regulations on safe operation, radwaste, and nuclear materials. However, in 2019 the new Ministry of Ecology and Environment Order No. 8 "Regulations on safety permit procedures for nuclear power plants, research reactors, and nuclear fuel cycle facilities" was issued. In this Ministerial Order three classes of research reactors are newly defined in Article 33. Class I of research reactors has the highest thermal power with an upper limit of 300 MW. So, a fusion facility above this power level is probably out of scope of this regulation.

The nuclear safety licensing systems and processes include five major stages about siting, construction, commissioning, operation and decommissioning.

Design safety regulations include general nuclear safety objective, radiation protection objective and technical safety objective. According to /SHEN 19/ the Defence-in-Depth (DiD) implementation and multiple barriers design of a fusion facility would be largely different from the descriptions in HAF201-1995 due to the wide distribution of mobilizable radioactive materials. These regulations also cover topics like the number of barriers, the independence of systems and components at different DiD levels and the main safety functions. The operation of a fusion facility is supposed to comply with Regulations on Safe Operation of Research Reactors (HAF202), but the detailed regulations on safe operation need to be modified to fusion specifics.

Safety guides and technical documents

These are general references rather than binding requirements. The safety guides are guiding documents and recommended practices, which are categorized into ten series with regards to department rules. The technical documents are technical insights and recommendations of experts, some of which are translated from the International Atomic Energy Agency's technical reports. These safety guides and technical documents are correspondently becoming more specific than department rules and most of them are not suitable for fusion facilities. Some may be applicable, e.g. for the primary cooling systems for blankets, divertors and vacuum vessel or the classification and management of fusion radwaste. For fusion specific topic, the corresponding guides and technical document must be developed newly.

3.1.7 Korea

In Korea, no special regulation for fusion facilities exists. According to /KIM 20/ the largest existing Korean fusion facility KSTAR was licensed in the existing radiation regulation framework.

The Nuclear Safety Act is of the highest legal effect. Presidential Decrees covering administrative matters for the implementation of the law, Ordinances of the Prime Minster about procedures, methods, etc., technical and safety rules to enforce the law and decrees at the principal level, and notifications are also of legal effect. They are substituted by Technical Criteria like technical baseline requirements, descriptions of acceptable methods, conditions and specifications and technical guidelines. Industrial codes and standards are the lowest level /IAEA 20/.

According to /KIM 20/ it is the characteristics of the Korean nuclear safety law, that it is a "positive regulation" defining what is allowed and prohibit everything else.

Currently installations are classified either as nuclear reactors, i.e. power reactors with relevant facilities or research reactors, or radiation generators. For reactors a construction and an operation permit are necessary. For radiation generators like accelerators or the fusion facility KSTAR, a license to use is required.

In /KIM 20/ it was found that for reactors the currently required format for the licensing documents would not fit that of a fusion power plant. It would also be unclear, which technical codes and standards need to be applied in such a licensing process.

The licensing process for the fusion facility KSTAR doing deuterium fusion tests required seven years to be completed. In 2000 it started with a review as a next-gen. nuclear power facility, in 2001 the production design was completed, in 2002 consultation about the licensing started and the license application was submitted, between 2002 and 2007 the license application was complemented and 7 deliberations took place. At the end of 2007 the license of KSTAR as a “radiation generating device” was issued.

3.1.8 Japan

No open literature about the regulatory approach to fusion facilities or fusion power plants were found for Japan. Also, the regulation of fusion facilities or power plants is not part of the Agreement between Japan and EURATOM for the joint implementation of the “Broader Approach Activities” in the field of fusion energy research.

3.2 Review and compilation of international approaches on fusion regulation

In the explanations above, several countries were identified which have already regulated fusion facilities (e.g. UK, France, Germany) including ones that use or will use tritium as fusion fuel. In addition, in several countries such as the USA, China, and Korea there are ongoing discussions and developments on how to advance the regulation of fusion facilities with different stakeholders /CALL 20/.

In several countries the application of the regulation for nuclear power plants is limited to those facilities using fissile material (uranium, plutonium, thorium), e.g. Germany, China. This corresponds to the different legal definitions of the term “nuclear facility”. Some countries limit it strictly to the use of fissile material, whereas other countries apply the term to facilities using any kind of radioactive material. Therefore, in some countries the regulation for fission facilities and reactors is formally not applicable to fusion facilities.

However, the regulations for radiation facilities and radiation protection apply and form the basis for licensing. It can be concluded, that in those countries, there seems to be a gap for licensing larger fusion facilities or fusion power plants /CALL 20/ with respect to

- the higher radiological hazard potential of fusion facilities and fusion power plants compared to typical radiation facilities and

- the lower radiological hazard potential of fusion facilities and fusion power plants compared to fission power plants.

The licensing of the ITER and JET demonstrate that the existing regulations for facilities like research and/or commercial reactors as well as radiation generating facilities can be applied to fusion facilities. The two facilities differ significantly in the amount of tritium to be used and stored: ITER approximately 4 kg on site, JET experiment 15 g in the vacuum vessel /BOY 16/.The experiences gained in these licensing process will very likely provide important input for future licensing processes and requirements of fusion facilities. Also, the challenges identified during these processes need to be considered in future fusion regulations.

As discussed above, the licensing of JET and ITER followed different approaches (“other non-nuclear licensed sites” versus “basic nuclear installation” (INB)).

It was found that in different countries the thermal power of reactors/research reactors is used in graded approaches in the existing nuclear regulation. For fission reactors the thermal power is also an approximate measure for the radioactive inventory and therefore for a potential source term in the case of severe accidents.

If this approach is applied to fusion facilities, then they are not given benefits due to less radio-toxic inventory compared to fission facility. So, the original purpose of using the thermal power as measure is lost if applying it to fusion facilities.

Comparing the different regulations approaches for fusion facilities found in different countries on way to categorize them is if they are following in principle a prescriptive approach or goal-oriented approach. Examples for a prescriptive approach are the regulations in Germany, Korea, China and the U.S., examples for a goal-oriented approach are the regulations in France and the U.K.

In a prescriptive approach the regulation contains explicit requirements the licensee needs to fulfil. The requirements are based on the technology used for the facilities the regulation is foreseen to be applied to. The level of detail can go down to specific safety systems to be installed in the facility.

A goal-oriented regulation sets safety goals like the containment of the radioactive inventory. It is the task of the licensee to prove to the authorities that the chosen design and way to operate the facility fulfil the given goals.

Both approaches have advantages and disadvantages.

In **prescriptive approach** simplifies the tasks of the licensee and the authorities, if applied to an established technology. In this case the licensee knows exactly what is necessary to fulfil the requirements. Also, the authorities can simply check if the given requirements are fulfilled. But in case of new technologies and before the design of such facility, the regulations for this new technology must be developed first. This might be a challenging way to go, because for new technologies the implementations of it might develop over time and therefore, also the regulation needs to develop. This process might take a long time and requires a very detailed knowledge of the technology, especially also in the organisations responsible for the development of the regulation and for issuing the licenses.

In a **goal-oriented approach** the regulations are usually technology neutral. Therefore, a regulation regime following this approach is likely to be applicable to new technologies (such as fusion facilities or advanced fission reactors). This should allow a licensee to apply for a license of a facility based on new technologies. But this also bears the risk for the licensee that the authority might not accept the claim that a new solution fulfils certain goals if the licensee does not provide an appropriate evidence of his safety claims. Review and assessment of applications under a goal-oriented regulatory framework requires a more intense and deep technical review compared to a review against a prescriptive regulatory framework. As this approach allows for more flexibility, care need to be taken so that the review of the technology is thoroughly enough. In practice, this usually leads to a hybrid solution, that even goal-oriented approaches include some prescriptive elements to emphasize certain safety aspects important to protect workers, the public or the environment against harmful radiological consequences.

The examples of the licensing of ITER and JET has shown that a more goal-oriented regulations has already been successfully used to license these facilities.

Different stakeholders have already expressed their opinions about how to further advance the regulatory approaches for fusion facilities. Based on the review above and the published feedback of the various stakeholders such as IRSN /PERR 17/, ITER /GAND 20/, /ELBE 20/, and others /KIM 20/, /RAED 16/, the following aspects need to be considered during the development of future fusion regulation (this does not mean that these will exactly appear in a regulation):

- To develop the safety requirements for
  - the residual heat removal during the operation of the plasma and during transfer and storage of activated components on site;
  - the radiation exposure risks for the operators and for the workers during maintenance;
  - the types of accidents to be considered for safety analyses;
  - the radioactive releases into the environment under normal operation;
  - the limitation of the amount of tritium on site;
  - the minimization, handling and processing of the waste produced.
- To consider the different sources and zones in a fusion facility with radiation exposure risks, e.g. in the main building for the plasma operation, transport and storage of activated components, and the tritium fuel cycle processing rooms.
- To develop a graded approach to balance the stringency of the regulatory framework with the radiological hazards of different fusion facilities.
- To establish a safety concept identifying fundamental safety function specifically for fusion facilities. These fundamental safety functions need to be supplemented by supporting safety functions with respect to purposes of different fusion facilities. The fundamental safety functions will very likely include the confinement of the radioactive inventory. Depending on the design, cooling of activated components might also be needed to be considered.
- To consider newer developments in the fission regulation like events resulting from external hazards, e. g. earthquakes and flooding, or very rare man-made external hazards (e. g. the crash of a large airplane).

- To consider internal hazards and initiating events specific for fusion facilities which could jeopardize the confinement function not covered by existing regulations. These include the release of energy stored in the plasma and in the magnets, or dust hydrogen reaction.

- To consider the non-radioactive hazards of a fusion facility (e. g. beryllium hazards or the release cryogenics).

- To consider the approach of Design Extension Conditions (DEC) as emphasized in several IAEA Safety Standards. The authors of this study expect that this will become an issue for designers of fusion facilities and fusion power plants in the near future.

- To develop regulatory approaches for licensing and oversight of a first fusion power plant of its kind. As a first-of-a-kind facility there will be no licensing experience available. This also results in uncertainties in the design.

- To identify available or develop new codes and standards applicable for fusion facilities.

- To enhance the regulatory readiness by building up fusion specific capacity at the regulator and its Technical Safety Organisation (TSO).

- To include the international and national operational experience feedback of fusion facilities and also relevant feedback from other non-fusion industries.

- To consider the need for interim storage and potentially final disposal of activated materials.

This study shows that nuclear regulators in several countries are facing the challenge of regulating new fusion facilities. So far, each country has its own regulatory approach to fusion facilities mostly based on their grown regulation for fission facilities. International collaboration in establishing guidance on the licensing approach for fusion facility could be beneficial. Any future fusion regulation also needs to be accepted by society for which there are important factors like the needlessness of evacuation for all plant states including design extension conditions, and the avoidance of high level waste.

4 IDENTIFICATION OF SAFETY REQUIREMENTS SPECIFICALLY NEEDED FOR FUSION FACILITIES

This section focusses on the identification of safety requirements specifically needed for fusion facilities. The performed work comprises an in-depth technical analysis of systems, structures and components (SSCs) and their associated importance for safety and a screening of existing international regulatory documents which may be considered for developing a specific regulatory framework for future fusion facilities utilizing tritium with magnetic confinement.

4.1 Main differences between fission and fusion facilities

The main difference between nuclear fission and nuclear fission is the nuclear reaction itself. In case of fission a thermal or fast neutron interacts with a fissile nucleus. If the energy of the compound nucleus exceeds the fission barrier, the nucleus will fission into two (radioactive) fission fragments and 1-3 fast neutrons. These neutrons can again initiate fission reactions and cause a nuclear chain reaction, which has to be controlled in a nuclear power plant. The large amount of radioactive fission fragments generate heat in the nuclear reactor requiring an ensured removal of decay heat.

Nuclear fusion reactions allow light nuclei to collide and fuse together into a heavier nucleus whilst producing energy. Among the possible fusion reactions, involving the two heavy hydrogen isotopes deuterium and tritium is currently preferred for a fusion reactor:

$$\mathrm{D} + \mathrm{T} = {}^{4}\mathrm{He} + n + 17.586\ \mathrm{MeV}$$

The D-T reaction creates ^{4}He and a free neutron with kinetic energies of 3.5 MeV (the alpha particle) and 14.1 MeV respectively /ZOHU 17/. To control fusion reactions two most advanced approaches are under development:

- Inertial confinement

Small pellets containing a mixture of deuterium and tritium are compressed and heated by high-intensity laser light to initiate nuclear fusion reactions. The National Ignition Facility (NIF) located at the Lawrence Livermore National Laboratory in Livermore, California U.S. uses laser-driven inertial confinement fusion /NIF 20/.

- Magnetic confinement

  Fusion reactions in the form of a hot plasma of fusion fuel (deuterium and tritium) enclosed in a magnetic field are approached under required temperature and density to generate thermonuclear fusion power. In the tokamak configuration, a powerful magnetic field is generated by superconducting coils to confine plasma in the shape of a torus. The ITER facility using magnetic confinement will complete the construction phase in 2025 /ITER 20/. It will produce 500 MW of fusion power for long pulses of 400 ~ 600 s with 50 MW of injected heating power to the required temperatures. ITER will not convert the energy it produces as electricity, but as the first of all fusion experiments in history to produce net energy.

Another possible approach to control fusion reactors is the magnetic confinement in stellarator machines (e.g. Wendelstein 7-X machine in Greifswald, Germany with modular superconducting coils), but this technology is less developed than the analogous in tokamak machines /WEND 20/.

GRS, together with the Max Planck Institute for Plasma Physics, KIT and the Institute for Applied Ecology did a review of the safety concept for fusion reactor concepts and transferability of the German nuclear fission regulation to potential fusion power plants /RAED 16/. The main findings regarding the differences between fission and fusion with regard to safety and regulation topics are listed in the following. Additional findings learned accompanying fusion development are included as well:

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

- In a fission power plant by far the largest part of the inventory which could be released to the environment in case of an accident is stored inside the fuel rods. This inventory consists of fission products and actinides. The fuel rods are either located inside the reactor core or in the spent fuel pool. The cladding of the fuel rods forms the first barrier of the radioactive inventory. The decay of the fission products and actinides produces residual heat which has to be removed to avoid melting of the fuel. Also, special care has to be taken to avoid an un-wanted (re-)criticality of the fuel.

- The source terms are substantially different between fusion and fission. In fusion power plants possible sources for the release of radioactivity to the environment are tritium (located in the vacuum vessel, in the cooling system (depending on the plant design), and in the tritium cycle system (breeding blankets, tritium processing systems)) and activated materials (steel and tungsten dust in the vacuum vessel and, depending on the coolant possibly activated corrosion products, in the cooling system). Excursions of the fusion reaction rate can be excluded, and any abnormal event will immediately lead to a termination of the fusion reaction. From the neutronic and plasma physical point of view the fusion reaction is self-terminating, irrespective of any postulated internal and external event plant considered.

- The technical safety concept of nuclear fission plants is based on the safe confinement of the radioactive materials. This is implemented by the multiple confinements of the radioactive materials by barriers, supported by retention functions and the protection of the barriers. The retention functions are implemented by different measures and installations which are independent between the consecutive levels of defence. The integrity of the first and second barrier is furthermore important to ensure the cooling ability of the fuel. The barriers in nuclear fission consist of the fuel rod, the reactor coolant pressure boundary and the containment. The containment is enclosed by a reactor building. The reactor building has the fundamental function to protect the containment against loads from external events, including human-induced external hazards. Internal hazards such as plasma transients (disruption) and magnetic and electromagnetic risks are fusion specific.

- In fusion power plants the fuel itself is a plasma, there are no fuel elements that serve as a first barrier. Instead, the vacuum vessel with the corresponding components (ducts, penetrations, in-vessel components cooling system) acts as a first confinement barrier. To ensure the integrity of this barrier under certain accidental conditions, passive measures to reduce possible overpressure are planned (pressure suppression pool or expansion volumes, depending on the actual plant design). The radiologic inventory of a fusion power plant is not concentrated in the fuel. Different potential sources have to be taken into account (mainly the vacuum vessel, the coolant system, the detritiation systems, the fuel cycle systems and the hot cells). For all of these, different barriers are to be implemented in a fusion power plant. With respect to other inventories, other first barriers do exist, like the piping of the heat transfer systems, tritium process lines or the hot cells. A second confinement barrier is provided by the reactor building itself. To ensure sufficient confinement integrity, the use of the different retention functions is necessary in fusion power plants like filtering and detritiation.

- The first commercial nuclear power plants have been connected to the grid in the second half of the 1950s. Since then, a considerable amount of operational experience has been collected. At the end of 2011, there were 435 nuclear power plants in operation world-wide. The total operational experience of commercial nuclear power plants at that date was about 13.000 reactor years /IAEA 12/. A system to evaluate the operational experience exists. E.g., the nuclear power plants in Germany faced between 100 and 500 reportable events each. For nuclear fusion facilities, in comparison only minor operational experience is available up to now. Since the 1960s, several experimental fusion facilities are in operation to test and verify the fundamental aspects of fusion physics and technology. Still, no experimental facility or prototype reactor exists today that is operating under the actual boundary conditions of a fusion power plant.

- In a fission power plant, the decay heat from used fuel elements has to be removed by active systems to avoid eventual fuel damage. Thus, the continuous cooling of the reactor core and the spent fuel pool is an important safety function in fission power plants. The current regulatory framework specifies requirements for all levels of defence to ensure the cooling of the fuel. In in a fusion power plant, the activated structures of the blanket, the divertor and other in-vessel components produce decay heat which can reach significant levels. However, in the blanket of a fusion power plant the decay heat per volume is orders of magnitude lower than in the fuel rods of a fission reactor. For a detailed design, analyses are necessary to show that any local decay heat production does not endanger the integrity of the first barrier.

- In fission reactors, for certain parts of the piping the component integrity is guaranteed by applying the "leak-before-break concept". The underlying safety importance is related to the fact, that in a case of a fast opening large break in the cooling system, dynamic forces may occur, such that the geometry of the core components cannot be guaranteed anymore. However, the integrity of the core geometry is required to ensure the coolability of the fuel. This effect is important if water (or other liquids) with high pressure is used for cooling. At the time of the study /RAED 16/, no information was found of the application of the "leak-before-break concept". Nevertheless, for a fusion power plant, especially if water with high pressure is used for cooling, either it would have to be demonstrated, that the pressure waves induced by a fast large break Loss of Coolant Accident (LOCA) would not challenge the integrity of barriers and the coolability of the reactor in a not tolerable way, or the "leak-before-break concept" would have to be implemented in fusion power plants.

- The comparison between the safety concepts for fusion and fission showed that the fundamental safety function "confinement of the radioactive materials" can be transferred directly in a methodical way. For a Fusion Power Plant (FPP) this fundamental safety function is based on both, physical barriers as well as on active retention functions. After the termination of the fusion process, residual heat is produced by the activated materials. Correspondingly, the fundamental safety function "cooling" is also applicable to fusion. The analyses performed so far have shown that in the case of an adequate design of a FPP the residual heat can be solely removed by passive means. For a fission power plant, the fundamental safety function "reactivity control" should prevent power excursion, guarantee that the fission process can be stopped, and re-criticality is prevented. The first aspect is not transferable to fusion, because such power excursions are excluded due to the physical nature of the fusion process. The requirement for the ability to terminate the power production can be applied in principle to a fusion power plant. It is fulfilled by the inherent features. In a FPP, it is by physical nature not necessary to consider re-criticality. Plasma control and inherent feature of the plasma are fusion specific.

- As in the safety concept of fission, postulated single initiating and multiple failure events, as well as severe plant states of a FPP are assigned to different levels of defence, covering the range from normal operation to very rare events. The assignment is based on probabilistic criteria and the possible radiological consequences. In a FPP measures and installations are foreseen to guarantee the compliance with radiological criteria. The measures and installations are based on inherent physical principles, and passive and active safety systems. For a FPP, the criteria for the measures and installations on the different levels of defence are not yet as detailed as for a fission power plant. The safety analyses for fusion performed so far have focused on plant-internal events. For these events in an adequately designed fusion power plant, only relying on inherent and passive safety features, the analyses showed that there will be no need for an evacuation outside the plant. Together with the development of more detailed plant concepts also events resulting from external hazards, e. g. earthquakes and flooding, or very rare man-made external hazards (e. g. the crash of a large airplane) have to be discussed in more detail.

- Radioactive wastes of FPP are significantly different from wastes of NPP. The fusion wastes contain large quantities of tritium, and different radioisotopes. Contaminated and tritiated fusion wastes have impact on the recycling and clearance processes. High-level waste is not expected to be produced in an FPP.

4.2 Assessment of safety issue for specific fusion SSCs

In this section, the characteristics of fusion systems with a focus on their safety aspects are described. The tokamak machine represents the most mature line of fusion reactor development. The DEMO concept is based on the "tokamak" principle that a fusion reaction taking place in a hot plasma, controlled by a magnetic confinement. It will bring fusion research to the threshold of a prototype fusion reactor beyond ITER. For specific aspects, examples are provided using the concepts of ITER and EU DEMO. Currently, different conceptual DEMO projects are under consideration. Detailed descriptions of the systems are provided in ANNEX B.

4.2.1 Magnet system

The magnetic energy stored in the magnet system has an impact on the confinement in accident situations /JIN 17/. The huge amount of magnetic energy accumulated in the superconducting coils has to be evacuated outside the coils and the tokamak building in case of malfunctions or coil failures. The safety risks associated with the magnets originate from quench development without energy discharge, and short circuit of the Toroidal Field (TF) coils and consequent arcing towards confinement barriers and release of 4 K helium. The credible magnet system failures under normal or abnormal conditions (including earthquake) must not cause damage to the confinement barriers.

A quench is caused by abnormal termination of magnet operation when the coil temperature rises above the superconductivity threshold that the windings suddenly develop a finite resistance. The circulating current passing through this elevated coil resistance creates heat, which leads to a sudden, explosive boil-off of liquid helium.

For example, fast discharge units (FDUs) applied in ITER provide protection of superconductive magnet system /FRID 11/, /SONG 11/. In case of failure, in particular quench in a superconducting coil, FDU provides fast and safe energy discharge from the magnetic system, which breaks the coil supply loop and provides energy dissipation in high energy resistors. In ITER, the TF FDUs are classified as items important to safety and perform the safety function of protection of vacuum vessel for confinement and limiting exposure. Also, for EU-DEMO, FDU studies are ongoing.

4.2.2 Vacuum Vessel and Vacuum Vessel Pressure Suppression System

The vacuum vessel (VV) provides the first confinement barrier against radioactive and hazardous inventories in the first confinement system. The entire vacuum boundary provides the confinement function as well, including seals, feedthroughs, (ceramic) windows, bellows etc. They are more vulnerable to failure than the vessel itself. The in-wall shielding plates of the VV provide the safety function of limitation of exposure to ionizing and electromagnetic radiation.

To ensure the structural integrity, the vessel (main vessel and ports) is designed with respect to allowable stress limits that it complies with the support function. Due to the loads of the VV boundary, the VV pressure of DEMO is limited to 200 kPa.

The cooling water flows in the interspace between the VV walls to remove heat deposited during plasma operation or decay heat in the in-vessel components (IVCs) in the event of a loss of coolant accident (LOCA) or loss of flow accident (LOFA) to those components via the decay heat removal system (DHRS) (see Section 4.2.6).

The relevant postulated initiation events (PIEs) identified for the VV are not only the VVA1 in Table 11 of Section 5.3.1.3, but also rupture / leak in the internal shell of the VV which leads to in-vessel LOCA of the VV primary heat transfer system (PHTS). The mobilizable source terms (tritium, dust, activated corrosion products (ACPs), sputtering products) located in the VV can be transported to rooms, building, and even released to the environment in accident case. The tritium inventory is controlled by the plasma chamber tritium monitoring.

In ITER, the main vessel, the ports and the VV supports are classified as SIC-1.

The vacuum vessel pressure suppression system (VVPSS) is the second confinement barrier of the first confinement system. It mitigates the VV pressure and confines source terms inventories in the VV and coolant inventories from the PHTSs in the event of first wall (FW) failure, divertor (DIV) plasma-facing unit (PFU) failure or feeding pipe failure (mainly in the upper port). Pressure (shock) waves or heat liberated by deflagration in the VVPSS must not adversely affect any other system. The VVPSS mitigates hydrogen explosion risk with the passive autocatalytic recombiners (PAR).

4.2.3 Breeding blanket system and concepts

4.2.3.1 Water cooled concept

The architecture of the water-cooled lithium lead breeding blanket (WCLL BB, see Annex B.5.1) presents several challenges in terms of safety. These challenges can be depicted through the analysis of the PIE identified in /PINN 17/. The most complex challenges arise from the PbLi loop:

- PbLi has an exothermic reaction with water, so the contact between the coolant (water) and the PbLi flow should be avoided /EBOL 16/. To cope with this issue, the PbLi piping system uses double-wall pipes and can be drained in case of accident /DELN 19/.
- Pb under neutron irradiation is transmuted in ^{210}Po and ^{203}Hg /MERR 14/. Both species are toxic, radioactive and volatile: ^{210}Po is an alpha-emitter, while ^{203}Hg is a beta-emitter /STAN 20/. The release of ^{210}Po and ^{203}Hg is especially dangerous during maintenance /PERR 16/.
- PbLi is highly corrosive, and this behaviour is exacerbated by the magnetic field created by the magnets /CORR 16/. This problem has two faces: the loss of integrity of the pipes, and the creation of clogging inside the pipes. To cope with this issue, protective layers, low flow velocities, and a purification system are adopted in EU DEMO /DELN 19/ /FEDE 19a/.
- PbLi freezes at about 234 °C, so the loop should be emptied during maintenance, or dedicated systems (heaters, pipe insulation, etc.) should be installed to avoid this phenomenon. The current strategy to cope with this issue is unknown.

Another important safety concern is the tritium inventory in the BB (PbLi loop). As a matter of fact, the releases of tritium toward the outer environment in both normal and off-normal conditions must be minimised. To cope with this issue the BB is designed with different permeation barriers to reduce as much as possible the tritium content outside the PbLi loop /DELN 19/.

The relevant PIEs identified for the WCLL are not only LFV1, LFB1, as well as FM1, LMO1, LMO2 and LMO3 for the PbLi loop in Table 11, but also LOFA due to internal clogging in the FW/BZ cooling channels, leak of FW or a sealing weld in the breeding zone (BZ), or local loss of heat sink.

The reliability block diagram (RBD) analysis for the WCLL BB/PHTS has been performed in /PINN 20/. The mean operational availability (OA) of 10.72% and mean inherent availability (IA) of 16.99% resulted for the WCLL BB are far below the targets. To enhance the reliability and availability of the WCLL BB its design must be improved deeply at first.

Concerning the BB solutions investigated outside Europe, the following safety issues can be listed /LING 20/:

- The activation of the coolant loop and the turbine in the concepts adopting water in typical boiling water reactor (BWR) conditions;

- The typical issues of super-critical water (corrosion, stresses on the pipes, etc.) /HIRO 07/.

- The issues with uranium for the concepts adopting a U-Zr alloy as breeding material. For this concept, several issues are still open, with the most challenging being: a) the strategy to control the criticality of the system; b) cope with proliferation issues; c), the production of actinides and more in general the production of high-level wastes; d) the decay heat removal during off-normal conditions.

4.2.3.2 Helium cooled concept

Radioactive source terms of the helium-cooled pebble bed (HCPB) blanket concept have been identified as tritium, activation products, dust, and neutron sputtering products, which have to be confined in the VV.

The plasma-facing component (PFC) carries plasma-wall interaction leading to dust generation, temperature resistance, chemical reactions (W-steam, W-air) resulting H_2 production in accident case. If the FW temperature exceeds the temperature limit of e.g. 1000 °C, the FW fails. This temperature is indeed less than EUROFER melting temperature (1325 to 1530 °C), but the EUROFER97 yield strength decreases to 100 MPa at 700 °C /GORL 19/, so plastic deformations can lead to a failure. The FW has the function of neutron shielding.

The relevant PIEs identified for the HCPB are not only LFV1, LFB1 in Table 11 of Section 5.3.1.3, but also LOFA due to internal clogging in the FW/BZ cooling channels, leak of FW or a sealing weld in the BZ, or local loss of heat sink.

The RBD analysis for the HCPB BB/PHTS has been performed in /PINN 20/. The mean OA of 14.66% and mean IA of 23.23% resulted for the HCPB BB are far below the targets. To enhance the reliability and availability of the HCPB BB its design must be improved deeply at first.

The activity and decay heat inventories have been assessed for the HCPB DEMO blankets in /PARK 21/. The concentration of uranium impurity in $Be_{12}Ti$ blocks above 0.1% results in the significant increase of the activity, which has impact on the reprocessing and storage of the waste management strategy.

4.2.4 Limiter

The limiters are physical barriers to protect the BB and the VV from high heat fluxes during incidental conditions. According to the description given in Annex B.5.3, both cooled and uncooled solutions seem under investigation. The main safety issues affecting the limiters should be:

- Ageing and activation due to irradiation;

- Tritium permeation in the PFC;

- Tritium permeation and transport in the cooling circuit (only for actively-cooled limiters);

- Corrosion in the primary system (only for actively-cooled limiters);

- Coolant and corrosion products activation due to neutron irradiation (only for actively-cooled limiters);

At the same time, the presence of the limiters inside the VV poses safety issues to the VV itself and the other area of the plants occupied by the coolant system. These issues are:

- Dust production in the VV in case of severe damage;
- Release of the water inventory in case of LOCA. This event can occur only for actively-cooled limiters, and it can affect the VV, the VV port, the cryostat, and more in general all the areas crossed by the coolant circuit;
- LOFA with consequent damage of the PFC.

For actively-cooled limiter, the safety requirements should be defined choosing if the coolant circuit is a second barrier to the release of the radioactive inventory or not.

4.2.5 Divertor

The divertor contributes to thermal shielding and neutron shielding of the VV. The divertor should withstand the thermal stress resulted from the temperature gradient of materials during the plasma operation phase or baking phase. The divertor structure must withstand electromagnetic forces caused by the halo and eddy current interaction with high magnetic field, especially in case of plasma vertical displacement events (VDE).

The divertor has to sustain very high heat and particle fluxes arising from plasma (up to 20 MW/m²) with proper cooling system to ensure its structure and lifetime. The PFC carries plasma-wall interaction leading to dust generation. Erosion measurement should be implemented in the divertor zone to monitor the dust production. Chemical reactions between steam and tungsten can lead to H_2 production.

The relevant PIEs identified for the divertor are not only LDV1 in Table 11 of Section 5.3.1.3, but also LOFA due to internal clogging in the divertor cassette cooling channels, leak of divertor cassette, or local loss of heat sink.

Activity and decay heat inventories have been assessed for the previous divertor models related to different BB concepts of DEMO baseline 2015 in /TIDI 20/. The PFCs and inner shell structure in divertor cassette exhibit highest activities and decay heats within 1 year of irradiation. In piping layer, ^{63}Ni and ^{60}Co are major contributors after 10 years of cooling. ^{56}Mn and ^{55}Fe are the key activity contributors in divertor cassette body.

4.2.6 Primary Heat Transfer System (PHTS)

All PHTSs including the primary side of heat exchangers (HXs) provide the first confinement barrier in the first confinement system for tritium and ACPs entrained in coolant outside the VV. Leak tight integrity must be maintained during all system states. Material selection of the PHTS has impact on the level of tritium permeation and diffusion into the cooling loop, ACPs, and water activation products. The need of safety isolation valves, which are considered at the interface to the VV to limit the coolant inventory in the accident case, is under discussion. Control valves may be used to regulate coolant flow during the dwell time.

The relevant PIE identified for the VV-PHTS is ex-vessel LOCA due to large rupture in the coolant manifold feeder in the PHTS. The DHRS is attached to the VV-PHTS that the VV is cooled down also in emergency conditions /CIAT 19/. It performs a supporting safety function. An emergency HX and a safety grade pump are installed in both VV-PHTS loops. In an emergency, when the other PHTSs are unavailable, the DHRS transfers the decay heat via radiation and conduction between BB and VV walls from the VV to the chilled water system (CHWS). The DHRS complies with fully redundant criterion that in case of the loss of one of two independent VV-PHTS loops, it is possible to rely on the DHRS attached to the intact loop. The low-flow pump is powered by the emergency diesel generator.

The relevant PIEs identified for the BB-PHTS of the HCPB are not only LBO1 and LBO3 in Table 11 of Section 5.3.1.3, but also small rupture of primary cooling loop in the cooling manifold feeder.

The relevant PIEs identified for the BB-PHTS of the WCLL are not only FB1, FF1, LBO1, and LBO3 in Table 11, but also small rupture of BZ primary cooling loop in the cooling manifold feeder, out-vessel LOCAs due to large rupture in the coolant manifold feeder / small rupture / large rupture of tubes in a primary HX in the FW-PHTS.

The relevant PIEs identified for the DIV-PHTS are not only FD1 and LDO1 in Table 11, but also out-vessel LOCA due to large rupture of tubes in a primary heat exchanger of the PHTS.

4.2.7 Balance of Plant (BoP)

Currently safety function is not identified for BoP components after the safety isolation valves on the secondary side of the HX. The most relevant PIE identified for the BoP is HA99 in Table 11. HA99 is the most severe PIE from the safety point of view because all primary cooling circuits integrated with PFCs are affected in the induced accident sequences.

4.2.8 Cryostat system

The cryostat is a structure surrounding the VV and the magnets, and according to the available data it should be operated below atmospheric pressure and at about 200 K in all DEMO concepts /CIUP 20/. In EU DEMO, the cryostat is not a self-supporting structure to reduce the amount of steel, but the pedestal supports both the VV and the TF coils. For this reason, the pedestal must be designed to withstand the impulsive loads coming from incidents.

As stated in Annex B.9, during normal operations the cryostat is not considered as a barrier in EU DEMO, but it is in CFETR /WANG 15/. Instead, during maintenance it forms the first barrier together with the Contamination Control Door /JIN 17/. To protect the integrity of the cryostat in case of overpressure, a venting system is installed. The cryostat is also considered as a possible heat sink to remove the decay heat in long-term fault scenarios /CIAT 19/. Studies performed for ITER demonstrated that the ingress of air in the cryostat is sufficient to allow a passive removal of the heat coming from the magnet structures and the cryostat wall /TAYL 14/.

The design characteristics of the cryostat prevents the formation of ozone, and they also provide: a) a barrier to limit the formation of ice on the internal structures, and b) a barrier between the air and the hydrogen formed during accidents.

According to /PINN 17/ the sole PIE affecting the cryostat is the loss of vacuum due to the ingress of gas. Although, the loss of vacuum could be also triggered by the ingress of liquids, such as water from the primary system of the WCLL BB or the limiter(s) cooling loop(s).

Considering the characteristics of the cryostat, the safety requirements should be measured according to its classification as barrier or not. If deemed as a barrier, stringent requirements should be put in place to ensure its integrity during accidents. Instead, if not deemed as barrier the requirements for the cryostat could be limited to: a) the capability of the pedestal to support the VV and the magnets, and b) the capability to avoid, or slow-down, the contact between the air and the hydrogen produced during accidents. In ITER, the cryostat is considered as a SIC-1 component /SEKA 15/.

4.2.9 Cryoplant and Cryodistribution

The cryoplant and the cryodistribution system produces and delivers cryogenic helium to the magnets, to the neutral beam injection (NBI) cell, to the thermal shields and the cryopumps. As stated in Annex B.10, the two systems have not been yet designed for any DEMO design. Safety considerations can be also performed looking at the ITER cryoplant and cryodistribution. In /HENR 07/ an analysis of the PIE affecting these two systems is performed. From this analysis, the following safety issues can be listed:

- Corrosion in the piping;
- Transport and activation of the corrosion products and hazardous materials. Tritium permeation is also an issue for the coolant routed to the cryopumps in the VV /TAYL 14/;
- Loss of insulation in one or more pipes, or more in general the inability to deliver He at nominal conditions to the magnets with the possible consequence of magnet quench /TAYL 14/;
- Release of cryogenic helium in the cryostat, in the NBI cell, in the tokamak building, or in the cryoplant. This event is capable of producing a combination of over-pressure and under-temperature loads, and it is of special concern if occurs in a room/area containing radioactive inventory /TAYL 14/;

In ITER, since the cryogenic system has to guarantee the protection of the magnets, the redundancy of some sub-systems is required /HENR 07/:

- The cooling loop for the TS;
- The pumping capacity to ensure the vacuum conditions also during He leaks;
- The associated safety sensors.

As for ITER, the cryodistribution for EU DEMO has no interfaces with the vacuum vessel, hence it is not a barrier to the release of radioactive substances. Figure 30 shows that the cryoplant is built near the tokamak building, i.e. far from any activated area. Except for the streams reaching the VV cryopumps, the cryodistribution also operates in non-activated or slightly activated areas of the plant. For these reasons, the same safety requirements defined for ITER could be applied to DEMO as well, such as /HENR 07/:

- Compliancy with the technical specifications and ISO standards for the single components /MONN 15/;
- Sufficient redundancy of both the cryoplant and the cryodistribution to ensure the safe control of the magnets in case of fast and soft PSD, and in case of magnet's quench;
- Capability to isolate portions of the cryodistribution to reduce the amount of released helium in case of leakages, and to preserve the magnet's cooling.

The relevant PIE identified for the cryoplant and the cryodistribution system is VCG1 in Table 11 due to large ingress of cryogenic fluid into the cryostat.

4.2.10 Tritium Extraction and Removal System

As the duty of the tritium extraction and removal (TER) system is to extract the tritium from the breeding blankets, the operation efficiency of the TER systems has direct impact on the tritium inventory in the breeding blankets and consequently on the tritium permeation in the interfacing systems, particularly in the cooling systems. Therefore, studies on both TER tritium inventory and permeation in the processes materials and the need for implementing Tritium Permeation Barriers (TPB) are of high priority and are investigated on various R&D programs and laboratories.

4.2.10.1 Water cooled concept

The main topics that have impact on the tritium inventory in the PbLi, permeation in the cooling system and consequently with impact on the working area and the environment, are as followings:

Magnetohydrodynamic effect on flowing PbLi

Among the neutronics requirements, in the case of the WCLL concept a stringent need for the reliable operation is to achieve the tritium self-sufficiency taking into account the various plant-internal losses that occur during DEMO operation. Therefore, the Magnetohydrodynamic (MHD) effect is thoroughly investigated, and its effect is addressed in the development of a tritium transport model aiming to correctly predict the amount of tritium inside the liquid metal, the retention in structural steels and the permeated flux into the coolant. These predictions play the fundamental role in guaranteeing tritium self-sufficiency in the fusion reactor (that means, tritium breeding ratio higher than one) and safety both for the workers and for the environment. In the EU R&D program devoted to DEMO, a 3D tritium transport model at breeder unit level for WCLL breeding blanket has been developed, considering the PbLi velocity field, implementing the buoyancy effect in the governing equations of PbLi flowing and including magnetic field. The available modelling results are related to the 2018 reference geometry of the WCLL. In this reference geometry the combined effect of buoyancies and MHD on the transport of tritium has been analysed. For this particular case a benchmark study on the MHD and magnetoconvection has been carried out and the effect of magnetic field has been implemented in a simplified case study of tritium transport. In particular, one-twelfth of the module including one coolant tube of the reference geometry have been studied for three different Hartmann number equal to 0, 5000, and 10000, both for the pure MHD and the magneto-convection cases. The velocity field of the PbLi has been modelled taking into account the MHD effect and compared to the magnetoconvection case. In the MHD case, the hydrodynamics profile is modified into the typical M-shape, with peak velocities increasing as Ha number increases. The tritium permeation rate is reduced by 30% from Ha = 0 to Ha = 10000. In the magneto-convective case, the velocity profile is strongly decreased with respect to the case Ha = 0, with peak velocities shifted in the toroidal direction and more asymmetric with respect to the MHD case. The tritium permeation rate is about twice the one in the no-buoyancy case for the case Ha = 5000 and Ha = 10000. The modelling showed highest permeation rate in the Eurofer baffle, less tritium is retained in the Eurofer pipes that implies a lower inventory (about 1/3) in the water /CIRO 20/. Nevertheless, the concern of the MHD effect on tritium inventory and permeation to the cooling shall be very carefully addressed during the design of a fusion device and experimental validation is compulsory as a prerequisite for the final design decision.

Fundamental properties of interaction tritium gas with PbLi

Regardless of the technology selected for the tritium extraction and recovery from the PbLi, the Sievert's constant related to interaction gas-liquid (T_2-PbLi) plays the crucial role in the lowest tritium partial pressure that can be achieved in the liquid PbLi, the efficiency of the process, and consequently on the tritium inventory in the PbLi. Several experiments aiming to measure the solubility of hydrogen isotopes in PbLi have been carried out all over the world but significant uncertainties remain /CALD 09/. Presently, the Sieverts' constant measurements are spread in two order of magnitude range. From the design point of view of the TER system this is the main drawback and sensitivity analysis are carried out aiming to overcome this issue. The uncertainties on Sieverts' constant, in addition to the impact of MHD effect on tritium transport in PbLi, represent the main safety topics related to the WCLL concept. Highest tritium inventory in the PbLi implies high tritium permeation rates in the cooling water and consequently in the environment. Though, it is not expected a significant isotopic effect on Sieverts' constant, measurements using tritium have not been carried out until now worldwide. In the EU DEMO program, a tritium infrastructure for measuring Sievert's constant has been developed at KIT-TLK (KIT Tritium Labor Karlsruhe) and the first measurements with tritium at partial pressures and temperatures that are relevant for DEMO are expected in 2022-2023.

Tritium permeation from PbLi processing components and removal/detritiation of the cooling system

The preliminary calculations of tritium permeation from the TER system into the Tokamak building showed that the amount of permeated tritium is not acceptable both from radiological hazard and from the technological challenges associated with the recovery process in view of providing self-sufficiency. Therefore, the permeation reduction factor (PRF) is considered as a working parameter in the evaluation of various options for the design of the TER system and of the detritiation systems of the cooling water, coolant purification system (CPS). Concerning the actual EU DEMO scenario related to WCLL operation it is considered a PRF of 100 and an off-line CPS configuration in the coolant system. Within these considerations, the permeation rates from breeder to coolant are moderate due to the permeation barriers (~400 mg/day). The tritium content in the cooling water never reach the saturation since there is no tritium sink in this sub-system (e.g. no CPS). As a result, tritium concentration and inventory in PbLi reach an asymptotic value after 1 day but the concentration and inventory in water grow linearly during the whole reactor lifetime (Figure 1).

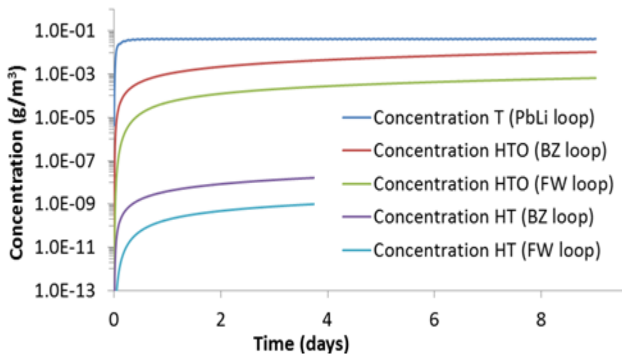

Figure 1 Tritium inventory in the loops /SPAG 20/

The highest source of permeation losses is found to be located in the PbLi piping system (~150 mg/day). As far as tritium permeation from the water-cooling pipes is concerned, it is considered that very small amount of HT and T_2 molecules are solved in water (most of tritium is in form of HTO) and therefore less permeation is found in the SG, the IHX and water pipes. On the other hand, removing of HT and T_2 from the cooling water is less challenging compared with the detritiation of the cooling water, meaning extraction of tritium from HTO form.

Therefore, various cases are evaluated in view of identifying and quantifying the tritium sources aiming to minimize the tritium inventory in various auxiliaries and consequently mitigating the release into the environment. In the EU DEMO program, the followings approaches are considered:

- Including an in-line CPS that process a small amount of water; this has a major impact on tritium concentration and inventory in water as represents a sink in the coolant system. In this case the tritium concentration will no longer grow linearly in time but reaches a steady state concentration.

- Enhancement of the TER efficiency as this parameter directly affects the tritium concentration in PbLi with direct impact on all model outcomes. Decreasing the tritium content in the PbLi will decrease the tritium concentration gradient in the solid materials that is translated into lower permeation rate from breeder to coolant and less inventory both in the steel and the water.

- Quantifying the effect of tritium permeation from the FW of the BB. The preliminary evaluation showed an approximatively 20 mg/day tritium permeation in the FW coolant loop, meaning an increase in the tritium inventories in the cooling loop by approximately 66%.

- Enhancement of the efficiency of the tritium permeation barriers with strong impact on general tritium transport mechanism. Presently in the EU DEMO project, a PRF factor of 100 is used as reference. Increasing the PRF by a factor 10 (from 100 to 1,000) the tritium permeation rate and the tritium inventories in the steels and water will decrees by the same factor. However, since the permeation losses are much smaller than the extraction rate in the TER, the tritium inventory in PbLi does not change significantly (less than 0.2%).

- Evaluation of the impact of hydrogen content in the water, mainly due to the radiolysis, on the tritium transport phenomena. Two counter effects have been identified: Hydrogen have an impact on the chemical equilibrium between the H_2, HT, T_2, HTO and H_2O molecules and on the other hand favours permeation via the recombination of HT molecules in the steel-water interfaces. The modelling results predict that increasing the hydrogen molar fraction (from 8 ppm to 100 ppm), increases the permeation rate from PbLi to water in the blanket only a small amount (~0.1%). However, it increases the permeation rate in the SG and IHX significantly (~75%).

4.2.10.2 Helium cooled concept

Due to high temperature operation of the breeding blanket, the tritium permeation issue is more stringent addressed in the HCPB concept. The TER system of the HCPB is in fact a helium purging loop that shall extract tritium from the pebbles beds followed by tritium removal from the purge gas. The references technologies for tritium removal from the purge gas are working at ambient temperatures or even lower, 77 K in some approaches, which will not promote tritium permeation from its tritium processing components. The main parameter that is related to the tritium permeation from the purge gas (TER) into the He coolant system is its partial pressure. From the design and operation of the TER system the tritium partial pressure is given by the gas purge flow rate. At the present purge gas flow rates, used in various DEMO designs, the tritium inventory in the purge loop is below 1 g of tritium, and the tritium partial pressure can be maintained at around 1.0 Pa /CRIS 20b/. Increasing of the purge gas flow rate has a crucial impact on the operational costs of the TER and therefore a trade-off between the investment and operation costs of the TER system and the amount of tritium permeation in the He coolant system shall be primary achieved. Nevertheless, developments on mitigation on tritium permeation in the cooling gas are of high priority together with the activities related to the cooling purification system. These topics are addressed on various R&D programs and laboratories in the following ways.

- Increasing the HTO content in the purge gas aiming to limit the HT contribution on the total amount of tritium extracted from the pebbles beds. There are still uncertainties related to the mechanism of tritium extraction from pebbles beds both from the modelling point of view and also from the experimental data base and validation. The preliminary calculations, based on conservative approach meaning HTO contribution around 1%

compared with HT, showed that after less than 100 hours of operation the pebble bed inventory is stable around a stationary value of approximatively 30 g T_2. This means that after this transient period, all the newly generated tritium is extracted and permeated in the different parts of the breeder (cooling plates, coolant, beryllium pebble bed, Li - ceramic pebbles). The tritium permeated inside the coolant, including the cooling plates, is estimated at approximately 0.8 g/day that represents less than 0.4% of the total tritium generated.

- The efficiency of the TER plays also a major role in achieving a low tritium partial pressure in the purge gas. The variation in extraction efficiency would strongly affect all the parameters related to the tritium migration in the entire HCPB configuration. In the reference case, the permeation figures from above are related to this, the TER efficiency was considered of 90%. In the case of TER efficiency decreasing down to 70% the amount of permeated tritium may be triple.

- Sensitivity analysis is on the performances of the CPS on the tritium permeated from the purge gas. Without a CPS the tritium partial pressure in the cooling gas equals the one from the purge gas. The necessity of a CPS is carefully addressed, and the driving parameter is the HT partial pressure in the purge gas. A lower HT content in the purge gas may exclude the CPS of the He cooling system. Nevertheless, there are technologies for tritium removal from the cooling gas and the trade-off between the tritium operation limit in the cooling system and operational will conduct the design.

- Minimization of tritium content in the neutron multiplier containing beryllium.

Based on the preliminary design status on EU DEMO, it is estimated that HCPB breeder blanket design seems to encounter less permeation issues, confirming its robustness in terms of tritium radiological risk.

4.2.11 Tritium, fuelling and vacuum system

The most important safety function related to the tritium processing systems, including fuelling and vacuum systems is related to the confinement and to the minimization of the tritium inventory. Tritium is retained in the vacuum vessel, on surfaces, permeated into structural materials, and absorbed into dust. It is also present in the entire fuel cycle equipment including pumping systems, fuelling systems and the fuel processing plant. The breeder blankets with their tritium extraction system represents another tritium inventory as far trapping into materials is concerned, and tritium will also contaminate remote maintenance systems used to replace in-vessel components. These components represent a significant tritium inventory into the active maintenance facility (AMF) where they are stored, maintained and decontaminated in view of mitigating the tritium release into the environment and tritium recovery.

During ITER and DEMO operation, due to the plasma–wall interaction, significant amount of particles/dust in sizes ranging from nanometers to tens of microns will be generated. The dust properties, especially their ability to be covered by a thin oxide electrostatic insulating layer, and surface topology deeply affect their tritium inventory. Consequently, physico-chemical properties specific to tritiated tungsten particles and consequence on particle behaviour in the facility and environment started to be carefully assessed. For size-relevant tungsten particles, the measured tritium inventory is ~10 GBq/g. However, it varies with the particle specific surface area. Due to tritium beta decay and the oxide-insulating layer, dust exhibits a positive electrostatic self-charging. For a 5 µm particle in diameter with a 10 GBq/g tritium inventory, self-charging rate could lead to $5.5{\cdot}10^4$ elementary electric charges per day. These electrostatic properties could change the adhesion of dust on walls. In the case of a single particle, the adhesion will be reinforced due to image and dielectric forces. However, if the tritiated particle is part of an aggregate, the adhesion remains unknown. Due to the limited free path of the β emission in material, the tritium inventory carried by airborne particles cannot be measured in real time by conventional continuous radioactive aerosols monitors, and a new measurement strategy is needed for atmospheric surveillance in the workplace and of facility exhaust. Toxicity studies dealing with exposure to untritiated/tritiated tungsten particles of 100 nm have been undertaken. It was shown that these particles are rapidly dissolved in biologic media. Finally, after collection, dust must be confined to avoid its spreading into the environment. Therefore, different technical solutions are under development /GRIS 19/.

During the design of the DEMO Tritium Plant, the minimization of tritium inventory is addressed along all systems covering the main functions such as:

a) receiving, accountancy and long-term storage of tritium shipped from external sources;

b) fuel gas supplying to the tokamak and the NBIs with required DT and D2 compositions, flow rate, pressure, duration, burn-dwell cycle;

c) processing of gas streams from torus vacuum pumps and NBI pumps;

d) processing of gas stream from the in-situ tritium recovery from co-deposited materials accumulated in the vacuum vessel, and ex-situ tritium recovery from in-vessel components handled in the hot cell.

In case of ITER, the largest tritium inventory is in the Isotope Separation System (ISS) based on cryogenic distillation and an upper limit of 700 g of tritium is considered as reference for the design /CRIS 07/. The large amount of hydrogen/deuterium/tritium inventory in the ISS represents an additional hazard as far as explosion is concerned with impact on the building design. As a result of the concept of Direct Internal Recycling (DIR) in the inner fuel cycle of DEMO, the tritium inventory is significantly reduced as only a small part is directed to the ISS. Nevertheless, the hydrogen/deuterium/tritium inventory in the ISS shall be carefully addressed in DEMO due to the large amount of tritiated hydrogen/deuterium gas supplied from the Water Detritiation System (WDS). Therefore, the detritiation technics including those from the hot cell facility shall be further developed in view of the amount of tritiated water minimization. Presently, it is very challenging to estimate the capacity of the DEMO WDS and input from ITER operation is required.

As far as tritium confinement in the DEMO Tritium Plant systems is concerned, the concept of double barriers including barrier with functional confinement can be successfully implemented. The high tritium permeation rate due to high temperature operation is limited in most of the systems, except the vacuum systems in the case of using mercury diffusion pumps. For this particular case, the detritiation technics of spent mercury shall be addressed and developed.

The relevant PIEs identified for the tritium plant systems are TGG1, TGO1, TGO3, THO1 and TWO1 in Table 11. Accident analyses for these PIEs will be investigated in the conceptual design phase.

4.2.12 Thermal shields

The thermal shields reduce the heat flowing to the magnets from the VV and the cryostat /KONC 17a/. Two shields are installed: one between the VV and the magnets, and one between the magnets and the cryostat. Thermal shields are cooled by the cryogenic lines (see Sections 4.2.9 and Annex B.10), so the risks associated with this aspect will not be re-discussed here. Apart from that, there are no risks associated with the thermal shields. Any system failures leading to increase of thermal shield temperature shall be detected to prevent pressurisation of the cryostat, heat-up of the coils and quench in the superconductors.

4.2.13 H&CD system

4.2.13.1 Neutral beam system

The NBI system is mostly aimed at heating up the plasma with the injection of D^0 neutrals, but it also contributes to the plasma rotation and the current drive /AGOS 20/. The NBI system is a large and complex system connected with the VV with one or more large openings. This disposition has three main drawbacks /MAIS 18/:

- Weakening of the VV mechanical resistance;
- Reduced shielding to the TF coils;
- Requirement of remote maintenance due to the activation of the whole NBI system.

Although, the NBI is the only system capable to inject high power in the plasma. Under the safety point of view, the NBI system poses several issues:

- The large opening(s) to connect the VV and the NBI system make it part of the first confinement barrier. Therefore, NBI has a safety function for confinement with a requirement to maintain and control vacuum at the interface with the VV Plasma Chamber (PC).

- The NBI system is operated in vacuum conditions. The loss of vacuum might lead to the release of D^0 neutrals, and so hydrogen explosion risks.

- The NBI system is completely activated, so it has a safety function to minimise the Shutdown Dose Rate (SDDR) for the components outside the bioshield with a requirement to provide appropriate shielding.

- RF drivers and magnetic fields are used to guide the beam, hence the minimization of the residual magnetic field during shutdown must be ensured.

- The grids used in the accelerator system require active cooling due to the high heat delivered by the impinging electrons. The safe handling of the coolant (water), the control of the coolant purity, and the avoidance of damage due to high temperatures are three safety issues /SONA 17/. The system should be also shielded to avoid the release of free electrons.

- The magnets installed in the surrounding of the ion source and the accelerator system are cooled by cryogenic liquid served by the cryodistribution (see Section 4.2.9) /GRIS 12/. Depending on the design of these magnets, the stresses triggered by a quench could be a safety issue.

- Each solution, proposed for the pumping system, aimed at reducing the losses in the accelerator section have different safety issues: cryopumps requires cryogenics fluids and are subject to the sudden release of D_2 captured in the getters. NEG pumps are not subjected to the sudden release of D_2, but they require a regular regeneration to release it /HEMS 19/. Mercury diffusion pumps have the same D_2 issue, plus additional issues such as mercury confinement, activation, and the safe waste processing at the end of the operative life.

- The NBI system requires a high voltage power supply. Two issues derive from this sub-system: the risks associated with the failure of the high voltage insulator, and the holding of high voltage in the accelerator section. The failure of the high voltage insulator might produce arching in the NBI, with the possible consequence of severe damage to the NBI itself /KOLB 96/. In turn, holding high voltage across gaps is one of the challenges in the design of any accelerator, but the understanding of the physics behind voltage holding is not fully understood. For ITER, the design was based on experience within the magnetic fusion energy neutral beam community. The length of the gaps was selected to hold a potential difference of a megavolt, and a maximum voltage gradient of about 3 kV/mm in relation to the electric field near the surfaces. These criteria were chosen to safely handle vacuum breakdown in the support structure and the accelerator /GRIS 12/.

- The NBI system must react quickly at the signals coming from the plasma D&C system. If a plasma disruption occurs and the NBI system is not deactivated, the incoming beam could damage the BB.

- In case of air or coolant ingress in the VV, the fast shutter valve should act promptly to avoid the transport of dust, ACPs, and tritium in the NBI cell.

In ITER, NBI Beam Source and Beam Line Vessels, and the transmission lines are identified as SIC-1 for confinement, and high voltage bushings as SIC-2.

4.2.13.2 EC system

The EC system is used for plasma heat-up and for the control of instabilities. Compared to the NBI system, the EC system has a main advantage: the complex components are installed far from radiological non-controlled areas, and only open wave guides and metallic mirrors are in the proximity of the plasma /MAIS 18/. As for the NBI system, the gyrotrons present risks associated with the RF sources, the residual magnetism, and the handling of a high voltage power supply. However, thanks to their position, eventual failure should not pose any concern regarding the safety of the VV. In turn, the presence of open wave guides poses a safety concern for the shielding of the magnets. In EU DEMO the development is ongoing to provide sufficient shielding behind the launchers. The components facing the plasma are also subjected to activation, as other IVCs.

Although, since the EC system crosses all the available barriers, the leak tightness of the connections must be ensured. The transmission lines are equipped with a mirror to avoid the flowing of dust, tritium, and ACPs. The failure of this mirror poses a safety concern for the release of hazardous substances. In EU DEMO the installation of the transmission lines inside the VV is proposed to ensure tritium segregation /AVRA 19/, while in CFETR a pumping system is in place /TANG 15/.

As for the other H&CD system, also the EC system is controlled by the plasma D&C system. Then, the EC system should be designed to promptly act at the request of the D&C system. In case of transients requiring a plasma shutdown, the EC system might be equipped with a dummy device onto which the beam can be directed, as it is in the CFETR plant. If arcing in the transmission line is detected, the D&C system should be also capable to turn off the power sources /TANG 15/.

The last safety concern is related to the active cooling of the mirrors guiding the beam through the transmission line and the launcher. In case of failure of the cooling system the mirrors might heat up and defocus the beam toward the transmission line or the launcher walls. At the same time, the release of coolant and the eventual failure of the window might lead to the ingress of water in the VV. In ITER, the launcher and the closure plate between the VV and the out-vessel waveguides are considered as SIC-1 components /SÁNC 18/, except the mirrors that are classified as Non-SIC components /SÁNC 20/.

4.2.13.3 IC Systems

The IC system is used to reach the flat-top phase and to control the plasma instabilities. As for the EC system, the launcher in the VV are considered as SIC components in ITER /LAMA 13/.The system presents most of the same safety risks described for the previous two systems:

- The RF source;
- The cooling of the mirrors and the risks associated with the release of the mirror's coolant;
- The handling of a high voltage power supply;
- Risk associated with arcing in the launcher system, or in the transmission lines /BOSI 01/;
- The prompt response to the signals coming from the plasma D&C system, and the possible needing to ensure the safe routing of RF waves to dummy components to prevent damages to the blanket;
- The integrity of the windows ensuring the tritium segregation. In this system the failure of the window is accompanied with air ingress in the VV due to the high pressure at which most of the IC system is operated;
- The activation of the IVCs.

Current developments in EU DEMO are aimed at reducing the risks associated at two additional safety issues /BADE 17/:

- The possible release of high Z impurity in the VV due to sputtering capable to lead to plasma disruptions;
- The possibility of arcs or breakdown in the antenna due to the high voltages.

4.2.14 Plasma Diagnostic & Control system

4.2.14.1 Plasma

Several transients can affect the stability of the plasma. Such scenarios can be divided into planned and unplanned transients depending if they are required to reach the flat-top (ramp-up and ramp-down) or not. The transients might lead to plasma disruptions, that can be mitigated or unmitigated depending if the PSD system is capable to safely terminate the plasma operations.

Generally, the plasma D&C try to mitigate and control the transients that could lead to a plasma disruption, so the PSD system can be considered as a last resort. Different strategies are adopted to control the transients:

- ELMs are controlled through the use of RMPs. However, this type of control is the source of the mode locking transients;
- Mode locking is controlled through the use of non-axisymmetric magnetic perturbations and the EC system /VOLP 15/;
- The grow of kink modes is controlled using conducting walls and external magnetic perturbations /MAUR 02/;
- The neo-classical tearing modes (NTM) are controlled by microwave injection;
- Upward and Downward VDEs are controlled through the magnetic field;
- H-L transition is controlled ensuring that the pressure in the plasma remains above a threshold;
- The divertor detachment is controlled through the injection of impurities and the use of the IC system. By converse, these two control strategies can be considered as the approach to avoid the divertor reattachment transient /LIPS 97/.

4.2.14.2 Diagnostic & Control

As discussed in the previous section and in Annex B.15, the D&C system is fundamental for the safe operation of DEMO. A flaw in the D&C system may reduce the performance of the machine or even lead to severe plasma disruptions. The D&C system should be capable to control the different transients and to reduce the likelihood of disruptions or any unrequested PSD. If disruptions occur, the integrity of the VV might be compromised and the radioactive substances could be released outside the first confinement barrier. Faults in the D&C system could also provide room for the propagation of accident sequences.

For ITER, /GONC 10/ states that similar regulatory requirements of the fission plant's D&C system are likely to be applied also for the D&C of fusion machines sustaining a burning plasma. For this reason, probably most of the diagnostic would need to be classified as SIC, and the redundancy of some (if not most) of the sub-systems should be ensured /BIEL 19/. Similar conclusions can be also drawn for DEMO.

According to the available data, the D&C system could be divided into four main parts: front-end components, cabling and antennae, hardware and software in charge of saving and processing the data, actuators for the control of machine. Each one of these parts has different safety issues:

- The front-end components, the cabling and antennae, and some actuators cross the VV, so their in-out VV interface should be designed to ensure the VV retention capabilities are not lost in case of damage to the D&C system.
- Some D&C systems require cooling, so the eventual loss of cooling within the VV or the BB, or the cryostat is a safety issue to be considered.
- The front-end components should be designed to ensure the robustness of their measurements under the harsh conditions faced in the VV. An eventual drift from the design specifications might lead to safety concerns on the capability to operate the plasma.
- Even if installed behind the BB, a portion of the cables is operated in an activated area. The same cables are also operated in a high-magnetic-field area. Specific counter-measures to avoid damaging or interferences in data exchange should be then put in place.
- The hardware and software solutions adopted to save and process data should be tested and validated to ensure the absence of faults. A comprehensive computer and information security protocol should be also in place.

- The actuators should act promptly and consistently according to the requests of the D&C system. Any improper action or fault is a safety issue.

If required, a PSD might be triggered. Then, the PSD system should follow robust safety requirements during the design, the building and the operative phases. Any fault in the PSD system could lead to the extensive damage of the IVCs, and so to the possible damage of the VV. Nevertheless, a PSD itself is a source of loads to the IVCs and the VV, so stringent protocols for each transient should be defined to ensure that fast PSDs are triggered only when strictly required.

4.2.15 Remote Maintenance system

Compared to a NPP, an extensive use of remote procedures to perform inspections and maintenance is expected in fusion reactors. The aim is to avoid the admittance of the personnel in areas having high dose rates and intense residual magnetic fields. To achieve this goal the extensive use of robots is foreseen. This approach is a good solution for the personnel safety, but it presents several safety challenges, especially in the time frame between the removal of an IVC from the VV and its deposition in the transport cask.

In /VALE 17/ a FFMECA analysis for the remote maintenance (RM) system is performed. In this work, the off-normal conditions are separated in two categories depending on the "recoverability" of the operations. If a failure triggers an interruption of the nominal operation, the "operation recover" is started. If the operation can't be recovered, the "cask rescue" with the deployment of additional systems is started. Summarizing the finding of /VALE 17/, /REED 18/, and the description provided in Annex B.16, the following safety risks can be listed:

- Losing of the VV confinement due to the opening of the upper, intermediate, and lower ports. During maintenance, only the tokamak building remains as available barrier to prevent the release of radioactive products. The unavailability of the VV barrier is especially dangerous in case of accident (falling of a robot in the VV, etc.). Then, the confinement barriers should be re-assigned.

- Cooling of the IVCs. Items not under maintenance are kept cooled through the cooling system and ventilation systems. In turn, items removed from their position are probably detached from any coolant loop. At any time, it must be ensured that no IVCs nor auxiliary systems (robotic arms) will suffer damage due to overheating. Up to now, it is not clear if the removal of the residual heat will be a safety function /PERR 16/. An incident where the IVCs are not properly cooled and with the transfer interrupted due to a failure is an accident of major concern.

- Mobilization of dust and release of gas and tritium. Opening the VV and removing the IVCs will probably lead to dust mobilization, and part of this dust will also remain attached to the removed IVCs. The release of gases and tritium is also a safety concern. Depending on the considered DEMO concept, the IVC cleaning is performed or in the cryostat or in the AMF. In this latter case, the transportation casks might become contaminated with dust, gases, or tritium. Suitable strategies must be then defined to manage this issue.

- Robust implementation of control, localization, navigation, and positioning systems, because any fault may trigger the interruption of nominal operations. As stated in ITER and fission nuclear reactors avoid placing Safety Functions on software, but due to the nature of the DEMO RM system this can't be excluded a priori. If Safety Functions are placed on software, a road map to regulatory acceptance including a qualification programme should be defined.

- Safe manoeuvrability of the IVC during the removal from the VV. The design of the system should allow the safe recover of the IVCs in case of incidents such as the falling of an IVC in the VV. If transported through cranes, it should be ensured that an eventual drop of the cask would not damage any of the systems in the VV or the tokamak building. If maintenance of more than one IVC at the same time, the system should be also capable avoiding collisions. The system should be also capable to withstand earthquakes and the loads coming from any external event.

- Reduced visibility in the VV. With no personnel in the VV or in the cryostat, the supervision of the activities can be done only through cameras. A suitable system of cameras should be in place to ensure that sufficient sightlines exist in the areas where the maintenance takes place. Proximity detectors may help to avoid collisions with walls etc.

- Activity inside the VV. Devices which are intended to help for in vessel maintenance have to withstand the radiation field inside the VV (e.g. the cameras have to be shielded in order to ensure a picture with acceptable quality).

- Redundancy and diversity of maintenance devices. The remote handling devices (e.g. cranes) have to be given at least twice with a different design. Cranes have to be able to savage crashed components inside VV, crane and robotic arms inclusive.

- Adequate shielding is required on removal, transport and storage of components following the safety function of Limit Exposure to Ionising Radiation. The identification of safe routes for the transportation casks is of utmost importance to ensure their easy recovery in case of problem/damage, and to avoid undue exposure to the components installed in the cryostat and the tokamak building.

- Safe connection between the tokamak building and the AMF. The connection should not hamper the retention of dust in the tokamak building, especially when the VV is open.

- Minimization of the fire risks associated with the remote cutting and welding systems mounted onto the robotic arms. The control system of these devices should be also capable to avoid the damage of neighbour elements while performing the required tasks.

4.2.16 Buildings

4.2.16.1 Tokamak building

The tokamak building has been proposed as the second confinement system in /JIN 17/ and it forms the final barrier between the tokamak and the environment. Release of radioactive source terms during normal operation and in accident case must be limited. Contamination with tritium, radioactive dust during in-vessel maintenance must be controlled, as well as fire, explosion, or flooding in the building. The building can be affected by neutron and/or gamma radiation emitted from hazard piping, in particular PFC cooling water systems and LiPb pipes. ORE during operation and maintenance must be minimised. Bioshield and main building structures have the safety function of limitation of exposure to ionizing and electromagnetic radiation. The very large tokamak building is divided into segregated rooms on the opposite side in order to limit radioactive and fluid enthalpy inventories and to protect redundant systems from common mode failures. As an example, the emergency diesel generators are each located on the opposite side of the tokamak building /GLIS 18/.

As the second barrier the building has to withstand loads such as missiles (internal hazard), aircraft crashes (external hazard.), too. Furthermore, a floating in case of external flooding has to be excluded by design.

Detritiation equipment is required to keep airborne tritium concentrations in cooling system equipment rooms and environmental releases within safe limits for both helium and water-cooling systems.

The lesson learned from ITER is that the nuclear regulator imposes significant requirements on the tokamak building that includes significant amounts of radioactivity /BACH 19/.

4.2.16.2 Tritium building

The tritium building has been proposed as the second confinement system in /JIN 17/ and it is the final barrier to the environment.

4.2.16.3 Machine Hall

In case of the use of rotation wheel for energy storage (as used in ASDEX for example), the machine hall has to be protected against a detached wheel.

Furthermore, the alignment of the wheel has to be chosen so that a detached wheel cannot hit the Tokamak building.

4.2.17 Active Waste Management Facility and Radwaste System

One of the top-level safety objectives of EU DEMO is the minimization of the radioactive waste hazards and volumes following the ALARA principle, not only during plant operation, but also during plant construction and waste processing. The IVC removed from the VV will be initially transported to the AMF where they will be cleaned, detritiated, and probably recycled (at least partially). Then, the waste arising from these procedures will be immobilized and disposed.

The radwaste system/building is the last step of these processes, but it’s unclear if the radwaste system will be only in charge of disposal or if it will be also in charge for their immobilization. Although, most of the wastes will be ILW and LLW /PORF 20/. The long-term disposal is foreseen only for: the PbLi because the recycling is not considered as a viable solution; and for the most harmful radionuclides of beryllium (3-4 kg) and tungsten (~60 tonnes). The PbLi inventory requiring disposal is not known for the WCLL concept, but for the HCLL concept it should be about 30,000 tonnes of lead and a maximum of 210 tonnes of lithium.

As stated in /PORF 20/, the waste management strategy is still an open issue, so a definition of the waste streams and the methods to process and store them is premature. For this reason, only generic safety issues can be defined:

- Combined opening of the VV and the tokamak building to remove the IVCs and to transfer the casks to the AMF. If this event occurs two of the main barriers to retain radioactive substances are lost. Then, additional barriers should be created, or the possibility to have both the VV and the tokamak building opened should be excluded.

- Minimization of the risks associated with hydrogen explosions in the AMF/hot cell before and during detritiation /PERR 16/.

The confinement of radiotoxic and dangerous substances, especially those volatile such as tritium. In any phase, the radioactive inventory should be monitored and controlled. No release to the outer environment should be permitted. To achieve this goal, the AMF and the radwaste building should be equipped with an HVAC system, and they should be operated at a lower pressure than the external environment;

- The minimization of the dose to the worker. A clear and comprehensive zoning of the AMF and the radwaste system should be performed, and a sufficient shielding should be installed in the areas where operators are supposed to work or transit. For the zoning of the AMF, the CFETR reactor is a step ahead compared to EU DEMO /GONG 15/. Four areas are identified: green, yellow, orange, and red areas. The green areas belong to the supervised zone, the yellow and orange areas to the specially regulated zones, and the red areas to the restricted zones.

- The management of the residual heat in the IVCs removed from the VV. It is probable that for some DEMO concepts both the blanket segments and the divertor cassette would require active cooling /PERR 16/.

4.2.18 Plant Electrical Supply

The plant electrical supply (PES) system is at a very early stage of development for each DEMO concept. So far, the development focused mostly on the adaptation of the solutions proposed for ITER. Although, some challenges have been already identified, and deep modifications start to be considered, as the removal of the thyristor converters in favour of additional coils to charge the magnets.

Compared to a PES of a NPP, those of a FPP presents additional challenges. During the dwell phases the PSS doesn’t provide energy to the electrical grid, but it is likely to be an “user”. This circumstance depends if the electrical storage available on the plant is sufficient to re-achieve the flat-top phase. Then, the PSS should be sufficiently versatile to ensure the safe operation in all the different conditions (power to the grid, power from the grid, power from the storage, and a mixing between power from the grid and from the storage). The compliance with the different options of the BB PHTS and PCS adds another layer of complexity to the PES /GAIO 20/.

Because of the scarce data available, only general safety issues can be defined:

- The same safety issues recognized for NPP in the IAEA Specific Safety Guide No. SSG-34 /IAEA 16f/ (Protection from external and internal hazards, minimization of the odds concerning station black-outs, availability of emergency supply systems as DC batteries and diesel generators, etc.);

- The control of the high active power peaks required during ramp-up and for plasma stabilization (in vertical direction) /GAIO 20/;

- The control of the reactive power if thyristor converters are adopted /GAIO 20/;

- The safe control of the energy released by the magnets during fast discharges /GAIO 20/;

- The safety concerns due to the operability, inspectability and maintainability of circuit breakers due to the harsh environmental conditions if installed in the tokamak building. Only a reduced access to the tokamak building is possible during normal plant operations, so robust strategies to ensure their safe serviceability should be put in place;

- The feeders penetration in the bioshield and busbars in the tokamak building should not hamper the capabilities of the confinement barriers to retain radioactive substances;

- The system should be also equipped with emergency power supplies to ensure a safe and continuous operation of the safety relevant systems in case of Loss of Offsite Power.

DEMO will require a recirculating electrical power in the range of 300-500 MW (almost one order of magnitude bigger than the recirculating power of an NPP) /CIAT 19/. This huge value, together with that relevant to the pulsed operation, necessitates a site close to very well interconnected electrical nodes of the EU grid. Mitigation solutions are being assessed with respect to ITER experience.

4.2.19 Plant Control System

The plant control system is not yet designed for EU DEMO. It is going to be constructed like in ITER /JOUR 20/ to consist three systems, as stated in Annex B.20: Control and data acquisition system, Central interlock system, and Central safety system.

4.3 Screening and categorisation of Council Directives / Directives of the European Parliament

At present, European Council Directives do not address specific requirements for fusion facilities, but place requirements which are generally applicable for all facilities.

The provisions in these Directives are mandatory for EU Member States and have to be transposed into national laws. The following Directives have been screened:

- Council Directive 2013/59/Euratom – European Basic Safety Standards;
- Council Directive 2009/71/Euratom – Nuclear Safety Directive;
- Directive 2011/70/EURATOM – Radioactive Waste and Spent Fuel Management;
- Commission Regulation (Euratom) No 302/2005 – Nuclear Safeguards;
- Council Directive 98/24/EC – Chemical agents at workplaces;
- Directive 2004/37/EC – Carcinogens or mutagens at workplaces;
- Directive 2013/35/EU – Worker exposure to electromagnetic fields;
- Directive 2011/65/EU – Hazardous substances in electrical and electronic equipment.

Whereas the first four Directives are specific for nuclear and radiation facilities and activities, the last four Directives address general requirements for conventional occupational health and safety.

4.3.1 Council Directive 2013/59/Euratom – European Basic Safety Standards

The European Basic Safety Standards Directive 2013/59/Euratom /EU 13/ of 5 December 2012 lays down uniform basic safety standards for protection of the health of individuals subject to occupational, medical and public exposures against the dangers arising from exposure to ionising radiation.

The Directive modernised European radiation protection legislation and consolidated the acquis at the time of Euratom radiation protection legislation into one single piece of legislation, merging five Directives and upgrading a recommendation to become legally binding. As a result, the following Directives were repealed:

- 89/618/Euratom: Council Directive on informing the general public about health protection measures to be applied and steps to be taken in the event of a radiological emergency;
- 90/641/Euratom: Council Directive on the operational protection of outside workers exposed to the risk of ionising radiation during their activities in controlled areas;
- 96/29/Euratom: Council Directive laying down basic safety standards for the protection of the health of workers and the general public against the dangers arising from ionising radiation;
- 97/43/Euratom: Council Directive on health protection of individuals against the dangers of ionising radiation in relation to medical exposure, and
- 2003/122/Euratom: Council Directive on the control of high-activity sealed radioactive sources and orphan sources.

The Directive respects the state-of-the-art of science and technology by taking into account the 2007 Recommendations of the International Commission on Radiological Protection (ICRP) /ICRP 07/ and harmonises numerical values with international standards, i.e. the IAEA Safety Standards Series No. GSR Part 3 /IAEA 14/. As a result of the consolidation process, the Directive was given a broader scope and a completely new structure improving coherence and clarity of both definitions and requirements.

Following the entry into force of the Directive on 6 February 2014, Member States were obliged to bring into force the laws, regulations and administrative provisions necessary to comply with the Directive by 6 February 2018.

The Directive establishes uniform basic safety standards for the protection of the health of individuals subject to occupational, medical and public exposures against the dangers arising from ionising radiation. This includes some fundamental principles of radiation protection, establishing legal systems including licensing and supervision, setting dose limits and reference values as well as emergency preparedness and response.

The requirements were not developed having in mind fusion facilities, but by providing basic safety standards they are applicable and have to be met according to occupational and public exposure. For fusion facilities this is in particular relevant in the context of handling and monitoring of large quantities of tritium, activation of irradiated materials and the connected risk of mobilizable activation products, dust from the plasma facing components and sputtering products.

Some of the general requirements need to be further elaborated in the national regulatory framework at subsidiary level taking into account the specifics of fusion facilities, for example the requirements for the licensing process.

Chapter I: Subject Matter and Scope

The Directive applies to any planned, existing or emergency exposure situation which involves a risk from exposure to ionising radiation which cannot be disregarded from a radiation protection point of view or with regard to the environment in view of long-term human health protection which include fusion facilities.

Chapter II: Definitions

For the purpose of the Directive, a set of definitions are defined.

Chapter III: System of Radiation Protection

The Directive requires the Member States to establish legal requirements and an appropriate regime of regulatory control which, for all exposure situations, reflect a system of radiation protection based on the fundamental radiation protection principles of justification, optimisation and dose limitation. Especially the fundamental radiation protection principles are fully applicable to fusion facilities.

Chapter IV: Requirements for Radiation Protection Education, Training and Information

The Directive requires general responsibilities for the provision of appropriate radiation protection education, training and information to all individuals whose tasks require specific competences in radiation protection by establishing an adequate legislative and administrative framework.

Chapter V: Justification and regulatory control of practices

Justification is one of the fundamental principles in radiation protection requiring that any practice resulting in exposure to ionising radiation is justified before being done, i.e. providing more good than harm. The practices are required to be subject to a graded approach of regulatory control for the purpose of radiation protection, by way of notification, authorisation and appropriate inspections. It has to be commensurate with the magnitude and likelihood of exposures resulting from the practice, and commensurate with the impact that regulatory control may have in reducing such exposures or improving radiological safety. Some practices may be exempted from regulatory control in compliance with general exemption criteria.

Chapter VI: Occupational exposures

The general requirements laid down for occupational exposures including responsibilities, operational protection, arrangements and classification of workplaces, radiological surveillance and individual monitoring, medical surveillance are directly applicable to fusion facilities.

Chapter VII: Medical Exposures

Medical exposures do not apply to fusion facilities.

Chapter VIII: Public Exposures

This chapter lays down the requirements for public exposure which has to be met for fusion facilities. This ensures operational protection of members of the public in normal circumstances and include inter alia the examination and approval of the proposed siting of the facility from a radiation protection point of view, estimation of doses to the public, establishing and monitoring authorised limits for the discharge of radioactive effluents as well as measures to control the access of members of the public. In addition, the undertaking is required to provide an adequate emergency response in case of an emergency in order to protect the public.

Chapter IX: General Responsibilities of Member States and Competent Authorities and Other Requirements for Regulatory Control

This Chapter provides general requirements for the institutional infrastructure which are not specific for fusion facilities. This includes requirements for the competent authority, recognition of services and experts (e.g. radiation protection officers and radiation protection experts) occupational health and dosimetry services, control of radioactive sources, significant events, emergency preparedness and response, system of enforcement

4.3.2 Council Directive 2009/71/Euratom – Nuclear Safety Directive

Council Directive 2009/71/Euratom /EU 09/ amended by Directive 2014/87/Euratom of July 2014 /EU 14/ is establishing a Community framework for the nuclear safety of nuclear installations.

The Directive pursues the goal of maintaining and promoting the continuous improvement of nuclear safety and its regulation. With the amendment, substantive technical provisions in the area of nuclear safety are included, such as on the safety objective and safety culture.

The Directive applies to any civilian nuclear installation subject to a license, i.e. enrichment plants, nuclear fuel fabrication plants, nuclear power plants, reprocessing plants, research reactor facilities and spent fuel storage facilities. The area of repositories for radioactive waste is covered separately by the so-called Radioactive Waste and Spent Fuel Management Directive 2011/70/EURATOM. It has to be emphasized that fusion facilities are not in the scope of the Directive, neither.

The Directive contains regulations on the creation of a legal and regulatory framework for nuclear safety, on the organisation and tasks of the nuclear authorities, on the obligations of the operators of nuclear facilities, on the education and training of the employees of all parties involved and on the information of the public.

The Member States are obliged - in addition to the self-assessment of the national legislative, enforcement and organisational framework and the competent authorities (so-called "peer review") already contained in the 2009 Directive - to conduct thematic peer reviews at least every six years. These thematic peer reviews on a safety topic to be jointly selected by the Member States are intended to set in motion a continuous system of mutual learning.

Even if fusion reactors are not in the scope, the general requirements of this Directive can be in principle applied to them. Here, a detailed examination has to be performed to determine whether corresponding adjustments have to be made for certain requirements.

4.3.3 Directive 2011/70/EURATOM – Radioactive Waste and Spent Fuel Management

Council Directive 2011/70/EURATOM /EU 11/ of 19 July 2011 establishes a Community framework for ensuring responsible and safe management of spent fuel and radioactive waste to avoid imposing undue burdens on future generations. It supplements the basic standards referred to in the Euratom Treaty as regards the safety of spent fuel and radioactive waste without prejudice to the Basic Safety Standards Directive. Member States are obliged to provide for appropriate arrangements for a high level of safety in spent fuel and radioactive waste management to protect workers and the general public against the dangers arising from ionising radiation. In addition, it ensures the provision of public information and participation in relation to spent fuel and radioactive waste management while having due regard to security and proprietary information issues.

The Directive applies to all stages of

- Spent fuel management when the spent fuel results from civilian activities;
- Radioactive waste management, from generation to disposal, when the radioactive waste results from civilian activities.

With the radioactive inventory in a fusion power plant including the radioactive waste produced through activation processes, the requirements of this Directive directly apply to this kind of facilities. Radioactive waste is defined in this context as radioactive material in gaseous, liquid or solid form for which no further use is foreseen or considered by the Member State or by a legal or natural person whose decision is accepted by the Member State, and which is regulated as radioactive waste by a competent regulatory authority under the legislative and regulatory framework of the Member State.

Since spent fuel refer to nuclear fuel that has been irradiated in and permanently removed from a reactor core, the requirements concerning spent fuel are not applicable to fusion power plants.

The Directive requires a national legislative, regulatory and organisational framework to be established and maintained by the Member States. This framework comprises establishing a competent regulatory authority in the field of safety of radioactive waste management and the needed resources and transparency. The responsibility for the management of radioactive waste lays within the Member States generated in them. The licence holder has to assure the safety of spent fuel and radioactive waste management under the supervision of its national competent regulatory authority.

4.3.4 Commission Regulation (Euratom) No 302/2005 – Nuclear Safeguards

Commission Regulation (Euratom) No 302/2005 /EU 05/ of 8 February 2005 on the application of Euratom safeguards applies to any person or undertaking setting up or operating an installation for the production, separation, reprocessing, storage or other use of source material or special fissile material.

Installation means in this context a collective term that includes a reactor, a critical installation, a conversion plant, a fabrication plant, a reprocessing plant, an isotope separation plant, a separate storage installation, a waste treatment or waste storage installation or any other location where source material or special fissile material is customarily used. Ores, source materials and special fissile materials as defined in Article 197 of the European Treaty is summarized with the term “nuclear materials”. Source materials refers to uranium containing the mixture of isotopes occurring in nature, depleted uranium, thorium and any substance containing the foregoing in a relevant concentration. The term ‘use’ of nuclear material is taken to include, inter alia: power production in reactors, research in critical or zero energy installations, conversion, fabrication, reprocessing, storage, isotope separation, and ore concentration, as well as treatment or storage of waste.

In order to enable the Commission to plan its safeguards activities, any person or undertaking setting up or operating an installation for the production, separation, reprocessing, storage or other use of source material or special fissile material has to declare to the Commission the basic technical characteristics of the installation. Additional communication duties on an annual basis and at least 40 days before taking a physical inventory are defined in Article 5.

On the basis of the basic technical characteristics submitted, particular safeguard provisions relating to the matters are adopted by means of a Commission decision addressed to the person or undertaking concerned, taking account of operational and technical constraints and in close consultation with the person or undertaking concerned and the relevant Member State.

Chapter III of the Regulation lays down the requirements concerning nuclear material accountancy including an accounting system, accounting and operating records and, in particular, information on the quantities, category, form and composition of these materials. Chapter IV regulates the transfer of nuclear materials between states and Chapter V is dealing in particular with ore producers and shipment, carriers, and temporary storage agents as well as retained or conditioned waste. Finally, Chapter VI defines an exemption of the provisions for defence purposes.

Since a fusion facility is not dealing with nuclear material, it does not fall under this regulation. However, due to the high inventory of Tritium expected in future fusion plants, a discussion on the extension of these concepts to such facilities may arise.

4.3.5 Council Directive related to non-radioactive hazards

4.3.5.1 Council Directive 98/24/EC – Chemical agents at workplaces

Council Directive 98/24/EC /EU 98/ of 7 April 1998 lays down minimum requirements for the protection of workers from risks to their safety and health arising, or likely to arise, from the effects of chemical agents that are present at the workplace or as a result of any work activity involving chemical agents and is the fourteenth individual Directive within the meaning of Article 16(1) of Directive 89/391/EEC. According to Article 1, the requirements of this Directive apply where hazardous chemical agents are present or may be present at the workplace, without prejudice to the provisions for chemical agents to which measures for radiation protection apply.

It is the duty of the Commission to evaluate the relationship between the health effects of hazardous chemical agents and the level of occupational exposure by means of an independent scientific assessment of the latest available scientific data. Based on this evaluation and after consulting the Advisory Committee on Safety, Hygiene and Health protection at Work, the Commission propose indicative occupational exposure limit values for the protection of workers from chemical risks, to be set at Community level. For any chemical agent for which an indicative occupational exposure limit value is established at Community level, Member States are obliged to establish a national occupational exposure limit value, taking into account the Community limit value, determining its nature in accordance with national legislation and practice. Along the same lines, occupational exposure limits and biological limits may be set up at Community level taking into account feasibility factors. In this case, Member States have to establish corresponding national occupational exposure or biological limits not exceeding the Community limits.

To first determine whether any hazardous chemical agents are present at the workplace has to be done by the employer. If so, he shall then assess any risk to the safety and health of workers arising from the presence of those chemical agents. The risk has to be assesses taking not account of all risks presented by all hazardous chemical agents occurring in combination.

In the case of a new activity involving hazardous chemical agents, work shall only commence after an assessment of the risk of that activity has been made and any preventive measures identified have been implemented. Furthermore, the Directive foresees additional general requirements on specific protection and prevention measures including a minimisation principle, meaning that the risk to the health and safety of workers at work needs to be eliminated or reduced to a minimum.

The Directive additionally contains requirements for arrangements to deal with accidents, incidents and emergencies, for the information and training for workers, for the health surveillance of workers for whom the results of the risk assessment reveal a risk to health as well as the prohibition of the production, manufacture, use at work of some distinct chemical agents listed in Annex III.

With this, the Directive provides general rules and requirements which are not specific to certain facilities and are therefore directly transferrable for fusion facilities utilizing chemical hazardous materials. According to Article 4, it is the duty of the employer to assess any risk to the safety and health of workers arising from the presence of those chemical agents.

4.3.5.2 Directive 2004/37/EC – Carcinogens or mutagens at workplaces

The Directive 2004/37/EC /EU 04/ of the European Parliament and of the Council of 29 April 2004 on the protection of workers from the risks related to exposure to carcinogens or mutagens at work has the aim to protect workers against risks to their health and safety, including the prevention of such risks, arising or likely to arise from exposure to carcinogens or mutagens at work. This applies to all workplaces without workplaces exposed only to radiation. For this, the Directive do not apply. As regards asbestos, the provisions of this Directive apply only whenever they are more favourable to health and safety at work.

The Directive lays down particular minimum requirements in this area, including limit values. It applies to all activities in which workers are or are likely to be exposed to carcinogens or mutagens as a result of their work. In the case of any activity likely to involve a risk of exposure to carcinogens or mutagens, the nature, degree and duration of workers' exposure needs to be determined in order to make it possible to assess any risk to the workers' health or safety and to lay down the measures to be taken. Where the results of the assessment reveal a risk to workers' health or safety, workers' exposure must be prevented

As it is done for chemical agents at workplaces in Council Directive 98/24/EC, this Directive foresees a minimisation principle, obliging the employer to reduce the use of a carcinogen or mutagen at the place of work, in particular by replacing it, in so far as is technically possible, by a substance, preparation or process which, under its conditions of use, is not dangerous or is less dangerous to workers' health or safety, as the case may be. Where it is not technically possible to replace the carcinogen or mutagen by a substance, preparation or process which, under its conditions of use, is not dangerous or is less dangerous to health or safety, the employer shall ensure that the carcinogen or mutagen is, in so far as is technically possible, manufactured and used in a closed system.

The Directive provides additional requirements for an unforeseen exposure, measures to access risk areas, hygiene and individual protection, information and training of workers as well as health surveillance. According to the duty of the Council to set out limit values, a few limit values for exposure are set out in Annex III.

With this, the Directive provides general rules and requirements which are not specific to certain facilities and are therefore directly transferrable for fusion facilities utilising carcinogens and mutagens. According to Article 3, the risk of exposure to carcinogens or mutagens needs to be assessed and regularly renewed.

4.3.5.3 Directive 2013/35/EU – Worker exposure to electromagnetic fields

The Directive 2013/35/EU /EU 13a/ of the European Parliament and of the Council of 26 June 2013 lays down minimum requirements for the protection of workers from risks to their health and safety arising, or likely to arise, from exposure to electromagnetic fields during their work. It covers all known direct biological effects and indirect effects caused by any electromagnetic fields. In this context, the Directive sets out exposure limit values (ELV), which cover only scientifically well-established links between short-term direct biophysical effects and exposure to electromagnetic fields. It does not cover suggested long-term effects, but in the case that well-established scientific evidence on suggested long-term effects becomes available, the Commission considers a suitable policy response to address such effects. Risks resulting from contact with live conductors are also not in the scope of the Directive.

The Directive requires employers to ensure that the exposure of workers to electromagnetic fields is limited to the exposure limit values for non-thermal and thermal effects set out in the Directive. To ensure this, the employer has to assess all risks for workers arising from electromagnetic fields at the workplace and, if necessary, measure or calculate the levels of electromagnetic fields to which workers are exposed. The employer needs to take the necessary actions to ensure that the risks are eliminated or reduced to a minimum. He has to implement an action plan including technical and/or organisational measures on the basis of the risk assessment performed.

The Directive provides additional requirements including worker information and training as well as health surveillance.

With this, the Directive provides general rules and requirements which are not specific to certain facilities and are therefore directly transferrable for fusion facilities operating high electromagnetic fields. According to Article 4, the risk arising from electromagnetic fields needs to be assessed and updated on a regular basis.

4.3.5.4 Directive 2011/65/EU – Hazardous substances in electrical and electronic equipment

Directive 2011/65/EU of the European Parliament and of the Council of 8 June 2011 /EU 11a/ lays down rules on the restriction of the use of hazardous substances in electrical and electronic equipment (EEE) with a view to contributing to the protection of human health and the environment, including the environmentally sound recovery and disposal of waste EEE.

It applies to all electrical and electronic equipment including industrial monitoring, and control instruments, IT and telecommunications equipment as well as electrical and electronic tools. For some EEE, listed in Article 2, the Directive does not apply, including large-scale stationary industrial tools and large-scale fixed installations.

The term 'large-scale stationary industrial tools' means a large-scale assembly of machines, equipment, and/or components, functioning together for a specific application, permanently installed and de-installed by professionals at a given place, and used and maintained by professionals in an industrial manufacturing facility or research and development facility. The term 'large-scale fixed installation' describes a large-scale combination of several types of apparatus and, where applicable, other devices, which are assembled and installed by professionals, intended to be used permanently in a pre-defined and dedicated location, and de-installed by professionals.

With these definitions, a fusion facility can be seen as a large-scale fixed installation. The Directive do not apply to them as it is the case for other industrial facilities. For electrical and electronic equipment, which is used in the facility, and bought on the free marked, the vendor is responsible to meet the requirements for this equipment.

4.4 Screening and categorisation of IAEA Safety Standards

The IAEA establishes safety standards to protect health and minimize danger to life and property. The IAEA must apply these standards in its own activities, and Member States of the IAEA can apply them by means of their regulatory provisions for nuclear and radiation safety. A comprehensive set of safety standards, reviewed regularly, and IAEA support and assistance for their application have become a key element of the global safety regime.

The findings of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and the recommendations of international expert bodies, notably the International Commission on Radiological Protection (ICRP), are taken into account in developing the IAEA safety standards. Some safety standards are developed in co-operation with other bodies in the United Nations system or other specialized agencies.

The principal users of safety standards in IAEA Member States are regulatory bodies and other relevant national authorities. The IAEA safety standards are also used by co-sponsoring organizations and by many organizations that design, construct and operate nuclear facilities, as well as organizations involved in the use of radiation and radioactive sources.

The IAEA safety standards are applicable, as relevant, throughout the entire lifetime of all facilities and activities — existing and new — utilized for peaceful purposes and to protective actions to reduce existing radiation risks. They can be used by Member States as a reference for their national regulations in respect of facilities and activities. IAEA's safety services — from engineering safety, operational safety, and radiation, transport and waste safety to regulatory matters and safety culture in organizations — assist Member States in applying the standards and assess their effectiveness.

In conclusion, the IAEA Safety Standards represent the international consensus on nuclear safety and form a regulatory framework for nuclear facilities and activities. However, no dedicated safety standards for fusion facilities have been developed by the IAEA as yet. The safety standards do not address specific requirements for fusion facilities, but requirements are rather applicable for all radiation facilities. IAEA's Safety Standards are hierarchically structured in safety fundamentals, safety requirements and safety guides. Here, a certain range of IAEA Safety Standards including the Safety Fundamentals will be reviewed and their applicability to fusion facilities will be evaluated.

Fundamental safety principles, and safety requirements that have been developed by the IAEA for **radiation facilities** are applicable to fusion facilities in which tritium will be used.

Specific safety requirements developed for the safety of fission facilities were not originally intended to be applied to fusion facilities. However, some of these requirements or related concepts could be applicable to all radiation facilities and enhance the safety of future fusion facilities. Should some of these requirements or concepts applicable to fission facilities be applied to future fusion facilities, it should be in a **proportionate and targeted manner**.

The following IAEA General Safety Requirements and Specific Safety Requirements are considered:

- Safety fundamentals:
  - IAEA Safety Standard Series No. SF-1 "Fundamental Safety Principles"
- General Safety Requirements:
  - IAEA Safety Standard Series No. GSR Part 2 "Leadership and Management for Safety"
  - IAEA Safety Standard Series No. GSR Part 3 "Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards"
  - IAEA Safety Standard Series No. GRS Part 4 "Safety Assessment for Facilities and Activities"
  - IAEA Safety Standard Series No. GSR Part 6 "Decommissioning of Facilities"
  - IAEA Safety Standards Series No. GSR Part 7 "Preparedness and Response for a Nuclear or Radiological Emergency"
- Specific Safety Requirements

- IAEA Safety Standard Series No. SSR-1 "Site Evaluation for Nuclear Installations"
- IAEA Safety Standard Series No. SSR-2/1 "Safety of Nuclear Power Plants: Design"
- IAEA Safety Standard Series No. SSR-2/2 "Safety of Nuclear Power Plants: Commissioning and Operation"
- IAEA Safety Standard Series No. SSR-3 "Safety of Research Reactors"
- IAEA Safety Standard Series No. SSR-4 "Safety of Nuclear Fuel Cycle Facilities"

4.4.1 IAEA Safety Standard Series No. SF-1 "Fundamental Safety Principles"

The objective of this IAEA publication /IAEA 06/ is to establish the fundamental safety objective and ten safety principles as well as their intent and purpose. SF-1 represents the fundamental concept providing the bases for the IAEA's safety standards and its safety related program. The safety principles provide the basis for requirements and measures for the protection of people and the environment against radiation risks and for the safety of facilities and activities that give rise to radiation risks. This also includes, in particular, nuclear installations and uses of radiation and radioactive sources as well as the transport of radioactive material and the management of radioactive waste. Considering fusion facilities, radiation risk aspects might be implicated in plasma operation, transport and storage of activated components, and the tritium fuel cycle processing.

Safety Objective

"The fundamental safety objective is to protect people and the environment from harmful effects of ionizing radiation."

This fundamental objective is in no way limited to individual technical aspects such as specific nuclear facilities or certain radioactive materials, but it is rather a basic credo for dealing with ionizing radiation in general. Thus, the fundamental safety objective applies to all circumstances that give rise to radiation risks – including fusion facilities.

Safety Principles

The following ten safety principles are applicable, as relevant, throughout the entire lifetime of all facilities and activities and to protective actions to reduce existing radiation risks. As well as the safety objective, these principles appear to be applicable to any kind of facility and activity that give rise to radiation risks.

- Principle 1: Responsibility for safety
  The prime responsibility for safety must rest with the person or organization responsible for facilities and activities that give rise to radiation risks.

- Principle 2: Role of government
  An effective legal and governmental framework for safety, including an independent regulatory body, must be established and sustained.

- Principle 3: Leadership and management for safety
  Effective leadership and management for safety must be established and sustained in organizations concerned with, and facilities and activities that give rise to, radiation risks.

- Principle 4: Justification of facilities and activities
  Facilities and activities that give rise to radiation risks must yield an overall benefit.

- Principle 5: Optimization of protection
  Protection must be optimized to provide the highest level of safety that can reasonably be achieved.

- Principle 6: Limitation of risks to individuals
  Measures for controlling radiation risks must ensure that no individual bears an un-acceptable risk of harm.

- Principle 7: Protection of present and future generations
  People and the environment, present and future, must be protected against radiation risks.

- Principle 8: Prevention of accidents
  All practical efforts must be made to prevent and mitigate nuclear or radiation accidents.

- Principle 9: Emergency preparedness and response
  Arrangements must be made for emergency preparedness and response for nuclear or radiation incidents.

- Principle 10: Protective actions to reduce existing or unregulated radiation risks
  Protective actions to reduce existing or unregulated radiation risks must be justified and optimized.

Due to the generic nature of the IAEA Safety Fundamentals with respect to fundamental radiation protection and basic safety culture aspects, this document is fully applicable to fusion facilities.

4.4.2 GSR Part 2 "Leadership and Management for Safety"

GSR Part 2 „Leadership and Management for Safety" /IAEA 16d/ emphasizes that leadership for safety, management for safety, an integrated management system and a systemic approach (i.e. an approach relating to the system as a whole in which the interactions between technical, human and organizational factors are duly considered) are essential to the specification and application of adequate safety measures and the fostering of a strong safety culture. /IAEA 16d/

Responsibility for safety

The license holder is obliged to ensure that the fundamental safety objective of protecting people and the environment from harmful effects of ionizing radiation is achieved. This obligation starts with the senior management and it is required to ensure that managers on all levels develop and maintain an understanding of radiation risks, potential consequences and how to manage radiation risks relevant to their responsibilities.

Leadership for safety

The senior managements shall demonstrate leadership for safety and a commitment to safety. Managers on all levels shall encourage and support all individuals in achieving safety goals and performing their tasks safely. The basis for decisions relevant to safety needs to be communicated clearly.

Management for safety

Senior management is responsible for establishing goals, strategies, plans and objectives for the organization that are consistent with the organization's safety policy. It is required that the management system integrates safety, health, environmental, security, quality, human and organisational factors, societal and economic elements in such way, that safety is not compromised. It is emphasized that the organization shall put in place arrangements with vendors, contractors and suppliers for specifying, monitoring and managing the supply to it of items, products and services that may influence safety. Necessary resources need to be provided by the senior management to perform activities of the organization safely.

Culture for safety

GSR Part 2 requires that all individuals in the organization, from senior managers downwards, shall foster a strong safety culture. The management system and leadership for safety shall contribute to foster and sustain a strong safety culture within the organization.

Measurement, assessment and improvement

With respect to the continuous improvement of safety also the effectiveness of the management system shall be measures, assessed and improved to enhance safety performance. It is the responsibility of the senior management to regularly commission assessments of leadership for safety and safety culture within the organization by utilizing independent assessments.

4.4.3 GSR Part 3 "Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards"

This General Safety Requirements publication, IAEA Safety Standards Series No. GSR Part 3 /IAEA 14/, includes requirements governed by the objectives and principles of the Fundamental Safety Principles (SF-1) and explains their context and concepts. Before explicitly introducing the

52 requirements set out in this document, GSR Part 3 points out main aspects about the system of protection and safety, types of exposure situation (see below), dose constraints and reference levels, protection of the environment and interfaces between safety and security – aspects and considerations applying to all phases of the lifetime of fusion facilities.

The requirements of GSR Part 3 are grouped into requirements applicable for all exposure situations ((0) General Requirements for Protection and Safety) and separate sets of requirements for

- (1) Planned Exposure Situations;
- (2) Emergency Exposure Situations and
- (3) Existing Exposure Situations.

For each of the three types of situation, the requirements are further grouped into requirements for occupational exposure, public exposure and (for planned exposure situations only) medical exposure. Of course, not all exposure situations might be applicable like medical exposure, but – vice versa – the document covers all possible exposure situations during all phases of the entire lifetime of fusion facilities. For fusion facilities, occupational radiation exposure probably appears to play the major role in the elaboration of requirements and safety standards.

GSR Part 3 is intended primarily for use by governments and regulatory bodies. Requirements also apply to principal parties, health authorities, professional bodies and service providers such as technical support organizations. GSR Part 3 stresses, that it establishes requirements to be fulfilled in all facilities and activities giving rise to radiation risks.

In the following, not all requirements will be examined individually, but the applicability and transferability of the key messages of (0)-(3) to fusion facilities will be reviewed.

(0) General Requirements for Protection and Safety

This section deals with

- the application of the principles of radiation protection;
- responsibilities of the government and regulatory body;
- responsibilities and management for protection and safety of person or organization responsible.

These requirements include assigning responsibilities of all parties involved: the government, the competent authorities, and the individuals or organizations primarily responsible for the safety of the facility or activity. with respect to the implementation of a protection and safety program and a proper management system, the promotion of safety culture and the consideration of human factors. These general considerations apply to any kind of facility and activity and can also be transferred to fusion facilities.

(1) Planned Exposure Situations

The requirements for planned exposure situations apply to several practices. Those practices set out in para. 3.1, which most likely resemble the nature of fusion facilities are:

- (b) The production and supply of devices that generate radiation, including linear accelerators, cyclotrons, and fixed and mobile radiography equipment.
- (c) The generation of nuclear power, including any activities within the nuclear fuel cycle that involve or that could involve exposure to radiation or exposure due to radioactive material.
- (g) Any other practice as specified by the regulatory body.

In any case, the latter justifies the applicability and transferability of the requirements to fusion facilities if the practice is specified by the regulatory body.

(2) Emergency Exposure Situations

The requirements for emergency exposure situations established in this section apply to activities undertaken in preparedness for and in response to a nuclear or radiological emergency.

These requirements include:

- Establishment of an emergency management system;
- Preparedness and response for an emergency for avoiding deterministic effects and reducing the likelihood of stochastic effects due to public exposure;
- Arrangements for controlling the exposure of emergency workers;
- Arrangements for the transition from an emergency exposure situation to an existing exposure situation.

Due to the fundamentally different physical processes in fusion facilities as opposed to fission reactors – these include among others no criticality, a lower inventory, heat removal as well as no highly active long lived waste – the potential radiological effects on the public differ to nuclear power plants. However, a protection strategy has to be developed and include reference level expressed in terms of residual dose, generic criteria for particular protective actions and operational criteria for initiating the different parts of an emergency plan. Such levels and criteria are recommended by the IAEA in the context of fission reactors and, for the most part, are integrated into the national safety concept by the Member States.

(3) Existing Exposure Situations

The requirements for existing exposure situations in this section apply to:

- Exposure due to contamination of areas by residual radioactive material;
- Exposure due to commodities, including food, feed, drinking water and construction materials;
- Exposure due to natural sources including, inter alia, Radon.

Items 1 and 2 might be applicable for fusion facilities if a nuclear or radiological emergency has occurred and been declared to be ended, or if radionuclides deriving from residual radioactive material would have been incorporated, respectively. Item 3 might not be applicable to fusion facilities at all.

The key massages and requirements for existing exposure situations are the responsibilities of the government and the regulatory body to ensure remedial and protective actions.

Schedule 1 of GSR Part 3 gives general criteria for exemption and clearance of a practice or a source within a practice from some or all of the requirements stipulated in this document. The general criteria are that:

- (a) Radiation risks are sufficiently low as not to warrant regulatory control, and there is no appreciable likelihood of occurrence for scenarios that could lead to a failure to meet the general criterion for clearance; or
- (b) Continued regulatory control of the material would yield no net benefit, in that no reasonable control measures would achieve a worthwhile return in terms of reduction of individual doses or reduction of health risks.

Exemption and clearance criteria are set out in detail in Directive 2013/59/Euratom and directly applies to fusion facilities.

4.4.4 GSR Part 4 "Safety Assessment for Facilities and Activities"

This IAEA General Safety Standard GSR Part 4 /IAEA 16e/ includes requirements governed by the objectives and principles of the Fundamental Safety Principles (SF-1). The requirements relate to any human activity that may cause people to be exposed to radiation risks arising from facilities and activities. The list of facilities and activities given in the scope of GSR Part 4 has been compiled from the lists provided in SF-1 and in GSR Part 2. Fusion facilities are not explicitly mentioned, but the list is explicitly mentioned to be not exhaustive.

In general, GSR Part 4 states that 'safety assessment' is the assessment of all aspects of a practice that are relevant to protection and safety. For an authorized facility, this includes siting, design and operation of the facility. Safety assessment is the systematic process that is carried out throughout the lifetime of the facility or activity to ensure that all the relevant safety requirements are met by the proposed (or actual) design. Stages in the lifetime of a facility or activity for which a safety assessment is carried out is also listed in this document and include

- Site evaluation for the facility or activity;
- Development of the design;
- Construction of the facility or implementation of the activity;
- Commissioning of the facility or activity;
- Commencement of operation of the facility or conduct of the activity;
- Normal operation of the facility or normal conduct of the activity;
- Modification of the design or operation;
- Periodic safety reviews;
- Life extension of the facility beyond its original design life;
- Changes in ownership or management of the facility;
- Decommissioning and dismantling of the facility;
- Closure of a disposal facility for radioactive waste, and the post-closure phase;
- Remediation of a site and release from regulatory control.

Chapter 2 of GSR Part 4 provides a basis for requiring a safety assessment, essentially based on SF-1. Since the IAEA Safety Fundamentals have a generic function in terms of fundamental radiation protection and basic safety culture aspects, this chapter of GSR Part 4 is directly applicable to fusion facilities.

GSR Part 4 provides 24 requirements for safety assessment for facilities and activities, presented in the following three chapters:

Graded Approach to Safety Assessment

The document states, that graded approach shall be used in determining the scope and level of detail of the safety assessment carried out at a particular stage for any particular facility or activity, consistent with the magnitude of the possible radiation risks arising from the facility or activity.

Safety Assessment

Overall requirements for safety assessment refer to an adequate representation of the scope of safety assessment, meaning that it shall be carried out for all applications of technology that give rise to radiation risks, to a definition of responsibilities and to emphasizing the purpose of the safety assessment, which is to determine whether an adequate level of safety has been achieved and whether the basic safety objectives and safety criteria established by the designer, the operating organization and the regulatory body, in compliance with the requirements in GSR Part 3 have been fulfilled.

Specific requirements for safety assessment are illustrated by means of a diagram within this document. It shows main elements of the process for safety assessment and verification. All features relevant to safety require a systematic evaluation to be carried out and incudes

- Preparation for the safety assessment, in terms of assembling the expertise, tools and information required to carry out the work;
- Identification of the possible radiation risks resulting from normal operation, anticipated operational occurrences or accident conditions;
- Identification and assessment of a comprehensive set of safety functions;
- Assessment of the site characteristics that relate to the possible radiation risks;
- Assessment of the provisions for radiation protection;
- Assessment of engineering aspects to determine whether the safety requirements for design relevant to the facility or activity have been met;
- Assessment of human factor related aspects of the design and operation of the facility or the planning and conduct of the activity;
- Assessment of safety in the longer term, which is of particular concern when ageing effects might develop and might affect safety margins, decommissioning and dismantling of facilities, and closure of disposal facilities for radioactive waste.

All those aspects in terms of preparation and implementation of specific steps in the safety assessment, weather regarding to radiological or non-radiological risks, are also part of the specific safety assessment for fusion facilities.

GSR Part 4 also addresses the concept of defence-in-depth. In this approach, a number of consecutive and independent levels of protection or physical barriers are provided such that, if one level of protection or barrier were to fail, the subsequent level or barrier would be available. The concept inherently plays an important role for fission reactors where criticality is given, nonetheless the principle can be applied to fusion reactors as well.

The document also requires adequate deterministic and probabilistic approaches, sensitivity and uncertainty studies, collection, assessment and documentation of operating experience data and of the overall safety assessment. These requirements are applicable for all types of facilities that give rise to any effects and risks to people and environment, including fusion facilities.

Management, Use And Maintenance Of The Safety Assessment

GSR Part 4 requires that safety assessments should be planned, organized, applied, audited and periodically reviewed. These processes are main parts of any kind of assessments procedures and therefore basic processes in the safety assessment of fusion facilities.

4.4.5 GSR Part 6 “Decommissioning of Facilities”

The General Safety Requirements, IAEA Safety Standards Series No. GSR Part 6 /IAEA 14c/ establishes general safety requirements for all aspects of decommissioning during the lifetime of a facility starting from the siting and design of a facility to the termination of the authorization for decommissioning. The term ‘decommissioning’ refers to the administrative and technical actions taken to allow the removal of some or all of the regulatory controls from a facility.

GRS Part 6 applies to nuclear power plants, research reactors, other nuclear fuel cycle facilities, including predisposal waste management facilities, facilities for processing Naturally Occurring Radioactive Material (NORM), former military sites, and relevant medical facilities, industrial facilities, and research and development facilities. It does not apply to radioactive waste disposal facilities or disposal facilities for NORM or for waste from mining and mineral processing. Fusion power plants are not explicitly mentioned, but could be included in industrial or research and development facilities.

Decommissioning is performed using a graded approach to achieve a progressive and systematic reduction in radiological hazards. Non-radiological hazards, such as industrial hazards or hazards due to chemical waste, can be significant during decommissioning, but are not covered by the scope of GRS Part 6. Decommissioning is undertaken on the basis of planning and assessment to ensure safety, protection of workers and the public, and protection of the environment. It is stressed that decommissioning is concerned with facilities, i.e. buildings including their associated land and equipment. The cleanup of areas, that may become contaminated during operation of a facility is part of decommissioning.

GSR Part 6 contain 15 general requirements which are grouped into the following topics:

- Protection of people and protection of the environment;
- Responsibilities associated with decommissioning;
- Management of decommissioning;
- Decommissioning strategy;
- Financing of decommissioning;
- Planning for decommissioning during the lifetime of the facility;
- Conduct of decommissioning actions;
- Completion of decommissioning actions and termination of the authorization for decommissioning.

Due to the very general character of the requirements, the requirements are directly applicable to fusion facilities.

4.4.6 GSR Part 7 "Preparedness and Response for a Nuclear or Radiological Emergency"

The General Safety Requirements publication, IAEA Safety Standards Series No. GSR Part 7 /IAEA 16h/, includes requirements governed by the fundamental safety objective and the fundamental safety principles of the Fundamental Safety Principles (SF-1). In particular, it addresses Principle 9, which is concerned with the arrangements that must be made for preparedness and response for a nuclear or radiological emergency and includes requirements for the transition to an existing exposure situation.

It establishes the requirements for an adequate level of preparedness and response for a nuclear or radiological emergency, irrespective of the initiator of the emergency, which could be a natural event, a human error, a mechanical or other failure, or a nuclear security event. The application of these requirements is also intended to mitigate the consequences of an emergency if such an emergency arises despite all efforts made to prevent it.

The requirements laid down apply for preparedness and response for a nuclear or radiological emergency in relation to all those facilities and activities, as well as sources, with the potential for causing radiation exposure, environmental contamination or concern on the part of the public warranting protective actions and other response actions.

Fusion facilities are not explicitly mentioned in GSR Part 7, but with the potential of the radioactivity contained in the facility are covered by the scope.

According to Requirement 4, the government shall ensure that a hazard assessment is performed. For the purposes of the safety requirements, assessed hazards are grouped into five emergency preparedness categories. These five categories establish the basis for a graded approach to the application of the requirements and for developing generically justified and optimized arrangements for preparedness and response for a nuclear or radiological emergency. The categorization with subsequent requirements of GSR Part 7 directly apples to fusion facilities.

4.4.7 SSR-1 Site Evaluation for Nuclear Installations

The requirements for site evaluation for nuclear installations in SSR-1 /IAEA 19/ are intended to contribute to the protection of workers and the public, and to the protection of the environment, from harmful effects of ionizing radiation. The predecessor document (IAEA Safety Standards Series No. NS-R-3) was revised in 2016 to take into account issues high-lighted after the Fukushima Daiichi accident in 2011. This publication is intended for use by regulatory bodies in establishing regulatory requirements, and by operating organizations or their contractors in conducting site evaluation for nuclear installations.

The objective of this publication is to establish requirements for:

- Defining the information to be used in the site evaluation process;
- Evaluating a site such that the site-specific hazards and the safety related site characteristics are adequately taken into account, in order to derive appropriate site-specific design parameters;
- Analysing the characteristics of the population and the region surrounding the site to determine whether there would be significant difficulties in implementing emergency response actions effectively.

This publication addresses the evaluation of site related factors such as external hazards and population distribution that have to be taken into account to ensure that the site-installation combination does not constitute an unacceptable risk to people or the environment over the lifetime of the nuclear installation.

SSR-1 lists nuclear installations which the established requirements apply to. Those nuclear installations are:

- Nuclear power plants;
- Research reactors (including subcritical and critical assemblies) and any adjoining radioisotope production facilities;
- Storage facilities for spent fuel;
- Facilities for the enrichment of uranium;
- Nuclear fuel fabrication facilities;
- Conversion facilities;
- Facilities for the reprocessing of spent fuel;
- Facilities for the predisposal management of radioactive waste arising from nuclear fuel cycle facilities;
- Nuclear fuel cycle related research and development facilities.

Up to now, fusion facilities are not explicitly covered by the scope of SSR-1. However, siting aspects are also important for fusion facilities: to protect items important to safety against external site-specific hazards as well as determining site-specific factors important for estimating the radiation risk to the public in case of planned discharges or accidental releases.

In the following, the individual sections of the document are reviewed for applicability and transferability to fusion facilities, and if appropriate, the individual requirements of each section are also examined in more detail.

Safety Principles and Concepts

This section is fully applicable as this section derives from the IAEA SF-1. The main requirement is the characterization of the safety objective in site evaluation for nuclear installations.

Application of the Management System for Site Evaluation

This section is fully applicable as this section derives from the IAEA GSR Part 2, which shows a rather generic character (see GSR Part 3). The main requirement is the conduct in a comprehensive, systematic, planned and documented manner in accordance with a management system.

General Requirements for Site Evaluation

Requirements of this section are mostly applicable with some adjustments:

- Requirement 3: Scope of the site evaluation for nuclear installations
  The amount, type and status of the radioactive inventory at the site to take into consideration in the application of a graded approach should be extended to Tritium and other associated materials concerning fusion facilities.

- Requirement 4: Site suitability and Requirement 5: Site and regional characteristics
  The characteristics of the site and its environment that could influence the transfer of radioactive material released from the nuclear installation to people and to the environment is different for fusion facilities. In general, the radiation risk to the public is expected to be lower for fusion facilities.

- Requirement 11: Special considerations for the ultimate heat sink for nuclear installations that require an ultimate heat sink
  This requirement takes into account air temperature and humidity as well as water characteristics. Specifications should be made concerning the heat sink characteristics and the cooling of materials.

- Requirement 12: Potential effects of the nuclear installation on people and the environment
  The radiological impact on people and the environment is very low for fusion facilities. Postulated accident scenarios including the resulting source terms are not applicable to the same extend as for fission reactors. Nevertheless, potential effects should be estimated in advance and the feasibility of planning effective emergency response actions at the site and in the external zone should be taken into account by a graded approach.

Evaluation of External Hazards

This section establishes requirements for the evaluation of external hazards. These requirements are to be applied as appropriate for the type of nuclear installation as well as the site under consideration. External hazards which should be considered in the site evaluation are seismic, volcanic, meteorological, flooding, geological, other natural as well as human hazards and all apply to the site evaluation of fusion facilities as well. Special concern should be given to external hazards regarding to fires and explosions due to larger amounts of hydrogen and dust, electromagnetic interferences and chemical components.

Evaluation of the potential effects of the nuclear installation on the region

Requirement 25, demanding that the dispersion in air and water of radioactive material released in operational states and in accident conditions is assessed, is directly transferable to fusion facilities, although there are differences in the result compared to nuclear power plants. The same applies to Requirement 26, demanding to evaluate and periodically update the potential impact of radioactive releases to the public, in both operational states and accident conditions. However, any possible health impacts on the public and the environment due to the release of other hazardous materials should be considered and evaluated.

Monitoring and periodic review of the site

Following the scope of applicability and transferability of the previous section, this section applies equally and in full extend to fusion facilities.

4.4.8 SSR 2/1 "Safety of Nuclear Power Plants: Design"

The specific safety requirement SSR 2/1 /IAEA 16/ deals with the specific design requirements for stationary light water reactors and is primarily developed for the design of new reactors and serves as a reference to identify potential safety improvements for existing reactors. This safety standard deals with the following topics.

*A*pplying the safety principles and concepts

Besides the consideration of radiation protection during the design the defence in depth concept is introduced. It is also emphasized, that all phases of the lifetime of the plant need to be carefully considered already during the design phase. Due to the generic character of this section, it is also applicable to fusion power plants.

Management of safety in design

This section emphasizes the need establish a management system already during the design phase to ensure that all safety requirements for the design of the plant are considered and implemented. This aspect is also important during the design phase of a future fusion power plant.

Principal technical requirements

The fundamental safety functions addressed in SSR 2/1 are specific for fission reactors, in particular the fundamental safety function ensuring reactivity control. Due to the different technology of fusion facilities and the associated safety issues, theses fundamental safety functions need to be modified before applied on fusion power plants. Requirement 6 "Design for a nuclear power plant" addresses high level requirements which are directly applicable to fusion power plants. Even if the implementation of the defence in depth concept differs between fission and fusion facilities, the high-level requirements are still valid for both types of facilities. A further important aspect is addressing the interface between safety and security as well as between safety and safeguards. However, safeguards may play a minor role as nuclear fuel is not utilized in fusion facilities. Nevertheless, tritium accountancy may be required and may lead to similar interface issues. Another important aspect is the inclusion of features to facilitate radioactive waste management and decommissioning during the design phase, which is also applicable to fusion power plants.

*General plan*t design

The categories of plants states are specific for light water reactors, in particular design extension conditions with core melt. This plant states are not directly applicable and need to be specifically adapted to fusion power plants. Whereas normal operation, anticipated operational occurrences, design basis accidents and design extension conditions without core melt can be transferred to fusion power plants, design extension conditions with core melt can be omitted for fusion power plants. The concept of design extension conditions has been introduced to enhance safety and to cope with less frequent postulated events. This section addresses the need to specify the design basis of the plant based on the list of postulated initiating events as well as internal and external hazards to be considered. The different plant states comprising normal operation, anticipated operational occurrences, design basis accidents and design extension conditions are defined. Fundamental design principles, like independence of levels of defence in depth, application of the single failure criterion, ensuring reliability of items important to safety, redundancy and diversity as well as fail safety design are addressed. The safety classification needs to be based on the safety significance of the safety functions to be performed. In addition, equipment qualification is an important aspect to ensure reliability of the items important to safety under the expected environmental conditions. Further aspects are the ageing management and the consideration of human factors during the design phase.

Design of specific plant systems

This section addresses specific design requirements for typical systems in nuclear power plants. Amongst others, these are the reactor core, shutdown systems, reactor coolant system, heat removal in operational states and accident conditions, the containment system, instrumentation and control (I&C) system, electrical systems and further auxiliary and support systems. This section is specific for NPPs and not applicable for fusion power plants.

It can be concluded that the general structure of SSR 2/1 can serve as a basis for safety requirements for fusion power plants. Whereas the general design principles can be easily adapted and transferred to fusion power plants, safety requirements for fusion specific systems need to be developed.

4.4.9 SSR 2/2 "Safety of Nuclear Power Plants: Commissioning and Operation"

The specific safety requirement SSR 2/2 /IAEA 16b/ /IAEA 16/ deals with the specific requirements for commissioning and operation of stationary light water reactors. This safety standard deals with the following topics.

Safety objective and safety principles

This section reflects the fundamental safety objective to protect people and the environment from harmful effects of ionizing radiation and emphasize the safety principles of IAEA SF-1 /IAEA 06/ relevant for operation of a nuclear power plant.

Management and organizational structure of the operating organization

This section deals with the management of safety. Basis is the management system. Further aspects are the structure and functions of the operating organization and the staffing of the operating organization.

Management of operational safety

This section emphasized the important processes to be addressed in the management system. It shall include the following topics: safety policy, operational limits and conditions, qualification and training of personnel, performance of safety related activities, monitoring and review of safety performance, control of plant configuration, management of modifications, periodic safety review, equipment qualification, ageing management, records and reports and the programme for long term operation.

Operational safety programmes

The operational programmes shall include regulations for emergency preparedness and accident management. Further operational programmes are necessary for radiation protections, management of radioactive waste, fire safety and non-radiation-related safety. The feedback from operating experience from the own plant but also from other facilities is an important driver to increase the reliability and safety of a plant. In this section, the interface between safety and security is also described.

Plant commissioning

This section deals with the commissioning of the plant before commencing regular operation. Commissioning aims for the demonstration that the behaviour of the plant as built is in compliance with the design assumptions and the licence conditions. Consequently, the commissioning programmes needs to be agreed with the competent regulatory body because the success of the sequential steps in the commissioning programmes is usually directly linked to the licencing procedure of the plant. For NPPs, fuel loading is an important step during commissioning due to the increase of the hazard potential of the new plant.

*Plant operatio*ns

Basis for the safe operation are the operating procedures. These procedures comprise normal operation, anticipated operational occurrences and accident conditions. For NPPs, guidelines to deal with severe accidents have to be developed. Procedures and guidelines shall apply event-based as well as symptom-based approaches. Further requirements are related to the habitability of the control room from which staff can operate and manage all situations of the plant in operational states as well as accident conditions. In addition, housekeeping, management of foreign materials, development of a chemistry programme, arrangements for core handling and fuel management are further important aspects to be considered for the operation of nuclear power plants.

*Mainten*ance, testing, surveillance and inspection

The operating organization shall ensure effective programmes for maintenance, testing, surveillance and inspection. These programmes are important to ensure the reliability and function of items important to safety in case of demand.

Preparation for decommissioning

The last section deals with the late decommissioning. Already during operation, a decommissioning plan shall be prepared. All information relevant for the later decommissioning need to be documented and considered in the decommissioning plan.

As for SSR 2/1, it can be concluded that the general structure of SSR 2/2 can serve as a basis for safety requirements for fusion power plants considering a graded approach.

4.4.10 SSR 3 "Safety of Research Reactors"

The specific safety requirement SSR 3 /IAEA 16c/ deals with the specific safety requirements for research reactors. In contrast to SSR 2/1 and SSR 2/2 this guide comprises all stages of the lifetime of a research reactor from siting, design, commissioning, operation to decommissioning in a single document. In addition to SSR 2/1 and SSR 2/2 this guide includes an additional section on regulatory supervision. The content of SSR 3 is very similar to SSR 2/1 and SSR 2/2. Application of a graded approach is foreseen for regulating research reactors commensurate with its hazard potential. In addition to the set of requirements for nuclear power plants, SSR 3 includes requirements for the utilization and extended shutdowns. Utilization, i.e. ensuring safety during experiments and modifications of the research reactors to serve scientific needs and extended shutdowns, i.e. a research reactor presently not in operation without a decision made to either to decommission or to restart the reactor, may not be an issue for fusion power plants.

4.4.11 SSR 4 "Safety of Nuclear Fuel Cycle Facilities"

The specific safety requirement SSR 4 /IAEA 17/ deals with the specific safety requirements for nuclear fuel cycle facilities. The structure of SSR 4 follows the structure already applied in SSR 2/1, SSR 2/2 and SSR 3.

*Applying the safety objective, concepts and principles for nuclear fuel cyc*le facilities

This section starts with a description of the fundamental safety objective and the ten safety principles of IAEA Safety Standard Series No SF-1 /IAEA 06/ and the principles for radiation protection. Application of the defence in depth concept is considered as an essential pillar for the safety of nuclear fuel cycle facilities. The same generic description of the five levels of defence in depth as already described in SSR 2/1 or SSR 3 is used but with the recommendation of applying a graded approach.

Regulatory supervision for nuclear fuel cycle facilities

This section reflects some of the requirements of IAEA Safety Standard Series No. GSR Part 1 /IAEA 16g/. The sequential and discrete licensing steps covering the whole lifetime of the facility are listed. As important licensing documents the safety analysis report as well as the operational limits and conditions are emphasized. The licensing documentation needs to be periodically updated to take into account any modification made to the facility. This section further emphasizes the need for regulatory inspection and enforcement as required in /IAEA 16g/.

Management and verification of safety for a nuclear fuel cycle facility

This section combines with two different topics: management of safety and verification of safety. First, it is clearly stated that the responsibility for nuclear and radiation safety rests with operating organization. The operating organization shall establish and implement safety, health and environmental policies that give protection and safety the overriding priority warranted by their significance. An integrated management system shall be established, implemented, assessed and continuously improved to ensure that all safety requirements at all stages of the lifetime are met. The second topic deals with the verification of safety of the facility. It is required that the operating organization analyses safety and that this assessment needs to be independently reviewed to demonstrate compliance with regulatory requirements. It is also required, that periodic safety reviews shall be performed to identify potential safety improvements during the lifetime of the facility. Furthermore, an independent safety committee shall be established. This internal committee shall advice the management of the operating organization on all safety aspects related to the facility.

Site evaluation for nuclear fuel cycle facilities

This section highlights the need for safety evaluation and the periodic reassessment of the site evaluation. In addition, SSR-1 provides more detailed and comprehensive requirements related to site evaluation. However, this section contains no specific aspects to be considered during the site evaluation for nuclear fuel cycle facilities.

Design of nuclear fuel cycle facilities

This section deals with design requirements for nuclear fuel cycle facilities. The first part deals with the main safety functions (i.e. fundamental safety functions), application of the defence in depth concept, radiation protections, safety classification, equipment qualification, plant states (including postulated initiating events) and general design principles (e.g. redundancy, diversity, protection against internal and external hazards). The second part of this section deals with more specific requirements for certain systems, structures and component like management of atmospheric and liquid radioactive discharges, control over the transfer of radioactive material and other hazardous material, criticality safety, protection against toxic chemicals, instrumentation and control systems, compressed air systems or monitoring and analysis of process chemistry.

Construction

This section includes requirements to establish processes to ensure that the design specification and the design intent will be met. It is important, that safety implications of design changes during construction need to be assessed and documented.

Commissioning

Within this section a commissioning programme is required to ensure that the facility as built meets the design objective and the performance criteria. It is distinguished between "cold" and "hot" commissioning. In contrast to commissioning of nuclear power plants or research reactors, where first fuel loading increases the hazard potential of the facility, for fuel cycle facilities the introduction of radioactive materials changes over from cold to hot commissioning. Such a criterion may be reasonable for fusion power plants, too.

Operation

This section starts with requirements regarding the operating organization, its structure and functions. Requirements for the operating personnel and its qualification, training and retraining are also included. Specific operational aspects are addressed as listed below

- Operational limits and conditions;
- Ageing management;
- Operational control of modifications;
- Records and reports;
- Conduct of safety related activities;
- Facility operations;
- Operating procedures;
- Operational housekeeping and material conditions;
- Maintenance, periodic testing and inspection;
- Nuclear criticality safety;
- Radiation protection programme and management of radioactive waste and effluents;
- Operational safety programmes;
- Operational accident management programme;
- Feedback of operating experience.

Preparation for decommissioning

The main requirement is to establish and update the decommissioning plan. However, comprehensive requirements for decommissioning are provided in GSR Part 6.

Interfaces between safety and security

This section addresses the interface between safety, nuclear security and the State system of accounting for, and control of, nuclear material. Even a fusion power plant does not contain any fissionable material falling under the safeguards regulations, accounting and control of the tritium inventory may be an issue for fusion power plants and thus its interface to safety and security needs to be considered.

4.4.12 IAEA Glossary

The IAEA Safety Glossary (2018 Edition) /IAEA 18/ defines some terms related to nuclear facilities, such as facility, building or equipment itself but also in terms of activities conducted and the materials used. The term "fusion facility" might be assigned and applicable to a certain definition in the IAEA glossary (e.g. to the rather generic term "facility"), nevertheless uncertainties appear in connection with other terms linked to it. It has to be kept in mind that the term "nuclear" or "nuclear energy" is always understood as the energy obtained by the fission of nuclei. However, the materials used in fusion or fission reactors are entirely different. In summary, the current definitions in the IAEA glossary do not clearly cover or include the term "fusion facilities". Table 1 compares some basic terms related to nuclear facilities and associated processes presenting definitions and characteristics of the IAEA Safety Glossary and their applicability to fusion facilities.

Table 1 Definitions of terms in the IAEA Safety Glossary and their applicability

Term	IAEA Glossary Definition	Comment
Facilities	A general term encompassing nuclear facilities, uses of all sources of ionizing radiation, all radioactive waste management activities, transport of radioactive material and any other practice or circumstances in which people may be subject to exposure to radiation from naturally occurring or artificial sources.	Applicable
Nuclear facility	1. A facility (including associated buildings and equipment) in which nuclear material is produced, processed, used, handled, stored or disposed of. (i)* Also nuclear fuel cycle facility.	Not applicable, since "nuclear material" is not applicable
Nuclear installation	Any nuclear facility subject to authorization that is part of the nuclear fuel cycle, except facilities for the mining or processing of uranium ores or thorium ores and disposal facilities for radioactive waste. (i) This definition thus includes: nuclear power plants; research reactors (including subcritical and critical assemblies) and any ad-joining radioisotope production facilities; storage facilities for spent fuel; facilities for the enrichment of uranium; nuclear fuel fabrication facilities; conversion facilities; facilities for the reprocessing of spent fuel; facilities for the predisposal management of radioactive waste arising from nuclear fuel cycle facilities; and nuclear fuel cycle related research and development facilities.	In principle not applicable, since "nuclear fuel cycle" is in principle not applicable
Fuel cycle	All operations associated with the production of nuclear energy. (i) Operations in the nuclear fuel cycle associated with the production of nuclear energy include the following: (a) Mining and processing of uranium ores or thorium ores; (b) Enrichment of uranium; (c) Manufacture of nuclear fuel; (d) Operation of nuclear reactors (including research reactors); (e) Reprocessing of spent fuel; (f) All waste management activities (including decommissioning) relating to operations associated with the production of nuclear energy; (g) Any related research and development activities.	Not applicable, since "nuclear energy" is associated with "nuclear fuel" (even if "nuclear energy" is not explicitly defined)
Nuclear fuel	Fissionable nuclear material in the form of fabricated elements for loading into the reactor core of a civil nuclear power plant or research reactor.	Not applicable

*In many cases, the recommended definition(s) is(are) followed by further information as appropriate in the IAEA Safety Glossary denoted by an information symbol, here presented by (i).

4.5 Screening of IAEA Safety Guides

IAEA publishes guidance documents which are usually more technically specific than the safety requirements. Theses safety guides describe (technical) possibilities to achieve the mandatory safety requirements. Within this section, the applicability of IAEA Safety Guides on future fusion facilities is screened. It is identified, if the guide is

- applicable (marked green),
- not applicable (marked red) or
- could complement requirements applicable to all radiation facilities to enhance the safety of future fusion facilities, provided it is done in a **proportionate and targeted manner** (marked yellow).

It has to be mentioned, that even a guide identified as applicable or applicable after adaption on fusion power plants, this should be understood as such that the topic of the guide is applicable. Possibly the guide should not be directly applied but adapted and potentially simplified taking into account lower hazards from fusion power plants. In this process there might also be some potential of grouping guides. In addition, the evaluation of the applicability might also change when going further into an in-depth evaluation and adaptation process. The applicability of guides will potentially also depend on the design of the fusion facility applying a graded approach taking into account the hazard potential, as well as on the overall national regulatory regime. The existing IAEA safety guides are grouped to the different stages of the lifetime of a facility:

- Site evaluation;
- Design;
- Construction and commissioning;
- Operation;
- Decommissioning and waste management.

In addition, the following topics are used for grouping the respective safety guides:

- Radiation protection;
- Leadership and management;
- Safety assessment.

4.5.1 Site evaluation

Requirements for site evaluations are provided in IAEA Safety Standard Series No. SSR-1 /IAEA 19/ and substantiated by the guides listed in Table 2. Site evaluation is an important step to identify the suitability of a candidate site to host a fusion facility. Site specific conditions are important to estimate the dispersion of released radionuclides in normal operation and accident conditions, the natural and human induced hazards which may impact the facility and the possibility to implement measure of emergency preparedness and response. The effort for the site evaluation process may require a graded approach to be commensurate with the radiological hazard potential of the future fusion facility.

Table 2 Applicability of safety guides related to site evaluation

Safety Standard No.	Title	Comment
SSG-9	Seismic Hazards in Site Evaluation for Nuclear Installations	Guidance specific for nuclear installations; need to be adapted to specific needs and the hazard potential of fusion facilities
SSG-18	Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations	Guidance specific for nuclear installations; need to be adapted to specific needs and the hazard potential of fusion facilities
SSG-21	Volcanic hazards in Site evaluation for Nuclear Installations	Guidance specific for nuclear installations; need to be adapted to specific needs and the hazard potential of fusion facilities
SSG-35	Site Survey and Site Selection for Nuclear Installations	Guidance specific for nuclear installations; need to be adapted to specific needs and the hazard potential of fusion facilities
NS-G-3.1	External Human Induced Events in Site Evaluation for Nuclear Power Plants	Guidance specific for NPP; need to be adapted to specific needs and the hazard potential of fusion facilities; guide under revision
NS-G-3.2	Dispersion of Radioactive Material in Air and Water and Consideration of Population Distribution in Site Evaluation for NPPs	Guidance specific for NPP; need to be adapted to specific needs and the hazard potential of fusion facilities; guide under revision
NS-G-3.6	Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants	Guidance specific for NPP; need to be adapted to specific needs and the hazard potential of fusion facilities; guide under revision

4.5.2 Design

No dedicated safety requirements and consequently no specific safety guides exist for the design of fusion facility. However, with IAEA Safety Standard Series No. SSR-2/1, SSR-2/2, SSR-3 and SSR-4 requirements for the design and operation of nuclear power plants, research reactors and fuel cycle facilities, respectively, exists and are substantiated by several safety guides. These safety guides are less technology neutral and are originally written having specific facilities in mind, like nuclear power plants with light water reactors, or fuel cycle facilities to produce nuclear fuel for nuclear power plants (see Table 3).

Table 3 Applicability of safety guides related to design

Safety Standard No.	Title	Comment
SSG-5	Safety of Conversion Facilities and Uranium Enrichment Facilities	Not applicable
SSG-6	Safety of Uranium Fuel Fabrication Facilities	Not applicable
SSG-7	Safety of Uranium and Plutonium Mixed Oxide Fuel Fabrication Facilities	Not applicable
SSG-15	Storage of Spent Nuclear Fuel	Guidance specific for LWR; need to be adapted to specific needs for storage of radioactive waste generated in fusion facilities
SSG-30	Safety Classification of Structures, Systems and Components in Nuclear Power Plants	Guidance specific for NPP; need to be adapted to specific needs of fusion facilities; the concept of Classification of Structures, Systems and Components is applicable
SSG-34	Design of Electrical Power Systems for Nuclear Power Plants	Guidance specific for NPP; need to be adapted to specific needs of fusion facilities; the importance of the electrical power system might depend on the design of fusion facility
SSG-37	Instrumentation and Control Systems and Software Important to Safety for Research Reactors	Guidance specific for research reactors; need to be adapted to specific needs and designs of fusion facilities
SSG-39	Design of Instrumentation and Control Systems for Nuclear Power Plants	Guidance specific for NPP; need to be adapted to specific needs of fusion and designs facilities
SSG-42	Safety of Nuclear Fuel Reprocessing Facilities	Not applicable
SSG-43	Safety of Nuclear Fuel Cycle Research and Development Facilities	Not applicable
SSG-52	Design of the Reactor Core for Nuclear Power Plants	Not applicable
SSG-56	Design of the Reactor Coolant System and Associated Systems for Nuclear Power Plants	Focussed on high pressure primary cooling loops of NPPs; applicability might strongly depend on the design of the fusion facility
SSG-62 superseded	Design auf Auxiliary Systems and Supporting Systems for Nuclear Power Plants	Guidance specific for water cooled reactors; need to be adapted to specific needs and design of fusion facilities;
SSG-63	Design of Fuel Handling and Storage Systems for Nuclear Power Plants	Not applicable, specific for LWR-fuel
NS-G-1.5 guide under revision	External Events Excluding Earthquakes in the Design of nuclear power plants	Guidance specific for water cooled reactors; need to be adapted to specific needs of fusion facilities; general approach applicable
NS-G-1.6 guide	Seismic Design and Qualification for Nuclear Power Plants	Guidance specific for water cooled reactors; need to be

Safety Standard No.	Title	Comment
under revision		adapted to specific needs of fusion facilities; general approach applicable
SSG-64	Protection against Internal Hazards in the Design of Nuclear Power Plants	Guidance specific for water cooled reactors; need to be adapted to specific needs and internal hazards of fusion facilities; general approach applicable
NS-G-1.13 guide under revision	Radiation Protection Aspects of Design for Nuclear Power Plants	Specific for NNP; need to be adapted to the specific radiation protection needs of a fusion facility
NS-G-4.6 guide under revision	Radiation Protection and Radioactive Waste Management in the Design and Operation of Research Reactors	Specific for research reactors; need to be adapted to the specific radiation protection needs of a fusion facility

4.5.3 Construction and commissioning

Four safety guides provide recommendations regarding the construction and commissioning phase of a nuclear installation. During construction it is important to ensure that all safety related Structures, Systems and Components (SSCs) will be constructed and installed in such a way, that the facility will be built as planed and that in particular civil constructions as well as installation of equipment is done in such way that the intended safety functions can be reliably performed. This needs to be considered during the construction of fusion facilities, too. Equipment qualification is a further important aspect during this phase. By an appropriate equipment qualification, it is ensured that the procured equipment can reliably perform the intended safety function under the expected environmental and seismic conditions. Before commencing operation, the new facility needs to be commissioned. This is an important step to verify that that fusion facility as built will perform all its functions, in particular its safety functions, as designed and demonstrated by the safety analysis. The applicability of the relevant safety guides is listed in Table 4.

Table 4 Applicability of safety guides related to construction and commissioning

Safety Standard No.	Title	Comment
SSG-28	Commissioning for Nuclear Power Plants	Specific for nuclear power plants; need to be adapted to the specific needs to commission a fusion facility
SSG-38	Construction for Nuclear Installations	Specific for nuclear installations; minor modifications necessary
NS-G-4.1	Commissioning of Research Reactors	Specific for research reactors; need to be adapted to the specific needs to commission a fusion facility, guide under revision
New guide	Equipment Qualification for Nuclear Installations	In principle applicable, To be published soon

4.5.4 Operation

Table 5 lists the relevant IAEA Safety Standards with respect to the operation of nuclear facilities.

Table 5 Applicability of safety guides related to operation

Safety Standard No.	Title	Comment
SSG-10	Ageing Management for Research Reactors	Specific for research reactors; need to be adapted to specific needs and ageing issues of fusion facilities
SSG-13 guide under revision	Chemistry Programme for Water Cooled Nuclear Power Plants	Focussed on water cooled reactors; need to be adapted to specific needs of cooling water in fusion power plants;
SSG-24 guide under revision	Safety in the Utilization and Modification of Research Reactors	Specific for research reactors; need to be adapted to specific needs of fusion facilities;
SSG-27	Criticality Safety in the Handling of Fissile Material	Not applicable
SSG-48	Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants	Guidance specific for NPP; need to be adapted to specific needs and ageing issues of fusion facilities
SSG-50	Operating Experience Feedback for Nuclear Installations	Specific for nuclear installations; minor modifications necessary; concept of OPEX applicable
SSG-51	Human Factors Engineering in the Design of Nuclear Power Plants	Specific for NPP; need to be adapted to specific needs of fusion facilities
SSG-54	Accident Management Programmes for Nuclear Power Plants	Strongly focussed on accident management programmes for existing nuclear power plants withs light water reactors and typical accidents, need to be adapted to specific accidents of fusion power plants and their management
NS-G-2.1 guide under revision	Fire Safety in the Operation of Nuclear Power Plants	Specific for NPP; need to be adapted to specific needs of fusion facilities; minor modifications necessary;
NS-G-2.2	Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants	Focussed on OLCs for NPPs, need to be adapted to address specific OLCs of fusion power plants.
NS-G-2.3 guide under revision	Modifications to Nuclear Power Plants	Specific for NPP; need to be adapted to specific needs of fusion facilities;
NS-G-2.4 guide under revision	The Operating Organization for Nuclear Power Plants	Specific for NPP; need to be adapted to specific needs of fusion operating organization;
NS-G-2.5	Core Management and Fuel Handling for Nuclear Power Plants	Not applicable

Safety Standard No.	Title	Comment
NS-G-2.6 guide under revision	Maintenance, Surveillance and In-Service Inspection in Nuclear Power Plants	Specific for NPP; need to be adapted to specific needs of fusion facilities
NS-G-2.8 guide under revision	Recruitment, Qualification and Training of Personnel for Nuclear Power Plants	Applicable by applying a graded approach
NS-G-2.14 guide under revision	Conduct of Operations at Nuclear Power Plants	Focussed on nuclear power plants; need to be adapted to specific needs of operating fusion power plants,
NS-G-4.3	Core Management and Fuel Handling for Research Reactors	Not applicable
NS-G-4.4 guide under revision	Operational Limits and Conditions and Operating Procedures for Research Reactors	Focussed on OLCs for research reactors; need to be adapted to address specific OLCs of fusion power plants
NS-G-4.5 guide under revision	The Operating Organization and the Recruitment, Training and Qualification of Personnel for Research Reactors	Specific for research reactors; in principle applicable

4.5.5 Decommissioning and waste management

Requirements for decommissioning and predisposal management of radioactive waste are included in IAEA Safety Standard Series No. GSR Part 6 and GSR Part 5. The safety guides listed in Table 6 substantiate these two requirements.

Table 6 Applicability of safety guides related to decommissioning and predisposal management of radioactive waste

Safety Standard No.	Title	Comment
GSG-1	Classification of Radioactive Waste	Applicable
SSG-15	Storage of Spent Nuclear Fuel	Not applicable
SSG-40	Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors	Need to be adapted to radioactive waste generated in fusion facilities
SSG-47	Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities	Specific for nuclear installations; need to be adapted to specific needs and design of fusion facilities
SSG-49	Decommissioning of Medical, Industrial and Research Facilities	Specific for medical, industrial and research facilities; need to be adapted to specific needs and design of fusion facilities
SSG-60	Management of Residues Containing Naturally Occuring Radioactive Material from Uranium Production and Other Activities	Not applicable
WS-G-5.1	Release of Sites from Regulatory Control on Termination of Practices	Applicable
WS-G-6.1	Storage of Radioactive Waste	Applicable

4.5.6 Radiation protection

IAEA Safety Standard Series No. GSR Part 3 /IAEA 14/ contains requirements for radiation protection in general. Additional requirements to protect the public in case of a nuclear or radiological emergency are laid down in IAEA Safety Standards Series No. GSR Part 7 /IAEA 16h/. The following safety guides addresses recommendations to meet those requirements. The applicability of those safety guides potentially relevant for fusion power plants are identified in Table 7.

Table 7 Applicability of safety guides related to radiation protection

Safety Standard No.	Title	Comment
GSG-7	Occupational Radiation Protection	Applicable
GSG-8	Radiation Protection of the Public and the Environment	Applicable
GSG-9	Regulatory Control of Radioactive Discharges to the Environment	Applicable
GSG-10	Prospective Radiological Environmental Impact Assessment for Facilities and Activities	Applicable
SSG-8	Radiation Safety of Gamma, Electron and X ray Irradiation Facilities	Not applicable
RS-G-1.7 guide under revision	Application of the Concepts of Exclusion, Exemption and Clearance	Applicable
RS-G-1.8 guide under revision	Environmental and Source Monitoring for Purposes of Radiation Protection	Applicable
RS-G-1.9	Categorization of Radioactive Sources	Not applicable
RS-G-1.10	Safety of Radiation Generators and Sealed Radioactive Sources	Not applicable

4.5.7 Leadership and management

Requirements concerning the leadership and management for safety are published in IAEA Safety Standard Series No. GSR Part 2 /IAEA 16d/. Table 8 list the relevant safety guides further substantiating this topic. All three guides are very similar in its structure and topics addressed. Those guides can be considered as a good basis for developing further guidance documents for the management system for fusion power plants.

Table 8 Applicability of safety guides related to leadership and management for safety

Safety Standard No.	Title	Comment
GS-G-3.1	Application of the Management System for Facilities and Activities	Applicable
GS-G-3.3	The Management System for the Processing, Handling and Storage of Radioactive Waste	Applicable
GS-G-3.5	The Management System for Nuclear Installations	Applicable

4.5.8 Safety assessment

The last topical area addresses safety assessments, which is important for safety demonstration of a future fusion power plants. IAEA Safety Standard Series No. GSR Part 4 Rev.1 /IAEA 16e/ contains the overarching requirements for safety demonstration. The following safety guides have been identified providing guidance to meet those established requirements. The applicability for fusion power plants is indicated in Table 9. It is emphasized, that the two PSA guides are strongly focused on severe accidents at nuclear power plants having core damage frequency (CDF, result of PSA Level 1) and larger early releases frequency (LERF, results PSA Level 2) as the main metrics. If probabilistic safety analyses are considered beneficial for fusion power plants, specific guidance for such kind of facilities need to be developed.

Table 9 Applicability of safety guides related to safety assessment

Safety Standard No.	Title	Comment
SSG-2	Deterministic Safety Analysis for Nuclear Power Plants	Modification necessary, no core melt accidents at fusion power plants
SSG-3 Guide under revision	Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants	Specific for NPP; need to be adapted to specific design of fusion facilities; concept of Level 1 PSA applicable
SSG-4 Guide under revision	Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants	Specific for NPP; need to be adapted to specific design of fusion facilities; concept of Level 2 PSA applicable
SSG-20	Safety Assessment for Research Reactors and Preparation of the Safety Analysis Report	Structure needs to be adapted to fusion power plants and fusion specific systems
SSG-61	Format and Content of the Safety Analysis Report for Nuclear Power Plants	Structure needs to be adapted to fusion power plants and fusion specific systems
GSG-3	The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste	Applicable
WS-G-5.2	Safety Assessment for the Decommissioning of Facilities Using Radioactive Material	Applicable

4.6 Review and evaluation of relevant fusion specific IAEA publications

4.6.1 IAEA TECDOC-1851 "Integrated Approach to Safety Classification of Mechanical Components for Fusion Applications"

This TECDOC /IAEA 18c/ deals with the safety classification of structures, systems and components (SSCs). This document reflects insights from IAEA Safety Standard Series No. SSR 2/1 /IAEA 16/ and the WENRA Report on safety of new nuclear power plant designs /WENR 13/ with respect to the defence-in-depth concept. With respect to fusion facilities, it is stated that design extension conditions may be considered to ensure that the design will be able to cope with common cause failures or multiple failure events. The document also reflects on the IAEA Safety Standard Series No. SSG-30 /IAEA 14b/ providing guidance to perform safety classification of SSCs in nuclear power plants. TECDOC-1851 is strongly governed by the conceptual design of ITER and EU-DEMO. A detailed description is provided to identify relevant SSCs and functions applying the methods of physical breakdown structure and functional break down structure. This supports the basic understanding of the plant design and is a prerequisite for an appropriate categorization of safety functions and classification of structures, systems and components.

This document provides valuable inputs concerning the location of radioactive material in a fusion facility. Tritium may be present in gaseous form, liquid as tritiated water, or absorbed within materials in the following locations /IAEA 18c/:

- Retained in the vacuum vessel: adsorbed on surfaces, permeated into the structure of in-vessel components, or absorbed in accumulated dust from erosion of plasma-facing surfaces;
- In fuel cycle equipment (fuelling, pumping, processing);
- In breeder blankets and the tritium extraction system;
- In remote handling equipment used to remove and transport in-vessel components;
- In storage of in-vessel components awaiting maintenance or disposal;
- In hot cells used to perform maintenance on removed in-vessel components;
- In coolants, due to permeation;
- In the atmosphere of rooms containing tritium systems.

The second group of radioactive materials results from neutron activation and occurs typically in the following locations /IAEA 18c/:

- In the materials of plasma-facing components;
- Accumulated in-vessel dust from plasma-facing surface erosion;
- Activated corrosion products in liquid coolants (e.g. water or lead-lithium);
- The vacuum vessel itself and ex-vessel components (at a lower level);
- Activated shielding materials.

It further discusses the usage of the terms safety functions and supporting functions for ITER and fundamental safety functions and supporting functions for DEMO. Both approaches are not consistent with the definitions of the IAEA safety glossary /IAEA 18//IAEA 18/.

The report indicates that for fusion power plants heat removal of in-vessel components may be required to avoid structural damage to those components and thus residual heat removal may be considered as a fundamental safety function.

Within TECDOC-1851 an overview of a radiation zoning scheme is provided based on dose rates. However, the applicability of such a radiation zoning scheme strongly depends on the national legal provisions.

It can be concluded that the methodology for classification of SSCs described in TECDOC-1851 is much more complicated than the methodology proposed in SSG-30. In contrast to SSG-30, the severity of a failure of a safety function is directly linked to a certain plant state and a certain level of defence-in-depth, respectively. In SSG-30 the consequences of a failure of a safety function are regarded independently of its assignment to a specific plant state.

4.6.2 IAEA TECDOC-440 "Fusion Reactor Safety"

The IAEA TECDOC-440 "Fusion Reactor Safety" is a report of the third meeting of the Technical Committee on Fusion Reactor Safety organized by the IAEA in November 1986 in Culham. It includes all papers presented at that meeting and the summaries of the different workshop sessions.

The following topics were covered through four sessions:

- Statements on national-international fusion safety programmes;
- Operating and system safety;
- Waste management and decommissioning;
- Environmental impact.

The sessions were followed by a sub-group discussion to identify high priority research and development issues in fusion safety. It was generally recommended that the subject of fusion safety receives continuous attention in the future. The document contains some basic considerations about the safety of a fusion reactor. However, due to the age of the document, it does not reflect the state of the art in science and technology.

4.6.3 IAEA Technical Report Series No. 324 "Safe Handling of Tritium"

The IAEA Technical Reports Series No. 324 "Safe Handling of Tritium – Review of Data and Experience" /IAEA 91/, published in 1991, provide practical guidance and recommendations and is the result of an IAEA Technical Committee Meeting on Safe Handling of Tritium held in Vienna from 13 to 17 October 1986 and a consultants meeting held in Chalk River, Ontario, in March 1987, with final compilation and review in December 1989. It covers topics like dosimetry and monitoring of tritium, protective clothing, safe practices in tritium handling laboratories, tritium compatible materials and management of tritiated wastes. It was anticipated that it will be valuable as resource material to all users of tritium, including those involved in fusion R&D. The environmental behaviour of tritium and its impact on the public at large are not addressed.

The technical report addresses primarily requirements for the safe handling of tritium in the following types of facility:

- Tritium handling laboratories;
- Industrial scale nuclear facilities such as heavy water reactors, tritium removal plants and fission fuel reprocessing plants;
- Facilities for manufacturing commercial tritium containing devices and radio-chemicals.

It is stated that the requirements of nuclear fusion reactors are not addressed specifically, because of a lack of handling experience with them. However, much of the material covered in this publication is expected to be relevant to them as well.

4.6.4 IAEA TECDOCs currently being developed

Currently a working group at the IAEA is developing two TECDOCs with the topics design safety of fusion facilities and on the regulatory framework for them. The first one on design safety will likely include recommendations or requirements for topics covering among other the design and safety assessment, the design basis, specific requirements for the design, provisions for the lifetime of a magnet fusion facility, radiation protection, requirements against non-radiological hazards, instrumentation and control systems, emergency systems, and other design considerations. The second one on the regulatory framework will cover topics like the safety objective and principles for magnetic fusion facilities, regulatory supervision, management and verification of safety, site evaluation, construction, commissioning, operation, preparation for decommissioning, and interfaces between safety and security.

Both TECDOCs are not yet published.

4.7 Review and evaluation of further fusion specific publications

4.7.1 DOE-STD-6002-96 – Safety of Magnetic Fusion Facilities: Requirements

The DOE Standard DOE-STD-6002-96 "Safety of Magnetic Fusion Facilities: Requirements" /DOE 96/ provide requirements for developing design and operations envelopes to ensure safety of magnetic fusion facilities. Also, safety principles are established to provide a framework within which the requirements can be implemented to build safety into fusion facility design and operations. These standards are formally only applicable to facilities operated by the DOE.

It was developed by the US Department of Energy and comprises the state of science and technology at the time it was published. The requirements have been derived from US Federal law, policy and other documents. In addition to specific safety requirements, it is stated that broad direction is given in the form of safety principles that are to be implemented and within which safety can be achieved.

The overall safety policy demand fusion facilities to be designed, constructed, operated and removed from service in a way, that will ensure the protection of workers, the public, and the environment. In detail, the safety policy is based on the following points need to be integrated:

- The public shall be protected such that no individual bears significant additional risk to health and safety from the operation of those facilities above the risks to which members of the general population are normally exposed.
- Fusion facility workers shall be protected such that the risks to which they are exposed at a fusion facility are no greater than those to which they would be exposed at a comparable industrial facility.
- Risks both to the public and to workers shall be maintained as low as reasonably achievable (ALARA).
- The need for an off-site evacuation plan shall be avoided.
- Wastes, especially high-level radioactive wastes, shall be minimized.

The DoE Standard provide general requirements for fusion facilities which should be considered by developing a regulatory framework for fusion facilities. Since the DoE standard was published a quarter of a century ago, the requirements should be reviewed in the light of the current state of science and technology, e.g. the occupational exposure dose limits provided are not in line with current EU legislation.

4.7.2 DOE-STD-6003-96 – Safety of Magnetic Fusion Facilities: Guidance

The DOE Standard DOE-STD-6003-96 "Safety of Magnetic Fusion Facilities: Guidance" /DOE 96b/ provide guidance for the implementation of the requirements identified in DOE-STD-6002-96. The aim is to provide a fairly complete, though not exhaustive, set of instructions that if followed will contribute to the achievement of safety.

The guidance is stated to be intended for managers, designers, operators, and other personnel with safety responsibilities for facilities designated as magnetic fusion facilities. In comparison to DOE-STD-6002-96, which is generally applicable to a wide range of fusion facilities, DOE-STD-6003-96 is described to be concerned mainly with the implementation of those requirements in large facilities such as ITER. The concepts presented may therefore also be applied to other magnetic fusion facilities. It is stressed that it is oriented toward regulation in the DoE environment as opposed to regulation by other regulatory agencies.

As for DOE-STD-6002-96, the guidance provides an important input and should be considered for developing a regulatory framework. However, the guidance would have to be adapted to the current state of science and technology and the overall legislative framework. This can be accompanied by taking the current discussions for an US regulatory framework for fusion energy systems into account

4.7.3 GRS 389 – Review of the safety concept for fusion reactor concepts and transferability of the nuclear fission regulation to potential fusion power plants

The study GRS-389 /RAED 16/ from the year 2016 summarizes the state of the art in science and technology of the safety concept for future fusion power plants (FPPs) and examines the transferability of the nuclear fission regulation to the concepts of future fusion power plants.

It is analysed whether the safety concept of fusion power plants fits to the existing nuclear regulation, which was developed for the current commercial nuclear (fission) power reactors. The study is based on different European studies, the licensing documentation of the ITER project as well as the current German nuclear safety regulation, i.e. safety requirements for nuclear power plants of 2012. These studies for fusion power plants are based on plant concepts using a plasma of deuterium and tritium confined by magnetic fields (tokamak and stellarator).

The study concludes that the radioactive inventory of a fusion power plant shows the necessity to confine the inventory safely. The potential release of tritium has been identified as a major potential hazard. The safety concept of a fusion power plant is based on the concept of defence-in-depth which emphasises the use of inherent characteristics or passive safety mechanisms. Each postulated event, the resulting plant states and the measures and installations which are foreseen to control design base accidents (DBA) and beyond design base accidents (BDBA) can be assigned to the different levels of defence.

According to the summary, the safety properties of a fusion power plant, however, appear to be much more advantageous compared with those of a fission reactor:

- Due to the nature of the fusion reaction only a small amount of tritium is available to sustain the burning fusion plasma, sufficient only for a few minutes of plasma burn. The power densities are comparably small (few MW/m^3).

- The power generation can be stopped reliably by switching off the fuel supply. During failures of systems (magnets, cooling, vacuum leaks etc.) a plasma quench will occur by inherent processes, and hence the energy production is stopped.

- Also, potential power excursions would lead to the termination of power production. Criticality accidents are principally excluded.

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

4.7.4 IRSN report "Nuclear Fusion Reactors // safety and radiation protection considerations for demonstration reactors that follow ITER facility"

The aim of the IRSN report "Nuclear Fusion Reactors // safety and radiation protection considerations for demonstration reactors that follow ITER facility" /PERR 17/ is to explain the safety and radiation protection issues which need to be examined while designing future nuclear fusion reactors after ITER. It is focused on fusion facilities using the "tokamak" concept.

Future demonstration reactors will mainly differ from the ITER reactor by seeking to attain tritium self-sufficiency and achieve significant longer operating times. The conceptual design of the "tokamak" is indeed the same for ITER and the currently planned nuclear fusion facilities, but according to the report, the differences will have a significant impact on design and a direct influence on safety.

After introducing the “tokamak” concept and general design aspects of a nuclear fusion facility, the ITER fusion facility and intermediate experimental reactors as well as fusion power rector concepts like DEMO are described. Chapter six describes the main differences between ITER and the planned reactors and Chapter seven safety and radiation protection issues to be examined from the design-phase of DEMO reactors.

Following the work obtained, “IRSN stresses that designers should examine the following subjects as a priority:

- Residual heat removal, taking into account the design envisaged for the tokamak cooling systems and those for the blanket sectors when they are transferred between the vacuum vessel and the hot cells and during their storage in these cells;

- Optimization of the doses which workers receive depending on robotization and the choice of materials;

- The types of accident considered for the ITER facility and the specific types of accident that could be associated with the design of DEMO rectors;

- Possibilities for limiting the overall quantity of tritium present in the facility and releases by various main paths for gaseous tritium releases. In this respect, the choice of tritium breeding blankets would appear to be a key factor;

- Identification of the waste management constraints based on the general policy of the country hosting the reactor”.

To conclude, the IRSN report provide a vital input for developing a general regulatory framework for fusion facilities and should be incorporated into the overall strategy.

4.7.5 CNSC Tritium Study Project

The Canadian Nuclear Safety Commission (CNSC) performed a large study on safety issues related to tritium. The results are documented in a series of publications:

- Standards and Guidelines for Tritium in Drinking Water /CNC 08/;

- Investigation of Environmental Fate of Tritium in the Atmosphere /CNSC 09a/;

- Tritium Releases and Dose Consequences in Canada in 2006 /CNSC 09b/;

- Evaluation of Facilities Handling Tritium/CNSC 10a/;

- Tritium Activity in Garden Produce from Pembroke in 2007 and Dose to the Public /CNSC 10b/;

- Health Effects, Dosimetry and Radiological Protection of Tritium /CNSC 10c/;

- Tritium Studies Project Synthesis Report /CNSC 11/;

- Environmental Fate of Tritium in Soil and Vegetation /CNSC 13/;

- Implementation of Recommendations from the Tritium Studies Synthesis Report /CNSC 19/.

In/CNSC 10a/ the practices in Canadian facilities handling tritium were evaluated. The report covers the Canadian facilities (the tritium removal facility of the Darlington NPP, the tritium laboratory of AECL Chalk River Laboratory, tritium handling at SRB technology, the tritium laboratory of Kinectrics Incorporated and the tritium removal facility of GE Hitachi Nuclear Energy Canada Inc) as well as facilities abroad (GE Healthcare Ltd /UK, NTP Radioisotopes (Pty) Ltd./ZA, and MB-Microtec AG/CH). Thus, this report covers a broad range of various tritium handling facilities. The report identified the following ten good practices in tritium handling which are also applicable for fusion facilities:

(1) High performance vacuum equipment should be used for gaseous tritium handling facilities.
(2) High-quality primary containment is the most important control feature for handling tritium.
(3) Uranium getter beds for operational storage of tritium gas are widely used, and considered good practice.
(4) Titanium getter beds are considered good practice for the long-term storage of tritium gas.
(5) Intentional release of tritium gas from pipe work and vessels is not good practice. Direct adsorption onto getter beds or inert gas purging and capture onto getter beds would be good practice.
(6) The use of oil-free scroll pumps facilitates the use of subsequent technology on the exhaust. There is some limited evidence that they may reduce the concentration of HTO being released in the exhaust.
(7) The removal of tritium gas and HTO from vacuum pump exhaust should be considered good practice, provided there is a treatment or disposal route for the abated tritium.
(8) Secondary containment of getter beds can reduce chronic releases, particularly for beds that are used at elevated temperatures over prolonged periods of time.
(9) The most effective use of abatement technology occurs at the point of generation of the release.
(10) Ventilation systems should have appropriately designed release points (stacks) to achieve good dispersion of tritium gas and tritiated water vapour.

4.8 CONCLUSION

A screening was performed on existing Council Directives and Directives of the European Parliament as well as on additional existing safety requirements, guidance documents and potential further relevant documents addressing safety requirements for nuclear facilities, such as nuclear power plants, research reactors or fuel cycle facilities. As fusion facilities are based on a different technology compared to the before mentioned nuclear installations utilising fissile material, the applicability of those requirements and guidance on fusion facilities was discussed.

At present, European Council Directives do not address specific requirements for fusion facilities, but place requirements which are generally applicable for all facilities. Due to the general character of the requirements in European Council Directives some of the directives screened are directly applicable to fusion power plants. These are:

- Council Directive 2013/59/Euratom – European Basic Safety standards;
- Directive 2011/70/EURATOM – Radioactive Waste and Spent Fuel Management;
- Council Directive 98/24/EC – Chemical agents at workplaces;
- Directive 2004/37/EC – Carcinogens or mutagens at workplaces;
- Directive 2013/35/EU – Worker exposure to electromagnetic fields;
- Directive 2011/65/EU – Hazardous substances in electrical and electronic equipment.

Two Directives apply only for nuclear facilities, meaning that fusion power plants are out of the scope of them. These are:

- Council Directive 2009/71/Euratom amended by 2014/87/Euratom – Nuclear Safety Directive;
- Commission Regulation (Euratom) No 302/2005 – Nuclear Safeguards.

Even if fusion reactors are not in the scope of the Nuclear Safety Directive, the requirements can be in principle applied to them even if a detailed examination do determine whether corresponding adjustments have to be made is deemed necessary. Concerning the Directive for Nuclear Safeguards, it does not apply to fusion power plants as this kind of facility is not dealing with nuclear material. According to the high inventory of tritium expected in future fusion plants, it has to be discussed if safeguard activities need to be extended for fuel containing tritium.

The IAEA Safety Standard Series provides a hierarchically structured set of safety requirements and substantiating safety guides. Whereas the high-level requirements are directly applicable to fusion power plants, specific safety requirements and guidance addressing typical systems of fusion power plants are not yet addressed. The Safety Fundamentals /IAEA 06/ contains the fundamental safety objective and 10 safety principles and are unrestrictedly applicable to fusion power plants. The same observation was made regarding the set of general safety requirements which can directly be applied to fusion facilities. In addition to the general safety requirements Part 1 to Part 5 the specific safety requirement SSR-1 is directly applicable to fusion power plants. The following list summarized directly applicable IAEA Safety Requirements:

- GSR Part 1, "Governmental, Legal and Regulatory Framework for Safety";
- GSR Part 2, "Leadership and Management for Safety";
- GSR Part 3, "Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards";
- GSR Part 4, "Safety Assessment for Facilities and Activities";
- GSR Part 5, "Predisposal Management of Radioactive Waste";
- GSR Part 6, "Decommissioning of Facilities";
- GSR Part 7, "Preparedness and Response for a Nuclear or Radiological Emergency";
- SSR-1 "Site Evaluation for Nuclear Installations".

IAEA has published specific safety requirements for different kind of facilities. These are the specific requirements for safety of nuclear power plants (SSR-2/1 and SSR-2/2), research reactors (SSR-3), nuclear fuel cycle facilities (SSR-4). All four specific safety requirements have in common that these documents deal with fission facilities utilizing or handling nuclear fuel. It was observed that all these four guides in principle follows a similar structure:

- Applying the safety objective, concepts, and principles
- Regulatory supervision and inspection
- Management and verification of safety
- Site evaluation
- Design
  - General design requirements
  - Specific design requirements
- Construction and commissioning
- Operation
- Preparation for Decommissioning
- Interface between safety and security

The topics regulatory supervision and inspection, management and verification of safety, site evaluation and decommissioning are comprehensively covered by GSR Part 1, GSR Part 2, GSR Part 4, SSR-1 and GSR Part 6, respectively. Considering that the before mentioned topics are comprehensively covered in separate documents and that the structure has been applied for several years in practice, the following structure can be proposed for a specific safety requirement for fusion power plants:

- Safety objective and safety concepts;
- General design requirements;

- Specific design requirements;
- Construction and commissioning;
- Operation;
- Interface between safety.

The requirements provided in the four specific safety requirements SSR 2/1, SSR 2/2, SSR-3 and SSR-4 provides a sound starting point for such fusion specific safety requirements. However, modifications are still necessary. The section dealing with specific design requirements for specific systems, structures and components in fusion power plants need to be developed taking into account the safety issues of fusion specific systems, structures and components.

The review and evaluation of additional fusion specific publication showed that they provide an important relevance for formulating requirements on fusion facilities.

It can be summarized that a legal and regulatory framework for fusion power plants needs to be established. The Directives of the European Commission establishes a common legal framework by implementing those Directives into the national legal frameworks of the Member States. With Council Directive 2013/59/Euratom and Directive 2011/70/Euratom the legal framework for radiation protection and waste management, respectively, has been established and applies to future fusion power plants too. In addition, conventional safety aspects are covered in various Directives dealing with hazardous substances or electromagnetic fields. In contrast to fission power plants, more specific legal requirements concerning the responsibility of the licensee for safety, technical safety objectives, review and assessment of safety have not yet been established on a European level. With respect to Council Directive 2009/71/Euratom a gap exists addressing similar high-level safety requirements for fusion power plants. It is also recommended to discuss the role of tritium accountancy within the nuclear safeguards, as Commission Regulation No. 302/2005 does not deal with tritium at all.

General safety objectives and principles as addressed in the IAEA Safety Standards are mostly applicable to future fusion power plants, as those requirements are written in a general and technology neutral manner. Nevertheless, requirements and guidance concerning fusion specific structures, systems and components have not yet been addressed in the current set of IAEA Safety Standards. It is recommended to close this gap by developing a regulatory framework for fusion power plants. In particular, IAEA safety guides dealing with certain safety aspects of structures, systems and components are strongly focused on those safety issues related to stationary (light) water cooled reactor designs. The description of the technical systems within this section and the safety aspects addressed in the screened documents provides a sound technical basis for developing such kind of fusion specific safety requirements.

It is expected that the adaptation and development of regulations and guides specific to fusion facilities will require significant work and time. Similar efforts are currently ongoing in different international fora (e.g. IAEA, NEA) for the development of appropriate regulations and guides for licensing of Small Modular Reactors (SMRs) and innovative reactors. These efforts potentially could have some input for some topics relevant for fusion (e.g. Emergency Preparedeness and Response).

5 RECOMMENDATIONS FOR THE IMPLEMENTATION OF A LEGAL AND REGULATORY FRAMEWORK SPECIFICALLY NEEDED FOR FUSION FACILITIES

The main objective of this section is to provide recommendations for the implementation of a legal and regulatory framework addressing the specific needs for regulating future fusion facilities utilizing tritium with magnetic confinement.

As described in Section 3 no dedicated regulatory framework for future fusion facilities exits so far. However, fusion facilities give rise to radiation risks and thus have to be regulated within a targeted and fusion specific framework. Safety issues of fusion facilities are neither comparable with those from nuclear power plants nor with those due to the use of radioactive sources in the medical and industrial area. Also, in the US for example there was a common opinion of the regulatory authority NRC and the fusion industry that a dedicated regulatory framework for fusion facilities would be an appropriate approach.

This study proposes a regulatory framework for future fusion facilities. In particular, safety requirements targeted to the specific safety issues of fusion facilities were drafted. These proposed requirements should be understood as a starting point for discussions with relevant stakeholders to develop such a regulatory framework. In addition, areas have been identified which require additional work to complete the regulatory framework for fusion facilities in the future.

A proposed structure of a regulatory framework for fusion facilities including examples for fusion specific requirements is given in ANNEX A.

5.1 Structuring a legal and regulatory framework for fusion facilities

The establishment of a legal and regulatory framework for fusion facilities can be directly derived from principle 2 of the IAEA Safety Fundamentals /IAEA 06/:

> *"An effective legal and governmental framework for safety, including an independent regulatory body, must be established and sustained."* /IAEA 06/

As fusion facilities are facilities that give rise to radiation risk, such facilities need to be regulated by a properly established legal and regulatory framework including an independent regulatory body. Requirement 2 of IAEA GSR Part 1 elaborates in more detail the elements of a legal and regulatory framework. Amongst others, the following topics shall be addressed by such a framework /IAEA 16g/:

- The safety principles for protecting people and the environment;
- The type of authorization that is required for (siting, construction,) operation (and decommissioning) of facilities;
- The authority and responsibility of the regulatory body for promulgating (or preparing for the enactment of) regulations and preparing guidance for their implementation;
- Provision for the involvement of interested parties and for their input to decision making;
- Provision for assigning legal responsibility for safety to the persons or organizations responsible for the facilities;
- Provision for the review and assessment of facilities;
- Provision for the inspection of facilities;
- Provision for preparedness for, and response to, a nuclear or radiological emergency;
- Provision for an interface with nuclear security.

In practice, the legal and regulatory framework can be represented as hierarchical pyramid. The top level is formed by acts or laws, which will be substantiated by orders or decrees on the second level. The third level can be formed by regulations, followed by guides. The lowest level is formed by codes and standards. Besides requirements for the siting, design, construction, operation and decommissioning of fusion facilities, requirements for the operating organizations, as well as for licencing and oversight of fusion facilities need to be addressed, including the establishment of the regulatory body. It is emphasized, that interfaces to existing legal and regulatory frameworks for nuclear facilities and radiation facilities already exists in many countries. Figure 2 illustrates a typical hierarchical structure of a legal and regulatory frameworks. The expectations concerning the information provided on each of the six levels is discussed in the following.

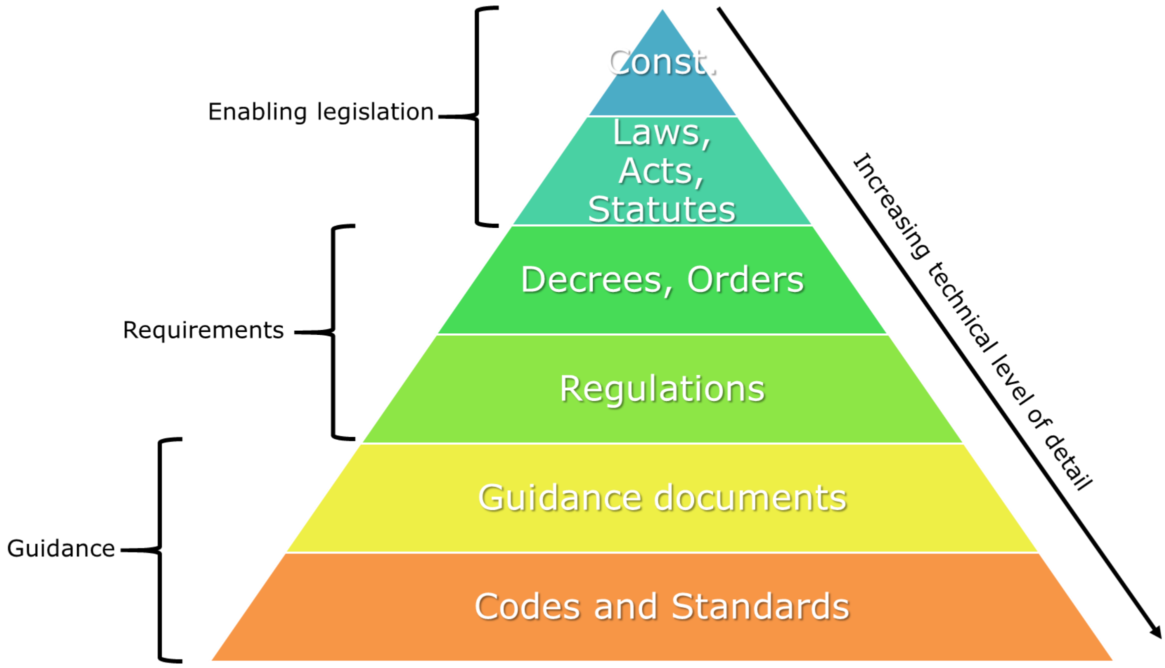

Figure 2 Typical structure of a legal and regulatory framework (based on /CHER 14/).

The first level is formed by the constitution. The second level encompasses laws, acts or statutes prepared by the government or parliament and adopted by the parliament. Here, the general framework governing activities involving nuclear energy and ionizing radiations is established and establishes the competent regulatory body with its functions and determines its responsibilities /CHER 14/. It further regulates main regulatory activities like notifications, authorizations, inspections, enforcement and penalties and includes high level requirements regarding radiation protection principles, primary responsibility and duties of authorized persons as well as provisions for occupational, medical and public exposures /CHER 14/. Specific provisions for nuclear facilities and radiation facilities, nuclear security, waste management, transport of radioactive material as well as emergency preparedness and response are established in /CHER 14/. These high-level provisions are further substantiated in decrees or orders issued by the government and sometimes adopted by the parliament. At this level more specific requirements for the protection of people and the environment against harmful effects of ionizing radiations and to achieve and maintain a high level of safety and security of radiation sources are established /CHER 14/. More specific and technical detailed requirements are formulated in regulations. These are usually issued by the government or regulatory body. Guidance documents are issued by the regulatory body and proposed solutions to achieve the established requirements. The last level consists of codes and standards. Codes and standards describe specific technical solutions and are usually issued by international or national standardization organisations with a strong participation of the industry. As the first three levels include only abstract provisions, this study focusses on the fourth level to propose specific requirements for the safe siting, design, construction, operation and decommissioning of D-T fusion facilities with magnetic confinement. As the technology of future fusion facilities is not yet finalized and different technical solutions are under discussions, a goal oriented regulatory framework is favoured over a prescriptive framework to be as technology neutral as possible and consequently will be applicable to different design solutions.

5.2 Legal framework

The legal framework is formed by acts, laws, decrees and orders. In Europe, several Council Directives provides the basis for the national laws and decrees. With respect to nuclear and radiation safety, this are mainly the two Directives

- Council Directive 2013/59/Euratom – European Basic Safety standards;
- Council Directive 2009/71/Euratom (amended by Directive 2014/87/Euratom) – Nuclear Safety Directive.

In particular, Council Directive 2009/71/Euratom establishes important aspects which are also applicable to fusion power plants. However, as fusion power plants are not within the scope of this Council Directive, the national legal framework shall establish at least the following important aspects to regulate fusion power plants:

- Establishing a legislative, regulatory and organisational framework (→ Article 4);
- Establishing the legal and regulatory framework to regulate fusion power plants;
- Establishing and maintaining an independent and competent regulatory authority (→ Article 5);
- Clear allocation of a competent regulatory authority for regulating fusion power plants;
- Obligations of the licence holders (→ Article 6);
- Prime responsibility for the safety of fusion power plants need to rest with the licencee;
- Demonstration of safety when applying for a licence;
- Reassessment of safety and continuous improvement;
- Implementation of a management system;
- On-site emergency preparedness and response;
- Provision of financial and human resources;
- Transparency (→ Article 8);
- Informing workers and public about normal operating conditions as well as prompt information in case of incidents and accidents;
- Nuclear safety objective;
- The avoidance of early or large radioactive releases shall be a safety objective for fusion power plants, too;
- Application of the defence in depth concept (→ Article 8b);
- Initial safety assessment and periodic safety reviews (→ Article 8c);
- Conducting a safety review when applying for a licence;
- Periodic reassessment of safety, at least every 10 years;
- On-site emergency preparedness and response (→ Article 8d).

Implementation of the above-mentioned articles would already establish a sound legal basis for regulating fusion power plants by providing the legal basis for

- Defining the competent regulatory authority;

- Establishing a licensing procedure for fusion power plants;
- Requiring initial assessment of safety and regular reassessment of safety (periodic safety reviews);
- A system for operational experience feedback;
- Assigning obligation and responsibilities to the licence holder;
- Defining a high-level safety objective and its implementation as high level requirements in the legal framework;
- Establishing an adequate on-site emergency organisation to cope with incidents and accidents.

As fusion power plants are out of the scope of this Council Directive it is recommended to discuss on a European level how to establish similar requirements for Europe. As there is at present no common approach to regulate fusion facilities, it is worthwhile to enhance discussion on this topic on the European level.

5.3 Regulatory framework

The regulatory framework consists of requirements and guidance documents substantiating and concretizing the more abstract legal framework. The level of detail increases and becomes more technology specific to better address the specific needs of facilities and activities. The regulatory framework should allow for a graded approach for the application considering the hazard potential of a specific fusion facility.

5.3.1 General Safety Requirements

As described in Section 4.4, IAEA provides already general safety requirements which are applicable to fusion power plants, too. In the following, these general safety requirements and its role in a future regulatory framework for fusion facilities are discussed.

5.3.1.1 Siting

Site survey, site evaluation and site selection

The suitability of the site has to be demonstrated by the applicant and needs to be assessed by the regulatory body in order to issue a required site licence (→ interface to the licensing process). Insights from the siting process also contributes to an environmental impact assessment. From a safety point of view, the following three main aspects are important to select a suitable site for a fusion facility:

- Natural and man-made hazards in the vicinity of the site need to be known to allow for a proper design of the facility to withstand impacts of such hazards without unduly challenging the safety of the fusion facility (e.g. earthquakes, flooding, volcanos, airports, nearby industrial facilities, etc.)

  → important for definition of the design basis of SSCs
- Environmental conditions, particularly meteorological, hydrological and topographical conditions, having an impact on the dispersion of released radioactive materials. These factors are important for the determination of the doses in normal operation and in accident conditions in the vicinity of the site.

  → important for radiological consequence analysis
- Population density and population distribution are important factors to estimate the possibility for implementing off-site emergency measures.

  → important for emergency preparedness and response (even if the design will not require off-site countermeasures, the competent authority may precautionary plan off-site countermeasures as part of its responsibility for emergency preparedness and response).

In addition, some security aspects need to be considered during the siting process. However, this is out of the scope of this study.

Although IAEA SSR-1 /IAEA 19/ is developed for nuclear installations excluding fusion facilities, it provides valuable requirements for evaluating and selecting a potential site to host a fusion facility. As the radiological hazard potential of fusion facilities vary, application of a graded approach is recommended. As described in IAEA SSR-1 the siting process can be divided in two steps: site survey and site selection. As described in IAEA SSR-1 /IAEA 19/ the siting process is divided into two stages to find a suitable site for an installation:

a) Site survey;

b) Site selection.

In a first step during the site survey candidate sites will be selected, starting with the investigation of a larger region. Unsuitable sites to host a fusion facility will be already rejected in this step. The second step is the site selection. In this step the candidate sites derived from the site survey will be assessed by screening, evaluation, comparing and ranking to select one or more preferred candidate sites. The result of the second step is the confirmation of the suitability of the preferred candidate sites to host a fusion facility. It is important that the suitability of the site is assessed early in a fusion project because it might have a large impact on the design basis of the future facility.

Environmental Impact Assessment (EIA)

A further important aspect interfacing the siting process is the environmental impact assessment (EIA) to investigate potential negative impacts of a planned facility on the environment.

Directive 2011/92/EU as amended by Directive 2014/52/EU deals with the assessment of the effects of certain public and private projects on the environment. According to Article 4 (1) of this Directive, referring to Annex I, an Environmental Impact Assessment (EIA) has to be carried out for certain projects. This includes nuclear power stations and other nuclear reactors including the dismantling or decommissioning of such power stations or reactors (except research installations for the production and conversion of fissionable and fertile materials, whose maximum power does not exceed 1 kilowatt continuous thermal load).

A fusion facility does not classify as a nuclear power station or nuclear reactor. Therefore, it does not fall under the projects listed in Annex I and II of the directives. A similar conclusion has been made by the EIB (European Investment Bank) /EIB 19/ with respect to the project of design and construction of the Divertor Tokamak Test facility ("DTT") in Italy.

The Espoo (EIA) Convention sets out the obligations of parties to assess the environmental impact of certain activities at an early stage of planning. It also lays down the general obligation of States to notify and consult each other on all major projects under consideration that are likely to have a significant adverse environmental impact across boundaries. The Espoo Convention was adopted in 1991 and entered into force on 10 September 1997. The second amendment of the Espoo Convention went into force on 23 October 2017. Furthermore, the Aarhus Convention addresses amongst others the access to information, public participation and access to justice with regard to in environmental matters. The convention was adopted on 25th June 1998.

Appendix I of the Espoo Convention and Annex I of the Aarhus Convention list the activities which are covered by the Convention. With respect to nuclear activities, it has similar specifications as the Directive 2011/92/EU as amended by Directive 2014/52/EU. It is emphasized that fusion facilities are not covered by these Conventions at present.

5.3.1.2 Leadership and management

Management of safety by establishing an integrated management system is an important requirement for the licence holder. It is expected and required that safety will be given highest priority over other aspects of managing facilities with nuclear or radiological risks. As having a documented integrated management system in place is not sufficient to establish a safety culture within the licensee’s organization, leadership is recognized to be important, too. New developments at IAEA as well as WENRA emphasizes and highlighted in recent publications the importance of leadership /IAEA 16d/ /WENR 20/. Already IAEA Safety Standard No. SF-1 /IAEA 06//IAEA 06/ establishes the need for leadership and management in its principle 3:

"Effective leadership and management for safety must be established and sustained in organizations concerned with, and facilities and activities that give rise to, radiation risks." /IAEA 06/

The review of the specific safety requirements for nuclear power plants, research reactors and fuel cycle facilities shows that all specific requirements addresses this topic. It can be concluded that requirements for leadership and management of safety need to be formulated for licence holders of future fusion power plants, too. The existing requirements can be used to implement such requirements in a regulatory framework dedicated to fusion power plants. As leadership and management is not technology specific, the existing requirements can be easily adapted to fusion power plants. Establishing, fostering and maintaining a high level of safety culture is an important element of leadership, starting already during the design phase and continuous until decommissioning.

5.3.1.3 Safety assessment

Safety assessment is an essential element to ensure nuclear safety by demonstrating that the facility complies with the legal and regulatory framework. IAEA has published a general safety requirement GSR Part 4 on this topic. In addition to GSR Part 4 IAEA published three safety guides providing further guidance to perform deterministic safety analyses (i.e. IAEA Safety Standard No. SSG-2) and probabilistic safety analyses (i.e. IAEA Safety Standard No. SSG-3 and SSG-4). All three guides are written for nuclear power plants. However, the methodology described can be adapted to fusion power plants following the plant states in accordance with the defence in depth concept.

Postulated initiating events

A plant specific list of postulated initiating events (PIEs) needs to be established covering all levels of defence in depth. PIEs shall be grouped and assigned to a specific level of defence-in-depth to control the event and avoid escalation to more serious accident conditions and to mitigate the consequences of the respective event on the assigned level of defence-in-depth. One option to assign an event to a specific level of defence-in-depth is its frequency. Table 10 provides a typical range of frequencies for PIEs on various levels of defence in depth. In addition, the scheme proposed for EU-DEMO is provided for comparison and are in compliance with expectations of IAEA. However, a slightly different terminology for the plant states is applied which should be harmonized in the future. Furthermore, the terminology applied in /WENR 13/ is shown. It is recommended to follow the nomenclature proposed by WENRA:

- Operational states
  - Normal operation
  - Anticipated operational occurrences
- Accident conditions
  - Postulated single initiating events
  - Postulated multiple failure events

The advantage of the terminology proposed by WENRA is that it is made clear, that all plant states shall be covered by the design and relevant impacts shall be included in the design basis of items important to safety to control such plant states.

A preliminary list of representative events is shown in Table 11 which has been established for EU DEMO using a certain breeding blanket design /PINN 17/.

Table 10 Frequency ranges for plant states

	Operational states		Accident conditions		
	Level 1 **Normal operation**	**Level 2** **Anticipated operational occurences**	**Level 3** **Design basis accidents**		**Level 4** **Design extension conditions**
Plant states according to /WENR 13/	Normal operation	Anticipated operational occurrences	Postulated single initiating events		Postulated multiple failure events
Indicative frequency ranges per reactor year proposed by IAEA /COUR 19/	Start-up, power operation, shutdown, maintenance	$f > 10^{-2}$	Infrequent faults $10^{-2} > f > 10^{-4}$	Limiting faults $10^{-4} > f > 10^{-6}$	$f < 10^{-6}$
Accident frequency per reactor year proposed for EU-DEMO /TAYL 19/	Normal operation -	Anticipated events / Incidents $f > 10^{-2}$	Unlikely events $10^{-2} > f > 10^{-4}$	Extremely unlikely events $10^{-4} > f > 10^{-6}$	Hypothetical events $f < 10^{-6}$

Table 11 List of the representative events for EU DEMO /PINN 17/

PIE	Event description
FB1	Loss of flow in the primary cooling loop of the breeder material because compressor or pump seizure
FD1	Loss of flow in the primary cooling loop of the divertor because pump seizure
FF1	Loss of flow in the primary cooling loop of the FW and blanket structures because compressor or pump seizure
FM1	Loss of flow in the liquid metal circuit because of electromagnetic pump seizure: the LiPb flow is lost in all blanket modules supplied by the LiPb circuit
HA99	Loss of heat sink in All FW, blanket structure, BZ and divertor primary cooling circuits because trip of both HP and LP turbines due to loss of condenser vacuum
LFB1	In-vessel LOCA from the cooling circuit of the FW or blanket structure inside the breeder blanket box due to a large rupture of a sealing weld
LBO1	Ex-vessel LOCA from the BZ primary cooling loop due to large rupture in the coolant manifold feeder located inside the PHTS Vault
LBO3	Ex-vessel LOCA from the breeder primary cooling loop due to large rupture of tubes in a primary HX (or SG)
LFV1	In-vessel LOCA due to a large rupture of the FW structure: complete rupture of the FW
LDO1	Ex-vessel LOCA from the Divertor primary cooling loop due to large rupture in the coolant manifold feeder located inside the PHTS Vault
LDV1	In-vessel LOCA from the Divertor primary cooling loop due to large rupture in the divertor cassette
LMO1	Ex-vessel loss of liquid metal from the LiPb circuit due to large rupture in the cold leg, downstream the electromagnetic pump (loss of LiPb feeding to all blanket modules supplied by the faulted LiPb line)
LMO2	Ex-vessel loss of liquid metal from the LiPb circuit due to a leak in the cold leg, downstream the electromagnetic pump
LMO3	Ex-vessel rupture of liquid metal circuit inside the heat exchanger (or steam generator), i.e. large rupture of tubes in the LiPb HX
TGG1	Break of tritium gas process line within secondary enclosure (e.g. glove box): cryogenic fluid and fuel gas released into pellet injector guard vacuum volume
TGO1	Ex-vessel release of tritium gas due to guillotine rupture in the process line of the Isotopic Separation System (ISS, tritium release inside the building)
TGO3	Release of tritiated effluents to environment due to misoperation in the tritium process systems, e.g. failure to operate the cryo-distillation columns in ISS
THO1	Guillotine break of the hydrogen gas pipe at the outlet of the electrolyzer. Direct tritium release into the WDS room. Risk of Q2 explosion
TWO1	Rupture of a high activity level holding tank in WDS
VCG1	Loss of cryostat vacuum due to large ingress of gas (He and/or air)

PIE	Event description
VVA1	Loss of vacuum (LOVA) in VV due to large ingress of air induced by rupture in a VV penetration

Acceptance targets and acceptance criteria

To demonstrate compliance with safety requirements acceptance targets and quantitative and/or qualitative acceptance criteria need to be defined against which the safety analyses can be assessed. For example, occupational dose limits and targets have been specified for EU-DEMO, see Table 12, as well as limits of the consequences of postulated accident scenarios in different categories of event frequency. However, it is emphasized that the on-site dose limit of 50 mSv/y exceeds the dose limits provided in EU Directive 2013/59/EURATOM and needs to be adapted to current dose limits. According to Art. 9 (2), the limit on the effective dose for occupational exposure shall be 20 mSv in any single year. However, in special circumstances or for certain exposure situations specified in national legislation, a higher effective dose of up to 50 mSv may be authorised by the competent authority in a single year, provided that the average annual dose over any five consecutive years, including the years for which the limit has been exceeded, does not exceed 20 mSv. It is emphasized, that radiological acceptance criteria have to comply with existing limits (such as dose limits, dose rate limits, activity limits) in the respective national framework. It is known from nuclear power plants that radiological acceptance criteria differ amongst the EU Member States. Moreover, some Member States provide quantitative acceptance criteria whereas other Member States only provides qualitative acceptance criteria. The development of a harmonized framework for fusion facilities would facilitate the definition of common radiological acceptance criteria in the national frameworks for all plant states to be considered in the safety analysis.

Table 12 Dose limits provisionally adopted for EU DEMO /TAYL 19/.

	Normal operation		Antici-pated events / Incidents	Unlikely events	Extremely unlikely events	Hypotheti-cal events
	Limit	Target				
Accident frequency /y	-	-	$f > 10^{-2}$	$10^{-2} > f > 10^{-4}$	$10^{-4} > f > 10^{-6}$	$f < 10^{-6}$
On-site Dose	50 mSv/y 100 mSv/5y	5 mSv/y	5 mSv/y	20 mSv/event	-	-
Off-site Early Dose	1 mSv/y	0.1 mSv/y	-	-	10 mSv/event	50 mSv/event
Off-site Chronic Dose	-	-	1 mSv/y	5 mSv/event	50 mSv/event	No cliff-edge effects. Limited countermeasures

Quantitative limits and targets of Occupational Radiation Exposure (ORE) for EU DEMO are set at current design stage /PORF 20/:

- Personnel maximum individual dose at 20 mSv/y and 100 mSv/5y;
- Maximum collective dose for the facility of 700 person.mSv/y (sum of all personnel individual doses in the power plant);
- Maximum dose rates in different zones of the facility according to a radiological zoning scheme as it used e.g. in France: 25 µSv/h in green zones, 2 mSv/h in yellow zones, 100 mSv/h in orange zones, above 100 mSv/h in red zone;
- An equivalent maximum dose target for releases to the environment is 50 µSv/y in normal operation.

In addition, an early dose (7-day uptake) of 50 mSv/event is assumed at the site boundary for the nearest resident population to meet the requirement for no evacuation.

ORE assessments are in progress for DEMO that an appropriate collective dose for DEMO and its allocation across the systems can be established.

5.3.1.4 Decommissioning of facilities

Even though fusion power plants are not yet available, first larger fusion facilities are under construction or in planning. On the first look, decommissioning may currently not be an issue, but an issue to be dealt with in decades from now. Experiences from nuclear power plants showed, that it is important to consider the later decommissioning and dismantling of nuclear facilities already during the design phase. IAEA Safety Standard No. GSR Part 6 /IAEA 14c/ highlights this aspect in para. 1.6:

"Planning for decommissioning begins at the design stage and continues throughout the lifetime of the facility." /IAEA 14c/

And more specifically in para. 7.3:

"For a new facility, planning for decommissioning shall begin early in the design stage and shall continue through to termination of the authorization for decommissioning." /IAEA 14c/

More details concerning decommissioning to be considered during the design phase can be found in Requirement 12 of IAEA Safety Standard SSR 2/1, Requirements 15 and 33 of IAEA Safety Standard No. SSR-3 and Requirement 33 of IAEA Safety Standard No. SSR-4. The following shall be considered during the design phase of a fusion facility:

- Choice of materials which will contribute to the minimization of radioactive waste and facilitates decontamination;

- Choice of materials resistant to chemical attacks and that have sufficient wear resistance to facilitate decontamination at the end of their lifetime;

- Selection of design options which will keep exposure of workers and the public arising from decommissioning as low as reasonably achievable;

- Avoidance of undesired accumulation of chemicals or radioactive materials;

- Minimization of number and size of contaminated areas to facilitate cleanup in the decommissioning stage;

- Provision of access and space for handling equipment during decommissioning, including later remote decontamination;

- Provision of facilities for the management of radioactive waste generated in the decommissioning of the facility.

A further important aspect is to manage relevant knowledge of the design as well as experiences during construction, commissioning and operation which will have an influence on the later decommissioning of the facility (contamination due to incidents, modification of the facility, etc.).

5.3.2 Deriving a Safety concept for fusion power plants

In this section a potential safety concept for fusion facilities is described. This has been developed from high level safety objectives considering the specific needs for D-T fusion facilities with magnetic confinement.

5.3.2.1 Safety objectives

The fundamental safety objective is applicable to all facilities and activities that give rise to radiation risks as expressed in /IAEA 06/. Thus, the fundamental safety objective has to be met also by fusion facilities and forms the sound basis of the safety concept.

"The fundamental safety objective is to protect people and the environment from harmful effects of ionizing radiation". /IAEA 06/

It is further stated that facilities and conducted activities must ensure the highest standards of safety that can be reasonably achieved. To ensure this fundamental safety objective, the following measures have to be taken according to /IAEA 06//IAEA 06/:

a) Control the radiation exposure of people and the release of radioactive material to the environment;

b) Restrict the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation;

c) Mitigate the consequences of such events if they were to occur.

From item a) it can be derived that a radiological safety objective for fusion facilities should be developed whereas from items b) and c) a technical safety objective for fusion facilities should be formulated. Both, the radiological and technical safety objective will substantiate the fundamental safety objective.

It is further important, that the fundamental safety objective has to be ensured for all stages over the lifetime of a facility, including planning, siting, design, manufacturing, construction, commissioning and operation, as well as decommissioning.

The radiological safety objective needs to include the three general principles of radiation protection as defined in Article 5 of Council Directive 2013/59/EURATOM /EU 13/. These three general principles are:

- Justification;
- Optimisation;
- Dose limitation.

Here, justification is a process to balance the benefits of a facility or activity utilizing ionising radiation with the detrimental negative effects on workers, public and the environment. This process will result in a list of acceptable or inacceptable use of ionizing radiation published by the government. The justification should be based on a scientific and technical rational. Whereas justification is usually a task for the government, optimisation and dose limitation is the responsibility of the licensee, holding or applying for a licence. One prerequisite for issuing a licence is the demonstration that a fusion facility can be constructed, operated and decommissioned in such a way that all legal dose limits will not be exceeded. It is important, that this will cover all plant states, operational states as well as accident conditions. Due to the nature of ionizing radiation and its biological effects on living tissues, the risk for mutations and cancer increases with increasing doses (stochastic radiation damage). Thus, to minimize the radiation risk it is important to apply the ALARA ("as low as reasonably achievable") principle to reduce the doses well below legal dose limits by balancing the achievable dose reduction with the necessary effort.

The technical safety objective will contribute to the fulfilment of the radiological safety objective and consequently to achieve the fundamental safety objective. Although the definition of nuclear installations in Council Directive 2009/71/Euratom amended by Council Directive 2014/87/Euratom /EU 09/, /EU 14/ excludes fusion facilities, the nuclear safety objective established in Article 8a is strongly recommended to be implemented in the safety concept for future fusion facilities:

> "*Article 8a*
>
> ***Nuclear safety objective for nuclear installations***
>
> *1. Member States shall ensure that the national nuclear safety framework requires that nuclear installations are designed, sited, constructed, commissioned, operated and decommissioned with the objective of preventing accidents and, should an accident occur, mitigating its consequences and avoiding:*
>
> *(a) early radioactive releases that would require off-site emergency measures but with insufficient time to implement them;*
>
> *(b) large radioactive releases that would require protective measures that could not be limited in area or time."*

The nuclear safety objective expresses two main aspects of nuclear safety, which are also expected to be fulfilled by fusion facilities:

- to prevent accidents and to mitigate its consequences once an accident occurs;
- to avoid early or large releases should an accident occur.

The first bullet point is fully in line with principle 8 of the IAEA Safety Fundamentals /IAEA 06/. This is more specifically substantiated for example in IAEA SSR-2/1 para 2.10 /IAEA 16/ and IAEA SSR-4 para 2.9 /IAEA 17/ where it is required, that the likelihood of an accident shall be minimized.

The second bullet point directly corresponds with the Vienna Declaration on Nuclear Safety. Although the Vienna Declaration on Nuclear Safety as well as Council Directive 2009/71/Euratom amended by Council Directive 2014/87/Euratom only address fission facilities and not fusion facilities, they define important safety objectives which should also be met by radiation facilities, in particular if the radiological hazard potential is much lower compared with a nuclear power plant based on fission. Acceptable radiological consequences for less hazardous radiation facilities, i.e. fusion facilities, shall not exceed those radiological acceptance targets for nuclear facilities with higher risk potential, such as conventional nuclear power plants. In addition, it would be difficult to communicate to the public that radiological consequences of a less hazardous fusion facility might by higher than for a fission power plant. For that reason, accident sequences leading to radiological consequences requiring measures of emergency response which are not limited in area and time have to be practically eliminated.

In addition, the technical safety objective has to address the protection of present and future generations as required by principle 7 of the IAEA Safety Fundamentals /IAEA 06/:

"People and the environment, present and future, must be protected against radiation risks."

In IAEA SF-1 para. 3.29 /IAEA 06//IAEA 06/ it is further required, that radioactive waste must be managed in such a way as to avoid imposing an undue burden on future generations. For that reason, the generation of radioactive waste shall be minimized as far as practicable by means of appropriate design measures and procedures, such as the recycling and reuse of material. Already during the design phase materials shall be selected which cannot be easily activated and thus reducing the amount of radioactive waste.

In /SEAF 95/ one of the two high level safety objectives for fusion facilities directly addressed the waste issue, requiring that no geological disposal shall be necessary for waste generated at fusion facilities:

"Radioactive wastes from the operation of a fusion plant should not require isolation from the environment for a geological timespan and therefore should not constitute a burden for future generations."

The fundamental safety functions for a fusion facility must be derived based on a functional approach to achieve the radiological and nuclear safety objective. To meet the radiological safety objective, ionizing radiation must be shielded, and radioactive materials must be confined within the facility to minimize exposures to workers, the public and the environment. Thus, the first two fundamental safety functions can be formulated as:

"Confinement of radioactive material and control of planned radioactive releases, as well as limitation of accidental radioactive releases."

"Shielding against radiation in normal operation and accident conditions."

The wording of the first two fundamental safety functions for fusion facilities is directly adopted from IAEA SSR 2/1 Rev. 1 Requirement 4 /IAEA 16/.

Challenges to the shielding or confinement functions may results in a violation of the first two fundamental safety functions and consequently of the radiological and fundamental safety objective. Therefore, it will be necessary to define further fundamental safety functions which prevents either a failure of the shielding function or challenges the integrity of a barrier for the confinement of radioactive materials. As the technology of a fusion facility, as described in Task 2, differs completely from a fission facility, the remaining two fundamental safety functions for fission reactors of SSR-2/1 Rev.1 are not applicable and fusion specific fundamental safety functions need to be formulated.

Whereas criticality is not a concern for fusion facilities, heat removal may be necessary depending on the design to remove decay heat from selected SSCs, in particular from the first wall and breeding blanket system. In contrast to fission facilities, this could be a safety function rather a fundamental safety function. To find fusion specific fundamental safety functions the radiation sources and those mechanisms, challenging either the shielding or the confinement function, need to be identified. Table 13 provides an overview of important radiation sources in a fusion facility relevant for the development of a fusion facility safety concept.

Table 13 Typical radiation sources within a fusion facility.

Radiation source	Primary origin (System, Structure, Component)		Safety function
Tritium	Vacuum vessel		Confinement
	Breeding blanket		
	Tritium fuel cycle system		
	Remote Handling Equipment		
	Storage of in-vessel components		
	Hot cells		
	In coolants (inparticular PbLi)		
	Atmosphere of rooms containing tritium systems		
	dissolved in water (e.g. in SG)		
Neutron radiation	Plasma Chamber (Vacuum vessel, Plasma)		Shielding
Gamma radiation	Vacuum vessel, first wall, breeding blanket		Shielding
Bremsstrahlung	Plasma Chamber (Vacuum vessel, Plasma)		Shielding
X ray	Plasma Chamber (Vacuum vessel, Plasma)		Shielding
Activation products	First Wall	Sputtering	Confinement
		Activation	
	Breeding blanket	Activation	
	Coolant	Activation	
		Corrosion products	
	Divertor	Corrosion products	
	In-vessel dust		
	Vacuum vessel and ex-vessel components		
	Activated shielding material		

The radiation sources exist in various material conditions. Tritium exists either as a gas or HTO steam in the vacuum vessel, the breeding blanket and tubes of supporting auxiliary systems. Interactions between plasma or high energetic particles with the first wall can cause sputtering and produce activated dust inside the vacuum vessel. Radioactivity in the coolant may be due to either activation of the coolant itself (e.g. built up of ^{16}N due to the nuclear reaction $^{16}O(n,p)^{16}N$, due to the cross section this reaction might be more important in case of fusion reactors than for light water reactors due to the harder energy spectrum) or due to activated corrosion products in the coolant loops.

The confinement function may be challenged by events impacting the integrity of the barriers. These impacts may lead to mechanical or thermal loads on the barriers.

There are planned and unplanned plasma transients. Planned transients are represented by the ramp up and down for the pulsed operation of a Tokamak and are not necessary for a stellarator design for example. Unplanned plasma transients are caused by some plasma instabilities (ELMs, NTMs, Kink etc.) and can lead to plasma disruptions which in turn can damage PFCs. So, the D&C system has to shut down the plasma in a controlled manner.

The first phenomenon which can have a strong thermal impact on the IVCs (FW, divertor target) and the vacuum vessel are plasma disruptions. To minimize such events the plasma has to be controlled, which is proposed as a supporting safety function:

Control of plasma stability

This means, that appropriate measures shall be provided to control plasma parameters like position, shape or stability to achieve stable plasma and avoid plasma disruptions. This relates to level 1 of defence in depth. If a disruption is unavoidable, a subsequent safety function has to ensure the radionuclide confinement (defence in depth).

In particular for future fusion power plants heat generated by the plasma has to be removed from the FW, divertor, BZ, limiters and further internal structures of the vacuum vessel. Depending on the nuclear heating in the structures it may be necessary to remove the nuclear heat (i.e. heat generated by decay of activated materials as well as heat up due to neutron scattering), either by active or passive means, from the inner structures of the vacuum vessel. This aims at keeping the temperature in the structures within the intervals for the normal operation for tritium retention (in NMM for example), TBR etc. Furthermore, it ensures structural integrity in order to avoid leakages (e.g. in relation to in vessel LOCA as result from LOFA/ex vessel LOCA etc.). Thus, a further supporting safety function can be formulated:

Removal of process and residual heat

In fusion facilities, utilizing strong superconducting magnets, quenching of one or more of these magnets might cause mechanical impacts on the barriers. A quench occurs if the temperature of a superconducting magnet exceeds the critical temperature. Due to the sudden resistance of the conductor, the stored energy will quickly heat up the solenoid. As quenching occurs in case of a misbalance of heat generation and heat removal, to ensure cooling of the magnets can be considered as an essential safety function. Thus, a supporting safety function can be formulated to ensure cooling of the magnets below the critical temperature to avoid a quench:

Cooling of superconducting magnets below critical temperature.

If this safety function fails, the barriers have to withstand the load of such an event (defence in depth).

The defined fundamental safety functions and supporting safety functions have been derived in view of the potential further development of fusion facilities to large scale power plants with an expected higher fusion power (e.g. DEMO facility with 2,000 MW) and longer operation times. For smaller fusion facilities some of these safety functions might be directly ensured by design, e.g. the plasma energies are such that plasma disruptions do not challenge the first wall or residual heat is limited so that structural integrity is not challenged. Nevertheless, the fulfilment of the fundamental safety functions needs to be demonstrated.

For EU DEMO, and for any fusion power plant, similar safety objectives, plant level safety requirements, safety functions have been defined at the top level /TAYL 19/. The safety objectives have been set consistently with international guidelines /IAEA 06/, in particular principle 5 of /IAEA 06//IAEA 06/:

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

- To minimize radioactive waste hazards and volumes and ensure that they are as low as reasonably achievable.

5.3.2.2 Defence in Depth Concept

The defence in depth concept is an essential safety concept in nuclear safety and is required in principle 8 of the IAEA Safety Fundamentals /IAEA 06/:

> "*All practical efforts must be made to prevent and mitigate nuclear or radiation accidents.*" /IAEA 06/

In addition, principle 9 of the IAEA Safety Fundamentals /IAEA 06/ has to be considered dealing with emergency preparedness and response:

> "*Arrangements must be made for emergency preparedness and response for nuclear or radiation incidents*." /IAEA 06/

The primary means of preventing and mitigating the consequences of accidents is the consequent implementation of the defence in depth concept. The main objective of defence-in-depth is the risk reduction of accidents with unacceptable radiological consequences by consecutive and independent levels of protection. Whereas for fission power plants the defence-in-depth concept is well elaborated /IAEA 16/, /IAEA 16b/, /INSA 96/,/WENR 13/ this concept needs to be adapted to the specific needs of fusion facilities. Application of the defence-in-depth concept is not only required for nuclear power plants, but it is also a requirement in the specific IAEA safety requirements for research reactors /IAEA 16c/ and fuel cycle facilities /IAEA 17/. Thus, it can be concluded that the defence-in-depth concept shall be also an essential element of the safety concept for fusion facilities.

Central elements, like consecutive levels linked to certain plant states will remain and are also applicable to fusion facilities. Table 14 shows the proposed levels of defence in depth for a fusion facility and the four associated plant states. In addition, the objective of each level of defence-in-depth as well as the acceptable radiological consequences are shown. In the following, the five levels of defence-in-depth and the associated plant states are discussed in detail.

Table 14 Safety levels of defence-in-depth for a fusion facility.

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

Level 1

Safety level 1 aims for maintaining the fusion facility in normal operation. A stable and reliable operation of the facility is ensured by a conservative design and high quality in construction and operation. It is emphasized that strengthening defence-in-depth aims also for ensuring a reliable normal operation. This primarily enhances safety, but it is also beneficial for the economic efficiency of a future deployed commercial fusion facility. Besides a conservative and reliable design, control systems ensure keeping the main plant parameters within defined limits. This includes controls which are able to return deviations from set point to zero e.g. microwave injection for destroying magnetic islands in a Tokamak. From a radiological point of view off-site radiological impacts are not acceptable. This means discharges must be kept below regulatory limits established in the regulatory framework or defined in the licence conditions. Usually, such operational limits for discharge ensure that the typical dose limits for public exposure will not exceeded taking contributions from several nuclear and radiation facilities in the vicinity into account. Dose limits for public exposure are established in Article 12 of Council Directive 2013/59/Euratom /EU 13/:

- Effective dose: 1 mSv/a;
- Equivalent dose for the lens of the eye: 15 mSv/a;
- Equivalent dose for the skin averaged over any 1 cm^2: 50 mSv/a.

In addition, dose limits for occupational exposure as established in Article 9 of Council Directive 2013/59/Euratom /EU 13/ shall not be exceeded to protect workers:

- Effective dose: 20 mSv/a;
- Equivalent dose for the lens of the eye: 20 mSv/a or 100 mSv in five consecutive years with maximum of 50 mSv in a single year;
- Equivalent dose for the skin averaged over any 1 cm2: 500 mSv/a;
- Equivalent dose for the extremities: 500 mSv/a.

The dose limits define the maximum acceptable doses. For a specific facility the actual exposures need to be optimized by utilizing the ALARA principle, dose constraints and regulatory limits for operational releases.

Examples for operation phases are start-up, normal operation, shutdown or maintenance phase due to blanket / divertor replacement. Accordingly, level 1 implies the planned plasma transients especially occurring in a Tokamak design.

In order to achieve the aim of reducing the radiological impact during normal operation, systems such as the TER have to be designed so that the mentioned reliability is met.

Level 2

On safety level 2 it is postulated that the normal operation is disturbed. The corresponding plant state is called anticipated operational occurrences and belongs like normal operation to the operational states. Anticipated operational occurrences are based on postulated initiating events expected to occur during the lifetime of the fusion facility such as unplanned plasma transients caused by instabilities (ELMs, NTMs, Kink modes, Mode locking, VDEs, H-L-transition, divertor reattachment). A further transient can be caused by the PSD system in order to avoid an extent in a plasma instability. The objective of defence in depth level 2 is to control anticipated operational occurrences and to bring back the plant into normal operation or in a stable shutdown mode. By controlling anticipated operational occurrences an escalation to accident conditions will be prevented so that components are protected, too. As both, anticipated operational occurrences and normal operation, belong to the operational states, the same radiological consequences are acceptable and the same dose limits as for level 1 shall not be exceeded. Again, the actual exposures need to be optimized by utilizing the ALARA principle, dose constraints and regulatory limits for operational releases. The demand to the systems for radiological retention may be extended during a transient because of temperature rise in the blanket for example which can lead to a higher release from structures. The design has to cope with the corresponding operational parameters being more demanding for the components of barriers.

Level 3

Safety level 3 is assigned to postulated single initiating events (also called design basis accidents / DBA). These are accident conditions with a single initiating failure. The occurrence of such accidents is postulated, but not expected to occur during the lifetime of the fusion facility. These accident sequences can be directly initiated by a postulated failure or could be the result of a failure to control operational states on the previous levels of defence in depth. Such accident sequences are controlled by dedicated safety systems. These safety systems will control the accident sequences and reliably prevent an escalation to more serious accident conditions. Postulated single initiating events can result in releases exceeding the regulatory operational limits for discharges and thus can lead to a more serious radiological impact. However, the resulting doses to the public shall not exceed doses which would require off-site countermeasures.

Examples for postulated initiating events are LOFA (Loss of Flow Accident), in-vessel LOCA, ex-vessel LOCA (Loss of Coolant Accident), loss of heat sink or a LOVA (Loss of Vacuum Accident), which are followed by a fast plasma shutdown initiated by the PSD system.

DBAs can lead to very demanding conditions to the components and barriers. For example, the plasma shutdown can be accompanied by a disruption which produces dust additional to the dust already produced during normal operation. This dust can be mobilized through coolant ingress into the vacuum vessel in a Tokamak in case of a LOCA. Furthermore, the mechanical load on the structures can be very large due to the collapse of the magnetic field. Also, the temperatures and the release of tritium from structures are increased. The design of the plant has to refer to these DBA a priority.

Level 4

The objective of safety level 4 of defence in depth is the control of accident conditions more serious than postulated single initiating events. This group of accidents includes postulated multiple failure events (also called Design Extension Conditions / DEC). Such events might be:

- Combination (consequential, causally linked or random combinations) of events;
- Failure of safety systems to control postulated single initiating events;

- Multiple failure events caused by internal and external hazards (e.g. internal flooding, explosions, induced vibrations caused by aircraft crash /IAEA 03/).

The same radiological consequences as for defence-in-depth level 3 shall be applied.

Controlling the accident means to mitigate the accident progression of such events /IAEA 16/. Due to the fact that such events are beyond design, appropriate procedures have to be developed.

Examples for postulated multiple failure events are air ingress into the vacuum vessel together with a failure from the blanket primary cooling system, or Loss of Flow (LOFA) accident without plasma shutdown inducing an in-Vacuum Vessel loss of coolant (in-VV LOCA).

Level 5

Level 5 describes off-site protective actions for emergency exposure situations and is based on principle 9 of the IAEA Safety Fundamentals /IAEA 06/. In contrast to levels 1 to 4 which mainly deal with the responsibility of the licensee to be addressed by the design of the facility, level 5 describes the measures of emergency preparedness and response. This is typically a task for the competent authority responsible for civil protection. The role of the licensee on level 5 is restricted in providing support to off-site response forces.

In general, all levels of defence in depth should be in place and effective. However, as fusion facilities may represent different radiological hazard potentials, the number of levels of defence-in-depth may be graded for lower risk facilities.

In order to mitigate radiological consequences, corresponding devices and appropriate procedures has to be set in place which can be a filter for unavoidable releases.

5.3.2.3 Concept of multi-level confinement of radioactive inventory

Radioactive materials present within the facility must be safely confined to avoid inacceptable exposures of workers and the public. The concept of multi-level confinement by barriers and retention functions directly contributes to fulfil the fundamental safety function confinement of radioactive materials. The concept of multi-level confinement is closely linked to the implementation of the defence-in-depth concept. Within this concept it is clearly distinguished between a barrier and a retention function. A barrier represents a gas-tight barrier, typically a metallic barrier (e.g. the metallic wall of the plasma chamber), whereas a retention function describes technical means to reduce releases of radioactive materials (e.g. filtering of radioactive materials in the exhaust air of the ventilation system).

As a failure of a barrier has to be postulated, at least as a design basis accident, a second concentric barrier to confine released radioactive materials is strongly recommended and should be required. Taylor and Cortes /TAYL 14/ already mentioned, that for ITER two confinement systems comprising one or two physical or functional barriers should be available.

Also for EU DEMO two confinement systems are proposed. The first confinement system prevents releases of radioactive and hazardous materials during normal operation into the accessible working areas in order to protect personnel. The second confinement system prevents environmental releases of the contamination to the working areas accessible by non-classified radiological workers, the general public and the environment during the failure of the first confinement system. Three sequential barriers are proposed to confine hazards and tritium release. The first confinement system includes two barriers, and the second confinement system includes the third barrier (building, etc.)

For fusion facility barriers that could be considered are for example the vacuum vessel wall and pipes containing tritium, the cryostat surrounds the toroidal and poloidal magnets, or the surrounding buildings.

For pipes carrying tritium outside the cryostat, the second barrier maybe formed by guard pipes. For other inventories outside the tokamak building equivalent barrier concepts are necessary.

For specific modes of operation, it might be necessary to intentionally open a barrier (e.g. replacing the first wall in the plasma chamber). In such cases it might be necessary to compensate the temporarily unavailability of a barrier by an adequate retention function.

As buildings are usually not gas-tight by itself, the barrier function needs to be supported by a filtered HVAC-system to ensure the confinement function of the outer barrier in operational states and accident conditions.

The concept of multi-level confinement shall be applied to all kind of radioactive materials present during operation (including in the tritium building). It shall also be applied to storage of radioactive waste. At least two concentric barriers shall ensure the safe confinement of radioactive waste. The projected design life of the storage facilities shall be commensurate with the half-live of the expected radioactive waste. It is expected that a storage time of radioactive waste generated in fusion facilities in the order of 100 years may be required.

5.3.2.4 Protection against internal and external hazards

Safety of future fusion facilities can be challenged by internal events, but also by internal and external hazards. In contrast to internal events, hazards have the potential to affect the plant (or even the site) as whole, in particular in case of external hazards. A seismic event or a human induced hazard, like an explosion blast wave or an aircraft crash, may challenge several and redundant safety functions. This may lead to a release of radioactive material. In such cases, where an impact of internal or external hazards may lead to unacceptable radiological consequences, the fusion facility must be protected against those hazards.

Internal and external, radiological and non-radiological hazards

Based on the internal hazards of the NPP in /IAEA 17b/, from ITER and proposed for DEMO in /JIN 17/ the **internal hazards** in fusion facilities are identified as:

- internal fire / explosion;
- thermal releases;
- hydrogen explosions (e.g. from released D^0 neutrals from NBI system in case of LOVA);
- dust explosions;
- plasma transients (disruption);
- pipe breaks (pipe whip and jet effect);
- internal flooding;
- missile effect;
- collapse of structures, falling objects and heavy load drop;
- chemical and toxic risk (e.g. dust incorporation);
- magnetic and electromagnetic risks;
- release of hazardous substances inside the plant.

Following /IAEA 18b/ all potential external hazards (natural and human induced hazards) that may affect fusion power plant site need to be identified. External hazards for DEMO have been identified from the natural environment or human activities /JIN 17/:

- Natural hazards: earthquakes, extreme meteorological conditions, notably severe temperatures, snow load, wind and lightning, external flooding, biological hazards and external fire.
- Human induced hazards: aircraft crashes, hazards associated with the industrial environment, transportation routes (seaways, tracks, roads), such as external explosions and unauthorized access (which is more a security aspect).

Accident risk resulted from an internal or external hazard needs to be assessed. Primary hazards by occurrence of creating conditions can induce a second hazard soon after, as induced hazards. Accident investigation due to external hazards is going to be started in the coming Horizon Europe (FP9) for EU DEMO. Assessment for the internal and external hazards for DEMO will be documented in the Preliminary Safety Report for DEMO, which is anticipated for the final selected blanket concept only. The report will be started after the on-going Generic Site Safety Report (GSSR).

5.3.2.5 Graded approach

Several projects on fusion facilities are world-wide ongoing, covering a broad bandwidth of different design options and plant sizes. Thus, future D-T fusion facilities with magnetic confinement might encompass facilities with quite different radiological hazard potential. The range might be from small experimental facilities up to demonstration facilities for energy production. Consequently, the radiological hazard potential of future fusion facilities may vary. To avoid applying too strict safety requirements not commensurate with the specific radiation hazard potential a graded approach can be applied. The graded approach is already introduced in the IAEA Safety Fundamentals /IAEA 06/ and should be an essential element of a fusion specific regulatory framework. This is clearly expressed in paras. 3.15 and 3.22 of IAEA SF-1 /IAEA 06/:

"3.15. Safety has to be assessed for all facilities and activities, consistent with a graded approach." /IAEA 06/

"3.22. To determine whether radiation risks are as low as reasonably achievable, all such risks, whether arising from normal operations or from abnormal or accident conditions, must be assessed (using a graded approach) a priori and periodically reassessed throughout the lifetime of facilities and activities." /IAEA 06/

More focussed on the regulatory framework, the application of a graded approach is required in para. 4.62 of IAEA GSR Part 1 /IAEA 16g/:

"4.62. The regulations and guides shall provide the framework for the regulatory requirements and conditions to be incorporated into individual authorizations or applications for authorization. They shall also establish the criteria to be used for assessing compliance. The regulations and guides shall be kept consistent and comprehensive, and shall provide adequate coverage commensurate with the radiation risks associated with the facilities and activities, in accordance with a graded approach." /IAEA 16g/

A first guidance on the application of the graded approach was published 2011 by IAEA on the use of a graded approach in the application of the safety requirements for research reactors /IAEA 11/. IAEA is still working on this topic and addresses the use of the graded approach on regulating the safety of radiation sources and on regulating nuclear facilities.

The key element of the graded approach is the categorization due to the intrinsic hazard potential of a certain facility, mainly the radiological hazard potential. This consideration shall not be limited to the tritium and radioactive material inside the vacuum vessel, but shall cover all radioactive material present on the site, including the tritium fuel cycles system and the waste management and storage facilities. To categorize a fusion facility, the radiological consequences need to be determined based on a postulated worst-case scenario without crediting engineered safety features, e.g. as an unmitigated risk approach. A guidance on how to apply a graded approach in the context of national regulatory framework could be beneficial for developers of fusion facilities.

Three risk categories based on the potential radiological impact are proposed (see Table 15).

Table 15 Risk categories for fusion facilities.

Risk Category	Radiological Consequences
I	Off-site radiological impact exceeding intervention levels for protective actions in emergency exposure situations
II	On-Site radiological impact, no off-site measure of emergency response necessary to protect the public
III	No countermeasures to protect workers outside radiation controlled aeras necessary

5.3.2.6 System for operating experience feedback

One experience from nuclear power plants was the need to exchange operating experience amongst operators, designers, regulators and technical safety experts to avoid similar incidents or accidents, to continuously improve safety and to early detect potential pre-cursor events which may lead to more serious accident sequences. Such a system needs to cover all levels, plant-level, national-level and international level to foster a mutual and transparent exchange of operating experience by open and honest communication. As less operating experience is available from fusion facilities compared with nuclear power plants and the deployment of fusion power plants is expected in the future, it would be especially important during the early phase of deployment of new technologies to collect, analyse and share operating experiences. IAEA Safety Standard No. SSG-50 provides guidance for operating organizations of nuclear power plants but the main aspects can be adapted to fusion facilities as well. To foster operational experiences it is further recommended to establish an international reporting system similar to the well established reporting systems for nuclear power plants (IRS, jointly operated by IAEA and OECD/NEA), research reactors (IRS-RR, operated by IAEA) or fuel cycle facilities (FINAS, jointly operated by IAEA and OECD/NEA) to make operating experiences with fusion facilities accessible to all designers, operators, and regulators world-wide and thus contributing to the process of continuous improvement of safety.

5.3.3 Fusion specific safety aspects

Currently specific safety requirements for fusion specific SSCs are missing and need to be developed in the future. These should be based on fusion specific safety aspects. In the following, first the energy sources and radioactive source terms of fusion facilities are identified, then specific safety aspects to be considered in deriving fusion specific requirements are elaborated.

In fusion facilities energy sources are identified as:

- **Plasma energy**: plasma disruption is a main issue here. Events such as a loss of coolant, a sudden inlet of impurities, a significant error in the measurement of some fundamental diagnostic can lead to a fast growth of destabilizing modes in the plasma and then to a disruption. During the plasma disruption event, huge EM forces can be induced on the surrounding structure, large thermal loads apply on IVCs, and sometimes a large amount of runaway electrons impacting on the first wall. Furthermore, arcing can occur.
- **Magnetic energy**: a huge amount of magnetic energy is accumulated in the superconducting coils (e.g. ~135 GJ in DEMO TF coils, /JIN 17/), which has to be evacuated outside the coils and the tokamak building in case of malfunctions or coil failures.
- **Enthalpy** in structure and coolant: Loss of Coolant Accident (LOCA) or Loss of Flow Accident (LOFA) can heat up the structure till high temperature level that the structure integrity are challenged.
- **Decay heat** after the plasma shutdown: If the power density is sufficiently high, energy released by the decay of activated materials can cause temperature excursions in the aftermath of LOCAs or LOFAs.
- **Chemical energy** from exothermal chemical reactions between materials in accident cases (e.g. Li-steam, W-air/steam etc.)
- **Energy release** due to postulated H_2 explosion (H_2 product by possible W-steam reaction if water cooling is used). Hydrogen is also a potential hazard in the fuel cycle / storages holding high quantities of deuterium / tritium.
- **Mechanical energy** stored in fly wheels for pumps and generator
- **External energy** from grid which might be converted and injected by faulty active plasma heating devices for example NBI or microwave injection from EC and IC systems.

The decay heat depends on specific irradiation scenario and is calculated from knowledge of activation inventories. Differences in materials and construction of the BB concepts lead to differing nuclear responses. Activation and decay heat analyses have been performed for the WCLL and HCPB blanket concepts with the previous reference designs in /EADE 17/. And it is updated for the current HCPB reference design in /PARK 21/ that the nuclear power generated in the blankets drops from 1,930 MW at normal operation to 22.25 MW at 1 s after the shutdown. The total decay heat in the blankets drops down to 0.99 MW after 1 week cooling time. Thus, the decay heat at 1 s after the shutdown is ~1.153% of the nominal power and one week later 0.0513% only.

The radioactive source terms are tritium, dust, and Activation Corrosion Products (ACPs). They are located mainly inside the VV (T, dust), in the breeding blanket (T, ACPs in water-cooled blanket concept), in the cooling loops (T, ACPs), and in the tritium building (T) /PORF 20/.

Tritium is built and located in the blanket, divertor, different PHTSs, TER, Tritium plant, etc. Tritium inventory is a safety and waste handling issue in a fusion facility. Aspect to consider in the control of the tritium inventory are:

- minimisation of tritium inventory of the fuel cycle particularly that of PFCs;
- accountancy of tritium within fuel cycle to meet both operational and regulatory requirements;
- release minimization in case of DBA and DEC scenarios from components.

The accumulated active dust in the VV is produced due to erosion of the PFC materials (e.g. tungsten) under plasma operation. Disruptions and other off-normal events cause significant erosion, and a sizeable portion of the eroded material generates dust. Off-normal events such as VDEs, may generate dust concentrations locally that are mobilized and adversely affect the overall system. Dust generated from the material may be copious, radioactive, chemically reactive, and/or chemically toxic, thereby, posing significant safety hazards. Shape, size, and mass are key factors in determining dust’s role in safety, its impact on machine operation and health of operators during maintenance.

Due to corrosion phenomena of mobilised activated materials in water coolant and liquid PbLi loops ACPs play an important role in the water-cooled blanket. ACPs will impact occupational exposure, routine effluents to the environment, and potential releases during accidents. Ten activation reactions have been selected for steel/water system in /DIPA 18/:

^{56}Fe (n, 2n) ^{55}Fe;	^{54}Fe (n, p) ^{54}Mn;
^{55}Mn (n, 2n) ^{54}Mn;	^{55}Mn (n, γ) ^{56}Mn;
^{56}Fe (n, p) ^{56}Mn;	^{58}Ni (n, p) ^{58}Co;
^{59}Co (n, γ) ^{60}Co;	^{52}Cr (n, 2n) ^{51}Cr;
^{58}Ni (n, 2n) ^{57}Ni;	^{58}Ni (n, np) ^{57}Co,

which give the following isotopes: ^{55}Fe, ^{54}Mn, ^{56}Mn, ^{58}Co, ^{60}Co, ^{51}Cr, ^{57}Ni, ^{57}Co.

Neutron sputtering is one mechanism for eroding the FW and contamination of the plasma. Due to high neutron energies structural and piping material atoms can be ejected from blanket and FW surfaces by neutron sputtering, and then transported throughout the cooling system. If such surface is of the inner wall of a cooling channel, the neutron sputtering products enter the cooling loop. They may be a source in He-cooled blanket, where corrosion is less important.

Inventory evaluation of mobilized source terms is an important issue for public and occupational safety. For DEMO they have been estimated with a semi-empirical approach to scale the radioactive inventory limits of ITER in /MAZZ 19/: 689 – 1,389 kg for dust and 671 – 4,676 g for tritium. However, such tentative scaling or direct extrapolation of experimental data result in large uncertainties. New assessment is in progress to link experimental data with plasma-wall interaction /COLL 20/ and tritium modelling /SPAG 20/. ACPs have been assessed for the DEMO WCLL with previous code PACTITER-v2.1 /DIPA 18/ and the new assessment with code OSCAR v3.1 /TERR 20/. For the WCLL accident analysis in /DONO 21/ the tritium (in form of HTO) of 2.726 g in the FW-PHTS and 4.82 g in the BZP-HTS, and the ACP quantity in a FW/BZ PHTS loop of 10 kg with a mobilization fraction of 7% are used. Inventory of sputtering products in use of W/EUROFER needs to be assessed with valid code for estimating sputtering rate.

In addition, polonium-210 is generated in the PbLi loop of the WCLL concept due to neutron interaction with bismuth impurities. Therefore, the level of bismuth impurities in the PbLi shall be below of a certain value, from which the generated polonium amount is acceptable to be released into the environment. In the DEMO PbLi purification section the polonium is removed in the stripping gas and is continuously monitored. Therefore, accumulation of polonium and uncontrolled release is avoided.

In blanket concepts containing beryllium, impurities from uranium leads to transuranic elements (see Task 2 ”Radwaste Categorization and Inventory”). These elements like plutonium are relevant for the disposal of waste and the radiation protection inside the plant. Correspondingly, the amount has to be limited by reducing the impurities in beryllium.

As depicted in Section 4, the specific aspects of a FPP are summarized as follows:

- Tritium: Tritium is one of the fuel components and is bred in the blanket.
    - The inventory as a main component of a potential source term
    - Tritium is able to permeate through nearly all material so that a confinement of this Isotope is difficult and the main issue for safety
    - Tritium is an isotope of hydrogen:
        - hydrogen in water and oil can be replaced by tritium and incorporated into the body
        - the danger of a hydrogen explosion can emerge
- Dust: During fusion reaction (D+T) neutrons emerge activating materials. Beside this activation the harsh conditions in form of particle impact onto FW and divertor (SOL) dust is produced which is then activated.
    - Dust is one main part of a potential source term
    - Dust is respirable and can cause pneumonia, lung cancer etc. beside the fact of having incorporated radiating material
    - Dust can lead to dust explosion under certain circumstances
- Magnetic field: In case of a magnetic cage for the plasma, a lot of energy is stored in it and mechanical forces are applied to the structures
    - release of magnetic energy can destroy the FW in case of disruption
    - mechanical load on structures means an elastic deformation so that mechanical energy is stored which can lead to missile effects in case of loss of structural integrity e.g. caused by a quenching magnet
- Flywheel: Many experimental plants (ASDEX, JET) use flywheels for energy storage in order to heat the plasma
    - in case of failure of the bearing, the wheel can be mobilized and cause damage to surrounding buildings
- Waste: Due to the activation of structures inside the VV, waste is produced
    - waste produced in FPP generates relative low activity
    - a certain fraction of waste has to be disposed for $10^2 \ldots 10^3$ y
    - in order to minimize the activity and to save fusion fuel, waste has to be detreated
- chemical inventory

The following proposals for fusion specific safety requirements consider the safety issues identified in Section 4.2.

5.3.3.1 Safety aspects for requirements for magnetic systems

The magnetic systems may cause the following hazards to be considered in the design and during operation of the magnetic systems in a fusion facility:

- Energy release in case of quench;
- Static and dynamic loads on structures;
- Exposure of workers to electromagnetic fields;
- Potential damage of structures due to objects attracted by the operating magnetic fields.

The main hazard is related to the quench of a superconducting magnet when the temperature in the coils exceeds the critical temperature. This will lead to a sudden release of stored energy and a potential evaporation of the cryogenic coolant. To deal with such an event the magnetic systems should be equipped with a fast quench detection system and means to quickly discharge the stored energy in a controlled manner, e.g. by fast discharge units.

The magnetic fields will create forces and loads on the surrounding structures. It is important that this loads and forces are carefully determined and considered during the design of SSCs.

Due to the high magnetic field strength of several Tesla an assessment against the exposure limit values (ELV) and action levels (AL) defined in Annex II and III of Directive 2013/35/EU should be made. The risk for workers arising from electromagnet fields at the workplaces should be assessed by appropriate calculation or measurement. Risks arising from electromagnetic fields at the workplace should be eliminated or reduced to a minimum during the design. In case the action levels (AL) are exceeded but still comply with the exposure limit values (ELV) technical and/or administrative measures should be implemented to prevent exposure exceeding the health effects ELVs and sensory effects ELVs.

Strong magnetic fields can attract magnetic or magnetisable objects (e.g. tools) which may cause damage to structures if uncontrolled attracted. This aspect is more important during maintenance and in-service inspection to use appropriate tools or secure objects not to be attracted by magnets when performing work in close proximity to magnets in operation.

5.3.3.2 Safety aspects for requirements for the vacuum vessel

The vacuum vessel is considered as part of the first barrier to confine radioactive materials generated within the plasma chamber. It will be necessary that the vacuum vessel should be designed, manufacture, constructed, located and operated such that the occurrence of rapidly propagating cracks and brittle fracture does not need to be postulated and a sudden failure of the vacuum vessel need not to be postulated. Such an approach is well known for the integrity of the reactor pressure vessel in nuclear power plants. In case of fusion power plants, the vacuum vessel needs to withstand low pressure and has to withstand loads during normal operation as well as loads generated during anticipated operational occurrences (AOO), design basis accidents (DBA), design extension conditions (DEC) as well as impacts resulting from internal and external hazards. The following design principles should be applied when designing the vacuum vessel and its adjacent structures being part of the first barrier:

- use of high-quality materials, in particular regarding toughness and corrosion resistance;
- conservative limitation of stresses;
- prevention of stress peaks by optimised design and construction, and
- assurance of the application of optimised manufacturing and testing technologies.

This includes the knowledge and assessment of possibly existing defects, which requires a sound quality assurance during manufacturing and construction to ensure early detection of potential defects. This is not only important during design, manufacturing, and construction, but also during operation to ensure and maintain the integrity of the vacuum vessel. For this purpose, additional measures and equipment for the monitoring of causes and effects of damage mechanisms, in particular of leakages during operation should be specified and installed.

As the vacuum vessel will be operated at low pressure, but can also experience pressure built up, e. g. in case of in-vessel LOCA, the vacuum vessel should be equipped with an effective and reliable system for pressure limitation and overpressure protection on levels of defence 1 to 4.

5.3.3.3 Safety aspects for requirements for the breeding blanket system and first wall

As described in Section 4.2.3 two different technologies are under discussion for the breeding blanket including the first wall: the water-cooled concept and the helium cooled concept. The water-cooled concept uses two separate loops, a PbLi loop for tritium breeding and a water loop for cooling.

To avoid an exothermic reaction both PbLi and water coolant loops should be reliably separated to prevent a chemical reaction of PbLi with water. PbLi loops should be designed as double-wall pipes. It is recommended that the anulus between both concentric pipes should be evacuated and monitored to early detect a potential leak and to initiate appropriate countermeasures. As PbLi is highly corrosive, the inner surface should have a protective layer and the mass flow should be designed to have low flow velocities. From an operational point of view, PbLi should be emptied from the loop before starting maintenance work. As PbLi freezes at about 234 °C heaters should be provided in the system to avoid clogging of the pipework due to freeze out phenomena. The pipe performs a barrier function for the confinement of tritium generated within the PbLi. The materials should be selected with respect to a low permeability of tritium. Evacuation of the anulus further contributes to control of tritium. The PbLi system should be equipped with a possibility to extract radioactive or toxic materials, like ^{210}Po or ^{203}Hg.

In such designs utilizing fissile material (e. g. using U-Zr alloy as breeding material) the design should ensure subcriticality by demonstrating that k_{eff} is lower than 0.95.

The materials for the first plasma facing components should be selected to minimize dust generation, withstand the expected temperatures and minimize chemical reactions generating reactions products (e. g. hydrogen) challenging the integrity of the confinement function in case of accident conditions.

5.3.3.4 Safety aspects for requirements for limiters and divertors

Safety requirements for limiters

The main safety issues regarding the limiters are described in Section 4.2.4. In case of actively cooled limiters the design should take into account the permeation of tritium into the coolant. Materials should be selected to hamper permeation of tritium and the limiters cooling system should be provided with means to remove tritium from the coolant. To minimize generation of radionuclides by activation materials should be selected which cannot be easily activated. In addition, the design should provide means to control the water chemistry to minimize corrosion of the cooling loops. Limiters should be made of materials which will not easily produce dust when interacting with plasma.

Safety requirements for divertors

The divertor has to sustain very high heat and particle fluxes arising from plasma as described in Section 4.2.5. The PFC carries plasma-wall interaction leading to dust generation. Chemical reactions between steam and tungsten can lead to H_2 production. The divertor contributes to thermal shielding and neutron shielding of the VV.

The divertor should be designed to withstand thermal stresses and electromagnetic forces. The divertor zone should be equipped with measurements to monitor the amount of generated dust due to erosion. Materials should be selected to withstand high heat and particle fluxes

5.3.3.5 Safety aspects for requirements for the primary heat transfer system

The primary function of the primary heat transfer system is to remove heat from the vacuum vessel to the heat exchangers. Safety issues of the primary heat transfer system are described in Section 4.2.6. As the coolant will contain tritium and activated corrosion products, the boundary should provide a barrier for the confinement of radioactive materials. Materials for the coolant loops should have a low permeability for tritium and should be resistant to corrosion to minimize the amount of activated corrosion products in the primary loops. The design should be such that the leak tightness can be tested to ensure integrity over the lifetime of the plant. The primary heat transfer system should be designed to withstand loads during normal operation and during accident conditions. To limit the ingress of water into the vacuum vessel in case of accidents the coolant loops should be equipped with isolation valves. For designs requiring decay heat removal in accident conditions to prevent structural damage to in-vessel components, a safety system for decay heat removal should be provided independent from the primary heat transfer system. The decay heat removal system should have an ultimate heat sink independent from the one of the primary heat transfer system. The decay heat removal system and the associated ultimate heat sink should be protected against external and internal events in such a way that the primary heat transfer system and the decay heat removal system will not fail due to a common cause or common mode failure. If decay heat removal will be necessary after plasma shutdown, a diverse heat sink should be provided to ensure heat removal in situations where the primary heat sink is unavailable.

5.3.3.6 Safety aspects for requirements for the cryostat and cryosystems

Cryogenic systems can be considered as auxiliary systems to operate the superconducting magnetic systems of a fusion facility. The use of large amounts of cryogenic coolants, in particular liquid helium, poses an additional risk to the safety of workers as well as to SSCs important to safety and thus challenging the performance of safety functions. This could lead to a challenge of the confinement system through over-pressure and under-temperature stresses. Therefore, the confinement of cryogenics has to be ensured, in particular for locations with radioactive material.

Table 16 provides an overview of the associated hazards relevant for workers as well as for plant safety. Workers can suffer from contact burns or frostbite in case of contact with liquid He or cold surfaces. In case of helium expulsion, inhalation of cold gases may damage the lunges. As evaporated helium will replace oxygen, in particular in elevated areas, there is a risk of asphyxiation. Areas, where larger amounts of liquid helium will be stored, processed or transferred, should be equipped with oxygen monitors and categorized as an oxygen deficiency hazard (ODH) area.

Besides the risk for the workers, accidental expulsion of liquid helium may impose consequential hazards on SSCs important to safety. In case of either a loss of helium cooling incident or due to an incident heating up the liquid helium, the pressure within the cryostat will increase. Thus, the vessels have to be designed to withstand a specified design pressure. In addition, a reliable safety relief valve system should be provided. To avoid threat to the workers the vent lines and exhausts should be designed and located in areas where workers are not normally working. Another risk of expulsed helium is the temperature decrease of surrounding SSCs important to safety. This may cause embrittlement and thermal contradiction (thermal stresses), challenging the structural integrity or execution of safety functions in case temperature will be below the qualified environmental conditions of a certain SSC. Therefore, SSCs important to safety need to be protected against impacts from accidentally expulsed cold liquids and gases. Due to the low temperature, atmospheric gases may condense and consequently increasing the oxygen concentration and thus increasing the risk for combustion within the plant. Finally, due to the low temperature of exposed SSCs humidity may condense and may cause icing of safety relevant components. It has to be ensured, that SSCs important to safety will be protected against the risk of icing.

Table 16 Hazards associated with cryogenic coolants.

Workers	Plant safety
Contact burns	Pressure build-up
Frostbite	Embrittlement
Asphyxiation	Thermal contraction
Inhalation	Condensation of atmospheric gases
	Icing

5.3.3.7 Safety aspects for requirements for the tritium extraction and removal system

As tritium is one of the main driver for the radiological risk, the safety of tritium handling is important for future fusion facilities. Due to the high permeability of tritium the design of the tritium handling, processing and storage systems should contribute to a minimisation of tritium releases into working areas as well as into the environment. The requirements have been derived from good practices identified by the Canadian Regulator CNCS /CNSC 09a/(see also section 4.7.5).The requirements have been derived from good practices identified by the Canadian Regulator CNCS /CNSC 09a/ (see also Section 4.7.5).

To reduce the concentration of HTO in the exhaust of pumps installed in the tritium handling systems, oil-free pumps should be used. The design should provide means to remove tritium gas and HTO from vacuum pump exhaust. A treatment or disposal route for the abated tritium should be included in the design.

SSCs of the tritium handling system contributing to the confinement function of gaseous tritium should be of high quality.

Getter beds for storing tritium should have a secondary containment to reduce chronic releases.

The tritium handling system should be equipped with a leak detection system and possibilities to isolated parts of the system where the leak occurred.

The ventilation system should be designed in such a way the tritium can be released in such a way that the amount of tritium can be accounted and that the release points (stacks) are designed to achieve good dispersion of tritium gas and tritiated water vapour.

The design should provide measures for inert gas purging and adsorption of tritium on getter beds before opening the pipework of the tritium system to minimize releases of tritium in case of maintenance.

5.3.3.8 Safety aspects for requirements for the H&CD system

The safety requirements for heating and current drive systems (neutral beam injection, electron cyclotron and ion cyclotron) can be commonly formulated because the safety issues of all three systems (NBI, EC and IC) are very similar.

All three systems contribute to the confinement function and thus the integrity of the barriers provided by these systems should be ensured. To prevent a release of dust, activated corrosion products and tritium from the vacuum vessel through the NBI, EC or IC the design of these systems should provide possibilities to isolate, either permanently by a reliable window or by fast shutters, the H&CD systems from the vacuum vessel. To ensure the confinement function the systems should be equipped with monitoring and control systems to maintain the vacuum condition to detect deviation from the specified operational limits and conditions.

Neutral beam injection

The neutral beam injection (NBI) systems contribute to the confinement function. It supports retention of radioactive materials generated within the vacuum vessel and prevents releases of hydrogen to the tokamak building (risk for hydrogen explosions). Outside the NBI hydrogen sensors should be installed to detect potential releases of hydrogen to detect potential formation of explosive atmosphere and to initiate countermeasures. The design should provide measures to monitor and control the vacuum conditions of the NBI interfacing the vacuum vessel. A permissible leak rate should be defined, and the leak tightness should be measured. Materials for the NBI should be selected to minimize activation. Due to the expected high activation of the components of the NBI the design should include provisions to facilitate remote handling. The residual electromagnetic fields in working areas (generated by the RF drivers and magnetic fields) should not exceed the exposure limit values (ELV) and action levels (AL) defined in Annex II and III of Directive 2013/35/EU. The neutral beam injection system should be shielded to minimize exposure of workers in adjacent work areas. The vacuum system should be designed to trap hydrogen and to release exhausts in areas without increasing risk for hydrogen explosions. NBI systems uses high voltages and thus these systems should be designed and operated in such a way that risk for arching will be minimized. The NBI should be equipped with a fast shutter to separate the NBI cell from the vacuum vessel in case of water or air ingress into the vacuum vessel to prevent transport of dust, activated corrosion products or tritium into the NBI cell.

*E*lectron Cyclotron system

The residual electromagnetic fields in working areas and action levels (AL) defined in Annex II and III of Directive 2013/35/EU. The electron cyclotron (EC) system is directly connected to the vacuum vessel and thus contributes to the confinement function. The design should provide measures to monitor and control the vacuum conditions of the EC systems interfacing the vacuum vessel. A permissible leak rate should be defined, and the leak tightness should be measured. EC systems uses high voltages and thus these systems should be designed and operated in such a way that risk for arching will be minimized.

Ion Cyclotron system

The residual electromagnetic fields in working areas and action levels (AL) defined in Annex II and III of Directive 2013/35/EU. The design should include the possibility of a safe routing of RF waves to dummy components to prevent damages to the blanket in case of signals triggered by the diagnostic and control system. EC systems uses high voltages and thus these systems should be designed and operated in such a way that risk for arching and breakdown will be minimized.

5.3.3.9 Safety aspects for requirements for remote maintenance

Remote procedures to perform inspections and maintenance are foreseen in fusion facilities to avoid staff entering areas having high dose rates and intense residual magnetic fields. Based on the insights described in Annex B.16 the following safety requirements can be derived. As there might be a risk of overheating of IVCs when detached from the coolant loops due to the decay heat, procedures should be in place (e.g. operational limits and conditions) specifying the time period between shutdown and detachment from the cooling loops to ensure that no damage to SSCs will occurred during remote handling of the IVCs. Load drop events while transporting IVCs with a crane should be prevented by the following design requirements related to the overhead lifting equipment stated in SSR 2/1 requirement 76 /IAEA 16/:

- Measures are taken to prevent the lifting of excessive loads;
- Conservative design measures are applied to prevent any unintentional dropping of loads that could affect items important to safety;
- The plant layout permits safe movement of the overhead lifting equipment and of items being transported;
- Such equipment can be used only in specified plant states (by means of safety interlocks on the crane);
- Such equipment for use in areas where items important to safety are located is seismically qualified.

A suitable system of cameras should be in place to monitor remote handling activities in areas where no operating staff is present. Cameras should be selected to withstand the expected radiation fields. Remote handling equipment should be equipped with sensors to avoid collisions with walls and IVSs. A second remote handling systems should be available to savage crashed components (including damaged cranes and robotic arms) inside the vacuum vessel.

Shielding should be provided during removal, transport, and storage of activated and contaminated components. Transportation casks should be foreseen for transporting components within the facility. Where permanent shielding is not feasible, mobile shielding should be foreseen in the design and described in the relevant procedures.

The design of remote handling equipment should take into account the fire risks associated with remote cutting and welding systems. Flammable materials should be avoided in the vicinity of areas where remote cutting and welding will be necessary.

It should be demonstrated for the entire life cycle of computer-based or programmable logic device (PLD) used in remote handling equipment that any manipulation of these systems is excluded by design or security measures.

5.3.3.10 Safety aspects for requirements for active management facilities and radwaste systems

Radioactive waste management

Radioactive wastes in a fusion power plant are significantly different from wastes of fission nuclear power plant with much larger quantities, huge amount of tritium, different radio-isotopes, but only tiny amounts of transuranic elements (from impurities of Be for example), no fission product, lower decay heat removal, lower radioactivity content. Radioactive waste in DEMO arises both from neutron activation of material and material contamination with tritium. Tritium will be one of the major isotopes in the radwaste generated in fusion devices. Tritium tends to migrate and diffuse through all standard materials (concrete and metals) that are the main constituents of radioactive disposal facility. With tritium recover environmental releases in storage and disposal facilities will be reduced, as current repositories are not designed for large amounts of tritium. Detritiation treatment prior to recycling is necessary for all fusion components and contributes to a saving of this material valuable to FPP (fuel component). Preliminary evaluation of radioactive waste arising from EU DEMO together with some selected techniques to reduce the radioactive waste burden (e.g. detritiation, smelting, chlorination, decarburization, reduction of impurities in the pre-use materials, etc.) have been investigated /PORF 20/. Operational wastes include gaseous waste, aqueous waste, organic liquid, and solid waste. Depending on the blanket concept the wastes will be composed by structure materials like irradiated steel and Eurofer, corrosion products (ACP) and more valuable substances such as beryllide (HCPB concept) or lithium (WCLL concept). Control of waste volume and hazard is in line with the safety function of limiting environmental legacy.

Following IAEA 2009 classification system /IAEA 09/ the preliminary classification of radioactive waste is proposed for EU DEMO: Non Active Waste (NAW) is a material with an IAEA clearance index of less than 1 /GILB 18/; a material is Low Level Waste (LLW) (incl. VLLW) if both its alpha-producing activity is below 4 MBq/kg and if the sum of beta and gamma activity is less than 12 MBq/kg /GILB 18/; Intermediate Level Waste (ILW) is particularly for long-lived radionuclides which require a greater degree of containment and isolation than that provided by near surface disposal but no provision / only limited provision /IAEA 09/; and High Level Waste (HLW) with levels of activity concentration high enough to generate significant quantities of heat by the radioactive decay process or large amounts of long-lived radionuclides that need to be considered in the design of a disposal facility /IAEA 09/. In addition, material is considered to be potentially recyclable if the calculated contact gamma dose rate is below 2 mSv/h /GILB 18/. Selection and adopted procedures decide on the management of radioactive releases and wastes. The results of the HCPCB in /PARK 21/ show that according to the U.K. classification tungsten and Eurofer structures would get LLW level already after 50 and 200 years respectively. Be12Ti will remain ILW during whole cooling time up to 1,000 years. The uranium impurities in the Be12Ti have effect on their reprocessing and storing strategy. Depending on the position of the blanket, about 75 tons of breeder ceramic becomes LLW after 1 week cooling time.

For fusion wastes, it is most desirable for the avoidance, followed by minimisation, reuse, recycle, and least desirable for disposal. The recycling/clearance strategy should be investigated at an early stage of any fusion design. HLW should be avoided in fusion. The amount of LLW is large, so recycle and clearance are essential to support fusion deployment. Decay storage before disposal could be the reference solution for tritiated waste. Radwaste burden for future generations needs to be minimised. The final disposal is only for the captured wastes which cannot be treated with any available technology for the reuse.

Waste classification should not incorporate dose rate limits since in practice these will depend upon facilities and regulations in the host country. The waste classes may depend on the other country specific regulations and the Waste Acceptance Criteria of specific waste repositories.

5.3.3.11 Safety aspects for requirements for electric power supply

Requirements for the emergency power supply are provided in SSR 2/1 Requirement 68 /IAEA 16/ or SSR 4 Requirement 49 /IAEA 17/. More details can be found in SSG-34 /IAEA 16f/. The electric power supply system should be designed following the approach of defence-in-depth. In normal operation, the AC power supply is provided by the grid connections. In addition, an emergency power supply system should be provided to supply all items important to safety with electric power to bring and maintain the facility in a safe state[1] in anticipated operational occurrences and design basis accidents in case of a loss of off-site power. The emergency power supply should be independent from the grid connection. Items important to safety requiring uninterruptible power supply should be identified and an uninterruptible AC or DC power supply should be provided in the design. The emergency power supply should be designed to be functional in case of impacts from internal or external hazards considered in the design basis of the facility. For design

[1] In a safe state the fusion facility can fulfil the fundamental safety functions, in particular the confinement of radioactive materials and, if necessary, the removal of decay heat.

extension conditions an alternative power supply should be provided compensating a failure of the emergency power supply. The alternate power supply should be independent from the emergency power supply. Sufficient supplies should be provided for the emergency power supply systems and the alternative power supply system to cover the time until re-establishing the power supply from the grid connection. Connection points for mobile equipment should be included in the design to provide necessary items important to safety with electric power to ensure the confinement of radioactive materials in case of impacts from external hazards exceeding the design basis events.

5.3.3.12 Safety aspects for requirements for balance of plant

As no safety functions to be performed by the balance of plant systems, is should be only requested to demonstrate that no detrimental feedback of the systems, structures and components will have an impact on fulfilling the safety functions of the fusion facility.

5.4 Codes and standards

Codes and standards provide guidance for the design of specific SSCs. The level of detail is usually very high and the scope is technology specific. Codes and standards contribute to the harmonization of technical design solutions. With the emergence of commercial fusion facilities, interest increases to establish fusion specific codes and standards rather than adapting codes and standards originally developed for fission facilities. Due to the nature of codes and standards to be very detailed and SSC specific, development of codes and standards for fusion requires expert with a deep technology background of the SSCs to be dealt with in the codes and standards. Experts from vendors, designers, operators, research and regulators need to cooperate to establish relevant codes and standards. Currently, there are ongoing projects e.g. within ASME to establish fusion specific codes and standards. It is important, that codes and standards comply with high level safety requirements and would not create contradictions to the legal and regulatory framework.

It is also emphasized that harmonised codes and standards would not only contribute to a harmonized safety level, but also facilitate the licensing procedure if a certain design will be deployed in several countries. Thus, harmonization of codes and standards will not only contribute to safety but also impacts economic aspects and contributes to facilitate the deployment of fusion facilities.

5.5 Interface between Safety, Security and Safeguards

A definition of nuclear safety and nuclear security can be found in the IAEA Safety Glossary/IAEA 18/. Nuclear safety is defined as the achievement of proper operating conditions, prevention of accidents and mitigation of accident consequences, resulting in protection of workers, the public and the environment from undue radiation risks. Nuclear security means the prevention and detection of, and response to, criminal or intentional unauthorized acts involving nuclear material, other radioactive material, associated facilities or associated activities. Although nuclear safety and nuclear security have several common approaches, also differences exist which may result in diametrical technical solutions, which should be avoided. That both, nuclear safety and nuclear security have the common aim of protecting human life and health and the environment, was emphasized during the IAEA's 52nd General Conference in 2008, as can be found in resolution GC(52)/RES/9, which was adopted by the Member States:

> *"[A]cknowledges that safety measures and security measures have in common the aim of protecting human life and health and the environment, calls upon the Secretariat to enhance its efforts to ensure coordination of its nuclear safety and security activities, and encourages Member States to work actively to ensure that neither safety nor security is compromised;"*

The International Nuclear Safety Advisory Group (INSAG) published its report INSAG-24 "The Interface between Safety and Security at Nuclear Power Plants" /INSA 10/ discussing the interface between safety and security in more detail. On the European level, WENRA initiated a task force to elaborate on the interface between safety and security. The findings of this task force were published in the WENRA report "Interfaces between Nuclear Safety and Nuclear Security" /WENR 19/ published 2019. Obviously, up to now the interface between nuclear safety and nuclear security was intensively discussed for nuclear power plants but up to now not for fusion facilities. Within this study, the available information on nuclear power plants will be reviewed and recommendations for addressing the interface between safety and security at fusion facilities will be derived recognizing the specific differences between nuclear power plants and fusion facilities.

The general requirement to address the interface between safety and security in a coordinated manner in such a way, that safety and security measures will not compromise one another covering the whole lifetime of the facility can be found in several IAEA Safety Requirements, such as IAEA SSR 1 para 1.14 /IAEA 19/, IAEA SSR-2/1 Requirement 8 /IAEA 16/, IAEA SSR-2/2 Requirement 17 /IAEA 16b/, IAEA SSR-3 Requirement 11 /IAEA 16c/ and IAEA SSR-4 Requirement 75 /IAEA 17/. In addition, SSR-2/1, SSR 3 and SSR-4 address the interface to the State system of accounting for, and control of, nuclear material. Due to the large amount of tritium used in a fusion facility, the inventory of tritium needs to be accounted. The French authority /ITER 10/ as well as the US Department of Energy /KLEI 15/ requires the surveillance of the tritium inventory also for fusion facilities. Although the French and American authorities require the accountancy of tritium, tritium is not in the scope of international safeguard obligations, e.g. Commission Regulation (Euratom) No 302/2005 – Nuclear Safeguards.

Based on the information above a general requirement for fusion facilities can be formulated:

- The interfaces between safety, security and the State system of accounting for, and control of, nuclear material shall be managed appropriately throughout the lifetime of the fusion facility. Safety measures and security measures shall be established and implemented in a coordinated manner so that they do not compromise one another.

Already in an early stage of a fusion project site specific aspects having implications for nuclear safety and nuclear security should be considered while selecting an appropriate site /INSA 10/ , /WENR 19/. This includes assessment of the natural and human induced site-specific hazards but also the vulnerability to assault of the site. The Finnish guideline YVL B.7 "Provisions for Internal and External Hazards at a Nuclear Facility" /STUK 13/ emphasizes to consider also nuclear security aspects when designing the layout of the site, as can be found in para. 301:

> *"Design of the site area layout shall be appropriate considering the facility's nuclear and radiation safety, preparedness and rescue arrangements as well as nuclear security."*

INSAG and WENRA highlighted in their reports that close cooperation between safety and security experts is mandatory to effectively address issues related to the interface between safety and security. This cooperation should be part of the organizational culture and should be supported, promoted and fostered by means of the management system integrating safety and security in mutually supporting manner. Conflicts between safety and security need to be addressed and solved in a timely manner by established processes.

A further conflict between safety and security exists with respect to balancing transparency and confidentiality. Transparency and open communication to exchange information is essential for improving safety and to build trust to stakeholders. In contrast to nuclear safety, nuclear security relies strongly on confidentiality. WENRA proposes in /WENR 19/, that releasing information, e.g. to promote public confidence, should be done in a measured way. Therefore, it is necessary to establish a process to identify, control and protect sensitive information.

During the design phase, but also periodically during the operational phase of a fusion facility, the vulnerability of the plant and SSCs important to safety or security need to be jointly evaluated by nuclear safety and nuclear security experts. This issue is also emphasized in the INSAG-24 report /INSA 10/. Insights from nuclear safety will guide nuclear security experts to identify security measures to be implemented. At least vital safety functions to prevent unacceptable releases need to be ensured by appropriate safety and security measures. Vice versa, nuclear safety experts should be involved in the evaluation of consequences of malicious acts, based on design basis threats, in order to implement safety measures to enhance the resilience of items important to safety to prevent undesired radiological consequences in case of malicious acts. For identification of vital areas from a nuclear security point of view guidance is provided in IAEA Nuclear Security Series No. 16 "Identification of Vital Areas at Nuclear Facilities" /IAEA 13/.

As computer systems will be part of future fusion facilities, information security measures need to be implemented for all computer systems important to safety or security. Coordination of safety and information security is necessary to avoid adverse effects of safety measures on security as well as adverse effects of security measures on safety as stated in IAEA Nuclear Security Series No. 20 /IAEA 13b/:

> *"Nuclear security and nuclear safety have in common the aim of protecting persons, property, society and the environment. Security measures and safety measures have to be designed and implemented in an integrated manner to develop synergy between these two areas and also in a way that security measures do not compromise safety and safety measures do not compromise security."*

It is essential, that the implementation of information security measures does not have adverse effects on safety objectives of the fusion facility. Hence, effects on safety should be considered in the selection, definition, implementation and - where applicable - operation of information security measures. Potentially adverse effects on safety should be analysed prior to implementation of information security measures. It is important to ensure that the implementation of information security measures

- does not introduce new vulnerabilities that could result in common cause failures between redundant and diverse systems

- does not have adverse effects on operator actions concerning safety and

- does not have adverse effects on fire safety and emergency response arrangements.

Additionally, it is important to ensure that neither normal nor abnormal operation of any information security measure has adverse effects on safety objectives of the fusion facility.

If an intended information security measure cannot be implemented because of safety concerns, the omission of information security measures is not acceptable. Instead, it is important to ensure that the information security measures with potentially adverse effects on safety are replaced by alternative information security measures.

Although the focus of coordination of safety and information security is on avoiding adverse effects of information security on safety objectives, adverse effects from safety-related actions on information security should also be avoided whenever possible. In this regard it should e. g. be ensured that periodic testing of safety functions has no adverse effects on the I&C system's information security.

All in all, there should be efforts to minimise potential conflict between safety and security already during design phase of a fusion facility as well as during later phases of the computer system life cycle.

In its report /INSA 10/ INSAG discusses the specific meaning of the construction phase for nuclear safety as well as nuclear security of the facility. It is emphasized that careful oversight has to be provided to ensure that the plant will be constructed as designed. Inadvertent or intentional introduction of weaknesses during the construction phase may result radiological releases during the operating phase, either to less resistant structures, systems and components in case of accidents or by introducing additional vulnerabilities to the plant which can be utilized by assailants in a later phase of the lifetime. This issue is challenging for the licensee as well as for the competent regulatory authority because during the construction phase a large number of workers even from sub-contractors having different professional backgrounds are working at the same time on the site. In addition, some inspections can only be performed once in a time during the construction phase. In the course of progressing construction some structures, systems and components will no longer be accessible for inspections and oversight.

WENRA emphasized in its report /WENR 19/ that all events concerning nuclear safety or nuclear security should be recorded and evaluated as an event related to nuclear safety may reveal a vulnerability related to nuclear security and vice versa. This requires joint evaluation programs between nuclear safety and nuclear security personnel.

Special obligations may arise during periods in which extensive plant modifications are under way. During such activities, many contractors may need to enter the vital area of the plant, resulting in the need for appropriate access controls for both safety and security purposes. Care must be taken to prevent the inadvertent or intentional introduction of vulnerabilities.

5.6 Conclusion

To establish a legal and regulatory framework for the safety of D-T fusion facilities with magnetic confinement the following steps and documents will be necessary. The development of a consistent and complete regulatory framework requires the involvement of various stakeholders, like international organizations, national regulatory authorities, technical safety organizations, research and industry providing the specific competences for drafting, reviewing and negotiating on relevant safety documents. As each government is responsible for establishing and maintaining an adequate national regulatory framework the proposed international regulatory framework will serve as kind of blueprint to be implemented in national regulatory frameworks and adapted to the specific boundary conditions of a country.

On the highest-level principle high-level safety objectives and requirements need to be established which can be implemented in the national laws and decrees. For Europe, such a document could be a Council Directive addressing such high-level safety objectives and requirements for fusion facilities. This can be achieved by either amending an existing directive (like the nuclear safety directive 2009/71/Euratom) or developing a new directive establishing a harmonized regulatory framework for fusion facilities in Europe. This will ensure a harmonized level of safety to be reached in Europe and facilitates the deployment of future fusion power plants in the European Union by avoiding too different requirements defined by the national regulators. It is recommended, that for the development of such a document IAEA Safety Standard No. SF-1 is taken into account. Such a document could be initiated by the European Commission together with the Member States (e.g. via ENSREG).

The legal framework needs to be substantiated and concretized further by a set of safety requirements. General safety requirements have already been published by IAEA and can be directly applied to fusion facilities as well. In particular, these are the following IAEA Safety Standards:

- IAEA Safety Standard No. GSR Part 2 "Leadership and Management for Safety";
- IAEA GSR Part 3 "Radiation Protection and Safety of Radiation Sources: International Basis Safety Standards";
- Here, also Council Directive 2013/59/Euratom needs to be considered;
- IAEA Safety Standard No. GSR Part 4 "Safety Assessment for Facilities and Activities";
- IAEA Safety Standard No. GSR Part 6 "Decommissioning of Facilities";
- IAEA Safety Standard No. GSR Part 7 "Preparedness and Response for a Nuclear and Radiological Emergency";
- IAEA Safety Standard No. SSR-1 "Site Evaluation for Nuclear Installations".

As the above mentions Safety Standards can be either directly or with small modifications adapted to fusion facilities, specific safety requirements for fusion facilities like

- IAEA Safety Standard No. SSR-2/1 "Safety of Nuclear Power plants: Design";
- IAEA Safety Standard No. SSR-2/2 "Safety of Nuclear Power plants: Commissioning and Operation";
- IAEA Safety Standard No. SSR-3 "Safety of research reactors" or
- IAEA Safety Standard No. SSR-4 "Safety of Nuclear Fuel Cycle Facilities"

need to be developed to address the specific safety needs for the design and operation of D-T fusion facilities with magnetic confinement. IAEA has already started an initiative to develop such a specific safety requirement based on IAEA Safety Standard No. SSR-4 (see also the Annex C). However, this approach lacks on the definition of specific design requirements for the various systems of a fusion facilities like it is provided in SSR 2/1 or SSR-3. This study provides recommendations for such specific design requirements targeted to the SSCs of D-T fusion facilities with magnetic confinement.

The safety requirements need to be substantiated by a set of safety guides providing guidance on how the safety requirements can be met. Typically, safety guides are not mandatory, but describes technical solutions to fulfil a certain safety requirement. Due to its nature, safety guides will become more technology specific compared with safety requirements. In addition to the safety guides codes and standards need to be defined. This requires a further discussion amongst regulators, research organizations and industry where industrial codes and standards can be applied and where the development of fusion specific codes and standards addressing fusion specific safety issues will be necessary. The contribution to develop the documents forming the regulatory framework will also vary between the different levels. On the highest-level, the role of the regulatory authorities is dominant, with an increasing level of technical details, the contributions shift more and more towards the technical experts and industry.

6 ACTION PLAN

This study concluded that presently no country applies a dedicated regulatory framework for the regulation of D-T fusion facilities with magnetic confinement. On the other hand, larger facilities are under construction or in planning including commercial fusion power plants. As the radiological hazard potential is higher compared to existing fusion research facilities, but still lower than for commercial large nuclear fission power plants, the development of a targeted and fusion specific legal and regulatory framework is considered necessary.

As a starting point, this study emphasizes the need to establish and agree on a safety concept based on international recommendations. Fusion specific safety issues have been identified and proposal for fusion specific safety requirements have been derived. This proposal needs to be discussed with the stakeholders, developed further and agreed upon.

The different actions of the actions plan are listed in Table 17.

The first action identified is to establish a common European legal framework to regulate fusion facilities as discussed in Section 5.2. This legal framework forms the basis on which the regulatory framework can be developed. Two options how this could be done were discussed in Section 5.2. The involved stakeholders are the European Commission and the Member States.

For the implementation of a regulatory framework, different fusion specific concepts have to be developed based on generic safety concepts also used in fission. These are tackled in the actions 2, 3. The involved stakeholders are the European Commission, the IAEA, national regulatory authorities, research organisations and the fusion industry. The findings of these two actions will influence all the following actions.

Based on the legal framework and the results of actions 2 and 3 the next step is to develop “Safety requirements for D-T fusion facilities with magnetic confinement”. In this action the stakeholders are the European Commission, the IAEA, the national regulatory authorities, research organizations, the fusion industry and TSOs.

The next actions would be the development of safety guides by increasing the technical content of the regulatory framework by the stakeholders national regulatory authorities, vendors, operators and TSOs. In parallel, the level of codes and standards needs to be discussed and, where considered necessary, the development of fusion specific codes and standards needs to be initiated by the stakeholders such as standardization organisations (ISO, EC, ASME, IEEE, etc.), fusion industry, national regulatory authorities, research organisations and TSOs.

During the workshop within this project, different participants raised the question, which of the steps of the action plan should be given the highest priority to start soon. One suggestion was that guidelines for designers of fusion facilities should be developed soon, another one that emphasis should be put on the development of high-level safety requirements.

Table 17 Action plan for the development of a targeted and proportionate regulatory framework

No.	Action	Stakeholder (lead underlined)
1	Establish a common legal framework to regulate fusion facilities • Option 1: develop a Council Directive like the nuclear safety directive 2009/71/Euratom dedicated to safety of fusion facilities • Option 2: adapt nuclear safety directive 2009/71/Euratom to extent scope to fusion facilities	• European Commission, • Member States
2	Discuss and agree on a defence in depth concept for fusion facilities • Applicable plant states • Technical acceptance targets • Radiological acceptance targets	• European Commission • IAEA • National regulatory authorities • Research organisations • Fusion industry
3	Develop a graded approach to be applied to fusion facilities to regulate such facilities commensurate with its radiological hazard potential	• Member States • IAEA • Technical safety organisations
4	Develop "Safety requirements for D-T fusion facilities with magnetic confinement" • Starting point: This study and work performed at IAEA • High level safety requirements addressing: o Leadership and management of safety o Siting o Design (general and specific design requirements) o Construction and commissioning o Operation o Decommissioning o Safety demonstration (initial and periodic safety assessments) o If necessary, emergency preparedness and response	• European Commission • IAEA • National regulatory authorities • Research organizations • Fusion industry • Technical safety organizations
5	Establish and implement a system for operating experience feedback • Establish the legal and regulatory basis for a national operating experience feedback system • Establish an international system to exchange operating experiences like IRS, IRS-RR of FINAS	• European Commission • IAEA • OECD/NEA • National regulatory authorities • Vendors • Operators
6	Develop a safety guide on design and operation of large superconducting magnets.	• National regulatory authorities • Vendors • Operators • Technical Safety Organizations
7	Develop a safety guide on the design and operation of cryogenic systems	• National regulatory authorities • Vendors • Operators • Technical Safety Organizations
8	Develop a safety guide on design and operation of tritium handling systems	• National regulatory authorities • Vendors • Operators • Technical Safety Organizations

No.	Action	Stakeholder (lead underlined)
9	Development of a guidance on application of a graded approach • This action requires that safety requirements for fusion power plants needs to be developed first	• <u>National regulatory authorities</u> • <u>Vendors</u> • <u>Operators</u> • <u>Technical Safety Organizations</u>
10	Development of a guidance on the design of electric power supply for fusion power plants • Requires discussion on the items important to safety necessary to control anticipated operational occurrences, design basis accidents and design extension conditions and the required reliability to ensure fulfilment of fundamental safety functions • This action requires insights from safety analysis of several postulated initiating events and their potential radiological consequences	• <u>National regulatory authorities</u> • <u>Vendors</u> • <u>Operators</u> • <u>Technical Safety Organizations</u>
11	Adaption of standards to design overhead lifting equipment in fusion facilities • To practically eliminate load drop events • For example, German KTA Standards 3902 ad 3903 might serve as a starting point	• <u>National regulatory authorities</u> • <u>Vendors</u> • <u>Operators</u> • <u>Technical Safety Organizations</u>
12	Develop a guidance to ensure safety during remote handling • Design of transport casks • Layout of transport routes • Security of IT-Systems	• <u>National regulatory authorities</u> • <u>Vendors</u> • <u>Operators</u> • <u>Technical Safety Organizations</u>
13	Develop fusion specific codes and standards	• <u>Standardization organisations</u> (ISO, EC, ASME, IEEE, etc.) • Fusion industry • National regulatory authorities • Research organisations • Technical safety organisations

7 WORKSHOP

7.1 PLANNING

From 22nd – 24th November 2021 GRS and KIT organized the on-line workshop "On the applicability of the regulatory framework for nuclear facilities to fusion facilities. Towards a specific regulatory framework for fusion facilities".

Prior to the workshop about 50 international experts on the safety of fusion facilities were identified and the invitation for the workshop has been sent to them on October 20th, 2021. They represented different organisations including research organisations and universities, national regulators, TSOs, consultants, plant constructors, and international organisations (EU commission, IAEA). Also, the invited experts represent a broad range of countries from Asia (China, Japan, South Korea), Europe (France, Germany, U.K., Italy, Russia) and North America (U.S., Canada). After agreeing with DG ENER on the agenda, a workshop agenda was distributed to the invited fusion safety experts (Table 18).

The workshop commenced at the early afternoon CET (12:00/13:00 to 16:00) to allow participation from both America and Asia. The agenda provided sufficient time to intensively discuss the findings of this study with the participating experts. Each of the four tasks was be presented followed by a discussion. As Task 3 is considered the main task of this project, so most of the time is reserved for this topic, especially for discussions. To allow for the preparation of the participants and to facilitate the discussion a draft version of the study report was distributed prior to the workshop to the participants.

Altogether 43 participants from licensees, research organisations, technical safety organizations, international organisations and national regulators took part at the workshop. Some participants provided written comments prior to the workshop. Some provided written feedback after the workshop.

The proceedings of the workshop were distributed to all participant (see Annex C)

7.2 DISCUSSIONS

After the presentation of each task, sufficient time for discussions were given. Main points of discussion and of comments received prior and after the workshop were:

- Current used licensing procedures, especially in France for ITER and in UK for JET; Benefits and challenges of the existing approaches were addressed.
- Planned licensing procedures, especially in the UK (green paper) and the US
- Current activities of the IAEA with regard to fusion regulation (TECDOC)
- Overall radioactive inventory of fusion power plants, also compared to ITER and JET
- Underlying basis of the regulatory approach, e.g. regulation based on the unmitigated risk of a fusion facility
- Application of a graded approach; also when applying fusion specific regulations taking into account various designs of FPP
- Application of specific IAEA guides; the way of transferring the content of specific guides to FPP
- Prescriptive vs. goal-oriented regulatory approach
- Process to develop the regulatory framework (e.g. high level objective and principles first, then requirements and guides); importance of design guides

Table 18 Workshop agenda

Time	Topic
November 22nd	**Chair: Björn Becker (GRS)**
12:15 – 12:30	Welcome, Introduction by EC and project team, topics of the day
12:30 – 13:15	Introduction of the participants
13:15 – 13:30	Overview of the project (Björn Becker, GRS)
13:30 – 14:00	Presentation of Task 1: International approaches to fusion regulation and existing safety requirements (Joachim Herb, GRS)
14:00 – 14:45	Feedback from participants on national approaches
14:45 – 15:00	Break
15:00 – 15:30	Presentation of Task 2: Assessment of main differences between fission and fusion and safety issue for specific fusion SSCs (Xue Zhou Jin, KIT and Joachim Herb, GRS)
15:30 – 15:45	Wrap-up of day 1 and adjourn
November 23rd	**Chair: Björn Becker (GRS)**
13:00 – 13:15	Welcome and topic of the day
13:15 – 13:45	Presentation of Task 2: Screening and review of international regulations, safety standards and guides and other documents (Thorsten Stahl, GRS)
13:45 – 14:15	Feedback from participants on Task 2
14:15 – 14:45	Presentation of Task 3: Recommendations for the implementation of the safety requirements specifically needed for fusion facilities (Isabel Steudel, GRS)
14:45 – 15:00	Break
15:00 – 15:30	Feedback from participants on Task 3
15:30 – 15:45	Wrap-up of day 2 and adjourn
November 24th	**Chair: Joachim Herb (GRS)**
13:00 – 13:15	Welcome and topic of the day
13:15 – 13:45	Feedback from participants on Task 3
13:45 – 14:15	Presentation of Task 4: Action plan for the development of a targeted and proportionate regulatory framework for the construction, operation and decommissioning of fusion facilities (Joachim Herb, GRS)
14:15 – 14:30	Break
14:30 – 15:30	Feedback from Participants and discussions of the action plan
15:30 – 15:45	Wrap-up of the meeting and next steps, closure of the meeting

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ANNEX A PROPOSED STRUCTURE OF A REGULATORY FRAMEWORK FOR FUSION FACILITIES

A proposed structure of a regulatory framework for fusion facilities including examples for fusion specific requirements is given in the following. These have been from the discussion in section 5.3.

1. SAFETY OBJECTIVES AND SCOPE

Article 1 – Fundamental, radiological and nuclear safety objective

Fundamental safety objective

(1) A fusion facility shall be planned, sited, designed, manufactured, constructed, commissioned, operated and decommissioned in such a way that people and the environment are protected from the harmful effects of ionizing radiation.

Radiological safety objective

(2) A fusion facility shall be planned, sited, designed, manufactured, constructed, commissioned, operated and decommissioned that in all operational states and accident conditions radiation exposure within the installation or due to any planned or accidental release of radioactive material from the installation is kept below prescribed limits and as low as reasonably achievable.

Technical safety objective

(3) A fusion facility shall be planned, sited, designed, manufactured, constructed, commissioned, operated, and decommissioned with the objective of preventing accidents and, should an accident occur, mitigating its consequences and avoiding

(a) early radioactive releases that would require off-site emergency measures but with insufficient time to implement them;

(b) large radioactive releases that would require protective measures that could not be limited in area or time.

(4) As far as practicable, the quantity and activity content of radioactive waste and discharges to the environment shall be minimized by design and application of control measures.

Article 2 - Scope

These safety requirements are applicable to D-T fusion facilities with magnetic confinement.

Article 3 – Fundamental safety functions

To achieve the safety objectives of Article 1 the following fundamental safety function has to be ensured:

Confinement of radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases.

Article 4 – Graded approach

(1) A graded approach shall be applied to apply these safety requirements commensurate with the radiological hazard potential of the fusion facility.

2. REQUIREMENTS FOR SITING

Article 5 – Site evaluation

(1) A meteorological description of the region shall be developed, including descriptions of the basic meteorological parameters, regional orography and phenomena such as wind speed and direction, air temperature, precipitation, humidity, atmospheric stability parameters, and prolonged inversions.

(2) A programme for meteorological measurements shall be prepared and carried out at or near the site with the use of instrumentation capable of measuring and recording the main meteorological parameters at appropriate elevations and locations. Data from at least one full year shall be collected, together with any other relevant data that may be available from other sources. Relevant site-specific data and data available from other sources shall be collected.

(3) On the basis of the data obtained from the investigation of the region, the atmospheric dispersion of radioactive material released shall be assessed with the use of appropriate models. These models shall include all significant site specific and regional topographic features and characteristics of the installation that may affect atmospheric dispersion.

(4) A description of the surface hydrological characteristics of the region shall be developed, including descriptions of the main characteristics of water bodies, both natural and artificial and including tidal effects, the major structures for water control, the locations of water intake structures and information on water use in the region.

(5) A programme of investigation and measurements of the surface hydrology shall be carried out to determine to the extent necessary the dilution and dispersion characteristics for water bodies, the reconcentration ability of sediments and biota, and the determination of transfer mechanisms of radionuclides in the hydrosphere and of exposure pathways.

(6) An assessment of the potential impact of the contamination of surface water on the population shall be performed by using the collected data and information in a suitable model.

(7) A description of the groundwater hydrology of the region shall be developed, including descriptions of the main characteristics of the water bearing formations, their interaction with surface waters and data on the uses of groundwater in the region.

(8) A programme of hydrogeological investigations shall be carried out to permit the assessment of radionuclide movement in hydrogeological units. This programme shall include investigations of the migration and retention characteristics of the soils, the dilution and dispersion characteristics of the aquifers, and the physical and physicochemical properties of underground materials, mainly related to transfer mechanisms of radionuclides in groundwater and their exposure pathways.

(9) An assessment of the potential impact of the contamination of groundwater on the population shall be performed by using the data and information collected in a suitable model.

(10) The distribution of the population within the region shall be determined.

(11) Information on existing and projected population distributions in the region, including resident populations and to the extent possible transient populations, shall be collected and kept up to date over the lifetime of the installation.

The radius within which data are to be collected shall be chosen on the basis of national practices, with account taken of special situations. Special attention shall be paid to the population living in the immediate vicinity of the installation, to densely populated areas and population centres in the region, and to residential institutions such as schools, hospitals and prisons.

(12) Before commissioning of the fusion facility, the ambient radioactivity of the atmosphere, hydrosphere, lithosphere and biota in the region shall be assessed so as to be able to determine the effects of the nuclear reactor on radioactivity in the environment. The data obtained are intended for use as a baseline in future investigations.

(13) The total nuclear capacity to be installed on the site shall be determined as far as possible at the first stages of the siting process. If the installed nuclear capacity is significantly increased to a level greater than that previously determined to be acceptable, the suitability of the site shall be re-evaluated, as appropriate. For assessing the feasibility of the implementation of the emergency plans, all nuclear installations to be installed on the site shall be considered.

3. GENERAL DESIGN REQUIREMENTS

Article 6 – Defence in depth concept

(1) The design of a fusion facility shall apply the concept of defence in depth. The following levels of defence-in-depth shall be implemented to prevent accidents and to ensure that appropriate measures are taken to mitigate harmful consequences if prevention fails:

- Level 1: normal operation;
- Level 2: anticipated operational occurrences;
- Level 3: postulated single initiating events;
- Level 4: postulated multiple failure events;
- Level 5: off-site emergency response.

(2) The levels of defence-in-depth shall be independent as far as is practicable. A failure of on level of defence-in-depth shall not impair the effectiveness of consecutive higher levels.

(3) The design shall take due account of the fact that the existence of multiple levels of defence is not a basis for continued operation in the absence of one level of defence. All levels of defence-in-depth shall always be kept available and any relaxations shall be justified for specific modes of operation.

Article 7 – Concept of multi-level confinement of radioactive inventory

(1) The design of a fusion facility shall provide concentric barriers and additional retention functions to ensure confinement of radioactive materials.

(2) The barriers and retention functions shall altogether be designed in such a way and maintained in such a condition over the entire life time of the fusion facility that, in combination with the measures and equipment of the respective levels of defence, the respective safety-related acceptance targets and acceptance criteria as well as the radiological safety objectives are met on the different levels of defence for all events or plant states and the associated mechanical, thermal, chemical and radiation-induced impacts.

(3) The number of physical barriers is defined in Article 8 "Safety design targets".

Article 8 – Safety design targets

The following safety targets shall be considered during the design phase and ensured during the operational phase of a fusion facility:

1. As far as reasonably achievable, radioactive inventories, activation and contamination of SSCs shall be minimized. The amount of radioactive waste shall be minimized as far as reasonably achievable.

2. Significant radioactive inventory shall be confined by three physical barriers in operational phases and by two physical barriers during maintenance and repair.

3. Sequences bypassing the physical barriers and leading to early or large releases shall be practically eliminated.

4. Passive or active means to safely shutdown the plasma in operational states and accident conditions shall be provided.

4. SPECIFIC REQUIREMENTS FOR DESIGN

Article 9 – Control of tritium inventory

(1) The design of a fusion facility shall provide means to retain tritium in closed systems. Diffusion of tritium through barriers shall be reduced as far as possible.

(2) For operational states, acceptable leak rates shall be defined commensurate with radiological acceptance targets for workers. Operational systems to reduce the tritium concentration in the reactor building shall be provided.

(3) In case of accidental releases of tritium safety systems to reduce the tritium concentration shall be provided.

Article 10 – Design requirements for the breeding blanket

(1) Breeding blankets utilizing PbLi shall have a double wall piping. The anulus shall be evacuated and monitored to early detect leakages and to extract tritium diffused through the inner pipes. The design shall be such that contact between PbLi and water is reliably prevented.

(2) The inner surface of the PbLi shall have a protective layer to avoid corrosion. Low flow velocities shall be ensured by design.

(3) The materials for the pipes shall be selected to minimize permeation of tritium.

(4) The PbLi loops shall be equipped with possibilities for removal of radioactive, toxic, and volatile materials (e. g. ^{210}Po or ^{203}Hg).

(4) Heaters shall be provided to avoid uncontrolled freezing and clogging of the PbLi loops. Possibilities for emptying the PbLi loops shall be provided.

(5) Materials for manufacturing plasma facing components shall be selected

- to minimize dust generation;
- to withstand high temperatures;
- to minimize chemical reactions generating reaction products (e. g. hydrogen) challenging the confinement function.

Article 11 – Design requirements for the limiters

(1) Materials for manufacturing plasma facing components shall be selected having a low permeability for tritium.

(2) Cooling systems of the limiters shall be provided with means to extract tritium and activated corrosion products. Systems shall be designed to control water chemistry to minimize corrosion.

(3) The quality of coolant carrying pipes shall be such that the frequency of loss of coolant accidents is minimized. The design shall allow for inspection to early detect ageing, corrosion and wear out.

(4) The design shall include means to monitor the flow rate to early detect loss of flow accidents to initiated countermeasures and prevent damage to plasma facing components.

Article 11 – Design requirements for the vacuum vessel

(1) The vacuum vessel shall be designed, manufacture, constructed, located and operated such that the occurrence of rapidly propagating cracks and brittle fracture does not need to be postulated.

(2) Adequate safety factors, justified from a safety point of view, shall be added in the design to the determined values of impacts to ensure that the specified limit values for the loads on the vacuum vessel resulting from impacts under specified normal operating and accident conditions are not exceeded.

(3) For the vacuum vessel and walls of components of adjacent systems the following design requirements apply:

- use of high-quality materials, in particular regarding toughness and corrosion resistance;
- conservative limitation of stresses;
- prevention of stress peaks by optimised design and construction; and
- assurance of the application of optimised manufacturing and testing technologies.

This includes the knowledge and assessment of possibly existing defects.

A concept to maintain integrity of the vacuum vessel shall be implemented to assure and evaluate the requisite quality of these components in operation. For this purpose, additional measures and equipment for the monitoring of causes and effects of damage mechanisms, in particular of leakages during operation shall be specified and installed.

(4) Effective and reliable equipment shall be provided for pressure limitation and overpressure protection on levels of defence 1 to 4.

Article 12 – Design requirements for cooling systems

(1) The primary heat transfer shall be designed, manufactured, constructed and operated to meet high quality requirements to ensure integrity of the boundary as a barrier for the confinement of radioactive materials. The design basis of the primary heat transfer system shall be designed to withstand loads and impacts from operational states as well as accident conditions.

(2) To avoid an uncontrolled ingress of water and a consequential pressure increase in case of a loss of coolant accident, the design shall include safety valves to isolate the affected part of the primary heat transfer system to minimize water ingress.

(3) The materials shall be selected to have a low permeability for tritium to avoid diffusion of tritium from in-vessel structures and components into the coolant and to minimize permeation of tritium outside the vacuum vessel. Corrosion resistant materials shall be preferred to minimize generation of activated corrosion products in the coolant. The system shall be provided with means to control water chemistry and to remove tritium and activated corrosion products to minimize activity of the coolant.

(4) The design shall be such that the leak tightness can be regularly tested to ensure the integrity of the barrier.

(5) In case active cooling is required to prevent structural damage to in-vessel structures and components in case the primary heat transfer system is unavailable, a decay heat removal system shall be included in the design. The decay heat removal shall be independent as far as possible from the primary heat transfer system and shall have an independent ultimate heat sink. Active components shall be provided with emergency power.

(6) The primary heat transfer system and the decay heat removal system shall be protected against impacts from internal and external hazards in such a way that a simultaneous unavailability of both systems due to common mode / common cause failures need not to be postulated.

Article 13 – Design requirements for magnetic systems

(1) Superconducting magnets shall be equipped with a reliable and fast quench detection. Means to rapidly discharge the stored energy after quench detection shall be provided.

(2) Electromagnetic fields at the workplaces shall be identified and assessed. If compliance with exposure limit values according to Directive 2013-35/EU cannot be reliably determined on the basis of readily accessible information, the assessment of exposures shall be carried out on the basis of measurements and calculations.

In such case where action levels according to Directive 2013/35/EU are exceeded but exposure limit values are not exceeded technical and/or administrative measures shall be taken to prevent exposure exceeding the health effects exposure limit values and sensory effects exposure limits.

(…)

Article 14 – Design requirements for cryogenic systems

(1) The confinement of cryogenics has to be ensured by appropriate design and operation of the cryogenic systems to avoid an uncontrolled explusion of cyrogenic liquids.

(2) Measures for reducing the pressure built up within vessels in case of in-vessel leaks shall be provided. Vent lines and exhaust shall be located in areas where neither workers nor public will normally stay.

(3) Items important to safety shall be protected against impacts from expulsion of cryogenic liquids to ensure performing the intended safety function in case of demand.

(4) Working areas shall be equipped with oxygen monitors to alarm works on the risk of asphyxiation (decreasing oxygen concentration, in particular at elevated working areas) or increased risk of fire and explosion (increasing oxygen concentration due to condensation of gases) in case of expulsion of large amounts of cryogenic helium and decreasing.

Article 15 – Control of flammable gases

(1) SSC containing hydrogen (including deuterium and tritium) shall be inertised to prevent explosions.

(2) Areas, where release of hydrogen (including deuterium and tritium) cannot be practically eliminated, sensors shall be installed to monitor the hydrogen concentration in the atmosphere.

(3) In the event of an accidental release measures shall be provided that there will be no deflagration processes of flammable gases inside the building that will put integrity of the building at risk.

Article 16 – Design requirements for primary coolant systems

(1) Provision shall be made for controlling the inventory, temperature and pressure of the coolant to ensure that specified design limits are not exceeded in any operational state of the fusion power plant, with due account taken of volumetric changes and leakage.

(2) Adequate facilities shall be provided for the removal from the reactor coolant of radioactive substances from the reactor coolant, including activated corrosion products, permeated tritium, and non-radioactive substances. Adequate facilities to control the water chemistry shall be provided to facilitate a water chemistry programme.

5. REQUIREMENTS FOR COMMISSIONING AND OPERATION

Article 17 – Commissioning of magnetic systems

(…)

Article 18 – Chemistry programme

(1) A chemistry programme shall be implemented to provide the necessary support for chemistry and radiochemistry.

(2) The chemistry programme shall be developed prior to normal operation and shall be in place during the commissioning programme.

(3) The chemistry programme shall provide the necessary information and assistance for chemistry and radiochemistry for ensuring safe operation, long term integrity of structures, systems and components, and minimization of radiation levels.

(4) Chemistry surveillance shall be conducted at the plant to verify the effectiveness of chemistry control in plant systems and to verify that structures, systems and components important to safety are operated within the specified chemical limit values.

(5) The use of chemicals in the plant, including chemicals brought in by contractors, shall be kept under close control. The appropriate control measures shall be put in place to ensure that the use of chemical substances and reagents does not adversely affect equipment or lead to its degradation.

Article 19 – Maintenance and in-service inspections

(1) In case of performing maintenance, in-service inspections or repair in close vicinity of magnetic fields non-magnetic tools shall be used. Magnetic or magnetizable objects shall be secured to prevent uncontrolled attraction of those objects by the magnet to prevent damage to items important to safety.

(2) Loops with PbLi shall be emptied before executing maintenance or in-service inspections to avoid chemical reactions.

Article 20 – Ageing management

(…)

6. DECOMMISSIONING REQUIREMENTS

Article 21 – Consideration of decommissioning during the design phase

(1) During the design of a fusion facility the following shall be taken into account to facilitate the decommissioning at the end of the lifetime:

a) Shall minimize the number and size of contaminated areas to facilitate cleanup in the decommissioning stage;
b) Shall choose materials for containment that are resistant to all chemicals in use and that have sufficient wear resistance, to facilitate their decontamination at the end of their lifetime;
c) Shall avoid undesired accumulations of chemicals or radioactive material;
d) Shall allow for remote decontamination, as necessary;
e) Shall consider the amenability to processing, storage, transport and disposal of the waste to be generated in the decommissioning stage;
f) Shall ensure that major system components and potential points of contamination, particularly in the facility structure, are readily accessible to facilitate decommissioning.

Article 22 – Documentation of relevant information and knowledge management

(1) The operating organization shall implement a system to document all relevant information and knowledge important for the decommissioning stage, starting from the design phase. The operating organization shall be aware, over the operating lifetime of the plant, of the needs in relation to future decommissioning. Experience and knowledge with regard to contaminated or irradiated structures, systems and components gained in modification and maintenance activities at the plant shall be recorded and retained to facilitate the planning of decommissioning. Complete and reviewed information shall be compiled to be transferred to the organization responsible for managing the decommissioning phase

(2) A human resource programme shall be developed for ensuring that sufficient motivated and qualified personnel are available for the safe operation of the plant up to final shutdown, for conducting activities in a safe manner during the preparatory period for decommissioning and for safely carrying out the decommissioning of the plant.

Article 23 –Preparation and updating the decommissioning plan

(1) The operating organization shall prepare a decommissioning plan. The decommissioning plan shall be updated in accordance with changes in regulatory requirements, modifications to the plant, advances in technology, changes in the need for decommissioning activities and changes in national policies [5]. 9.2. A human resource programme shall be developed for ensuring that sufficient motivated and qualified personnel are available for the safe operation of the plant up to final shutdown, for conducting activities in a safe manner during the preparatory period for decommissioning and for safely carrying out the decommissioning of the plant.

7. WASTE MANAGEMENT

Article 24 – Minimization of radioactive waste

(1) During the design phase materials which cannot easily be activated by neutron induced nuclear reactions shall be preferred.

Article 25 – Waste processing facilities

(…)

Article 26 – Interim storage facilities

(1) The design shall provide interim storage facilities for a safe storage of radioactive waste for a period of at least (…) years.

(2) The storage shall provide adequate shielding of the radiation emitted from the waste.

(3) The interim storage shall be equipped with measure for access control.

(4) The storage facilities shall be equipped with radiation monitoring, in particular to detect possible leak of the storage casks.

8. REQUIREMENTS FOR SAFETY DEMONSTRATION

(…)

ANNEX B OVERVIEW ON FUSION SPECIFIC STRUCTURES, SYSTEMS AND COMPONENTS

The characteristics of fusion systems and their safety aspects are described in the following by using the DEMO concept as an example. The DEMO concept is based on the "tokamak" principle, i.e. a fusion reaction taking place in a hot plasma controlled by a magnetic confinement. It will bring fusion research to the threshold of a prototype fusion reactor beyond ITER. It addresses the technological questions of bringing fusion energy to the electricity grid:

- Full tritium self-sufficiency (Tritium Breeding Rate TBR > 1.0);
- Stability of plasma operation;
- Lifetime of materials.

Currently, different conceptual DEMO projects are under consideration by all of the member nations participating in ITER. The main parameters of the design are shown in Table 19. DEMOs from EU, Japan and Korea intend to approach the net electric power of ~500 MW. The CFETR proposes operating scenarios among A.1 for the fusion power of 120 MW, A.2 of 229-482 MW, A.3 of 974 MW and A.4 of 2192 MW at DEMO level /ZHUA 19/. A.3 and A.4 have the net electric power of 232 #MW and 738 MW respectively. In the USA, the Fusion Nuclear Science Facility (FNSF) is considered as the first significant fusion nuclear step beyond ITER which is used for the development and testing of fusion materials and components for a DEMO-type reactor /KESS 18/, /KESS 18a/. The US-III pathway in DEMO configuration is not yet available /ELGU 19/.

Table 19 DEMO projects worldwide and main parameters

DEMO	EU	Japan	China	Korea	U.S.
	EU DEMO	JA DEMO	CFETR phase II (A.4)	K-DEMO	FNSF
Aspect ratio	3.1	3.5	3.27	3.24	4.0
Major radius (m)	9.0	8.5	7.2	6.8	4.8
Minor radius (m)	2.9	2.42	2.2	2.1	1.2
Elongation	1.59	1.65	2.0	2.0	2.2
Triangularity	0.33		-	0.625	0.63
Magnetic field (T)	5.9	5.94	6.5	7.4	7.5
Plasma current (MA)	18	12.3	13.78	> 12	7.9
Fusion power (MW)	2,000	1,420	2,192	2,200 ~ 3,000	unknown
Net electric power (MW)	500	500	738	400 ~ 700	Not targeted
Reference	/FEDE 18//FEDE 18/	/TOBI 19/	/LI 19/,/ZHUA 19/	/KIM 15/	/KESS 18/

In the EU DEMO the plasma generates a fusion power of ~2 GW_{th} (500 MW_{el}) and is operated in long pulses (~2 h). The confinement of the plasma is achieved through a toroidal magnetic field of ~6 T, and the machine is operated adopting an edge localized H-mode scenario with high plasma edge density and radiation power. This confinement strategy ensures no or very-low Edge Localized Modes (ELMs) required to facilitate the plasma detachment, to limit the power to the divertor, and to reduce the wall damage. The baseline divertor operates in SN configuration. The EU DEMO plasma key features are well defined, but they have not been yet simultaneously demonstrated.

The research and development required to provide the basis for an electricity-generating Fusion Power Plant (FPP) are outlined in the European Fusion Roadmap /EURO 18/. The near-term approaches are:

- Construction of ITER (– 2025);
- Research & Development in support of ITER;
- D-T operation of JET;

- Conceptual Design phase of DEMO (2021 – 2027);
- Research & Development for DEMO;
- Construction of a fusion materials testing facility, IFMIF-DONES;
- Scientific and technological exploitation of the stellarator concept.

The road to EU DEMO construction is divided into three stages:

Stage 1	The pre-conceptual design (2014 – 2020) to seek out options that are developed, compared and assessed.
Stage 2	The conceptual design (2021 – 2027) to approach a single overall concept taken into the third stage.
Stage 3	The Engineering Design Activity leading to a construction decision (2030 – 2040).

The main systems of EU DEMO and options have been progressed in design since 2014. The current systems design is based on DEMO baseline 2017 (Table 19). The general layout in shown in Figure 3. Eight Key Design Integration Issues (KDIIs) have been addressed at Stage 1 /BACH 19/:

KDII#1	Feasibility of limiter concepts during plasma transients;
KDII#2	Integrated design and feasibility of breeding blanket and ancillary systems concepts;
KDII#3	Engineering and Integration design risks arising from advanced divertor configurations;
KDII#4	Design and feasibility of breeding blanket vertical segment-based architecture;
KDII#5	Design and feasibility of Power Conversion System Options, i.e. direct or indirect;
KDII#6	Design and feasibility of tokamak building concepts incl. ex-vessel maintenance;
KDII#7	Design and feasibility of a pumping concept based on tritium direct recirculation;
KDII#8	Development of a reliable plasma-operating scenario including supporting systems, H&CD, and plasma diagnostics/control systems.

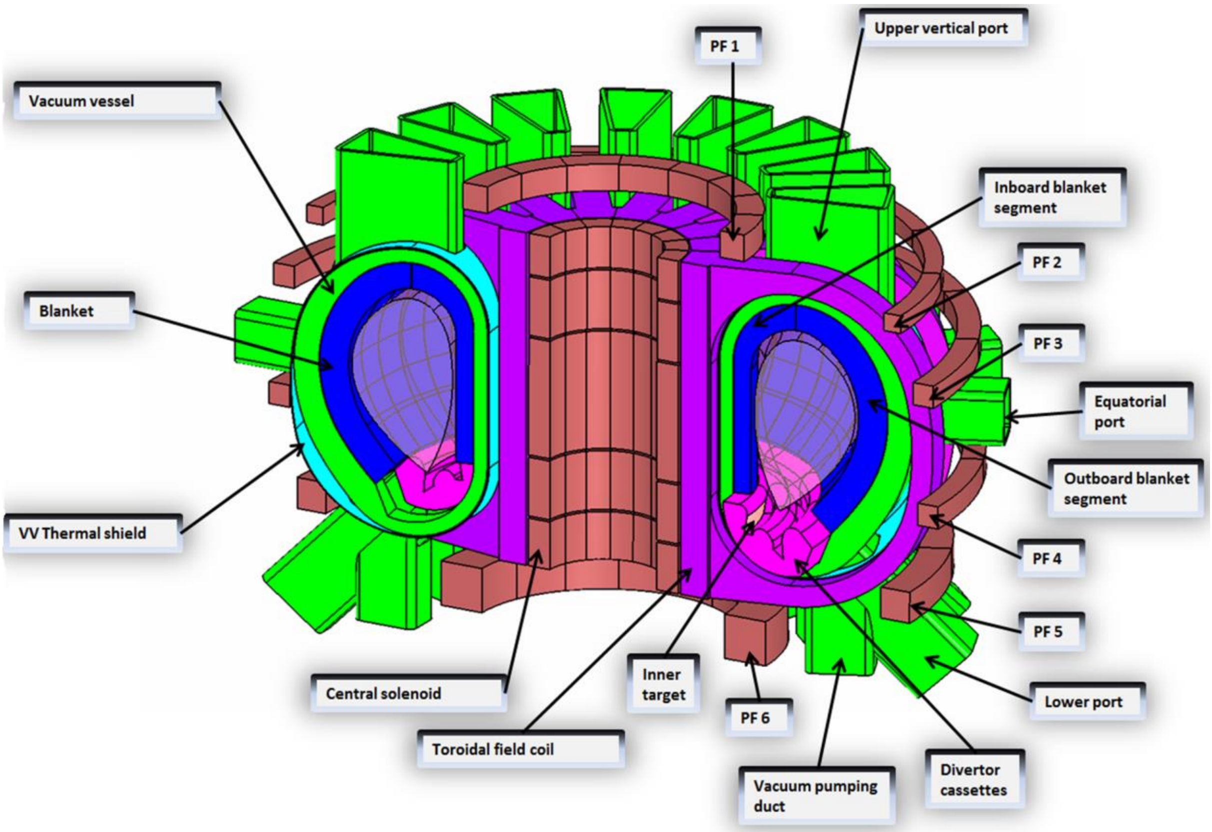

Figure 3 General layout of EU DEMO /TASSO 19/

The main systems at Stage 1 according to the Plant Breakdown Structure (PBS) are listed in the following. Before the description for the design status of these systems in the sub-sections, the basic plasma physics and fusion plant operational modes will be illustrated firstly.

- Magnet system
- Vacuum Vessel (VV)
- Breeding Blanket (BB)
- Divertor (DIV)
- Primary Heat Transfer System (PHTS)
- Balance of Plant (BoP)
- Cryoplant and Cryodistribution
- Tritium Extraction and Removal (TER) system
- Tritium, fuelling and vacuum system
- Cryostat system
- Thermal shields
- Heating and Current Drive (H&CD) system
- Plasma Diagnostic & Control system
- Remote Maintenance (RM) system
- Radwaste System
- Buildings (tokamak building, tritium building, Active Maintenance Facility (AMF))
- Plant Electrical Supply (PES)
- Plant Control System

B.1 Basic plasma physics

In a tokamak machine, the hot plasma needed to ignite the fusion reaction is enclosed in a magnetic field which is generated by superconducting coils and a current flowing in the plasma. The great challenge of each fusion machine is to control the plasma, which is more than 100 million degrees hot, and to preserve the components facing the plasma being constantly bombarded by heat and particles escaping from the plasma.

Up to now, the EU DEMO is, for example, designed to operate as pulsed machine concept based on modest extrapolations from the ITER physics and technology /FEDE 19b/, i.e. an inductive plasma operated in H-mode and affected by large ELMs /VINC 21/. The operational modes for a tokamak machine can be depicted from Figure 4. In first approximation, the pressure, the density, the temperature, etc. decrease in function of the radius of the plasma (the centreline is in the centre of the plasma, as in Figure 5). While heating up a plasma (through an injection of neutral particles, microwaves or through Ohmic heating) the plasma confinement time decreases, i.e. the time taken by the plasma to lose all of its energy if the fuel is not replenished and decreases /CEA 01/. Although, above an energy threshold – depending on the size of the tokamak and by the presence of some impurities – the plasma confinement time abruptly increases of about a factor two /WAGN 84/. These two operational modes are called L-mode (low containment time) and H-mode (high containment time), respectively. The H-mode is characterized by a "pedestal", i.e. a rapid decrease of the plasma properties at its boundary (see Figure 4). The drawback of this H-mode is the creation of ELMs, i.e. disturbances in the plasma releasing concentrated burst of energy on the tokamak surfaces that are also capable to disrupt the plasma. Additional modes for plasma operations can be reached controlling the plasma shape, or the current and the electric field profiles in the pulse through the use if Ion Cyclotron (IC) systems or through the injection of impurities into the plasma /CEA 01/. These modes – usually called "advanced tokamaks" – are:

- I-mode: an intermediate confinement mode presenting the most favourable elements of the L and H-modes /FEDE 19b/. In this mode a temperature pedestal is formed (as in H-mode), but not a density pedestal (as in L-mode) allowing: a) a control of the plasma density only through the divertor cryopumps, and b) a plasma operation ELM-free /WHYT 10/.
- QH-mode: a Quiescent H-mode with a wide pedestal and improved confinement, but without ELMs. This mode is reached by ramping down the injected neutral beam torque /CHEN 17/.
- Negative triangularity: a plasma having a mirrored shaped compared to that shown in Figure 5. The main advantages of this shape variation are: a better ELM behaviour and the increase of the target area on the divertor (with a constant heat flowing to the divertor, an increase of the target area leads to a decrease of the heat flux) /XUE 19/, /MEDV 15/.

None of these solutions are under consideration for EU DEMO, except for an alternative solution based on the physics extrapolated from the latter-stage ITER operational scenario /FEDE 19b/, /VINC 21/. This solution called flexi-DEMO (or DEMO2 depending on the source) would be a machine capable to move to steady-state operations.

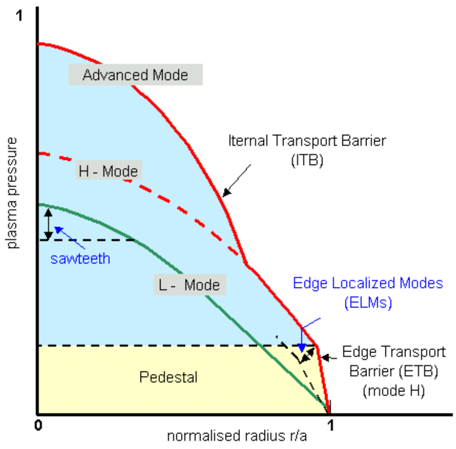

Figure 4 Operational modes of a tokamak machine /CEA 01/

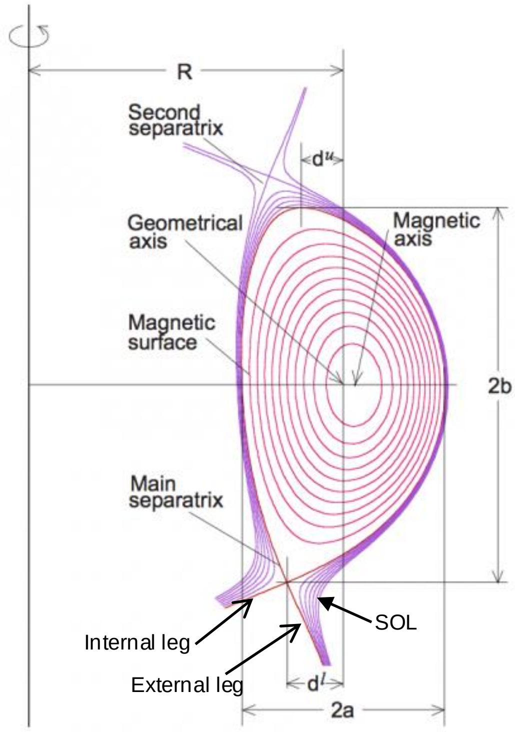

Figure 5 Magnetic configuration in a tokamak with a double null divertor /CIE 14/

B.2 Fusion plant operational modes

As stated in the previous section, all the DEMO concepts under development foreseen a pulsed machine. As for a Nuclear Power Plant (NPP), DEMO will undergo an initial commission phase followed by an operating phase /PINN 20/, /KONI 17/. Even if detailed commissioning plans are not available, /KONI 17/ describes a generalized model for the commissioning phase. This phase is likely to be characterized by Power Ascension Tests (PAT) performed employing only a Deuterium-Deuterium (DD) fusion reaction. This reaction can be considered as the equivalent of the criticality in a NPP because it operates the entire plant at a minimum power for a time sufficiently long to test the "flat-top" phase (see Figure 6). In this condition, the plant would be operated almost at nominal conditions: i.e. all the systems in nominal conditions, and plasma parameters similar to those of a DT plasma, except for the alpha and neutron heating. After the first short DD PAT, the following PAT will gradually increase the length of the pulses, and small amount of tritium will be added to the plasma.

Terminated the commissioning phase, the operating phase takes place. In EU DEMO, this phase is subdivided into three sub-phases:

- First operating phase;
- Blanket replacement phase;
- Second operating phase.

The first operating phase is planned to last 3 years with the 63% of this phase spent on plasma operation, and the remaining in maintenance periods /PINN 20/. The second operating phase should take between 3.9 and 5.5 fpy. The fundamental difference between the two operating phases is the blanket: in the first phase a blanket with a neutron damage capability below 20 dpa is used, while in the second phase the blanket should be capable of achieving 50-70 dpa /ARRO 15/.

During these operating phases, the pulses will be managed as shown in Figure 6. Each pulse is characterized by three phases /BUST 17/: the initial dwell phase, the flat top phase, and the final dwell phase. The initial dwell phase is in turn subdivided into two sub-phases: in the first one the Central Solenoid (CS) is charged and the required VV conditions are established (pump-down), while in the second one the plasma current is increased (ramp-up). The pump down phase takes approximatively 500 s and the CS recharge between 200 and 900 s. When the plasma current has reached the nominal value the "plasma current flat top" begins. Again, this phase is divided into two sub-phases: the heating phase to reach the conditions to ignite the fusion reaction, and the burning phase. The two phases take between 5-300 s and 1,800-30,000 s (target value of 7,200 s, i.e. 2 h of burning operation), respectively. When the pulse is over, a new dwell phase begins characterized by the ramp-down of the plasma. The ramp-down phase is not symmetric with the ramp-up phase, and it takes between 260 and 900 s. The operations made during the initial and the final dwell phases are complementary, so they can be performed in parallel. The aim in EU DEMO is to reduce the dwell time between two pulses below 600 s (10 min). Even if details for the other DEMO concepts have been not found, this description should hold also for them.

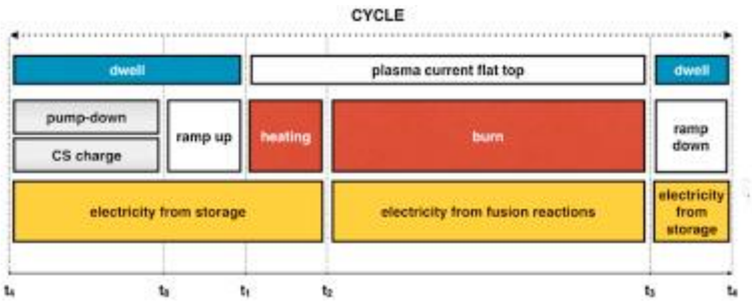

Figure 6 Events occurring during a pulse /BUST 17/

B.3 Magnet system

The magnet system aims to confine, shape and control the plasma inside the vacuum vessel (VV), for DEMO by using 16 superconducting Toroidal Field (TF) and 6 Poloidal Field (PF) coils, a central solenoid (CS), and a set of correction coils. The latest DEMO baseline 2018 has major and minor radius of 9.1 m and 2.9 m, plasma current 17.9 MA, toroidal field on the plasma axis 5.2 T, and the peak field in the TF conductor 12.0 T /SEDL 20a/.
The primary function of TFs is to confine the plasma particles. The TF coil assembly is the second backbone structure of the tokamak (the VV being the first) supporting PF coils, CS, VV thermal shield and cryostat thermal shield (outboard part). The four Low-Temperature-Superconductor (LTS) variants based on Nb_3Sn are feasible with available technology. They differ in three different aspects:

1) choice of wind-and-react or react-and-wind technique for the Winding Pack (WP) manufacture;
2) design decision concerns layer-winding or pancake-winding;
3) presence or absence of the radial plates.

The four TF WP options differ in these design choices. The fifth variant base on High Temperature Superconductors (HTS) is presently considered as prohibitively expensive for the huge TF fusion coils. However, it is worth to investigate the feasibility of the HTS TF coil. At high magnetic field it is an issue to be addressed that the coil mechanical design has to deal with increased electromagnetic (EM) loads.

The PF magnets pinch the plasma away from the walls and contribute in this way to maintaining the plasma's shape and stability. In the PF coils PF2 – PF5 the peak field is lower and NbTi superconductors can be used. However, PF1 and PF6 cannot be reasonably made of NbTi due to the peak field close to 7 T and requested design current sharing temperature (T_{CS}) exceeding 6 K (a temperature margin of 1.5 K is required on top of the inlet temperature of 4.5 K). They will have to be made of Nb_3Sn. Nb_3Sn would allow a substantial reduction of the coil size and a higher tolerance to AC loss. The AC in most fusion conductors drops significantly after EM cyclic loading. The AC loss is contributed originally from the bundle of superconducting strands (hysteresis and coupling loss) and copper-matrix stabilizer located around the cable /SEDL 20b/.

The CS is a coil positioned in the central hole of the torus. It is responsible for driving the plasma current. This current generates another magnetic field to pinch the plasma current towards the centre and so keep it away from the walls. Two variants of the CS coils have been proposed: a purely LTS one based on a Nb_3Sn pancake-wound design, and a hybrid coil (HTS-Nb_3Sn-NbTi) based on layer winding. In the latter one, the innermost layers made of HTS allow the designers either to increase the magnetic flux, and thus the duration of the fusion pulse, or to reduce the outer radius of the CS coil. The mechanical analysis performed on the CS coil addresses also the fatigue load, which turned out to be the main design driver for the CS coil. For the required and assumed operation limit to 20,000 plasma cycles, the design restriction due to mechanical fatigue translates into an allowable hoop stress of 300 Mpa if stainless steel 316LN is used in the jackets. Thus, the space allocation for the CS coil in the baseline 2018 seems not to be sufficient to reach the target magnetic flux of 250 Wb. Four approaches are proposed to solve this problem:

1) to increase the space allocated for the CS coil, leading to larger and more expensive tokamak;
2) Bucking of the CS coil by the TF coils. It prevents vertical coil segmentation into independently powered modules, thus seriously restricting flexibility of the plasma control;
3) A usage of high-strength composite materials, e.g. zylonepoxy composite, etc. the preliminary investigation shows that the potential benefit for the CS coils seems to be quite limited;
4) A double-wall conduit. The inner wall made of softer, fatigue resilient metal acts as a helium containment, while the stiff outer conduit provides the mechanical support. Even if the outer wall cracks, the coil integrity would be preserved.

Some of these four options will be further investigated. If no solution is found, the fusion pulse length in DEMO will have to be shortened.

All magnets are cooled by a forced flow of supercritical helium at ~4.2 K to maintain its superconducting properties. The helium removes the heat received by the coil from neutrons, thermal radiation of the VV and cryostat thermal shields, and conduction at thermal shield support structures or at the TF coil supports. The TF and PF coils are cooled and shielded from the heat generating neutrons of the fusion reaction (in-vessel shielding).

The magnet system is loaded by electromagnetic loads. It provides also support structures to thermal and structural loads. Three main key-components of the structural system are identified: the outer inter-coil structures, the inner inter-coil structures, and the gravity supports.

In DEMO magnet system, the promising technologies seem to be react-and-wind Nb_3Sn flat conductor and HTS conductors in the CS coil. There are challenges arising from the large size of the coils with respect to the manufacture, transportation, costs, and peak electric power needed to control the plasma or fast discharge of the coils in case of quench.

B.4 Vacuum vessel (VV) and VV pressure suppression system (VVPSS)

The vacuum vessel (VV) is one of the two backbone structures of the tokamak and provides a supporting structure both for the in-vessel components (IVCs) such as the breeding blanket (BB), the divertor (DIV), the limiters, Heating and Current Drive (H&CD) systems, diagnostics, fuelling system, and impurity seeding system. The VV consists of the main vessel (plasma chamber), upper ports, equatorial ports, and lower ports mainly /BACH 18/. Shielding plates in the space between the inner and outer shells, shield plugs, port closure plate, sector splice plates are also the components of the VV.

For DEMO, a water-cooled ITER-like vessel was chosen. The VV is a torus-shaped double-walled steel vessel with the design pressure of 200 kPa and the structure temperature of 40 °C. It provides the primary vacuum (~1.0E-5 Pa) and shields the magnet system from neutrons. The thermal loads are mainly caused by temperature gradients inside the VV structure: nuclear heating on inner side and water cooling between two shells. The nuclear heat loads on the vessel inner shell are about one order of magnitude higher compared to ITER. The structural integrity of the VV inboard wall was checked applying all provided rules in the relevant vessel design code RCC-MRx /MOZZ 20/. Both the primary (P) and secondary (S) type damages /DEM 16/ for the inboard wall in case of high nuclear heat load, VV coolant pressure and TF coils fast discharge radial pressure were evaluated.

The neutron irradiation limit for the VV structure is defined as 2.75 dpa.

The maximum acceptance leak rate of the VV (including ports but excluding attachments) is defined as 1.0E-7 Pa.m^3/s in ITER /PEAR 19/.

Openings, ports in the VV provide access for remote handling operations, diagnostics, heating, and vacuum systems.

The VV must support EM loads during transient events plasma disruptions, Vertical Displacement Events (VDEs), magnet current fast discharge, and withstand Design Basis Accident (DBA) to keep the confinement. Thus, the Vacuum Vessel Pressure Suppression System (VVPSS) is required to avoid overpressurization inside the VV.

The VVPSS protects the VV from overpressurization in case of in-vessel Loss of Coolant Accident (LOCA) /PORF 20/. The solution for the water-cooled blanket concept and for the helium cooled blanket concept has to be different. The Suppression Tank (ST) for the water-cooled blanket or the Expansion Volume (EV) for the helium-cooled blanket is used to expand the coolant and contain it. The bleed line connecting the VV to the EV / ST will be triggered at a sub-atmospheric pressure limit of 90 kPa; for large leak size the Rupture Disc (RD) connecting the VV to the EV will be triggered passively at 150 kPa /CARU 16/, /JIN 18/. The number of the RDs is defined by leak size.

For the water-cooled blanket, the size and the number of the STs are defined by the inventory lost. A preliminary sizing of the ST was proposed in /CARU 16/ and it is updated in /DONO 21/. Passive Autocatalytic Recombiners (PARs) installed in the atmosphere of the STs are being investigated to mitigate hydrogen explosion risk.

Wet EV including water for improved heat exchange and dry EV for effective pressure suppression with respect to helium inventories have been proposed in the on-going deterministic accident analysis.

Design of the VVPSS will be improved based on the results of accident analyses and solutions from the NPP.

B.5 Breeding blanket system and concepts

The Breeding Blanket (BB) has several functions such as tritium self-sufficiency, tritium removal, plasma-wall interaction, heat exhaust, nuclear shielding and access for auxiliary systems. It is a key IVC. Table 20 shows the current BB concepts for DEMOs. Based on different coolant technologies, water-cooled and helium-cooled blanket concepts are two options. They are described in Annexes B.5.1 and B.5.2 respectively.

In EU DEMO, the whole blanket system in all-round toroidal direction is divided in 16 sectors. Each sector includes 3 segments on outboard (OB) side and 2 segments on the inboard (IB) side in the toroidal direction. Each segment is designed as Single Module Segmentation (SMS), which is modularized in the poloidal direction. In total 48 OB segments and 32 IB segments are defined for the blanket system. Each segment is an assembly of several breeding modules individually supported by a manifold / backplate that supplies coolant to and removes tritium from the breeding modules.

Tungsten is the selected functional material placed on the plasma exposed surfaces as Plasma Facing Component (PFC) in the BB (first wall) and divertor (targets). It has to face high H/He plasma ions, high heat flux, high energy neutrons and energetic ions that escape from the plasma /GORL 19/. High-quality tungsten is required to retain sufficient bondage to the underlying materials, sufficient thermal and mechanical properties, and maintain fusion-specific interfacing performance such as plasma erosion, high heat flux stability. However, its reproducible supplier is missing. A reliable and reproducible high-quality tungsten which can form the baseline armour material for EU DEMO is being investigated within the EU-Japan Broader Approach.

The structural material of all BB components inside the VV of EU DEMO is the 9Cr-1WVTa Reduced Activation Ferritic Martensitic steel (RAFM) (EUROFER97) /GORL 19/. EUROFER97 is undergoing a codification process within the RCC-MRx nuclear code, to meet the requirements of the ITER test blanket modules. There is significant missing data on the materials performance of EUROFER97 to cover EU DEMO requirements. Some of the key failure mechanisms for the IVCs are anticipated after irradiation ageing and there is insufficient data to date on the neutron irradiation effects on EUROFER97, especially at higher doses, with correct fluence or under fusion neutron spectrum. There is limited data on the interaction of EUROFER97 with proposed coolants and breeder materials.

B.5.1 Water-cooled concepts

The salient characteristics of the Water Cooled Lithium Lead (WCLL) Breeding Blanket (BB) developed for the EU DEMO reactor concept (see Figure 7) are /DELN 19/, /MAIO 16/:

- Liquid lithium-lead (PbLi) at high ^{6}Li enrichment as breeder, neutron multiplier and tritium carrier;
- EUROFER as structural material;
- Water-cooled First Wall (FW) and breeder unit.

The design of the WCLL-BB consists in several identical segments repeated along the poloidal direction (Single Module Segment approach) (see Figure 7). Differences exist between the inboard and outboard segments because of the different loads they must withstand. Both the WCLL-BB inboard and the outboard segments can be divided into three zones (in radial direction):

- The FW;
- The Breeding Zone (BZ);
- The manifold region and the Back Supporting Structure (BSS).

The segments are attached to the VV through the BSS. Because of the volumetric power density deposited on the inboard segments, the water manifolds are integrated between the BSS and the PbLi manifolds. The water manifolds provide the coolant to the FW and the breeder units, while the PbLi manifolds the breeder medium. Two types of water manifolds are foreseen in the current concept: the high and low-pressure manifolds. This design ensures a symmetric thermal field in the BZ and the FW whilst maximizing the shielding performances of the BSS.

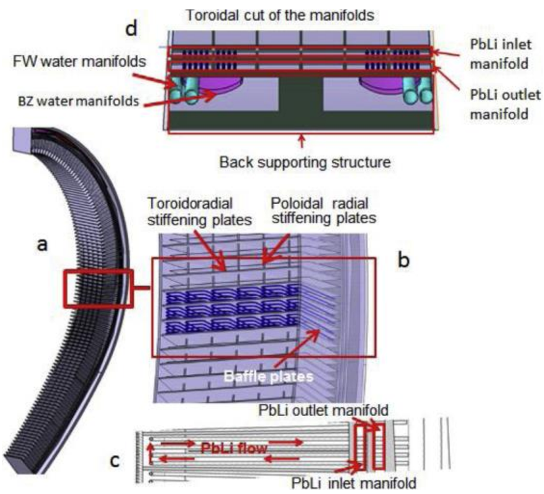

Figure 7 Reference WCLL design of the EU DEMO with SMS breeding blanket and breeding zone stiffening approach /FEDE 19a/

The PbLi manifolds are based on coaxial square structures: the outer EUROFER plates are in charge of the structural function, while the inlet structures are responsible for the separation of the inlet and outlet manifolds. The breeding zone is on front of the PbLi manifolds. This zone is composed by two main components: the stiffening plates; and the breeding cells. The stiffening plates are arranged in vertical and horizontal directions with the aim to reinforce the BZ to withstand thermo-mechanical and electro-magnetic loads. The breeding cells consist of Double-Wall Tubes (DWT) where the PbLi flows.

Table 20 Main specifications of the current breeding blanket concepts for DEMOs

DEMO	EU DEMO		JA DEMO	CFETR phase II		K-DEMO	FNSF
Blanket option	WCLL	HCPB	-	HCCB	WCCB HBRB SCWB	-	DCLL (HCLL, HCCB/PB)
Plasma-Facing Component	Tungsten		-	Tungsten		Tungsten	Tungsten
Interlaying material	-		-	-		Vanadium	-
Structural material	RAFM (EUROFER97)		RAFM (F82H)	RAFM (CLF-1/CLAM)		RAFM	RAFM / SIC FCI
Coolant	pressurized water 295 ~ 328 °C, 15.5 Mpa	Helium, 300 ~ 520 °C, 8 Mpa	pressurized water, 290 ~ 325 °C, 15.5 Mpa of heavy water	Helium, 300 ~ 500 °C, 8 Mpa	pressurized water, 285 ~ 325 °C, 15.5 Mpa (WCCB & SCWB) pressurized water, 280 ~500 °C, 25 Mpa (SCWB)	pressurized water, 290 ~ 330 °C, 15.5 Mpa	He, < 550 °C, 5 and 8 Mpa, PbLi
Breeder	PbLi	ACB pebbles (mixture of Li_4SiO_4 & Li_2TiO_3)	Li_2TiO_3 pebbles, mixture with NMM	Li_4SiO_4 pebbles	Li_2TiO_3 pebbles (WCCB & SCWB) Li_4SiO_4 (HBRB)	Li_4SiO_4 pebbles	PbLi (PbLi, ceramic)
Neutron Multiplier Material	PbLi	$Be_{12}Ti$ block	$Be_{12}Ti$ or $Be_{12}V$ or $Be_{13}Zr$ pebbles	Be pebbles	Be pebbles (SWCB) $Be_{12}Ti$ pebbles (WCCB) U-10Zr (HBRB)	$Be_{12}Ti$ pebbles	PbLi
Reference	/DELN 19/	/HERN 20/	/TOBI 19/	/LI 19/, /LING 20/		/KIM 15/	/KESS 18/

The FW protects the BZ from the heat loads coming from the plasma. The FW is made of Eurofer with a Tungsten layer on the plasma-facing side. Horizontal square channels (radial-poloidal direction) – created within the FW – provide the required water cooling. Design characteristic of the WCLL concept is the use of water at Pressurized Water Reactor conditions (Table 20).

A water-cooled BB concept is also under investigation for the JA DEMO /TOBI 19/, /SOME 19/. Several concepts have been proposed in the last 25 years, but the most relevant one is the Water-cooled Solid Breeder (WCSB, see Figure 8). This concept adopts water at ~15.5 Mpa and 290-330 °C as coolant, RAFM F82H steel as structural material, Li_2TiO_3 as tritium breeder, a tungsten layer to protect the FW, while different options are considered for the neutron multiplier: $Be_{12}Ti$, $Be_{12}V$ and $Be_{13}Zr$. Key aspect of this design is the use of mixed tritium breeder/neutron multiplier pebbles in the breeding region. Studies are on-going to change the coolant into heavy water to improve the tritium breeding ratio /SOME 19/.

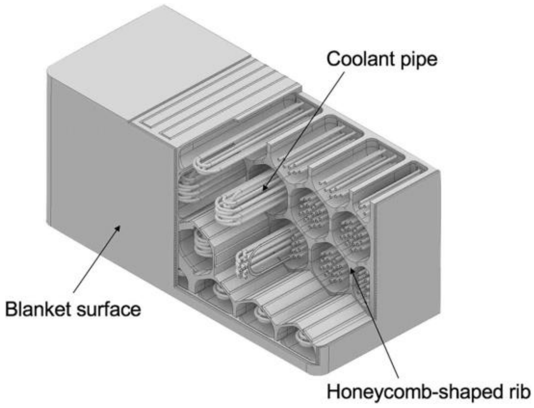

Figure 8 Reference WCSB design of JA-DEMO /TOBI 18/

As in the EU DEMO WCLL-BB, a segmentation of the blanket is also proposed for this WCSB concept /SOME 15/. These segments are with different modules along the poloidal direction. The design of the modules at different poloidal positions is slightly different in order to have a uniform temperature field in all the modules. Each module has a back structure to connect them with the VV. In front of the back structure the water manifolds are installed routing the coolant to the FW and the breeding region. The cooling pipes of both the FW and the breeding region are welded. The breeding region is composed by mixture of pebbles beds, cooling pipes, supporting plates and rib structures. The supporting plates allow the exact positioning of the coolant pipes, while the rib structures provide the required stiffness to the blanket. The FW in front of the breeding region is again made of RAFM F82H steel and protected by a tungsten layer on the plasma-facing side. The cooling of the FW is ensured through square channels excavated within the FW thickness.

Different water-cooled breeding blanket concepts are also under investigation for the Chinese CFETR. Up to now, the most developed concept is called WCCB (Water-Coolant Ceramic Breeder) /WANG 20a/, /LIU 17a/ and foresees the use water at ~15.5 Mpa and 290-355 °C, but for low-powered machines (up to 200 MW fusion power) the use of water in Boiling Water Reactor conditions is considered as well /LING 20/. Li_2TiO_3 and $Be_{12}Ti$ are used as breeder material and neutron multiplier, respectively, and the RAFM CLF-1 steel is chosen as structural material. The FW is also protected by a thin layer of tungsten. Again, a segmentation and a modularization of the design is adopted as in the WCSB. Key aspect of this design are:

- The use of stiffening grid plates to subdivide the breeder zones into sub-modules. Each sub-model is in turn divided into "mixed breeder zones" through the use of cooling plates. Cooling channels crosses both the cooling plates and the stiffening plates.
- Since CFETR is designed to investigate different fusion power levels (from 200 MW up to 1.5 GW), three independent cooling systems are proposed in the latest designs: Two circuits for the breeder region, and one circuit for the remaining components (FW, stiffeners, etc.).
- A minimized manifold area to enlarge as much as possible the breeder region.
- The presence of purge gas system to extract tritium from the breeder pebble bed. The channels for this system are integrated into the module' lateral walls and in the stiffening plates. The purge gas used is helium at low pressure and low velocity.

The remaining characteristics are similar to that of the WCLL-BB and the WCSB concepts.

Two other concepts are under investigation for CFETR phase II: the High-Breeding Ratio Blanket (HBRB) and the Super-critical Water-Cooled solid Breeder (SWCB) blanket concepts /LING 20/. Scarce data are available for HBHRB because it has been deeply modified in the latest years. The current design adopts water in typical PWR conditions as coolant, China Low Activation Martensitic (CLAM) steel as structural material, Li_4SiO_4 as breeder material, a U-10Zr alloy at low enrichment as neutron multiplier, and a tungsten layer for the protection of the FW. This blanket concept uses also two not-independent coolant systems: one for the FW and the breeding region, and one for the neutron multiplier region /TAO 19/. This choice is driven by the necessity to ensure maximum temperatures on the structural material (CLAM) below 823 K, and on the U-Zr region below 890 K to avoid phase change /LEI 17/.

In turn, the SWCB concept adopts water at 25 Mpa and 280-500 °C as coolant for both the FW and the breeding region, and CLAM steel as structural material /LING 20/, /CHEN 19/. Neutron multiplication is achieved by means of Be pebbles protected by two thin steel claddings, and Li_2TiO_3 is used as breeder material. The produced tritium is extracted through a low-pressure purge gas system using a mixture of He and H_2 as carrier gas. This purge gas system foresees the flowing of the carrier gas in upward direction to benefit from natural convection and reduce the pumps power consumption. The breeder zone is arranged as alternate layers: a breeding layer, and a Be layer. The cooling of the breeder zone is ensured through cooling channels crossing the different layers. The fundamentals of the SWCB concept have still to be proven because the preliminary safety analysis showed too high peak stresses and deformations.

As part of the Korean efforts on the K-DEMO reactor, a water-cooled breeding blanket concept has been developed /PARK 15/. The concept adopts a segmentation of the blanket into outboard and inboard sectors and a modularization of each segment. Water in typical PWR conditions is adopted as coolant, RAFM steel as structural material, Li_4SiO_4 as breeding material, and $Be_{12}Ti$ as neutron multiplier. The FW has a sandwich arrangement: on the plasma-facing side a thick tungsten layer, a vanadium interlayer, and the RAFM steel structural wall. Several shielding materials are investigated for the back supporting structure, such as B_4C layers, borated steel, and tungsten carbide.

B.5.2 Helium-cooled concept

The current Helium Cooled Pebble Bed concept (HCPB) for DEMO is completely different from the ITER HCPB TBM after several design iterations in the past years. It is based on an arrangement of the fuel-breeder pins architecture built in Single-Module Segments (SMSs) (see Figure 9). The pin consists in two concentric tubes, forming the pin inner and outer cladding /HERN 20/. The volume formed in the pins is filled with Advanced Ceramic Breeder (ACB) pebbles which is a ceramic breeder mixture of Li_4SiO_4 and 35 mol % Li_2TiO_3 (60% ^6Li). The pins are embedded in prismatic blocks of $Be_{12}Ti$ acting as Neutron Multiplier Material (NMM). He gas is used as coolant at the inlet pressure of 8 Mpa and a temperature window of 300 – 520 °C due to its neutronic and chemical inertness. A Purge Gas (PG) sweeps the NMM and ACB to remove the tritium produced in the functional materials. To enhance the heat transfer the FW channels are equipped with V-ribs or at least with a surface roughness of 40 50 µm. The FW is a U-shaped and actively cooled plate facing the plasma, which incorporates a 2 mm thick W armour as sacrificial layer against plasma erosion from fast particles. The BZ is formed by the volume between the FW and the backplate containing fuel-breeder pins. A Back Supporting Structure (BSS) on the rear side of each blanket segment sustains inertial and EM loads and gives structure support. The segments are closed with top and bottom caps.

The main loads acting on the DEMO HCPB BB are: inertial loads (dead weight, seismic loads), pressure loads (incl. LOCA accidents), thermal loads (FW heat flux loads, nuclear heating, power excursion, off normal transients, accidents), EM loads (during plasma pulse, off-normal), and pretension loads. Due to the different nature of the WCLL and the HCPB in progress, the associated system (BB-PHTS), auxiliaries (Coolant Purification System, TER, etc.), Balance of Plant will have different solutions.

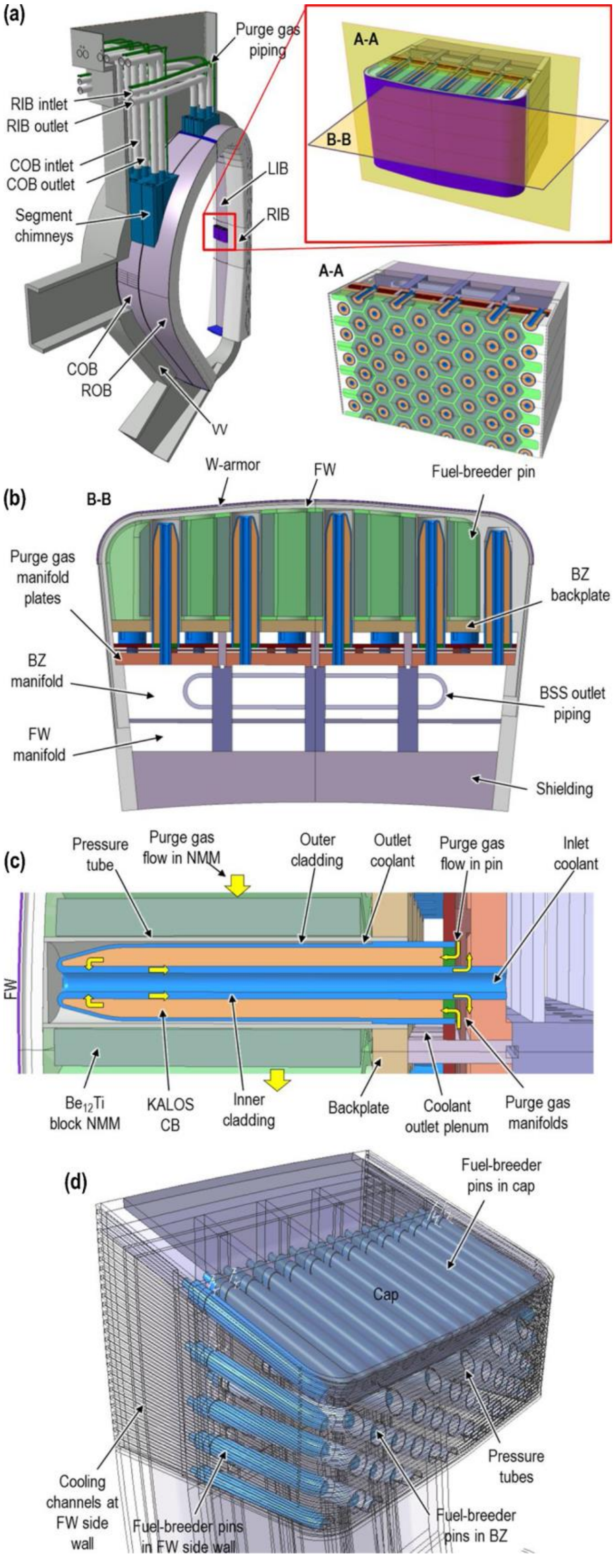

Figure 9 Reference HCPB design of the EU DEMO /HERN 20/

B.5.3 Limiter

The current BBs developed in the frame of EU DEMO adopt the FW as an integral part of their design. Although, studies are on-going to evaluate the possibility to have detached FW and limiters because the current FW solutions adopted are unable to withstand the heat fluxes reached during plasma displacement events (above 2 MW/m^2) /VIZV 20/. The rationale behind this choice is to simplify the substitution of the FW area/limiters suffering damages during these events.

Even if the path to pursue has not been yet decided, the design of the limiters is at a more advanced stages /VIZV 20/. Four types of limiters are under consideration to withstand four different plasma displacing events (see Figure 10):

- Outboard Mid-plane Limiters (OMLs) to protect against ramp-up events (5-10 MW/m² for 20-60 s). Two solutions are under investigation: four OMLs to protect the port plugs, or three OMLs to protect the equatorial ports.
- Upper Limiters (ULs) to withstand upward VDEs (~25 GW/m² for about 4 s). The solution under consideration sees the replacement of the central module of each outboard blanket segment with a UL. Being these limiters those facing the most demanding conditions, their design is mainly pushed to ensure the integrity of the heat sink since the tungsten monoblock would surely melt/evaporate (see Figure 11);
- Outboard Lower Limiters (OLLs) to face downward VDEs (~150 GW/m² for ~4 s). These limiters are placed below the OMLs;
- Inboard Mid-plane Limiters (IMLs) to withstand loss of confinement events (10-20 MW/m² for about 5 s). These limiters are the sole installed in the inboard segments and are usually placed in front of the equatorial ports to ease their maintenance/remote handling.

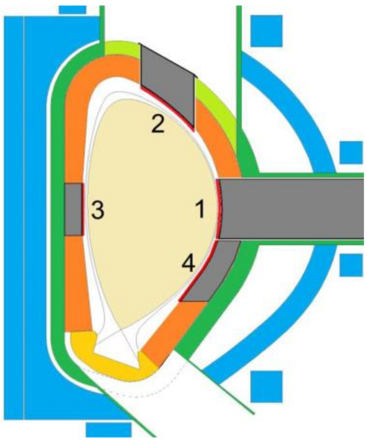

Figure 10 Positions of the limiters in EU DEMO. Position 1. Limiter(s) against Ramp-up events, Position 2. Limiter(s) against Upward VDE, Position 3. Limiter(s) against Downward VDE, and Position 4. Limiter(s) against H-L transition /VIZV 20/.

Discording data are available on the design of these limiters: /MAVI 21/ states that only the OMLs are actively cooled, and the remaining are only "energy dampers"; while /VIZV 20/ lists some common characteristics of these limiters, such as:

- The use of a thick tungsten monoblock as PFC;
- A CuCrZr heat sink;
- Water as coolant.

According to /VIZV 20/ the current solutions propose a sort of multi-layered structure: the CuCrZr pipe (heat sink), wrapped in an "thermal brake" pipe which is in turn inserted in a "filling material" pipe that is in physical contact with the tungsten monoblock. Investigations are also in progress to test different plasma facing materials (P91 steel, OFHC copper, and CuCrZr), thermal brake materials (EUROFER and P-91 steel), and different filling materials (Orobraze™ 950, Orobraze™ 1025, Pallabraze™ 950, NBLM™, H-Bronze™).

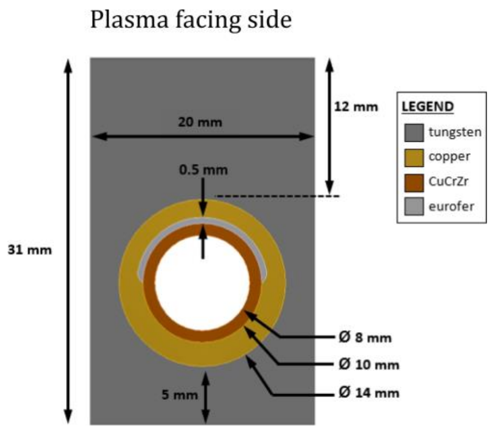

Figure 11 Example of an OML /VIZV 20/

As stated in /FRAN 20/, the OML and UL are connect to a Eurofer steel structure (port plug) that provides the required shielding to the port plug and the VV. The cooling concept foresees two different cooling circuit: one for the PFC (in this case a CuCrZr monoblock), and one for the shield block and the port plug. The operative conditions of both systems are those already proposed for the divertor to reduce the DEMO plant cost and to ease the plant layout.

No information is available about the possible use of limiters in the CFETR phase II, K-DEMO, and JA DEMO concepts.

B.6 Divertor

The divertor is a key IVC, as it is responsible for power exhaust, impurities and helium ash removal. Handling the high heat flux (HHF) on the divertor is one key issue of fusion research, for which more resistant materials are being developed, and to reduce the heat flux on the divertor target plate advanced magnetic configurations are being investigated. Table 21 shows the divertor configuration for DEMO investigated worldwide. Single Null (SN) is the baseline configuration in EU, Japan and China.

Table 21 Divertor concepts

DEMO DIV	EU EU DEMO	Japan JA DEMO	China CFETR phase II	Korea K-DEMO	U.S. FNSF
Baseline	SN	SND	SN	DN	DN
HHF (MW/m²)	10 – 20	10	20	10	10
Material	W / EUROFER / CuCrZr	RAFM / Cu-alloy pipes		RAFM / CuCrZr	W / RAFM
Option / advanced DIV	DN	SXD / SFD	x-DIV / SF	SXD / SFD	-
Reference	/FEDE 18/	/TOBI 19/	/LI 19/	/KIM 15/	/KESS 18/

The divertor configuration depends on the shape of the separatrix. The separatrix is the virtual boundary between the zone of the magnetic field having closed field lines and those with open field lines (see Figure 5). The area outside the separatrix is usually called Scrape-Off Layer (SOL). Therefore, the divertor is the component onto which the field lines terminate. For this reason, the divertor must face intense heat loads and the effects of the erosion due to the dust. Figure 12 shows the three possible divertor configurations: the SN configuration (left), the Double Null (DN) configuration (centre) and the island divertor (right) used in stellarator machines /CIE 14/. Additional divertor configurations are the x-divertor, the super-x-divertor, and the snowflake divertor /RYUT 15/. All these divertor configurations are achieved controlling the field lines in the divertor target area by using additional polodal field coils. In the x-divertor the SOL in the region of the legs (see Figure 5) is enlarged to spread the heat flowing to the divertor in a larger area. Instead, the idea behind the super-x divertor is to move the divertor to the largest radius allowed by the geometrical constraint of the magnets while keeping the main typical plasma geometry /KOTS 08/. The increase of the radius of the divertor would increase the target area for the external leg by a factor 2-3. Finally, the snowflake divertor is successive evolution of the super-x divertor /RYUT 15/. In this divertor the poloidal coils are used to create a null configuration of the second order, i.e. the legs would follow a r^2 law instead of linear law as shown in Figure 5, thus increasing the divertor target area (see Figure 13).

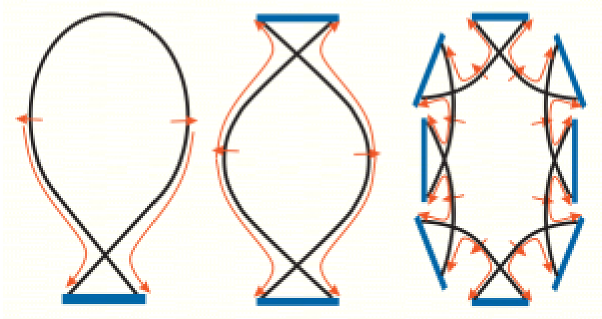

Figure 12 SN, DN and island divertor configurations /CIE 14/

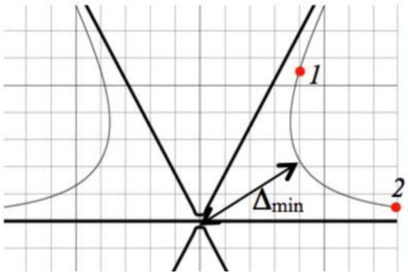

Figure 13 Snowflake divertor configuration /RYUT 15/

The SN divertor for EU DEMO consists of 48 cassettes due to 16 ports, and each port is segmented into 3 cassettes /MAZZ 20/. Each cassette includes one Cassette Body (CB) with reflector plates and three PFC layers, namely, the Inner Vertical Target (IVT), the Outer Vertical Target (OVT) and the shielding liner. The main function of the shielding liner is to provide neutron shielding for the VV and magnet coils. It is made of Eurofer with a first W-layer of 3 mm and cooled by the cassette cooling circuit. ITER-like target design is the baseline concept. Each cassette carries two vertical targets IVT and OVT. One OVT includes 43 Plasma Facing Units (PFUs) and one IVT includes 31 PFUs. The PFUs are made of W-monoblocks cooled by water flowing in CuCrZr tubes. A Cu interlayer of 1 mm is foreseen between W and CuCrZr. The W-monoblocks are brazed to support leg made of Eurofer. A pin connection link the support leg to a support plug welded in the divertor cassette. The main function of the reflector plate is to provide thermal, alpha particle and impurities shielding for the PFC cooling manifold / distributor. As the liner it is make of Eurofer with a first W-layer of 3 mm.

The lifetime of the PFCs is affected by possible material degradation, cracking, plastic fatigue, erosion of armour and corrosion of cooling pipe /YOU 18/. For the near-term water-cooled design four PFC target concepts have been investigated (Table 22). A local thermal break layer between the PFC and the cooling pipe is introduced to mitigate the local concentration of heat flux. This break is a thick Cu interlayer to reduce the thermal conductivity. The Wf-Cu composite pipe has high mechanical strength at elevated temperatures. A very thin Functional Graded Material (FGM) interlayer (20 – 25 mm) can reduce the risk of fast fracture in interlayer copper embrittled by neutron irradiation. The tungsten flat-tile particle-reinforced composite-block concept and helium-cooled divertor concept are considered for exploratory purposes only.

Table 22 Divertor target concepts of EU DEMO /YOU 18/

Target concept	Coolant	Interlayer	Heat sink
ITER-like (baseline)	water	Cu	CuCrZr
Thermal break	water	Oxygen-free Cu cast	CuCrZr
Wf-Cu composite pipe	water	Cu	Wwire/Cu
FGM	water	Cu / W	CuCrZr
Composite block	water	Cu	W-Cu
Helium-cooled	He	Cu	W-Cu

The EM loads during transient electromagnetic events (plasma disruptions and magnet current fast discharges) are the most important mechanical loads on the divertor components and in particular for the supports design. Pressure loads are the coolant pressure at operating conditions, baking and testing. Thermal loads induced by the heat flux from plasma and by the neutron heat generation are a strong design driver.

The challenge of reducing the heat load on the divertor targets has been addressed through the investigation of divertor configurations alternative to the standard SN, such as the Snowflake (SF), DN, X and Super-X (SX) divertors /MARZ 20/. DN divertor uses two divertors, at the top and bottom of the vacuum enclosure. The main advantage of the split divertor is that the outboard legs are relatively small components, which can be maintained more easily than full cassette-like divertors, and are therefore more suited to the more frequent replacement required of a divertor in general. The DN requires a different Remote Handling (RH) scheme for blanket than the SN reference baseline (removal through the equatorial port instead of upper port).

Snowflake divertor (SFD) flares the hot plasma at the divertor surface and reduces the residual heat flux per wall area. The technique has the potential to create heat load that can be tolerated by existing materials.

Super-X divertor (SXD) is a divertor design in which the power per unit area striking material surfaces is reduced greatly. It requires a set of divertor coils that extends and controls a long plume of exhaust plasma. The length of the plume allows high radiative cooling before the plasma reaches the target.

Implementing SXD and SFD would be difficult at the present stage due to the limitation of superconducting magnets.

The overall configuration of the K-DEMO divertor system based on the ITER-like water-cooled tungsten technology is a double-null type symmetric divertor subdivided into 32 toroidal modules for the vertical maintenance /KWON 20/. RAFM steel and CuCrZr are major candidates of heat sink materials in the divertor.

B.7 Primary Heat Transfer System (PHTS)

PHTSs are designed to extract thermal power from the IVCs (BB, divertor) and the VV structures during the plasma operation and transfer it to the secondary BoP side /BARU 18/, /BARU 21/. Water is the coolant for the PHTS of the WCLL, VV and divertor, while helium for the PHTS of the HCPB.

B.7.1 BB-PHTS

Each cooling loop of the BB-PHTS includes main components such as pump/circulator, Intermediate Heat Exchanger (IHX) or Steam Generator (SG) due to indirect or direct coupling option, pressurizer (PRZ) for water, inlet/outlet headers for coolant distribution or collection, and various valves (isolation, control), etc. Due to the pulsed operation of DEMO, an Intermediate Heat Transfer System (IHTS) equipped with an Energy Storage System (ESS) to be placed between the BB-PHTS and the Power Conversion System (PCS) is introduced for a continuous power conversion and electricity production. The IHX of the indirect option or the SG of the direct option transfers heat to the IHTS or the PCS respectively.

B.7.2 WCLL BB-PHTS

For the WCLL the Direct Coupling Design (DCD) is suggested to be the reference for the next DEMO conceptual phase, where the BZ and FW Once Through Steam Generators (OTSGs) are directly connected with the steam turbine of the PCS /BARU 21/. The PHTS is designed in two independent primary systems: the BZ PHTS and the FW PHTS. Each cooling loop delivers power to the PCS by means of one of four OTSGs. The reference thermodynamic cycle is based on pressurized water at 15.5 Mpa and inlet/outlet temperatures of 295 °C and 328 °C respectively. Each PHTS is designed in two cooling loops. Each cooling loop serves eight BB sectors. Each sector contains 3 OB segments and 2 IB segments. Two Main Coolant Pumps (MCPs), an OTSG, and the connecting piping between components are contained in each BZ cooling loop. One PRZ is shared in two loops of the BZ PHTS. One MCP, an OTSG, and the connecting piping between components are contained in each FW cooling loop. One PRZ is shared in two loops of the FW PHTS. The PHTS has interface to the auxiliary systems such as the Coolant Purification System (CPS), Chemical and Volume Control System (CVCS), draining and refilling system, and drying system. Functional feasibility of the SG and steam turbine is a main issue of this design.

Alternatives to the above DCD is the DCD no storage and ICD (Indirect Coupling Design) at low load. The DCD no storage minimizes the energy storage while ensuring a safe operation of the steam turbine in dwell. The ICD adopts a small energy storage to operate the steam turbine in dwell at low load.

B.7.3 HCPB BB-PHTS

For the HCPB the ICD is suggested to be the reference for the next DEMO conceptual phase. The HCPB BB-PHTS is designed in eight separate cooling loops. Each cooling loop serves 2 BB sectors. Each sector contains 3 OB segments and 2 IB segments. Each loop requires two circulators, one IHX and piping between components. The IHX uses Shell&Tube type and molten salt on the secondary side. Piping is equipped with thermal insulation. In operation, helium coolant is at 8 Mpa and inlet/outlet temperatures of 300 °C and 520 °C respectively. The PHTS has interface to the auxiliary systems such as the CPS, helium supply and storage, pressure control system; detection of helium leakages; pressure suppression system, evacuation of the gaseous contaminant (e.g. nitrogen) before the start-up; cooling of the main active components (e.g. circulators). Commercial unavailability of circulators is a main issue in this design.

Alternative to the ICD is the DCD with small ESS plus electrical heater to maximize electrical power production during pulse and to maintain the electrical generator synchronized to the grid during dwell period, operating the steam turbine at a minimum operational load of 10%.

B.7.4 VV-PHTS

The VV-PHTS for the WCLL DCD BoP or the HCPB ICD BoP is segmented into 2 separate cooling circuits, one serves eight even sectors, and the other feeds the odd ones.

For the WCLL, the VV-PHTS is equipped with the Decay Heat Removal System (DHRS). Each cooling circuit contains a main Heat Exchanger (HX), a PRZ, a MCP, a DHR HX, a DHR pump, and connecting pipes between components. The VV-PHTS is integrated in the feedwater train of pre-heaters. The DHR HXs transfer the extracted decay heat to the Chilled Water System (CHWS). The HX uses the shell&tube technology.

For the HCPB, each cooling circuit contains a main HX, a PRZ, a MCP, and connecting pipes between components. The main HX is integrated as pre-heater of the feedwater. The DHRS is annexed to the VV-PHTS to transfer the decay heat removed from all IVCs to the CHWS during emergency conditions. It is present in each VV cooling loop and consists of a HX, a pump and connecting pipes. The HX transfers the extracted decay heat to the CHWS. The HX uses the shell&tube technology.

The VV-PHTS has interface to the auxiliary systems such as the CPS, the CVCS and Component Cooling Water System (CCWS, e.g. MCP).

B.7.5 DIV-PHTS

Two DIV-PHTSs are designed: DIV-PFU PHTS and DIV-CAS PHTS.
The DIV-PFU PHTS / DIV-CAS PHTS for the WCLL DCD BoP is divided in two independent loops. Each cooling loop contains a HX, a MCP, a PRZ and the connecting piping between the components. The HX is integrated in the feedwater train of pre-heaters and it uses the shell&tube technology. Each cooling loop feeds the CAS / PFUs of eight tokamak sectors.

The DIV-PFU / DIV-CAS PHTS for the HCPB ICD BoP is segmented into 2 separate cooling circuits. Each cooling loop contains a HX, a MCP, a PRZ and the connecting piping between the components. Each cooling circuit provides pressurized water in forced convection to the DIV cassette bodies / PFCs of eight tokamak sectors, and heat removed from the PFCs is transferred to the PCS via a HX. The HX uses the shell&tube technology.

The DIV-PHTS has interface to the auxiliary systems such as the CPS, the CVCS and CCWS (MCP).

B.7.6 Auxiliary systems

Auxiliary systems such as CVCS, CHWS and CCWS, which have application in ITER, are envisaged also in DEMO plant. The CVCS is conceived for the water-cooled PHTSs and BoP. The main functions of the CVCS are to maintain the PHTS coolant purity and activity level within acceptable limits, to regulate the coolant inventory, to control the water chemistry within the required pH limits and the concentration of dissolved oxygen to the minimum. Two main functions of the CHWS are identified in ITER: to remove heat from safety-important components including heat removal from the VV during Loss of Offsite Power (LOOP) events using dedicated chiller package units that reject this heat directly to the atmosphere, and to provide normal cooling to building by Heating, Ventilation Air Condition (HVAC) units. The CCWS will manage the rejected heat.

B.8 Balance of Plant (BoP)

The DEMO BoP is a chain of systems devoted to the extraction of the plasma thermal power and to its conversion in electrical power. At the present stage, specific efforts and proper solutions are being investigated to cope with the pulsed operation, and two separate BoP options due to the WCLL and the HCPB blanket concepts are conceived /BARU 21/. This boundary condition due to the time profile of the plasma heat sources challenges the feasibility and the operations of the BoP. The BoP design must accommodate the loads during very frequent transients to ensure safe and reliable operation. The PCS provides pressurized water/steam to the steam turbine to convert the thermal power extracted from DEMO into mechanical, and then into electrical power through the synchronous alternator. The PCS foresees all equipment presented in a commercial power plant, in particular: SGs, reheaters – in between high and low pressure turbine stages, feedwater pre-heaters, deaerator, condenser and feedwater & condensate pumps. The BoP components are located in the turbine electrical generator hall.

The heat from the DIV- and VV-PHTSs is used to preheat the PCS feedwater by means of the integration of DIV and VV heat exchangers in the feedwater train of heaters in order to improve the overall plant efficiency.

B.8.1 WCLL BoP

Three main variants have been conceived. The main requirements for the variant design are to avoid disconnection from the grid for each pulse/dwell phase and to limit the impact of frequent temperature transients to structures, taking into account the feasibility of the solutions proposed, performance, safety and cost aspects at the same time.

The **WCLL ICD BoP** uses an IHTS+ESS operated with molten salt (HITEC) to decouple the regular operational transients of the tokamak from the PCS operation. The energy from the BZ is delivered to the PCS, while the power of the FW PHTS is delivered to the IHTS, then to the PCS. The power from BZ and FW is used to produce the main steam at condition suitable to feed steam turbine. The ESS is designed to deliver 100% of the nominal power during the dwell time. This option delivers continuous and nearly constant electrical power to the grid in both pulse and dwell time. However for this purpose very large dimensions of energy storage tanks are needed (~11,000 m^3 each).

The **WCLL DCD BoP** is based on the direct cycle that the BZ and FW OTSGs are directly connected to the steam turbine of the PCS (see Figure 14). A small ESS with two tanks (1,500 m^3 each) is adopted to operate with molten salt (HITEC) and is heated with electrical heaters. It can feed the steam turbine during the dwell with ~10% of the nominal steam flow rate, maintaining the connection with the electrical grid to have a minimum production of electric power which is enough for the PHTSs and BoP auxiliaries. The small ESS loop consists of molten salt pumps and tanks, electrical heaters, steam generator and the piping between the components. The effectiveness of this option has been studied with detailed transient analysis and stress assessment that it is the reference variant for the next conceptual design phase.

The **WCLL DCD BoP with auxiliary boiler** is the second DCD solution, in which the BZ and FW PHTSs are directly coupled with the PCS through two OTSGs. The gas-fired boiler connects the turbine directly to keep the minimum steam flow rate at 10% of the nominal value, and the turbine works at 10% of nominal power during dwell time. Large power of the auxiliary boiler is required to have an external source for operating the BoP at minimum load in dwell time.

Further option such as **WCLL DCD NO STORAGE** will be investigated in the next conceptual design phase. It adopts basically the same architecture of DCD with small ESS, minimizing the energy storage while ensuring a safe operation of the steam turbine in dwell time.

An additional **WCLL ICD BoP variant** with small storage system is to introduce an intermediate loop with a small ESS to limit the SG units so that to limit related regulation and stability issues; the ESS would be much smaller than the one in the WCLL ICD BoP. and the steam turbine can work at low load during dwell time.

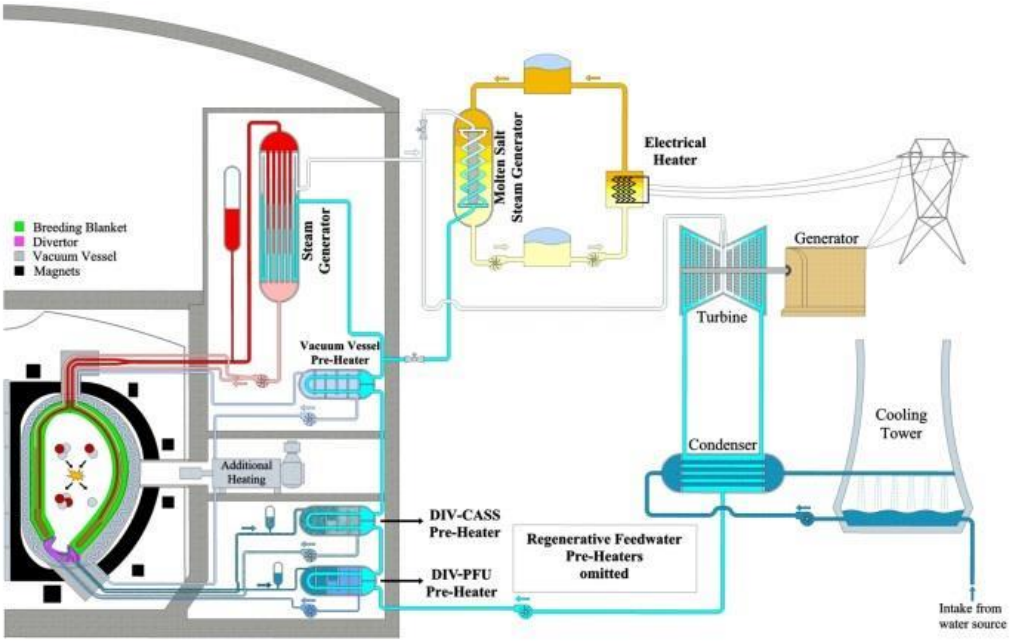

Figure 14 Overview of the WCLL DCD BoP /BARU 21/

B.8.2 HCPB BoP

The **HCPB ICD BoP** uses an IHTS equipped with an ESS operating with molten salt (HITEC) to decouple regular plasma strokes from the PCS (see Figure 15). It is the reference variant for the next conceptual design phase. The IHTS design uses qualified technology coming from concentrating solar power plants.

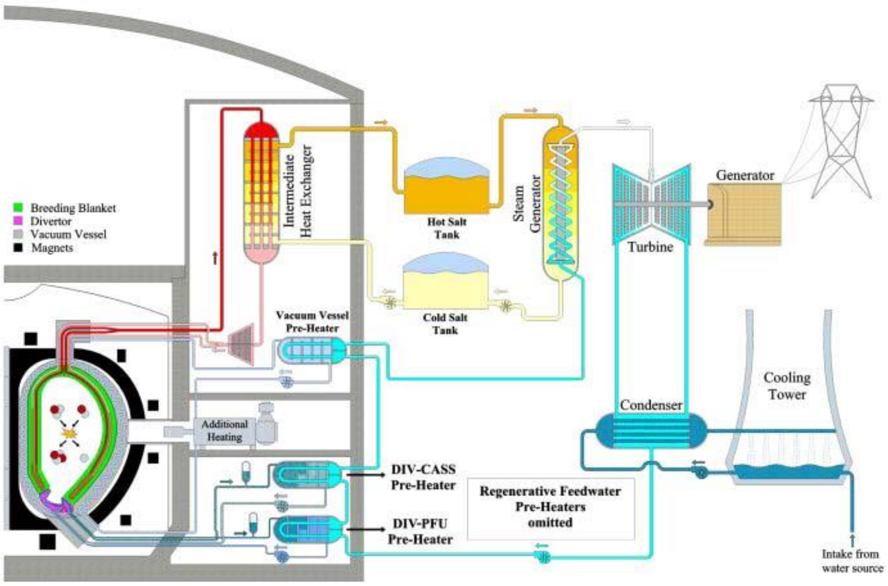

Figure 15 Overview of the HCPB ICD BoP /BARU 21/

The **HCPB DCD BoP variant 1** uses a gas-fired boiler to provide steam flow to keep the power train in operation. The main challenge is to cope with the fast power transient while the turbine works in a safe operational state. Also to keep the required power of the auxiliary boiler during dwell time an additional auxiliary heater section is needed. It is comparable to a small gas-fired power station with a large gas pipeline. Hereby temperature and pressure transients during pulsed operation are difficult to manage. This option is considered as a potential back-up solution due to the costs, size and heat transfer constraints.

The **HCPB DCD BoP variant 2** collects fusion energy during pulse and stores it in a solid state ESS. The collected thermal energy is then released to the PCS during the dwell time. This reduces boiler size, however, the ESS realized as high-temperature concrete is not able to release the energy within the short dwell time. This option is considered as a remote back-up solution due to safety function of the PHTS with respect to the spatial request of the ESS and the complicated piping and control system.

The **HCPB DCD BoP variant 3** uses HITEC molten salt and an electrical heater in order to maximize electrical power production of the PCS during pulse and to maintain synchronized the electrical generator to the grid in dwell time during the operation of steam turbine at a minimum operational load of 10%. This option appears to be the one with the lowest integration and feasibility risks, allowing to adopt control strategies that might minimize the impact of thermal-hydraulic transients on main equipment. Further studies focussed on creep assessment and start-up evaluation are needed to confirm that this solution can be considered as the first back-up choice.

The PCS has been optimized base on the different variants and available energy sources. The detailed design proposed by an industrial partner gave a breakthrough as a consequence of the optimization of the turbine-feedwater train. The gross output of the turbogenerator during dwell time is even higher than that available during the pulse period due to the reduced BB-PHTS circulation power need.

B.9 Cryostat system

The cryostat is a structure encapsulating the VV and the superconducting magnets. It has two aims: a) support the VV, the TF coil system and the cryostat thermal shield; and b) provide the vacuum and cryogenic conditions required to operate the magnets. It is itself supported by the tokamak building. In the EU DEMO, the cryostat is made of AISI 304 steel and operates at about 10^{-4} Pa and 200 K /CIUP 20/. Design limits for the internal and external pressure are 2 bar and 1 bar, respectively. The cryostat is designed according to ASME VIII, Div. 2.

On the contrary of the ITER cryostat, the DEMO one is design as a not-self-supported structure to reduce the amount of steel required for its construction. The structural support is instead provided by the bioshield enclosing the reactor. This design allows also the use of flat lid (instead of a dome one as in ITER), thus reducing the height of the cryostat (about 6 m) and the reactor building (see Figure 16).

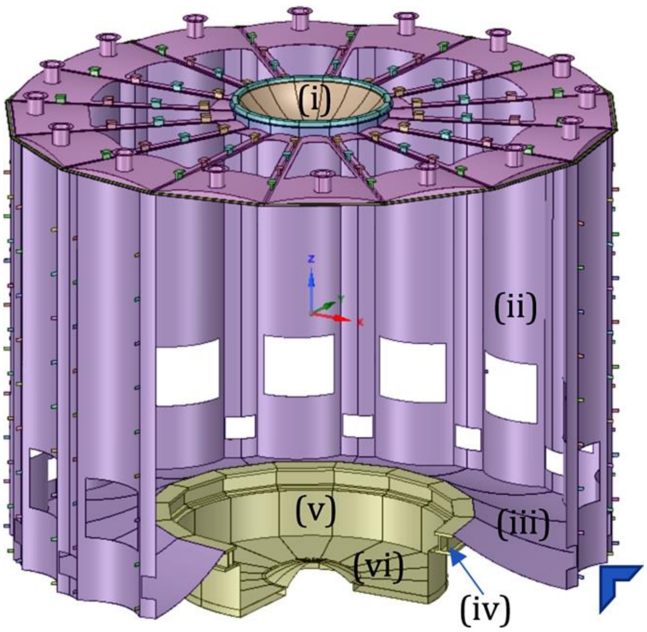

Figure 16 Schematic of the EU DEMO cryostat – (i) top lid & central lid, (ii) main cylinder, (iii) skirt segments, (iv) pedestal ring, (v) basement cylinder and (vi) basement plate /CIUP 20/

The actual cryostat design adopts a cylindrical structure with flat top lid. The cylindrical part is divided into 16 segments each connected with the bioshield. Each segment is made of two tube segments: a large tube segment, and a small tube segment. The large tube segments are aligned with the lower parts, while the small tube segments with the lower port pipe annex. The top lid is divided into two parts: the central lid and the radial beams. The central lid is designed as in ITER. The radial beams are connected to the bioshield roof and are made of curved steel plates. The segmentation of the radial beams is required to allow the access to the upper ports. The cryostat is closed at the bottom by the pedestal ring and the skirt. The pedestal ring is segmented in 16 straight sectors and supports both the magnets and the vacuum vessel. The skirt connects the pedestal ring with the cylindrical part of the cryostat. Again, the skirt is made of 16 welded steel plates having a single curvature. The skirt is welded to both the cylindrical part and the basement cylinder. The basement cylinder is not supported by the thermal shield but it is supported by the basement plate. The basement plate is made like in ITER.

In turn, two cryostat designs were under development for the CFETR phase II reactor in 2015 /WANG 15/, but only one looked in actual development in 2017 /WAN 17/. AISI 304L steel is adopted as structural material, and it is designed to operate at 10^{-4} Pa and at atmospheric temperature with a limit on the leak rate of 10^{-4} Pa m^3/s. The ASME Section VIII Division-2 was selected as the reference code for design. Differently from the EU DEMO, the CFETR cryostat is intended to be the second barrier for the radioactive inventory. The proposed CFETR cryostat layout is similar to ITER one. The cylindrical zone is subdivided into lower and upper cylinder. These two parts are welded together and reinforced through the use of vertical and toroidal stiffening ribs (welded to cylinder). The top lid adopts a dome-like shape and it is bolted to the upper cylinder. The bottom head is welded to the lower cylinder. The support plugs connect the bottom head with the concrete base to react to seismic events and to allow the cryostat radial movement. These support plugs are welded to the bottom head. 16 tokamak support pillars are also connected to the cryostat bottom head. The numerous openings on the cryostat and the harsh operative conditions pushed a study on the adoption of metallic seals instead of rubber ones /WANG 15/.

Detailed studies on the cryostat proposed for K-DEMO are not available. /UTOH 13/ briefly mentions the cryostat while discussing the loads on the TF coils. The document states that the structural strength of the cryostat is ensured by the reinforcement of the wall and the flanges of the maintenance ports, but the feasibility of this option has still to be proven. From this scarce information it is difficult to state the characteristics of the cryostat in K-DEMO, but probably their concept is more similar to the ITER and CFETR ones than the EU DEMO one.

No data have been found regarding the cryostat of JA DEMO.

B.10 Cryoplant and Cryodistribution

The Cryoplant and Cryodistribution are in charge of producing and providing the liquid helium required to cool down the superconductive magnets and the thermal shields. This system also provides helium to the cryopumps installed in the divertor region to keep the VV in vacuum conditions. EU DEMO needs a vast cryoplant devoted to the production and distribution of the cryogenic fluids, but its design is not yet available. As a reference, ITER needs about 25 tons of He at 4 K during operation. For CFETR a conceptual design exists but only for phase I, not for phase II. No relevant data have been found for K-DEMO and JA DEMO.

The ITER cryoplant has been designed to provide a cooling power of 75 kW at 4.5 K through three liquid helium (Lhe) plants (see Figure 17) /MONN 15/. An additional loop provides the helium cooling to the thermal shields /CHAN 17/. These cooling loops are supported by two liquid nitrogen refrigerators connected to the ITER nitrogen production facility /MONN 15/. An additional tank is also installed in parallel to the Lhe plants to sustain the cooling of the magnets during the high heat load phases/CHAN 17/. Helium is used in a closed loop; hence a purification system is in place. Both the storage and the recovery of helium is managed via cold and warm tanks (helium inventory of 27 ton). The tanks adopted in ITER are summarized in Table 23.

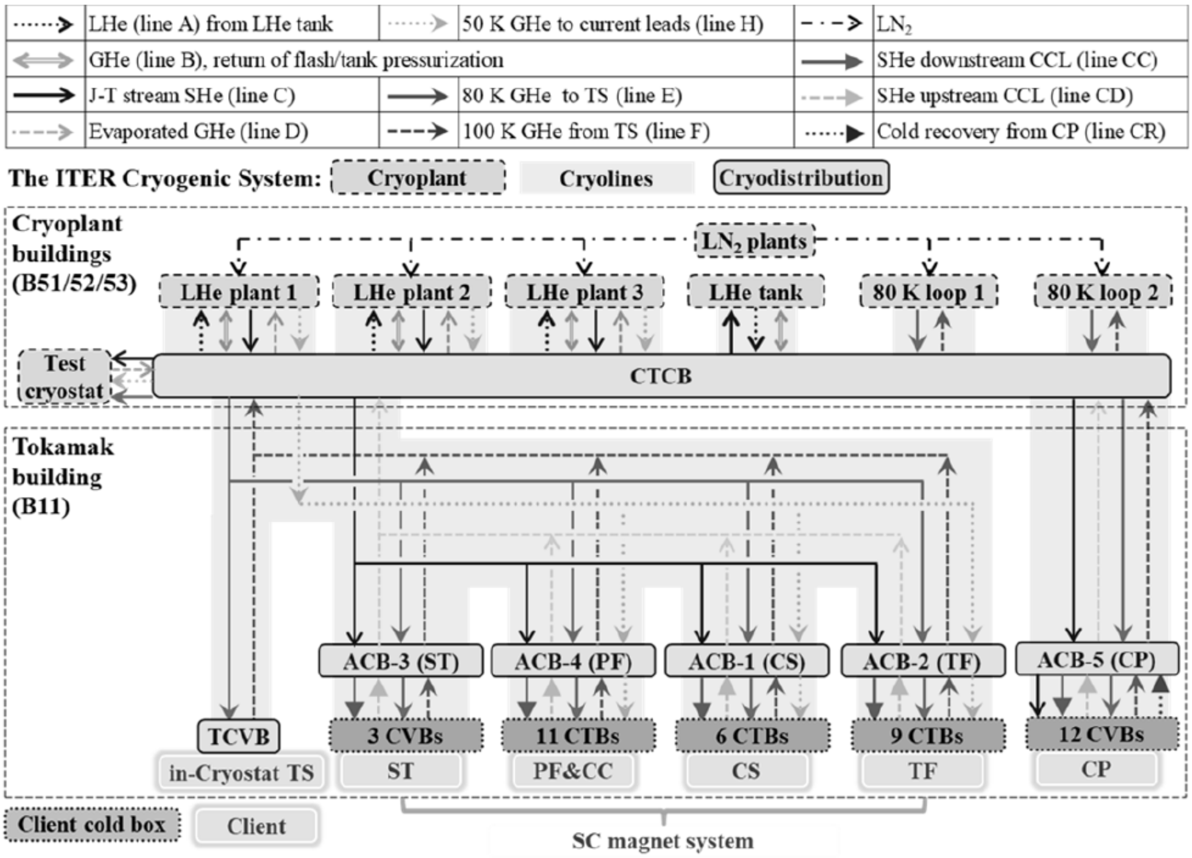

Figure 17 ITER cryoplant /CHAN 17/

Table 23 ITER storage tanks

Description	No. of tanks	Single tank volume (m³)
Pure helium storage at ambient temperature	5	400
Impure helium storage at ambient temperature	1	400
Lhe storage	1	175
He quench storage	2	360
He gasbag	7	120
Liquid nitrogen storage	1	300
Gas nitrogen storage	1	100

As stated in /MONN 15/: "the cryoplant technical specification refers to the European or International standard such as EN ISO 10440-1 for rotary-type positive-displacement compressor or EN ISO 10438 for lubrication, shaft-sealing and control-oil systems and auxiliaries. All rotating machineries have to follow either an ISO standard or its equivalent from the American Petroleum Institute. The heat exchanger will follow the TEMA (Tubular Exchanger Manufacturer Association) or ALPEMA (Aluminium Plate Fin Exchanger Manufacturer Association) standards".

In turn, the ITER Cryodistribution is composed by seven cold boxes: one Cryoplant Termination Cold Box (CTCB), one Thermal Shield Cold Valve Box (TCVB), and five Auxiliary Cold Boxes (ACBs) /CHAN 17/. Each ACB is dedicated to a specific sub/system (user). These sub/systems are: the CS coils, the TF coils, the magnet structures, the PF & Correction Coils, and the Cryopump system. The CTCB is installed in the cryoplant building: it interfaces the cryoplant with the ACBs and the TCVB installed in the tokamak building. Each ACB foresees a cold circulator and a heat exchanger to pressurize and cool the liquid helium to the desired conditions. The heat exchanger is also connected to a phase separator. Coil Termination Boxes (CTBs) and the Cold Valve Boxes (CVBs) manage the mass flow distribution from the ACB toward its clients. In turn, the TCVB has a simpler arrangement being directly connected to its client.

The conceptual cryoplant design of CFETR phase I is based on the solutions adopted by ITER because they face similar heat loads /LIU 17/. The flow diagram of the system foresees a three steps process: the warm helium gas goes to the compressor station and then to the cold boxes. Here, with the aim of a liquid nitrogen precooling, the helium is cooled down to the requested temperature and send to the users. Four output temperature levels are available: 4.2 K, 4.5 K, 50 K and 80 K. The 4.2 K and

4.5 K lines supply the superconducting magnets, the 50 K line to the high temperature superconducting current leads, and the 80 K line the thermal shields.

B.11 Tritium Extraction and Removal (TER) system

The breeding blanket with integrated FW is the key nuclear component for power extraction, tritium fuel sustainability, and radiation shielding in fusion reactors. The breeding blanket concepts that are under development consider the both options of breeding tritium, in ceramic solid breeders and in the liquid breeders such eutectic PbLi. Currently, various processes related to the tritium extraction from solid and liquid breeders are under development and decision on the processes selection is expecting in the following years. The developments on the ceramic breeders and the full characterisation of the tritium transport in various operation conditions have a significant impact on the related technology for tritium extraction. Preliminary tritium experiments related to tritium extraction from solid breeders are available in contrast with the liquid breeders where the experiments only with hydrogen and deuterium are available for limited operation conditions /CRIS 20a/.

B.11.1 Water cooled concept

The eutectic alloy $Li_{17}Pb_{83}$ (LiPb in the following) is the breeder and neutron multiplier of the WCLL. In addition, LiPb is in fact the carrier of the tritium outside of the VV in a dedicated loop (LiPb loop) and transports the tritium to the extraction units. Following the extraction, the tritium is routed to the Tritium Plant, in particular to the Tokamak Exhaust Processing (TEP) of the Fuel Cycle.

The WCLL TER System consists by a LiPb loop connected to the BB BZ, a pumping system to recirculate the LiPb, a Tritium Extraction Unit (TEU) that extracts the tritium from the LiPb flow and auxiliaries to transfer the tritium from the TEU to the Fuel Cycle. In the configuration of the TER system, several auxiliary systems for the LiPb purification aiming to remove the corrosion and activation products, to control the LiPb chemistry, to refurbish the ^{6}Li content, for heating and cooling of the LiPb flow, to store the LiPb, draining of the BB segments and shield the equipment due to LiPb activation are included /UTIL 19/.

Several TER processes that have potential for industrialization have been investigated but a decision on selection of the technology is still pending. The following TER processes are under evaluation:

- Tritium stripping in Gas-Liquid Contactor (GLC) packed column;
- Permeator Against Vacuum (PAV);
- Liquid Vacuum Contactor (LVC)

and the criterion for the selection are: efficiency of the technology, design integration topics, operation, safety, waste generated and costs.

The main features of the technologies/processes that have potential for implementation in WCLL TER are as follow.

The PAV operating principle is simple: PbLi with a certain concentration of tritium flows in a channel delimited for a given length by a membrane permeable to tritium. Vacuum is maintained on the other side of the membrane, so that the difference between the tritium partial pressure on the two sides of the membrane drives the flux of tritium from the PbLi side to the extraction, leaving a lower tritium concentration at the outlet of the channel. As opposed to the undesired permeation of tritium through the walls in the rest of the circuit, which may lead to safety issues and reduced TBR, here the relatively low solubility and the resulting high partial pressure of tritium in PbLi, together with a suitably permeable and corrosion resistant material for the membrane, are at the basis of the PAV potential for operation. The two main options for the membrane material are iron and niobium. Nb has a higher tritium permeability with respect to Fe, thus allowing to improve the extraction performance, but it tends strongly to oxidize at high temperature. For this reason, a Pd coating should be considered as an option. However, the additional resistance given by the Pd coating should not affect the result significantly because of its high permeability and small thickness /DAUR 17/.

The GLC process use the mechanisms of diffusion between gas and liquid phases to extract tritium from the breeder. In this process, PbLi and the process gas are mixed together and their contact surface is maximized, in order to obtain a higher value of the gas flux at the interface gas-liquid. The GLC are vertical columns filled with packing or other device providing a large interface between liquid and gas phase in both counter-current and co-current flow. The main advantages of the GLC are related to the reliability of the injection system, it is not necessary to inject small size bubbles, the reliability in operation due to the kinetics of mass transfer and due to the packing material that could be manufactured of high corrosion resistance materials to PbLi /UTIL 19/.

The method LVC is based on the passive extraction of tritium from millimetre-scale PbLi droplets falling in a vacuum tank. The main drawback of the method is related to the tritium transport under vacuum in large volumes with large surface of PbLi that may promote several adsorption and desorption processes until tritium release. Effects related to the droplets oscillations that may improve the tritium transport from the droplets are under investigations /OKIN 16/.

B.11.2 Helium cooled concept

The TER system for the helium cooled concept is related to the HCPB approach. The tritium release from the ceramic materials accommodated inside the solid BB is a complex mechanism. While tritium is originally produced in its atomic form inside the ceramics structure, it is rapidly to be found also as tritiated water at the surface of the breeder material due to interaction with residual moisture / water layer on the ceramics and free oxygen atoms after Li burn-up. The tritium is removed from the ceramic breeders using a purge gas that enhances its transfer.

Depending mainly on the purge gas chemistry and the temperature, the tritium is then liberated and extracted in its both molecular (Q_2) and oxidized (Q_2O) forms (Q means one of the hydrogen isotopes H, D, or T). Therefore, in contrast to the liquid BB where tritium is extracted as Q_2 solely, the TES from any solid BB must handle both the molecular tritium and the tritiated water. The ratio Q_2/Q_2O strongly depends on many parameters such as the ceramic material, the temperature, and above all the purge gas composition. It has been commonly accepted to use helium as purge gas containing hydrogen and steam that help the tritium to be released from the ceramics and thus enhancing the tritium extraction with a direct and significant reduction of the tritium residence time. The H_2 and H_2O at the level of 10-100 Pa proved to be very effective in enhancing tritium release and its later extraction from the breeding zone.

In order to minimize the tritium permeation from the purge gas into the cooling system, the tritium partial pressure inside the purge gas shall be kept at minimum reasonable figure. The purge gas flow rate has the main contribution on the tritium partial pressure in the purge gas.

In the case of helium purge gas that contains Q_2 and steam, at ratios that will be further defined, the TER system shall be designed to allow tritium extraction from the two phases: the gas phase containing tritium Q_2 and the tritiated water Q_2O.

Following the purging of the breeder blanket, the stream will be diverted through the TER to remove tritium from the helium stream and recycle helium back to the purged blanket. The design of the DEMO TER is based on the trapping/adsorption of Q_2O on the Reactive Molecular Sieve Bed (RMSB) and adsorption of Q_2 on Cryogenic Molecular Sieve Bed (CMSB) at 77 K.

Tritium is recovered from the RMSB via catalytic isotope exchange (isotopic exchange between a purge gas (H_2) and the Q_2O trapped on bed) during the regeneration phase. The release of tritium from the CMSB is realized during the regeneration phase which consists on heating up the tritium trapping beds.

The main drawback of the Q_2 recovery from purge gas using the CMSB is the large consumption of LN_2 that in the case of high pressure of purge gas, 8 Mpa, become prohibited. Therefore, in view of minimization of the operational costs and the load on the infrastructure, alternatives options for removing tritium as Q_2 from the purge are under evaluation. One option is the getter beds technology that is further investigated due to the benefits such as investment and operational costs.

Developments in view of mitigation of the tritium permeation in the cooling gas are on-going and one of them is the operation of TER with steam in purge gas at partial pressure in the range 10-100 Pa. This operation allow decreasing of the Q_2 content in the purge gas with large benefits on the CMSB or getter beds design and operation /CRIS 20a/.

B.12 Tritium, fuelling and vacuum system

The development of DEMO fuel cycle that includes tritium recovery from unspent fuel and the integration in the process loop of bred tritium, the fuelling and vacuum system addresses the main issue that is the tritium inventory for DEMO operation. Large tritium inventory is required when the batch technology choices for primary pumping (cryopumping), rough pumping (cryopumping and mechanical pumping) and isotope separation (cryogenic distillation) are adopted as in ITER case. In the ITER configuration ´once through´ architecture in which all tokamak exhaust gas is routed through the tritium plant and finally separated to the level of the pure hydrogen isotopes (H_2, D_2, T_2), for re-injection in the torus or for temporary storage is implemented /DAY 19/. The addition of an outer circuit for tritium breeding at the level needed to achieve tritium self-sufficiency (at a fusion power of 2 GW and a tritium breeding ratio of 1.05) and the technology to recover the tritium from the blanket coolant is adding another contribution to the integral tritium inventory of the DEMO plant. Therefore, the fuel cycle of a DEMO differs from the ITER configuration not only in terms of the blanket systems but has to meet additional requirements while not expecting significantly higher burn-up fractions than anticipated for ITER. As a result, the concept of Direct Internal Recycling (DIR) was integrated, leading to a three-loop architecture with the fast DIR loop, an inner tritium plant loop, and an outer tritium plant loop integrated with the blanket systems as illustrated in Figure 18. The DIR shortcut re-routes at short processing time the major part of the tritium in the exhaust gas, so that only a minor fraction will have to be routed through the tritium plant.

This fraction again is distributed in a faster loop with Protium Removal (PR) and Isotope Re-balancing (IR) which produce D-T at the required composition, and an outermost loop for the tritium recovery duties of the remaining gas. The later loop also includes the tritium recovery from the tritiated water.

Replacing of the batch wise processes with continuously processes is largest implemented all over the tritium processing systems, avoiding the tritium accumulation and immediate reuse of the tritium from the breeding blanket. This includes the change from discontinuous torus cryopumping to mercury based continuous vacuum pumping, with zero demand on cryoplant power, and implementing of a thermal adsorption process instead of the cryogenic distillation process that implies large tritium inventory. Super permeable metal foils are introduced in the divertor ports to separate the DT streams for direct internal recycling and to feed the pellet injector system.

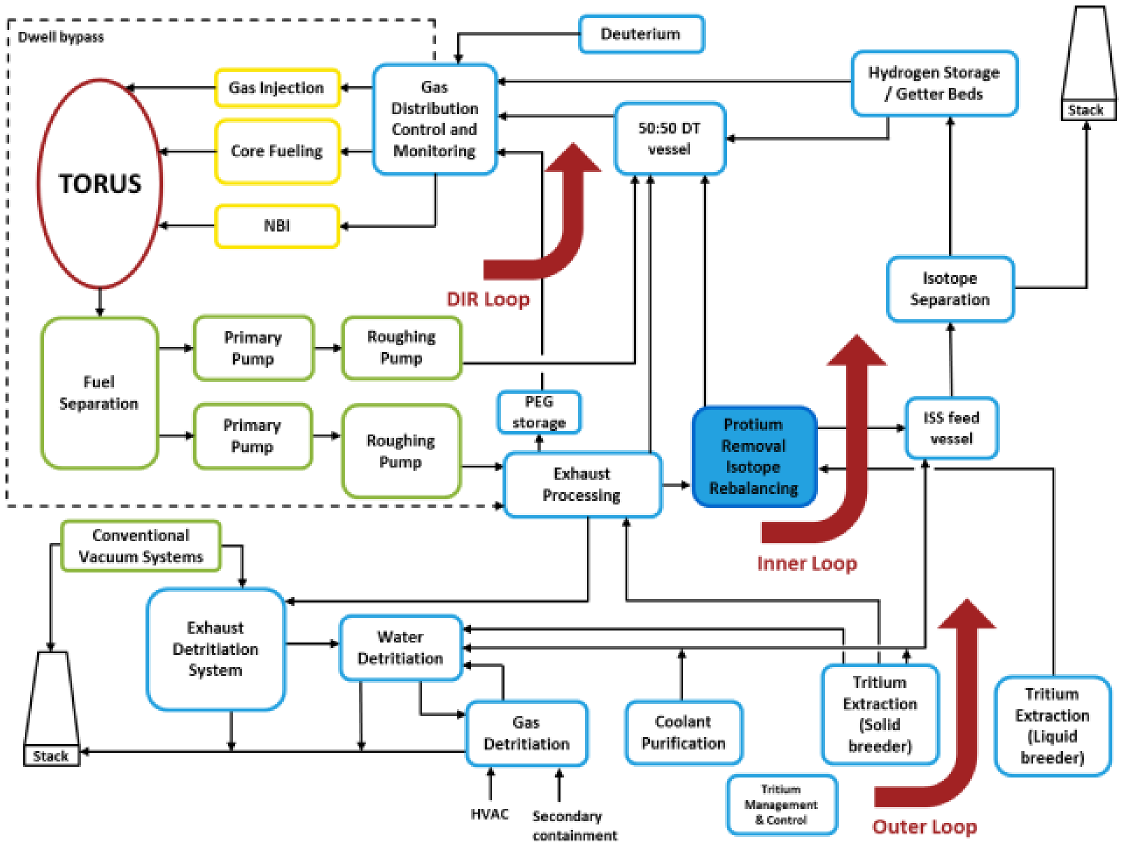

Figure 18 Schematic of the EU DEMO fuel cycle architecture

B.13 Thermal shields

Two thermal shields are installed in EU DEMO to protect the magnets system from the radiative heat coming from the VV, the cryostat, and the in-cryostat components. The first thermal shield is installed between the VV and the magnets (VV thermal shield), while the second one is installed between the magnets and the cryostat (cryostat thermal shield). No detailed design exist for the Thermal Shields, but preliminary studies show that they need to be actively cooled to reduce the thermal loads on the magnets /KONC 17a/. In /KONC 17b/ it is suggested to use a single circuit with helium at ~80 K to ensure thermal shield temperatures of ~100 K. About 80% of the total heat reaching the magnets come from the VV. This pushes for the evaluation of different designs for the two thermal shields

/KONC 17b/. The rationale behind the cryostat thermal shield is to follow the ITER design: a single wall structure made of AISI 304LN steel covered by a layer of silver to reduce the emissivity below 0.05. Instead, for the VV thermal shield the installation of an additional 15 mm-radiative Multi-Layer Insulation (MLI) on its warm side is proposed. This concept is borrowed from Wendelstein 7-X where MLIs are made of 90 layers of aluminium foil with glass silk spacers /NAGE 11/.

The design of the thermal shields for CFETR seems to be at a more mature stage of development. For the cryostat thermal shield the use of AISI 304LN steel has been selected together with a subdivision into 16 sectors /WANG 19/. Each sector is composed by 20 sub-parts connected by bolts. In the cryostat side the cryostat thermal shield is refrigerated by a complex piping system employing helium at 1.8 Mpa and 80 K. A similar cooling arrangement seems adopted also for the VV thermal shield, but no further details are available /ZHUA 19/.

No data have been found regarding the cryostat of K-DEMO and JA DEMO.

B.14 H&CD system

The study and development of the H&CD system in EU DEMO involve the area of Neutral Beam (NB), Electron Cyclotron (EC) and Ion Cyclotron (IC) systems. Heating of the plasma is required to bring the plasma to the temperature necessary for fusion reaction.

B.14.1 Neutral Beam Injection (NBI) system

The NBI system injects high energy beams of deuterium neutrals (D0) in the plasma. These beams are injected in tangential direction in the plasma and their aim is to support plasma heat-up and to contribute to the plasma rotation and current drive /SONA 17/. In EU DEMO the functional requirement of the NBI system is to provide about 50 – 70 MW of core plasma heating /AGOS 20/. A Negative ion-based Neutral Beam Injector (NNBI) is under development based on the experience from ITER. The NBI adopted for ITER consists in /GRIS 12/:

- An ion source to provide H-/D- atoms;
- An accelerator system to accelerate D- ions and to remove the free electrons;
- A neutralizer to convert the D- beam into a D0 beam;
- A Residual Ion Deflection unit (RID) to remove residual positive and negative ions through the application of an electrostatic field;
- A calorimeter to measure the beam power;
- A shutter valve to isolate the NBI from the tokamak building;
- A system to connect the NBI system with the VV;
- A beam duct to inject the D0 beam into the plasma.

A surface-plasma ion source similar to that used in ITER /GRIS 12/ is proposed also for EU DEMO /SONA 17/, but instead of a single source the adoption of several sources in parallel to achieve a higher availability is considered. The main advantage of this solution is the better alignment of the beams when deformations occur due to the thermal expansion (smaller source size means smaller deformations). On the other hands, this solution increases the complexity of the design and the construction of the NBI system /AGOS 20/. Each sub-source is also equipped with a Radio Frequency (RF) source to drive the extraction of ions. Two RF concepts are under consideration: race track RF and helicon plasma. It is necessary to replenish the Cs layer on the plasma grid continuously due to deconditioning of the ion source. Effective Cs distribution schemes are investigated to solve the problem with Cs depletion during long pulses when the conventional distribution of Cs through Cs ovens is adopted /SONA 17/.

At the outlet of the ion source, an accelerator system is installed. The accelerator system is based on the electrostatic acceleration principle and it consists of several copper plates at different potential aimed at accelerate D^- ions while removing free electrons /SIMO 97/. This system is equipped with a vacuum pump to reduce the losses. Three concepts are under investigation: Non-Evaporable Getter (NEG) pumps, cryopumps, and mercury diffusion pumps. Among the three options, the NEG pumps is the preferred one because: it is more resistant to neutron radiation, it doesn’t require liquid nitrogen or helium to operate, and it has a lower operative cost. The mercury diffusion pumps are another solution proposed to avoid the use of cryopumps, but both solution (mercury diffusion and NEG pumps) are still in the validation process. In the surrounding of the beam source and the accelerator system magnets are also installed to compensate the magnetic field deriving from the main magnets around the VV.

The neutralizer is installed after the accelerator system. Two concepts were under investigation in 2017 /SONA 17/: the photo-neutralizer (see Figure 19) and the gas neutralizer concepts (as in ITER). The photo-neutralizer concept uses laser beams /FASS 16/, while in the gas neutralizer concept the neutralization is reached through the collision of the D^- beam with gas molecules /WEI 13/. As stated in /AGOS 20/, the photo-neutralizer concept was abandoned for EU DEMO due to its lack of maturity, but it is still considered for the FPPs after DEMO. At the same time, studies on a beam driven plasma neutralizer have begun. In this concept, the plasma created by the passage of the beam through the neutral gas is used to enhance the neutralization efficiency /SURR 13/.

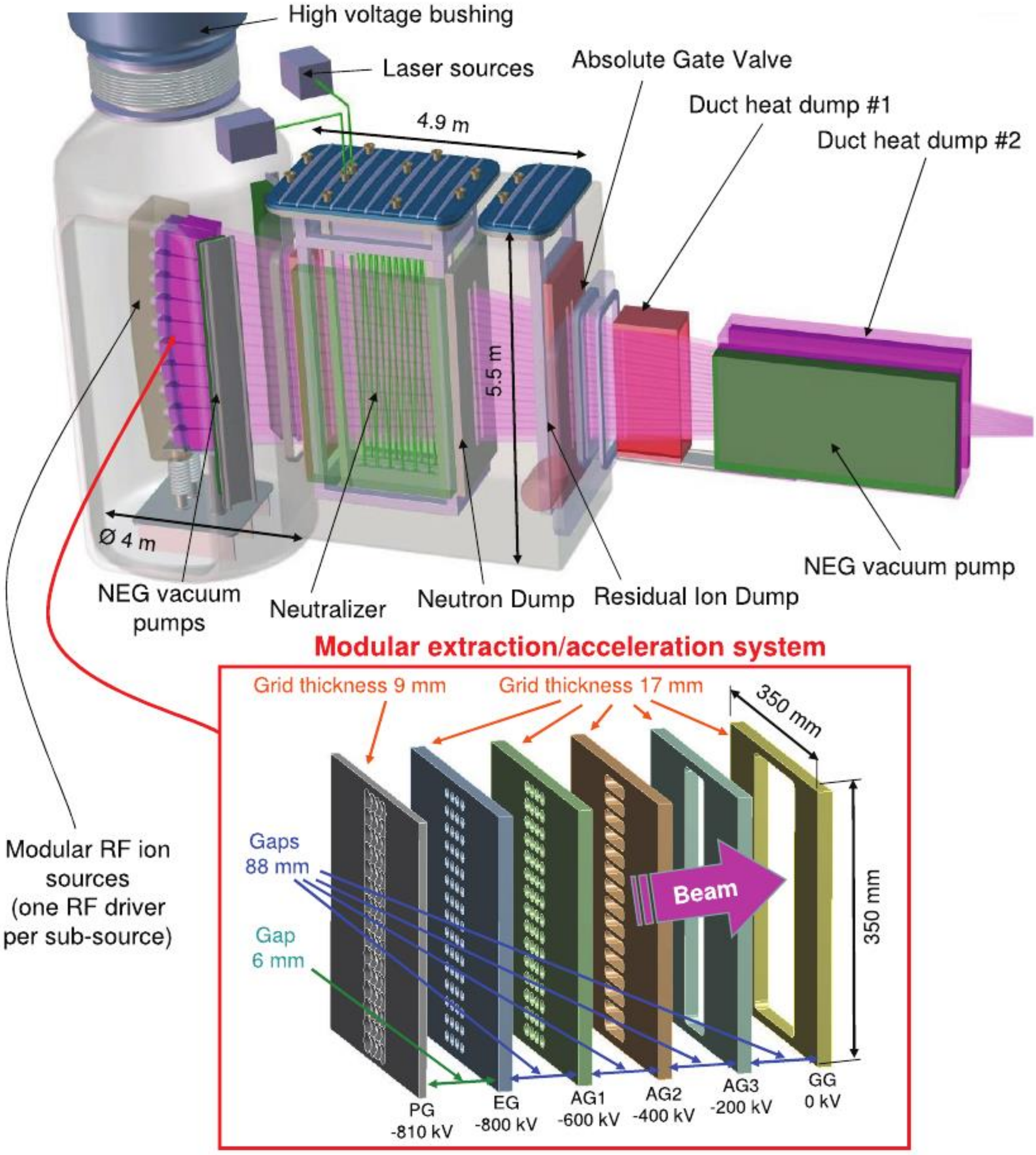

Figure 19 Schematic of the EU DEMO NBI equipped with photo-neutralizers /SONA 17/

The installed RID unit magnetically deflects the remaining ions to a flat water-cooled CuCrZr plate. The design of the RID unit is still on-going.

Both the ion source(s) and the accelerator are installed in the beam vessel, while the neutralizer and the RID structures are contained in the beam line vessel. The beam duct connects the beam line vessel with the tokamak chamber. This duct is kept in vacuum conditions to avoid re-ionization of the D^0 beam /SONA 17/. As for the accelerator system, the same three pump concepts are under investigation.

A NBI system is also under development for the CFETR plant. The solutions proposed are similar to those adopted by ITER reactor: a single surface-plasma ion source /LIU 20/, cryopumps to keep the vacuum conditions /ZHAN 19/, and a gas neutralizer. No relevant data have been found for K-DEMO and JA DEMO.

B.14.2 Electron Cyclotron (EC) system

The EC system is used for plasma heat up and control of the plasma instabilities. The physics behind this system is based on the wave propagation and absorption phenomena: an electromagnetic wave having the same polarization of the plasma is launched against the plasma itself to heat it up thanks to the resonance phenomenon /ERCK 94/. In a tokamak machine like DEMO, the EC system is required for: plasma start-up, ramp-up, burn, ramp-down, and the control of plasma instabilities /GARA 20/.

At the present stage /GARA 20/, only a pre-conceptual performance study has been performed for the EU DEMO. So far, the EC system development was mostly based on reaching the operative conditions required for DEMO, which are beyond the actual state-of-the-art for most of its components /AVRA 19/. According to the actual pre-conceptual design, the EC system is divided into clusters each having several coaxial gyrotrons, several Multi-Beam Transmission Lines (MBTL), and a launcher.

Gyrotrons are unique components aimed at creating microwaves by electron resonance in a magnetic field. A detailed design of the gyrotrons is not yet available /GARA 20/, but only target design conditions have been specified: capability to operate at two different frequencies (170 and 204 GHz) with a power level of 2 MW, a 60% efficiency, a 98% Gaussian output content, and a reliability of ~98%. The gyrotrons belonging to the same cluster are fed by 2 or 4 high voltage power supplies.

The MBTL connect the gyrotrons with the launcher (see Figure 20). MBTL are equipped with several mirrors and their aim is to guide and split the beam from the gyrotron to the launcher. Again, the design at early stage of development /GARA 20/ and only target design conditions are available: multi-frequency capability, handling of 2 MW continuous wave beams, 90% transmission efficiency, and nuclear safety. Two solutions are currently proposed for the MBTL: the W7-X and ITER lookalike solutions. In the first solution the design is similar to that of W7-X, but the lines operate under vacuum conditions instead of air as in W7-X. The second solution is based on the use of evacuated corrugated waveguides as done in ITER.

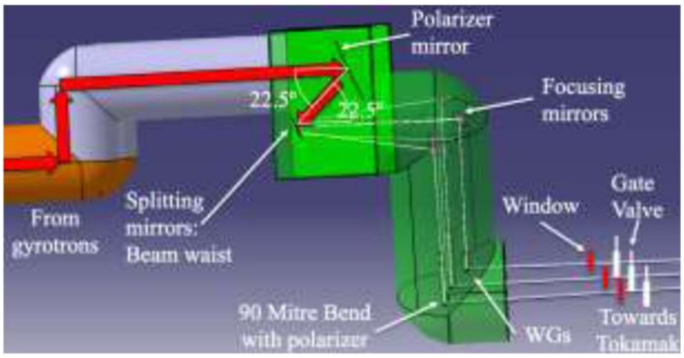

Figure 20 Schematic of the EC MBTL in EU DEMO /GARA 20/

As for the gyrotrons and the MBTL, also the launcher design and its antenna are still in active development /GARA 20/ (see Figure 21 a) and b)). The antenna options under study are remote steering antenna and Mid-Steering Antenna (MSA) for plasma stabilization, simply Open Ended Waveguide (EOWG) antenna for plasma heating, and multi-beam mirror antenna for both tasks. Among these options, the MSA and EOWG solutions are at a more mature stage of development, and for them a pre-conceptual design of the launchers is available /SPAE 20/ (see Figure 21 c)). The launcher is a steel block having several channels (antenna) drilled inside where mirrors are installed to guide the beam from the MBTL to the plasma. In DEMO, these mirrors require active water cooling. The launchers are installed in a recessed position behind the BB and in correspondence of the equatorial ports. Most of the launcher is protected behind the BB, but a portion has to face the plasma to launch the beams onto it. Because of its position, the launcher must ensure the tritium segregation. The launcher is fixed on the port extension that goes from the VV equatorial port to the bioshield. As for the mirrors, also this extension requires active cooling. The material of the launcher is not yet defined, but the port extension is supposed to be AISI 316 L(N).

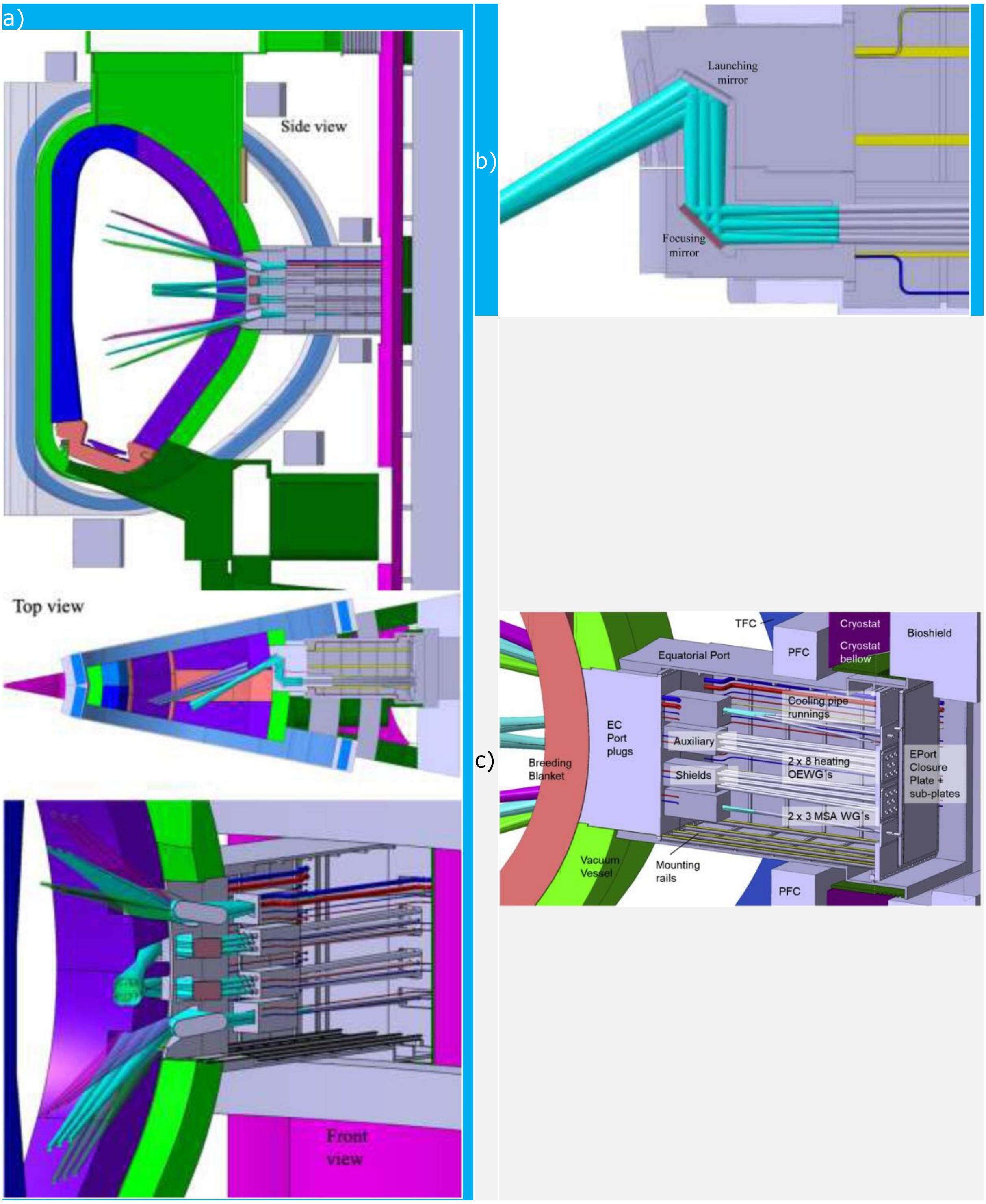

Figure 21 Schematic of the EC launcher in EU DEMO /GARA 20/ /SPAE 20/.

Studies are also on-going for the integration of the EC system in the tokamak building. Preliminary studies not considering the tokamak magnetic shielding prescribes a distance of 120 m between the RF buildings and the DEMO hall. The gyrotron hall is near the tokamak hall and the two buildings are connected by the MBTL system.

For CFETR a conceptual design was introduced in 2015 in /TANG 15/, and further updated in 2018 /TANG 19a/, 2019 /TANG 19b/, and 2020 /LU 20/. At the present stage, it is not clear what is the current design because the latest documents (/TANG 19b/ and 2020 /LU 20/ provide contradictory data. According to /TANG 19b/, gyrotrons are required to operate at two frequencies (170 and 230 GHz) as in EU DEMO, but the transmission lines are made as circular corrugated waveguides as in ITER, and only a single launcher is required. /LU 20/ describes a completely different design and provide few data concerning structural material: waveguide made of AISI 316 LN, mirrors made of Oxygen Free Cooper (OFC), and mirror support actively cooled with water and made of CLAM.

No detailed data have been found for K-DEMO, except for the requirement of a high-frequency EC system (> 200 GHz) as in EU DEMO /KIM 19/.

B.14.3 Ion Cyclotron (IC) System

The IC System is used for the FW conditioning, removal of impurities, heating and current drive, and plasma start-up, burn, and ramp-down /VANE 19/. A typical IC system – as the ITER one /LAMA 13/ (see Figure 22 and Figure 23) – consist of RF sources fed by high-voltage DC suppliers, transmission lines with their matching units, and antennas equipped with feeding lines and integrated in a VV port or in the BB. The waves produced by the RF sources are transported to the antennas through the transmission lines. The connection between the transmission lines and the antennas is performed with the matching units.

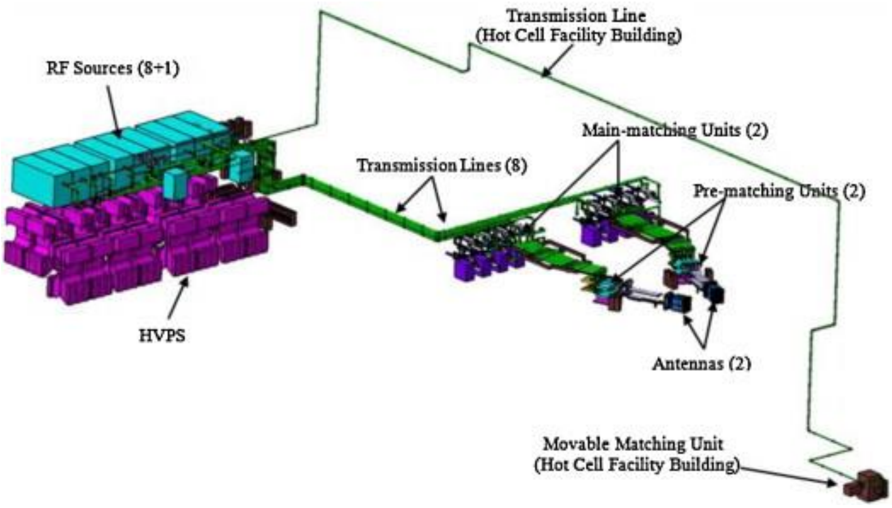

Figure 22 Schematic of the IC system in ITER /LAMA 13/.

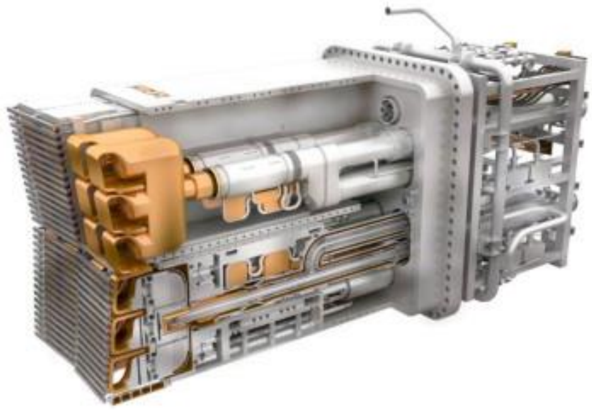

Figure 23 Schematic of the IC launcher in ITER /LAMA 13/.

The IC system is at a very beginning stage of development in the EU DEMO /BADE 17/. Currently, Travelling Wave Antennas (TWAs) integrated in the BB are considered for EU DEMO. EUROFER is considered as material for TWAs, while for the transmission lines the CuCrZr alloy is candidate. The structural material for the feeding lines is EUROFER with a thin conducting layer (CuCrZr). Most of transmission line is at high pressure, but the final part is in vacuum conditions. To ensure the separation of the two parts, ceramic windows are introduced. A ceramic material is also adopted as insulating material in the transmission lines. Except the antennas and the final part of the transmission lines, the other components are installed in dedicated buildings outside the cryostat.

For CFETR, several studies are on-going. The strategy pursued regarding the H&CD system seems unclear, especially concerning the implementation of an IC system. In /HU 20/ an IC system together with a Low Hybrid wave Current Drive (LHCD) system is considered, while in /ZHU 19/ the substitution of the LHWC with a High Harmonic Fast Wave (HHFW) current drive is studied. No details on the design of the IC system, LHWC, and HHFW current drive have been found.

In K-DEMO the IC system is called Helicon Current Drive (HCD), and it is based on the TWA concept /JANG 20/. The design of the system is still in active development, but recent works suggest the use of ceramic windows made of high-purity alumina (Kyocera A479B), a water cooling to keep the system below 200 °C, and a design frequency of about 480 MHz.

No relevant data have been found for JA DEMO.

B.15 Plasma Diagnostic & Control system

Since the plasma key features have not been yet simultaneously demonstrated, the Diagnostic & Control (D&C) is still an early stage of development for each DEMO concept. In EU DEMO the current trend is to develop the D&C basing on the ITER experience /BIEL 19/. The D&C system is not only in charge of the control of the burning plasma, but it is also an operation supervisory tool. As for ITER, the following solutions will be adopted: port plug integration approach, maintenance through remote handling, and neutron-resistant materials. Although, some challenges that EU DEMO will face are beyond the ITER situation: the harsher operative conditions (higher neutron and thermal loads, erosion and deposition of dusts), the higher reliability to avoid plasma disruptions that could damage the machine, the higher accuracy near its operation limits, the space limitations to ensure a sufficient tritium breeding. So far, the development of the D&C focused only on the full-power scenarios, with the ramp-up, ramp-down, control of instabilities, and incidental scenarios left for a later stage of the project.

The operative experience of ITER is a fundamental requisite to develop and design the DEMO diagnostic because it will show the capabilities of the whole Control, Data Access and Communication (CODAC) architecture /DONN 07/. The plasma diagnostic systems are required to provide accurate measurements of the plasma characteristics for the plasma control including plasma current, position, shape (vertical and horizontal stability), density, radiation, fusion power, temperature, impurities, etc. Figure 24 and Figure 25 summarize the design approach adopted for ITER: the first figure shows the controlled parameters vs the actuators used to manage them, while the second figure the diagnostics proposed for each controlled parameters. The most critical components of each diagnostic system are installed far from the VV, with only viewing ducts, microwave antennae, and waveguides facing the plasma. Optical mirrors, beam paths, magnetic sensors and their cabling are installed behind the blanket. Further signal routing components are installed in the port plug region. Instead, the implementation of a sheath voltage or thermo-current measurement is planned for the divertor. The control of the plasma will not be only performed through detailed measurements, but it will be also based on advanced modelling, especially for off-normal conditions.

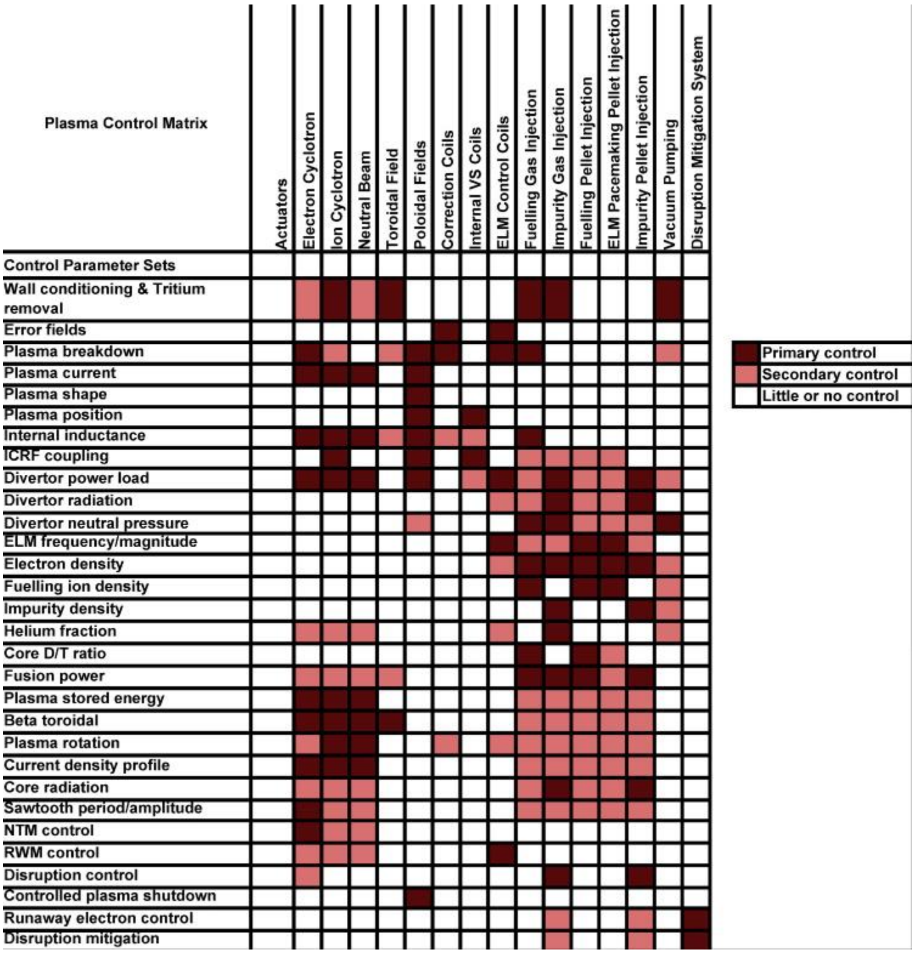

Figure 24 Controlled parameters vs Actuators /SNIP 10/

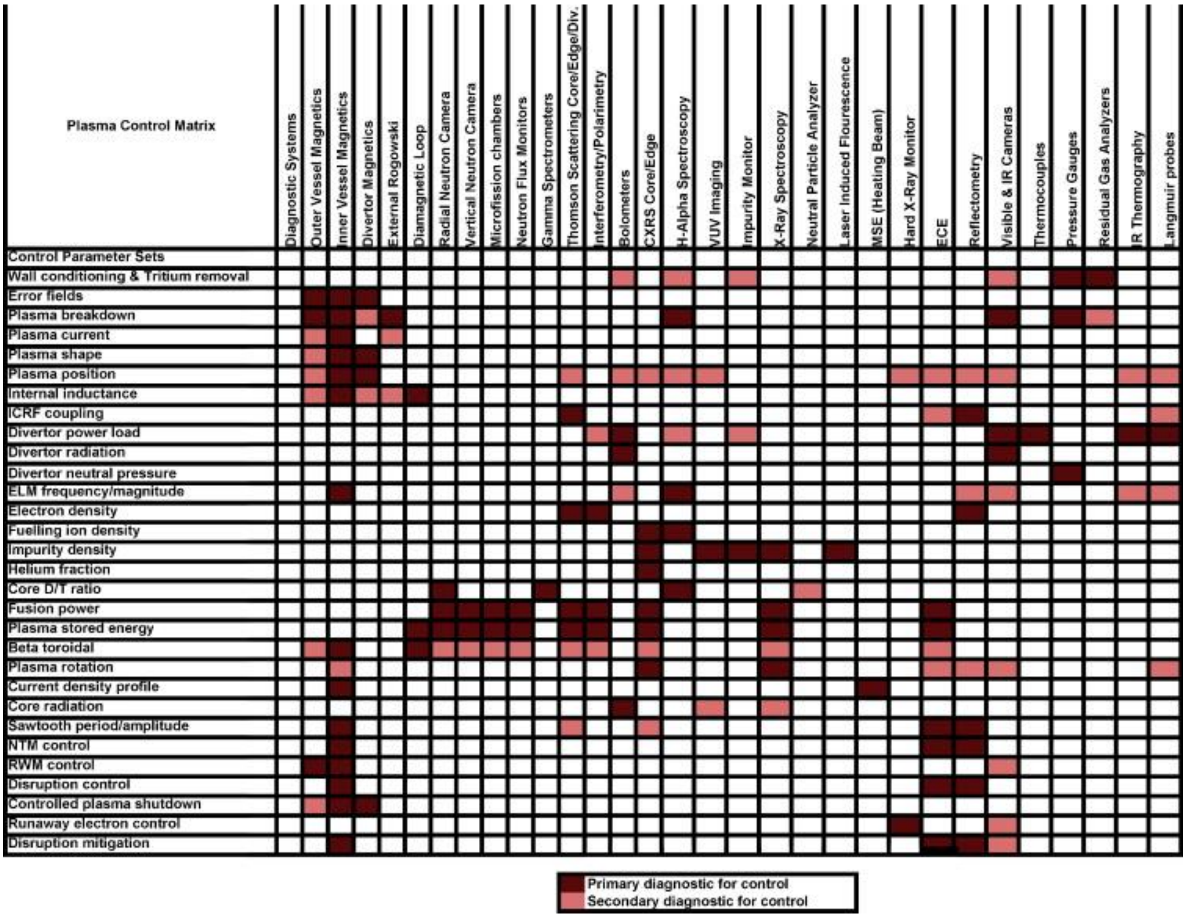

Figure 25 Controlled parameters vs Diagnostic systems /SNIP 10/

It can be expected that most (if not all) of the diagnostics considered for ITER will be employed as well in DEMO. Although, the development in view of EU DEMO seems started for only few diagnostic systems:

- In-vessel magnetic coil based diagnostics for equilibrium control. Two isolation materials are under investigation of the in-vessel sensors to reduce the noise: mineral insulated cables, and low-temperature co-fired ceramics. Both solutions are under investigation also for ITER, but the harsh conditions of DEMO pose the serious risk that the measurement devices will degrade over time. Two backup solutions to solve this issue are proposed: ex-vessel magnetic coil based diagnostic and metallic Hall sensors. The first solution implies the problem of being slow, so it can't be used to control rapid plasma movement. For the second solution the structural material is the key issue. In ITER Bismuth based Hall sensors are developed, but for DEMO Bi based alloys and Gold sensors are under investigation because of their higher melting point compared to pure Bi (271 °C).

- Microwave reflectometry and electron cyclotron emission measurements for plasma density and electron temperature, respectively. The front-end components of both solutions are horn antennae and waveguides made of EUROFER with tungsten coating.

- Infrared (IR) laser interferometry and/or polarimetry for the measurement of central plasma density. The mirrors required for these solutions are installed behind a duct, and an IR window made of crystal is required to isolate the high-pressure zone from the vacuum zone (as in the IC system).

- Spectroscopic and radiation measurements required to balance the heating power and the core radiation power. The heating power can be calculated basing on the neutron flux, while for the core radiation power the core plasma bolometers are used. Instead, the high-resolution visible spectroscopy is used to measure the plasma density and temperature in the outboard divertor target area. The vacuum ultra-violet spectrometry is also part of these measurement methods, which is required to control the amount of the impurities in plasma. This last solution is also investigated as a first backup solution of the high-resolution visible spectroscopy. The second backup solution is the thermographic observation, but it has still to be proven.

- Divertor thermo-current measurements are made at several divertor target plates to control the power exhaust. This solution requires the use of ceramic insulators between the divertor target and the divertor cassette or the VV. Two options are under investigation for the measurement of the thermo-current: use shunt resistors connected to the ground, or use the coolant tubes itself. The latter case requires a more complex design of the coolant pipes, which would also require a ceramic insulation.

- Neutron/gamma diagnostic made through cameras. The front-end of such cameras consist in a long duct with a small inner diameter. At the far end of the collimators one or more detectors mounted outside the bioshield. The collimator tube is made of EUROFER surrounded by boron carbide.

As already stated, the design of the whole D&C system is still in progress, but it is based on pushing forward the approaches adopted for ITER. The adopted diagnostic is only a part of this system, with the hardware and software composing the remaining D&C system. Modern tokamaks such as ITER and JET use several in-house developed software together with a robust hardware composed of Multiple-Input-Multiple-Output (MIMO) controllers, Field Programmable Gate Arrays (FPGA), and Digital Signal Processors (DSP) to integrate in real-time: a) measurements from different sensors, b) real-time plasma modelling, and c) the effects of multiple actuators /GONC 10/ (see Figure 26).

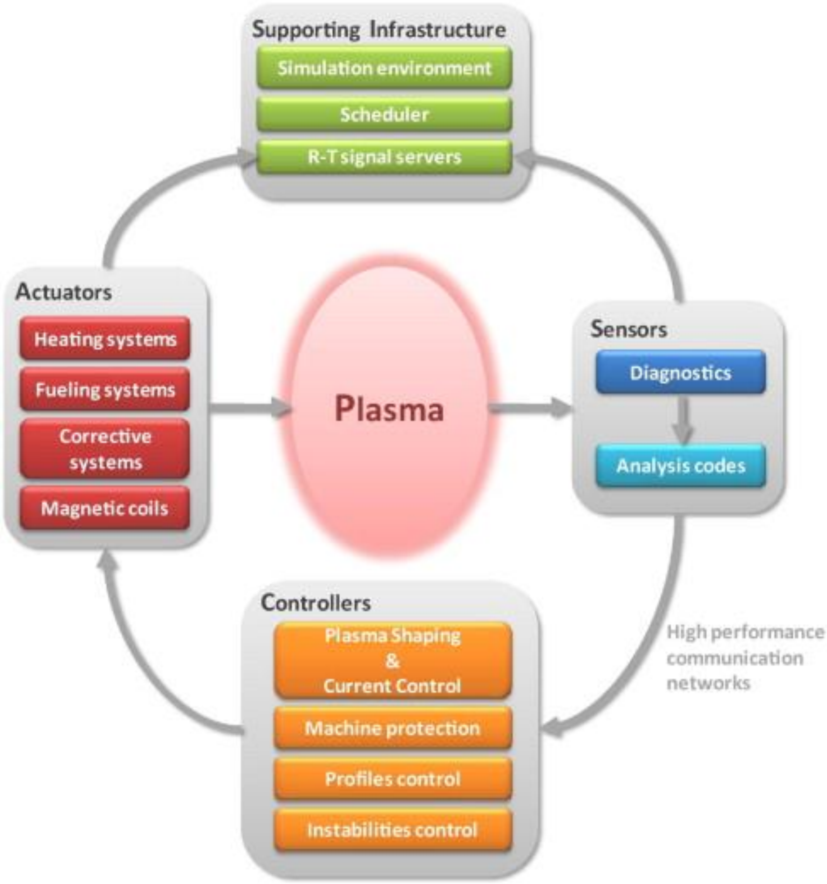

Figure 26 Workflow of the control system in EU DEMO /GONC 10/

For DEMO, the challenges for the D&C system are not only related to the harsher conditions in which it operates, but also on its capability to achieve advanced tokamak regimes capable of sustain high plasma pressure, long energy confinement time, and non-inductively driven plasma current. These goals can be reached controlling the plasma position (especially the vertical position), shape, density, current, and temperature /GARC 19/, /GONC 10/, /SNIP 17/. These plasma characteristics can be controlled promptly acting on other parameters of the machine, such as:

- The fuelling rate to have a stable plasma burning and so a steady-state power exhaust;
- The auxiliary heating to heat-up the plasma at the required temperature. In ITER this system is fundamental also during steady-state operations because of the non-burning nature of its plasma, but in DEMO a self-sufficient heat should be achieved through the fusion reactions themselves;
- The neutral pressure and the impurities amount to ensure plasma stabilization.

The plasma D&C system is also in charge of the prompt termination of the plasma if transients or instabilities arises during normal operations /PINN 20/. This system is called Plasma Shut Down (PSD) system, and it can trigger a fast shutdown (about 3 s) or a soft shutdown (in the order of 100 s) /MAVI 21/. The PSD system should act through the injection of inert gas or injecting a shutter pellet /PINN 20/, /MAVI 21/. This system is the equivalent of the SCRAM system in a NPP.

For CFETR phase I a preliminary design of the implementation of the plasma D&C is available /HU 20/. For phase II, the diagnostic will be reviewed in light of the findings of phase I. Table 24 presents the diagnostic solutions proposed for CFETR phase I.

No relevant documents have been found for JA DEMO and K-DEMO, but the requirements for plasma D&C can be expected to be similar/the same.

Table 24 Plasma D&C proposed for CFETR phase I

1	Magnetics (flux loop, Rogowiskis/Diamagnetic/ Mirnov/Sad. coils)	8	Visible spectrometer	15	Visible survey/ imaging system
2	Polarimetry and Interferometry POINT	9	Fission chamber & microfission chamber	16	IR survey/ imaging system
3	MW reflectometer	10	Neutron camera (Vertical & Radial)	17	Divertor Erosion monitor
4	Metal Bolometer	11	Gamma Ray spectrometer (GRS)	18	Thermal couple
5	LIDAR TS	12	Neutron Activation System (NAS)	19	Dust monitor
6	CXRS	13	Energy-Res. Farad Cup	20	Tritium surface concentration intensity, Hall
7	XICS	14	LVIC camera	21	Dosage inspection on outflow of the machine (solid, gas, liquid), T in the air

B.16 Remote Maintenance (RM) system

The components facing the plasma require regular replacement due to the erosion and the neutron damage. For this reason, EU DEMO is designed taking into consideration the requirements for a remote maintenance system. At the present stage, the design of the remote maintenance equipment is at an early stage of development, and also the strategies to perform the maintenance services are under investigation /CROF 16/, /BONG 19/. In any case, the guiding approach is to avoid the presence of personnel during each phase of the maintenance. /FARQ 19/ suggests three types of maintenance outages: short, minor, and major term outages. Short term outages are performed each 0.02 fpy, minor term outages each 0.49 fpy, and major term outages each 1.58 fpy. Given these conditions, it is clear that the design of a robust remote maintenance system and strategy is of utmost importance to demonstrate the feasibility of fusion power as a reliable source of energy production.

Because of the harsh conditions inside the VV, the remote maintenance will be performed through robotic arms entering from the upper, equatorial and lower VV ports. These arms are equipped with tools to cut/weld pipes, to assess pipe and IVCs integrity, and to lift the IVCs in case of the replacement. The robotic arm employed for the blanket segments removal is entered and anchored in the upper port to create a stiff system capable to handle the weight of a blanket segment (see Figure 27). In turn, the arm for the divertor cassettes extraction is inserted through a lower port, but it not fixed to the port itself because of the lower weight of this IVC (see Figure 28). The equatorial port would be used to insert a robotic arm capable of performing inspections and checks of the different IVCs /LOVI 14/ (see Figure 29). The whole system is also designed to allow the recovery and rescue of the IVCs and the Remote Handling (RH) equipment in case of failure.

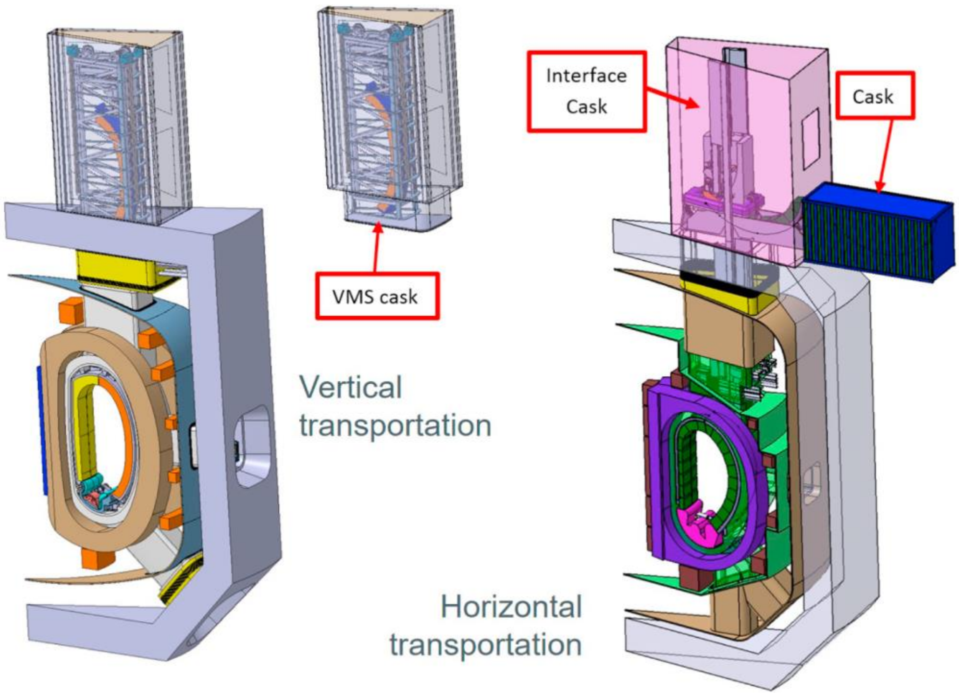

Figure 27 Two possible strategies (vertical and horizontal transportation) for the removal of the blanket segments from the upper port /VALE 17/

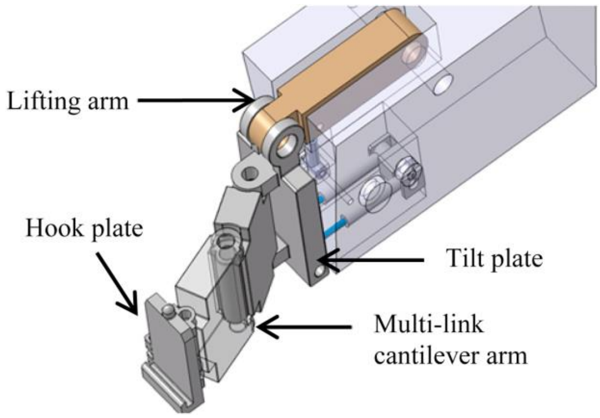

Figure 28 Divertor cassette deployer /CROF 16/

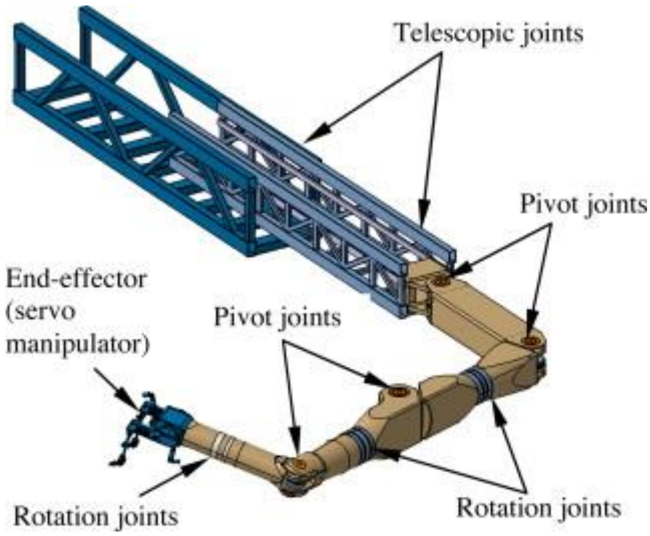

Figure 29 Robotic arm for inspections through the equatorial port /LOVI 14/

The segmentation of the blanket and the divertor ensures that a part of each IVC is visible through the ports to ease the handling of the components. The ports are also designed to allow the service connections within them to face environmental conditions less severe than those in the plasma chamber.

A plan is also in place to keep cooled surfaces of IVC, both under and not under maintenance, below 100 °C. This condition would be met using the coolant system and ensuring a ventilation rate in the VV of about 10 kg/s.

Several transportation schemes are under evaluation to transfer the IVCs to the Active Maintenance Facility (AMF), but the proposed technologies can be divided into three categories /VALE 19/: overhead, ground, and a combination of both. In any case, it is foreseen to place the IVC and its handling equipment in a cask, which is then moved to the AMF gate(s). /VALE 17/ suggests that the casks may be temporarily placed in parking, storage, and exchanging areas, but no further details are available.

The CFETR RM system seems at a more mature stage of development /WAN 17/. The removal of the IVCs components is plant to be done through the upper, equatorial/horizontal, and lower ports /GONG 15/, /ZHUA 19/. The divertor and the cryopumps (and their RH equipment) are removed from the lower ports, the blanket segments from the upper ports, and the remaining equipment from the horizontal ports. Tele-operated devices moving on rails installed in the VV and the ports extract the IVCs and place them into the cask system connecting the tokamak building with the Hot Cell.

The handling of the divertor cassettes is planned to be done through two devices /SONG 14/:a Cassette Multifunctional Mover (CMM), and the Cassette Toroidal Mover (CTM). The CMM moves along the port rails and it is in charge of: cut and weld the pipes crossing the lower ports, open/close the casks used to transport the IVCs to the Hot Cell, place the IVC in the cask, and transport the CMM device. The CTM moves along the toroidal rails inside the VV and it is in charge of: remove and install the divertor cassettes, cut and weld the divertor pipes, manipulate and lift the divertor cassettes. Both systems are designed to operate in an environment contaminated with tritium and dusts (C, Be, and W) at a temperature of 20-50 °C, and under a radiation intensity of 500 Gy/h.

In turn, the blanket remote handling is done through three systems /SONG 14/. The main system is made of a lifting structure, a pedestal structure, and a transfer structure, and it is aimed at transporting the blanket modules through the upper ports. The remaining two systems are in charge of lifting the blanket modules from their position in the VV and cutting/welding the blanket pipes, respectively.

The transfer of the IVC from the VV to the docking station is performed through the cask system /SONG 14/. The casks are directly connected to the upper, equatorial and lower ports flange through a double seal door system. Each cask is designed to face internal pressures up to 0.9 – 1.05 bar (they operate at a lower pressure than the external environment), temperatures up to 65 °C, and radiation field of about 100-300 Gy/h. The casks will be remotely controlled and self-propelled vehicles. A light gamma shielding will be installed only to protect the components of the transfer system that are sensible to the gamma field. Two different casks designed are investigated because of the different layout of the upper and the intermediate/lower ports.

Also for K-DEMO a RH system is under development /BONG 19/. The actual design is a mix between the solutions adopted for EU DEMO and CFETR. The blanket segments are lifted and removed through the upper port, but the machinery to cut/weld the coolant pipes is introduced in the VV through the equatorial ports. Instead, the lower ports are used to introduce the device to move the blanket segments in the VV. These movements are controlled anchoring the blanket segments to the VV rail system, as in the CFETR reactor concept.

A relevant difference of K-DEMO compared to the other DEMO concepts is the use of an intermediate maintenance enclosure above the upper ports /BROW 19/. Instead of the placement of each single IVC into a cask, for K-DEMO the building of a temporary metal-constructed intermediate maintenance enclosure is foreseen. This enclosure would be a demountable unit to allow the maintenance and the removal of the TF coils. The IVCs removed from the VV would be temporarily stored in the intermediate maintenance enclosure, and then sent to a Hot Cell where the IVCs would be connected to a coolant system to keep them cooled. After a period in this Hot Cell, the IVCs would be sent to the AMF.

For the JA DEMO two different remote maintenance strategies were considered in the past /UTOH 17/, but recently the solution similar to the EU DEMO one was selected as the preferred one

/UTOH 19/. A preliminary design of the system is available, but it is quite similar to the EU DEMO one.

B.17 Radioactive Waste System

B.17.1 Radioactive Waste Categorization and Inventory

In Europe, studies done in 2018 based on the DEMO 2015 baseline /GILB 17/, /GILB 18/ suggested that:

- For the WCLL blanket concept most of the blanket components become Low-Level Waste (LLW) in a 100-300 year timescale according to the IAEA classification system. Instead, for the HCPB blanket concept the blanket remains an Intermediate Level Waste (ILW) for more than 1,000 years. These figures are mostly due to the transmutation of N14 into C14 which is part of the EUROFER composition. In the HCPB concept an additional contribution is given by the impurities in the beryllium.
- The divertor body becomes - in all blanket concepts – a LLW after 100 years, but the W armour (PFC) remains a ILW for more than 1,000 years.
- More than 50% of the VV becomes ILW in 20 years in the HCPB and WCLL concepts. The VV become significantly LLW only after 200-500 years after shutdown.

More recent studies suggest that /PORF 20/:

- Less than 1% of the total beryllium used (< 5 kg) represent the most harmful radionuclides. For beryllium the "prevention" phase is considered of utmost importance being the long-lived radionuclides created mostly by its impurities, i.e. uranium and cobalt;
- About 3% of the total tungsten used will turn into long-lived radionuclides;
- The Pb used in the WCLL blanket concept will be disposed (no recycling).

Studies performed for JA DEMO show that 50,000 tons of waste might be generated in 20 years operation if most of the materials are not reused /SOME 16/ Y. Someya, et. al, Safety and Waste management studies as design feedback for a fusion DEMO reactor in Japan, Contribution given to the 26th IAEA Fusion Energy Conference, 17-22 October 2016, Kyoto, Japan. Paper ID: SEE/P7-5. https://nucleus.iaea.org/sites/fusionportal/Shared%20Documents/FEC%202016/fec2016-preprints/preprint0600.pdf.

/SOME 17/. This figure can be reduced to 10,000 tons reusing the blanket and divertor structural materials. Recycling the beryllide and lithium composite, the remaining wastes are classified as LLW after 10 years according to the Japanese regulation /TOBI 18/. The residual presence of Uranium (few kg) in beryllium is an issue of main concern due to the creation of transuranic elements. /SOME 18/ suggests that after 10 years the activities of these radionuclides are so low to allow shallow land disposal.

For K-DEMO the removal of the CuCrZr alloy from the divertor was proposed as solution to minimize the activity of the waste /PARK 17/. For this concept the operative life of each component is also studied to ensure that all the waste created are LLW /KIM 19b/.

No relevant data have been found for CFETR phase II.

B.17.2 Radioactive Waste System

The waste management in EU DEMO is still considered an open issue due to the unclear component replacement policy /PORF 20/. As a consequence, the waste stream and the methods to store the waste have been not yet defined, but it is clear that the radwaste system will adopt remote handling techniques to manage them. To date, only a multi-steps generic plan to reduce the waste amount has been proposed /PORF 20/, /DIPA 19/, /GILB 17/:

- Prevention, i.e. reduce as much as possible the impurities in the pre-use materials being in most cases the causes of higher activation. This is specifically the case of tungsten and beryllium;
- Detritiation. As a matter of fact, tritium will permeate all the IVCs and its removal would avoid the long-term disposal of the vast majority of the materials;

- Recycling of the Li_4SiO_4 pebbles and impurities minimization;
- Recycle structural materials having a contact dose below 2 mSv/h. In this phase the decarburization of ^{14}C in both EUROFER and AISI 316L steels is considered;
- Reclamation;
- Class reduction;
- Disposal.

Except for the prevention step, the remaining ones will be performed in the AMF.

Similarly, also for K-DEMO, JA DEMO, and CFETR phase II no designs of the radioactive waste system are available, but for JA_DEMO a shallow land burial is proposed **Error! Reference source not found.**.

B.18 Buildings

The tokamak building is the main building, while the tritium and diagnostic buildings are two outbuildings of the tokamak complex. The Active Maintenance Facility (AMF) is required for DEMO maintenance. Figure 30 shows a plausible arrangement of the buildings composing the EU DEMO plant.

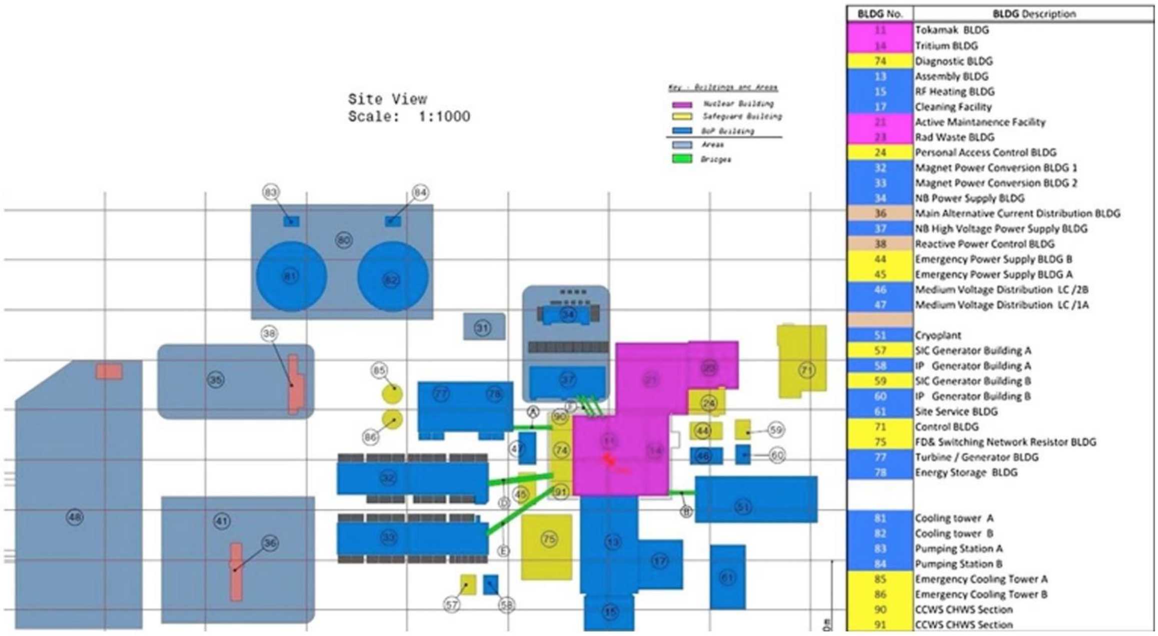

Figure 30 Overview of EU DEMO buildings /GARA 20/

B.18.1 Tokamak building

The design criteria of the tokamak building are mainly the functions, the overall safety, and maintenance approach /GLIS 18/. The tokamak building houses the tokamak and numerous plant systems, e.g. the PHTS, the H&CD system, the cryo-distribution system, the electrical power supplies, and HVAC system. Safety-related protection and mitigation systems are located in the tokamak building such as the TF coil quench detection system and part of the TF fast discharge system, the decay heat removal system, and the detritiation systems /BACH 19/. Also, the fire detection and suppression system is placed in the building. The building is designed to permit assembly and operation of the DEMO tokamak with 16 sectors, each with radial ports at three building floor levels and one vertical upper port. A large maintenance hall above the tokamak provides access for initial assembly, vertical BB maintenance and decommissioning. The large building volume is divided into smaller volumes to limit radioactive and fluid enthalpy inventories within the rooms. The pressure of the building is sub-atmospheric to avoid potential releases to the environment.

Around the tokamak a cylindrical bioshield with a thickness of 2 m is implemented encasing the cryostat and floor levels corresponding to the cryostat penetrations. The bioshield is a reinforced heavy concrete structure that fulfils three main functions: reduce the gamma radiation level to allow man-access to external areas during maintenance period, provide access to the cryostat, and – as part of the tokamak building structure – provide support to the equipment and building structures on top of the bioshield roof /CIUP 18/.

The tokamak building is designed to withstand seismic events with an expected reoccurrence every 10,000 y and overpressurization from the ex-vessel LOCA.

The radiation level in the tokamak complex due to activated flowing water has impact on the architecture of the building that improvement of concrete and shielding are expected /PALE 20/.

B.18.2 Tritium building

In the development of the Tritium building, two main issues shall be primarily addressed: develop the ventilation system according to general rules for hydrogen/tritium hazardous atmosphere and avoiding cross tritium contamination between various rooms and also between the utilities. /DOE 15/ is the main reference.

Due to high temperature operation of various components, tritium penetrates its barriers and can be released into the worker breathing area. Therefore, the ventilation system should be designed to meet the following main objectives:

- Remove the released tritium from the worker breathing space as soon as possible but minimizing the contamination of other areas while moving released tritium; the exhaust gas from each room should dump into a central exhaust duct and the exhaust gases from several rooms should not be combined before being dumped into the central exhaust duct;
- The ventilation system can be a single-pass ventilation system. Outside air is brought in through a supply fan, conditioned for the comfort of the workers, passes through the ductwork to the ventilated spaces one time, goes through the exhaust ductwork to an exhaust fan, and is released to the environment through the facility stack;
- The gases for each type of function, such as room ventilation, high velocity air hood ventilation, and glovebox ventilation, from a single room should not be combined until they reach the central exhaust;
- The ventilation control system shall be designed to hold the spaces occupied by the tritium operations at a negative pressure relative to the spaces surrounding the facility. If the whole building is a tritium facility, then the building is at a slightly negative pressure relative to the environment;
- The walls separating adjacent rooms should be reasonably sealed to minimize tritium released from contaminating an adjacent room. Airlocks shall be extensively used and administrative controls need to be put in place.

For facilities handling gram quantities of tritium, a rule of thumb is 6 to 10 air changes per hour as the standard of performance; As far as utilities are concerned such as floor drains, gas supplies and chilled water systems, the sharing with the adjacent non tritium areas shall be avoided. In the case of the same water-cooling system for cooling the tritium-related equipment is used to cool the non-tritiated office spaces, the leaks of tritium contaminated chilled water result in contamination of clean areas. In addition, if the floor drains from the non-contaminated areas drain into the same system as the floor drains from the tritiated areas, gases may flow from one area to another.

B.18.3 Active Maintenance Facility (AMF)

At the present stage, a design of the AMF is not available for the EU DEMO. The design will probably occur in the following design phase when the understanding of the waste inventory will be in a more mature stage of development. Although, in /THOM 14/ a credible overview of the main AMF zones is discussed (see Figure 31). The AMF will have three aims: a) store the casks containing the removed IVCs; b) dismantle the IVCs; and c) maintain the remote handing equipment /CROF 16/. The size the AMF building is likely to be larger than 700,000 m^3, i.e. almost 6 times the size of the ITER hot cell /THOM 14/. The AMF is likely to be divided into five zones: a IVCs transfer area, a IVCs processing area, a remote handling equipment storage area, a remote handling equipment maintenance area, and a waste conditioning area (see Annex B.17). The movement of the IVCs through the different areas is likely to be performed with casks. Two types of casks would be required, one to handle blanket segments and one divertor cassettes. Salient aspects foreseen for this facility area /THOM 14/, /LOVI 14/:

- The casks containing the IVCs should arrive already cleaned in the AMF (or cleaned once entered the building);

- The facility should ensure an adequate cooling of the irradiated IVCs. IVCs are likely to stay in this area for a period up to 18 months. After this period the decay heat should be sufficiently low to ensure a maximum component temperature of about 50 °C without active cooling system;
- A thorough decontamination must be ensured for each IVC. This cleaning is likely to be done into dedicated self-contained cells designed to handle specific components (cells for divertor cassettes, cells for blanket segments, etc.). The facility is likely to be designed to allow parallel operations in order to reduce the duration of the maintenance activities;
- The AMF must have the capacity to cope with breakdowns and maintenance of the remote handling facilities and equipment;
- Up to this phase, all these activities should be done remotely, without any operator in place. Operators might be involved in the inspections and maintenance activities only after the cleaning of the IVC;
- Each zone of the building should be shielded to cope with the high dose rates of some IVCs;
- A physical separation between irradiated and fresh IVCs (independent routes from-to the tokamak). At any time, it is foreseen to have a full load of fresh IVCs in the AMF;
- The facility should be prepared for the next long-term maintenance outage when the reactor operates at full power to reduce the maintenance duration.

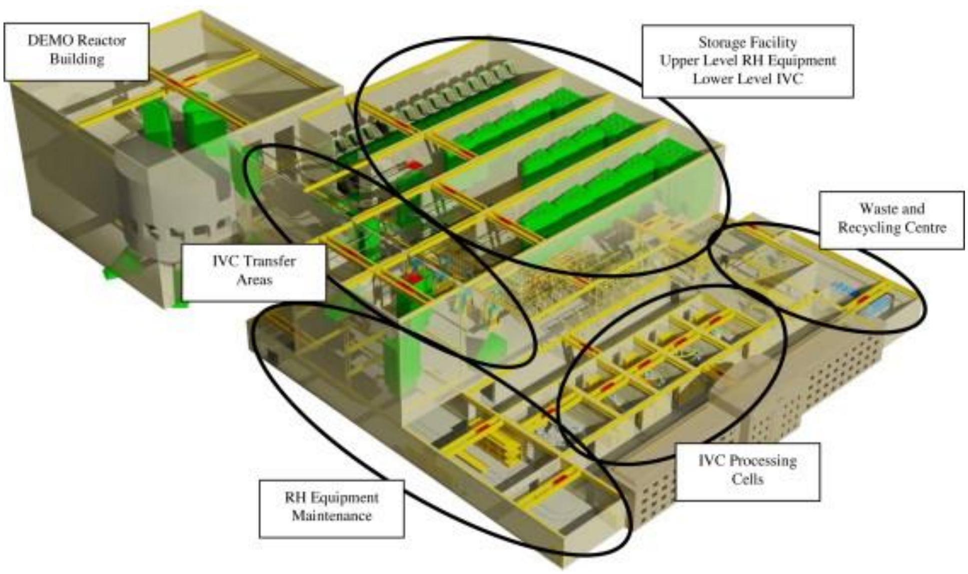

Figure 31 Overview of the AMF in EU DEMO /THOM 14/

Instead, the design of the AMF in the CFETR reactor (called Hot Cell as in ITER) seems in a more mature stage of development /GONG 15/ (see Figure 32 and Figure 33). The Hot Cell is a 4 floor building nearly 200 m long and 150 m wide. The layout of the building follows the workflow of the operations that have to be performed: the IVC shipment arrived at the docking station is sent to the specific floor handling the type of the IVC (blanket, divertor, control & diagnostic equipment, etc.) where it is cleaned. Then, the shipment is sent to the decay heat removal cell equipped with an air Heating, Ventilation Air Condition (HVAC) system. The use of water in the HVAC is not foreseen due to the presence of tritium trapped inside the IVCs. The shipment stays for a long period in this area, and then it is sent to the sub-maintenance cell where the IVC is cleaned from dust and decontaminated. Once this pre-treatment is over, the IVC is sent to the maintenance cell where inspections, reparations, and refurbishments take place. The waste created in this cell are sent to the RTF, while the refurbished components are sent to a test section to assess their properties. If the IVC passes the required tests, it is taken by the remote handling system and transported back into the tokamak. As in EU DEMO, also in CFETR the maintenance of the remote handling equipment is done in this building. The process is similar to that of the IVCs, but no storage in the decay heat removal cell is foreseen due to low activation expected in this equipment. Differently to the EU DEMO, there is no mention about the route separation of new and refurbished elements.

In turn, for JA-DEMO only a radwaste management scenario has been found **Error! Reference source not found.**. This scenario foreseen four phases:

- The first phase takes place in the reactor. Here the IVCs to be removed are de-tritiated and cleaned from dust prior their replacement.

- The second phase is in the hot cell. Here two different routes are followed depending if the IVC is a divertor cassette or a blanket segment. If it is a divertor cassette, it is stored for about 1 year, then it's disassembled and sent to a temporary storage for 4 years. In case of a blanket segment, the storage in the hot cell lasts for 6 years, and the temporary storage after disassembly lasts for 10 years.
- The third phase takes place in the packing/interim storage. Here the waste is classified and packed. The wastes remain in the interim storage for a time between 20 and 50 years, and then they are sent to disposal (shallow land burial, fourth phase of the process).

No relevant data have been found for K-DEMO.

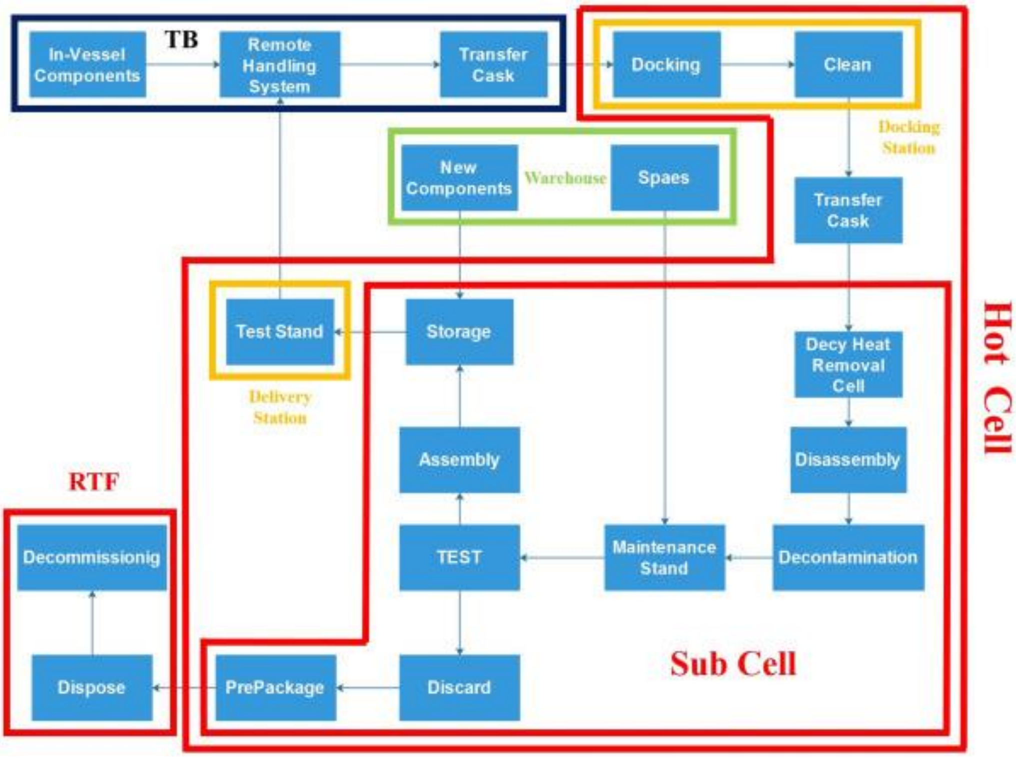

Figure 32 Arrangement of the hot cell in CFETR phase II /GONG 15/.

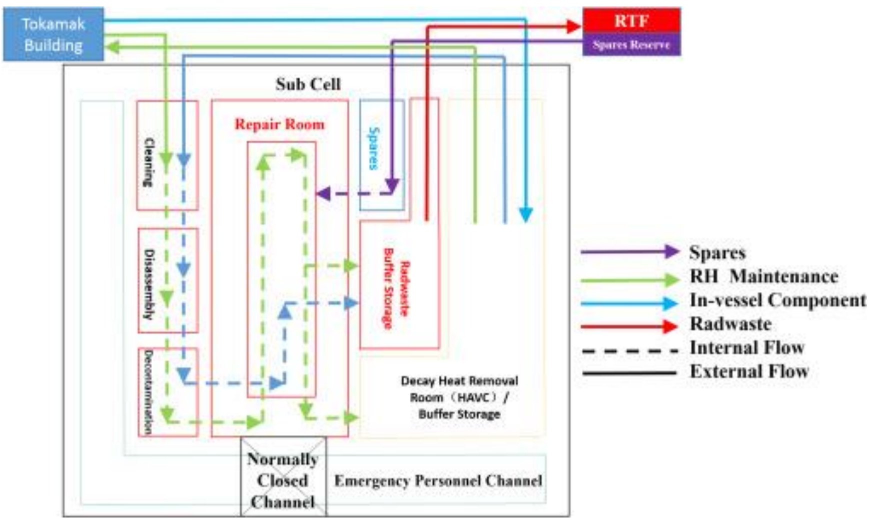

Figure 33 Streams crossing the hot cell in CFETR phase II /GONG 15/.

B.19 Plant Electrical Supply (PES)

The Plant Electrical Supply (PES) system has a dual aim: to supply the plant power to the electrical loads, and to deliver electrical power to the electrical grid /GAIO 20/. The PES has been investigated to estimate the amount of power required to sustain the plasma burning, and to verify eventual limit to provide it from the Turbine Generator (TG) and the Power Transmission Grid (PTG). The outcomes of the first analysis showed that the solutions adopted for ITER have to be reconsidered because of the higher power peak envisaged, and to the handling of the generated and recirculated power issues. Moreover, the reconsideration of the adopted solutions allows the introduction of new technological progresses. The PES includes also the emergency power supply for safety classified electrical loads and on-site emergency diesel generators and electric batteries in case of loss of off-site power.

Up to date, a detailed design of the PES is not available, but the preliminary design adopts a subdivision in 5 subsystems /GAIO 20/:

- TG to generate electrical power;
- AC High Voltage switchyard/power distribution to deliver the electrical power to the electrical grid;
- Coil Power supply system to provide the required power to the SC coils;
- H&CD power supply system to provide the required power to the H&CD systems;
- Steady state electrical network to provide the required power to the remaining users within the plant.

Due to the pulsed nature of the DEMO machine, two solutions are under investigation to connect the PCS with the PHTS /GAIO 20/ directly coupling, or indirect coupling with the use of an IHTS equipped with an ESS (see Figure 34). The interface with the High Voltage (HV) external PTG will be studied once the site for construction will be defined. The preliminary design of the HV power supply for NBIs is similar to that of ITER (the EU DEMO and ITER systems have almost the same requirements), but investigations are on-going to remove the thyristors and to introduce Modular Multilevel Converters (MMC).

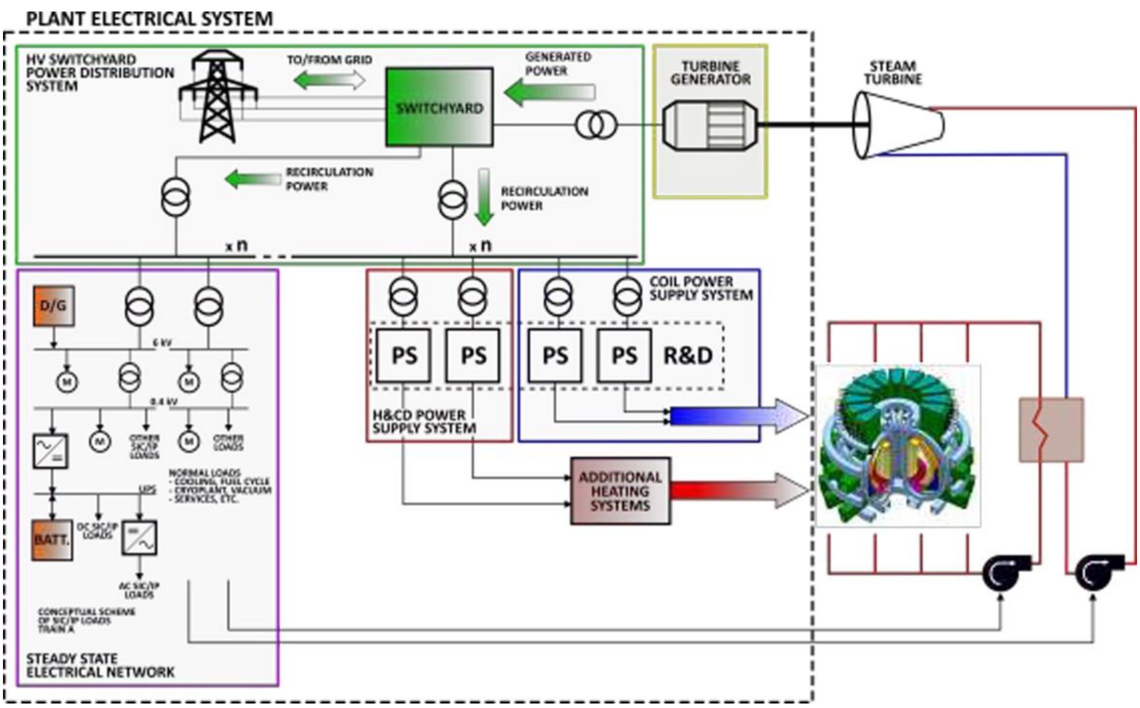

Figure 34 Schematic of the EU DEMO PES /GAIO 20/

The main issues found in the applicability of the ITER design to the EU DEMO reactor are /GAIO 20/:

- The pulsed nature of the DEMO machine lead to a repetitive request of power to allow the plasma formation. This request can be more than 1 GW with the CS and the PF coils requiring about 45 kA and 20 kV. The approach followed in ITER based on the thyristor converters seems not practicable due to the high reactive power (>2 Gvar). The main alternative under development for EU DEMO is the Magnetic Energy Storage and Transfer system (MEST). In this technology, the energy required by a "Load Coil" (LC) is stored in a "Sink Coil" (SC) and exchanged between the two when requested. Another technology under investigation is the voltage source converters (VSC) with active front end (AFE) consisting in a cascade connection of bridges with fully controllable power switches;

- The number of feeders penetration in the bioshield and busbars in the Tokamak Building. CS and PF coils are two independent systems, while the subdivision of the TF coils into independent system has been not yet defined. The number of sectors has not been yet defined since it is linked to the improvements in the FDUs in handling the maximum voltage across the coil and at the coils terminal versus ground (10 kV and 5 kV, respectively).

- The location of the FDU Circuit Breakers. In the current DEMO design they are placed in the tokamak building, but the environmental conditions pose a challenge in the serviceability of such components. Moreover, specific paths in the tokamak building should be created for the operators carrying out inspections and maintenance activities to the circuit breakers.

Outside Europe, preliminary considerations on the adoption of the ITER solutions to CFETR phase I have been found /LI 14/. As for EU DEMO, studies are planned to remove the thyristors with alternative technologies, but they are all in an early stage of development. No relevant data have been found for JA DEMO and K-DEMO.

B.20 Plant Control System

Like in ITER /JOUR 20/, DEMO plant control system is going to be designed including three systems:

- Control and data acquisition system, to provide overall control of the DEMO plant

- Central interlock system, to provide plant-wide investment protection functions of all SSCs but SIC

- Central safety system, to provide plant-wide nuclear and occupational safety functions through the plant safety system for the SSCs safety classified.

ANNEX C PROCEEDING OF THE WORKSHOP

In the following the proceedings of the workshop are given. The annex B-H to the workshop consisting of the slides shown, the the list of participants and the issues to be updated in the report are not included.

Proceedings to the online workshop

On the applicability of the regulatory framework for nuclear facilities to fusion facilities. Towards a specific regulatory framework for fusion facilities.

November 22nd – 24th, 2021. 12:00/13:00 – 16:00 CET

GRS and KIT have been awarded a contract by the European Commission to perform a study on the applicability of the regulatory framework for nuclear facilities to fusion facilities. The results of this study have been presented at a dedicated workshop to international fusion experts. Experts from licensees, research organisations, technical safety organizations, international organisations and national regulators participated to the workshop.

The main goal of the workshop was to obtain the feedback from the participants on the presented results of the study. Time for in-depth discussion of the presented study was included in the agenda. To allow for the preparation of the participants and to facilitate the discussion a draft version of the study report was distributed prior to the workshop to the participants.

Summary of the Study

The study was carried out in four different tasks. In the first task, the current regulatory framework has been screened and reviewed for different countries: China, Korea, Russia, Germany, France, U.K., and USA. None of these countries has a dedicated comprehensive fusion-specific regulatory framework for the whole lifecycle from siting to decommissioning. Thus, requirements for fission facilities as well as radiation protection requirements were applied to regulate fusion facilities.

In the second task of this study specific safety requirements for fusion facilities have been derived, focusing on facilities using magnetic confinements and tritium as plasma component. The safety requirements are based on the generic differences between fusion and fission facilities, an assessment of the safety issues of fusions specific systems, structures and components, the screening of European Directives, IAEA safety standards, IAEA safety guides, other fusion relevant IAEA publications, and further international fusion specific publications.

The results of the third task are recommendations on the implementation of a legal and regulatory framework specifically for fusion facilities. They are based on the existing IAEA safety standards and European Directives. The general safety requirements should cover topics like siting, leadership and management, safety assessment, decommissioning. The safety concept of fusion facilities should include safety objectives, defined by fundamental and supporting safety functions, the defence in depth concept, the concept of multi-level confinement of the radioactive inventory, the protection against internal and external hazards, a graded approach based on the intrinsic hazard potential of a certain facility, mainly the radiological hazard potential, and a system for operating experience feedback. Also, the interface between safety, security and safeguards should be considered.

An action plan of the steps necessary to implement the legal and regulatory framework has been developed in task 4 which identifies the steps and the stakeholders involved.

Presentation: Overview of the project (Björn Becker, GRS)

Mr. Björn Becker from GRS gave in his role as project lead an introduction to the workshop and a general overview over the project carried by GRS and KIT. He briefly presented the different tasks and laid out their objectives.

The slides of the presentation are given in Annex B.

Presentation of Task 1: International approaches to fusion regulation and existing safety requirements (Joachim Herb, GRS)

The second presentation was delivered by Mr. Joachim Herb from GRS, lead of task 1. Mr. Herb presented the approach to the assessment of the international approaches to fusion regulation. He summarizes the results of the screening of the national approaches of:

- France,
- United Kingdom,
- Germany,
- Russia,
- USA,
- China, and
- Korea.

The general conclusion of the assessment was that:

- Some countries have regulated fusion facilities (e.g. UK, France, Germany) with/without tritium
- There are ongoing discussions and developments on how to advance the regulation of fusion facilities with different stakeholders (e. g. USA, China, and Korea)
- In some countries the application of the regulation for nuclear power plants is limited to facilities using fissile material (uranium, plutonium, thorium), e.g. Germany, China
- There are different legal definitions of the term "nuclear facility
- Regulations for radiation facilities and radiation protection apply and form the basis for licensing
- There seems to be a gap for licensing larger fusion facilities or FPP with respect to the higher radiological hazard potential of them compared to typical radiation facilities and with respect to the lower radiological hazard potential of them compared to fission power plants
- In the licensing of the ITER and JET existing regulations for facilities like research and/or commercial reactors ("INB") and like radiation generating facilities ("other non-nuclear licensed sites") were be applied to fusion facilities

Furthermore, Mr. Herb discussed the importance of the application of a graded approach and the different characteristic of a prescriptive or goal-oriented approach for regulation. He concluded his presentation by highlighting aspects which might be considered by future fusion regulations.

The slides of the presentation are given in Annex C.

Following the presentation various participants commented on specific aspect of the national approaches (UK, USA, Russia, France). Participants highlighted to further clarify the approach undertaken by the regulators NRC and ASN. Experiences of the regulation of ITER in France was shared by several participants. In addition, a discussion on the green paper of the UK as well as on the classification of JET could be added to the report. It was underlined that the hazard potential of JET is much lower than that of a fusion power plant.

Participants also addressed the recent international efforts by the IAEA such as drafting two TECDOCs and comparable studies for the application of the current regulatory framework for SMR.

Presentation of Task 2-1: Assessment of main differences between fission and fusion and safety issue for specific fusion SSCs (Xue Zhou Jin, KIT and Joachim Herb, GRS)

The third presentation of the workshop was a combined presentation of Ms. Xue Zhou Jin from KIT and Mr. Joachim Herb from GRS. In his part of the presentation Mr. Herb discussed the assessment of main differences between fission and fusion. He addressed aspects such as consequences of a purely hypothetical release of large amounts of radioactive inventor and the radioactive inventory of the facility. The confinement concept of NPPs was mentioned and which similar aspects are relevant for fusion facilities. Furthermore Mr. Herb addressed briefly relevant aspects such as operational experience, decay heat removal, Leak-before-break concept, as well as the fundamental safety functions. Finally, he discussed the basics of safety concepts.

In the second half of the presentation Ms. Jin discussed safety issues for specific fusion SCCs of Tokamak machine with magnetic confinement. Based on the example of DEMO Ms. Jin addressed the following aspects

- Energy sources
- Radioactive source terms
- internal hazards
- external hazards
- Ionizing radiation
- no-radiological hazards
- Activated materials & activation products
- Functional Failure Mode and Effect Analysis (FFMEA) & Postulated Initiating Events (PIEs)
- Accident analyses (WPSAE)
- Confinement strategy
- Radioactive waste management

The presentation continued with explanations of the functionalities and the safety issues of each of the main system of EU DEMO. Due to the significant number of systems, not all the slides were presented during the workshop.

The slides of both parts of the presentation are given in Annex D.

The presentation was followed by a short discussion on the radioactive inventory of fusion power plants. Both inventory of the vacuum vessel and the Tritium building was addressed. Different estimations of the total amount were stated.

Presentation of Task 2-2: Screening and review of international regulations, safety standards and guides and other documents (Thorsten Stahl, GRS)

Mr. Thorsten Stahl of GRS presented the second part of task 2 presenting the screening and review of international regulations, safety standards and guides and other documents. In his presentation Mr. Stahl laid out the approach used for review. Regarding the EU Directives he discussed the directives specific for nuclear and radiation facilities and activities, i.e.:

- European Basic Safety Standards (2013/59/EURATOM)
- Nuclear Safety Directive (2009/71/EURATOM)

- Radioactive Waste and Spent Fuel Management (2011/70/EURATOM)
- Nuclear Safeguards (Commission Regulation (Euratom) No 302/2005)

He concluded that with exception of the European Basic Safety Standards, the other directives are not directly applicable for fusion facilities. Mr. Stahl also addressed briefly directives dealing with conventional occupational health and safety. These directives provide general rules and requirements which have to be fulfilled by fusion facilities.

Mr. Stahl continued by presenting the screening of IAEA Safety Standards. These represent the international consensus on nuclear safety and form a regulatory framework for nuclear facilities and activities. While the safety fundamentals (SF-1) and the general safety requirements (GSR) can be applied since they are high level and general requirements, the specific safety requirements of the IAEA have been established for specific types of facilities and are not directly applicable to fusion facilities. Mr. Stahl continued with the presentation of the screening of IAEA guides. These were classified into three categories (Green: Guide applicable, Yellow: Could complement requirements if done in a propriate and target manner, Red: not applicable)

Mr. Stahl briefly discussed the screening of fusion specific publications such as IAEA TECDOCs and DOE standards. He concluded his presentation by stating that a legal and regulatory framework for fusion power plants needs to be established and that it is recommended to close this gap by developing a regulatory framework for fusion power plants in a proportionate and targeted manner.

The slides of the presentation are given in Annex E.

Following the presentation various participants commented on the screening of IAEA safety standards. It was stated that the classification of the guides needs more clarification. Some stated applicability needs to be reconsidered in the report, in particular with regard to the application of PSA (SSG-3, SSG-4). In addition, it was mentioned that the evaluation of the applicability might also change when going further into an in-depth evaluation. The applicability of guides will also significantly depend on the design of the fusion facility, as well as on the overall national regulatory regime. Participants also commented that even if a guide is considered as applicable or applicable after adaption, this should be understood as that the topic is applicable. Applicable topics should be grouped and simplified taking into account the lower hazards from fusion power plants.

Participants saw the adaptation of guides to the reality of a fusion facility as a challenge. Participants also pointed out the specific guidance for Tritium industry might be needed when going to larger implementations of fusion power plants.

Furthermore, it was discussed among participants to which extent fusion regulation should be based on the unmitigated risk of a fusion facility and that it is important to recognize that fusion is not fission.

Presentation of Task 3: Recommendations for the implementation of the safety requirements specifically needed for fusion facilities (Isabel Steudel, GRS)

Ms. Isabel Steudel of GRS presented the results of task 3 on recommendations for the implementation of the safety requirements specifically needed for fusion facilities. In the first part of her presentation, Ms. Steudel addressed the recommendations for the basic safety concept for fusion power plants consisting of fundamental safety objectives, fundamental safety functions, supporting safety functions, and technical safety concepts.

Ms. Steudel stated that the fundamental safety objective of SF-1 to protect people and the environment from harmful effects of ionizing radiation applies to fusion facilities. Following SF-1, to ensure this fundamental safety objective, the following measures have to be taken to control the radiation exposure of people and the release of radioactive material to the environment; to restrict the likelihood of events that might lead to a loss of control over a radioactive source or any other source of radiation; and to mitigate the consequences of such events if they were to occur. As fundamental safety functions the confinement of radioactive material and control of planned radioactive releases, as well as limitation of accidental radioactive releases and the shielding against radiation in normal operation and accident conditions were identified. As supporting safety functions control of plasma stability, removal of process and residual heat and cooling of superconducting magnets below critical temperature were proposed. Ms. Steudel stressed that depending on the design safety function might inherently be fulfilled.

Ms Steudel briefly discussed the application of the technical safety concepts of Defence-in-depth, multi-level confinement of radioactive inventory, protection against hazards, and the graded approach based on risk categories.

In the second part of the presentation, Ms. Steudel discussed fusion specific safety aspects which might be considered in requirements of fusion regulations addressing:

- Vacuum vessel
- breeding blankets and FW
- limiters and divertor
- primary heat transfer system
- cryostat and cryosystems
- tritium extraction and removal system
- heating and current drive systems
- remote maintenance
- active waste management facilities and radwaste systems
- electric power supply

She concluded that to establish a legal and regulatory framework for the safety of fusion facilities the following steps will be necessary:

- involvement of various stakeholders (int. organisations, regulatory authorities etc.) providing specific competences
- proposed int. regulatory framework will serve as kind of blueprint to be implemented in nat. regulatory frameworks and adapted to specific boundary conditions
- high-level safety objectives and requirements, which can be implemented in nat. laws and decrees (for Europe e.g. council directive)

The contribution to develop the documents forming the regulatory framework will also vary between the different levels. On the highest-level, the role of regulatory authorities is dominant, with an increasing level of technical details, the contributions shift more and more towards the technical experts and industry.

The slides of the presentation are given in Annex F.

Following the presentation various participants commented on the general approach towards fusion regulation. Participants stated that it is important that the regulatory approach takes into account the risk profile of the fusion power plant. It was stated that a goal oriented, technological neutral approach for new technologies has more flexibility, but requires more efforts by the applicant. It was mentioned that goal oriented technological neutral regulation is also the trend for new advanced fission reactors. Regarding the fusion specific aspects, participant commented that when developing the regulatory approach, high level objective and principles should be developed first. During this process the need for more detailed requirements and guides will be clearer. It was mentioned that setting Safety Requirements at the current stage might be too soon, as they presume a certain design and level of hazard of fusion plant.

Presentation of Task 4: Action plan for the development of a targeted and proportionate regulatory framework for the construction, operation and decommissioning of fusion facilities (Joachim Herb, GRS)

Mr. Joachim Herb of GRS presented the results of task 4 on the steps of an action plan how to establish a common European legal and regulatory framework for fusion devices. For each step he identified the main responsible stakeholders and the ones whose support is needed.

In the presented action plan the first step is to establish a common European legal framework to regulate fusion using the experience of the existing 2009/71/Euratom Council Directive. Other steps to establish the legal framework should be to discuss and agree on a defence in depth concept for fusion facilities and to develop a graded approach concept for fusion facilities. Stakeholders involved in these steps were the European Commission, the EU member states, the IAEA, the national regulatory authorities, research organisations, the fusion industry and technical safety organizations.

Then Mr. Herb outlined the steps necessary to create a regulatory framework for fusion facilities. These steps include the development of high-level "Safety requirements for D-T fusion facilities with magnetic confinement", the establishment of a system for operating experience feedback, the development of safety guides for different aspects of fusion facilities, and the development of fusion specific codes and standards. The stakeholders in these steps were identified by Mr. Herb as the European Commission, the OECD/NEA, the IAEA, the national regulatory authorities, research organizations, the fusion industry, technical safety organizations, and standardization organisations (ISO, EC, ASME, IEEE, etc.).

The slides of the presentation are given in Annex G.

Following the presentation various participants commented on the proposed action plan. Some excerpts of the ITER safety report for beyond design basis accidents were discussed that were showing the calculated source terms of tritium and radioactive dust and the corresponding doses at two different positions outside ITER. It was questioned by some of the participants of the workshop to which extent future fusion facilities will need a detailed regulatory framework as proposed in this study.

Other participants suggest the first step of the action plan to be completed should be the development of design guidelines, because they would be need soon and a design based on them could be presented to a regulator following a goal-orientated approach for licensing. Another comment was the step to define the high-level safety requirements should started soon.

Wrap-up of the meeting (Joachim Herb, GRS)

Mr. Joachim Herb of GRS wrapped up the presentations and the discussions during the three days of the workshop. The participants of the workshops came from eight countries (Canada, China, France, Germany, Russia, South Korea, UK, USA) and three international organizations (IO, EC, IAEA). They represented different stakeholders (plant constructors, research institutions, regulators and associated organizations).

Mr. Herb concluded that the aim of the workshop, namely to have detailed discussions also including controversial views was reached. He thanked the participants for their active participation at the workshop and emphasized the need for future international collaboration on the topic of the regulation of fusion facilities. He summarized the feedback for the different talks and how GRS and KIT will include this feedback in their final report.

The slides of the presentation are given in Annex H.

Annex A: Agenda of the Workshop

Time	Topic
November 22nd	**Chair: Björn Becker (GRS)**
12:15 – 12:30	Welcome, Introduction by EC and project team, topics of the day
12:30 – 13:15	Introduction of the participants
13:15 – 13:30	Overview of the project (Björn Becker, GRS)
13:30 – 14:00	Presentation of Task 1: International approaches to fusion regulation and existing safety requirements (Joachim Herb, GRS)
14:00 – 14:45	Feedback from participants on national approaches
14:45 – 15:00	Break
15:00 – 15:30	Presentation of Task 2: Assessment of main differences between fission and fusion and safety issue for specific fusion SSCs (Xue Zhou Jin, KIT and Joachim Herb, GRS)
15:30 – 15:45	Wrap-up of day 1 and adjourn
November 23rd	**Chair: Björn Becker (GRS)**
13:00 – 13:15	Welcome and topic of the day
13:15 – 13:45	Presentation of Task 2: Screening and review of international regulations, safety standards and guides and other documents (Thorsten Stahl, GRS)
13:45 – 14:15	Feedback from participants on Task 2
14:15 – 14:45	Presentation of Task 3: Recommendations for the implementation of the safety requirements specifically needed for fusion facilities (Isabel Steudel, GRS)
14:45 – 15:00	Break
15:00 – 15:30	Feedback from participants on Task 3
15:30 – 15:45	Wrap-up of day 2 and adjourn
November 24th	**Chair: Joachim Herb (GRS)**
13:00 – 13:15	Welcome and topic of the day
13:15 – 13:45	Feedback from participants on Task 3
13:45 – 14:15	Presentation of Task 4: Action plan for the development of a targeted and proportionate regulatory framework for the construction, operation and decommissioning of fusion facilities (Joachim Herb, GRS)
14:15 – 14:30	Break
14:30 – 15:30	Feedback from Participants and discussions of the action plan
15:30 – 15:45	Wrap-up of the meeting and next steps, closure of the meeting

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