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Integrated Approach to Safety Classification of Mechanical Components for Fusion Applications

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International Atomic Energy Agency

INTEGRATED APPROACH TO SAFETY CLASSIFICATION OF MECHANICAL COMPONENTS FOR FUSION APPLICATIONS

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INTERNATIONAL ATOMIC ENERGY AGENCY
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FOREWORD

Since the 1970s, fusion safety has been on the IAEA's agenda of safety activities. Over the past four decades, the IAEA has organized several workshops and Technical Meetings on fusion safety, and has produced a number of publications on the topic, including Fusion Safety (IAEA-TECDOC-277), published in 1983; Fusion Safety Status Report (IAEA-TECDOC-388), published in 1986; Fusion Reactor Safety (IAEA-TECDOC-440), published in 1987; ITER Safety (ITER DS/36), published in 1991; and Technical Basis for the ITER Final Design Report, Cost Review and Safety Analysis (FDR) (ITER EDA DS/16), published in 1998. However, none of these included an assessment of safety classification of components for fusion applications.

This publication addresses the need for information on fusion specific applications considering existing industry practices. This publication is expected to be reviewed and updated regularly to reflect the progress in the field. It represents the current state of the art thinking on safety classification of components for fusion applications. It is hoped that it will be used for stimulating studies and further enhancing international collaboration in this subject area.

The IAEA wishes to express its appreciation to all the contributors to this publication, and, in particular, M. Barbarino (Italy) who compiled, elaborated and reviewed the first complete version of the text. The IAEA officer responsible for this publication was S.M. Gonzalez de Vicente of the Division of Physical and Chemical Sciences.

CONTENTS

1. INTRODUCTION

1.1. Background

Nuclear fission is currently the process mainly used to generate nuclear energy. Nuclear reactors operate on the principle of nuclear fission, the process in which a heavy atomic nucleus splits into two smaller fragments. The fission fragments are in very excited states and emit neutrons, other subatomic particles, and photons. The emitted neutrons may then cause new fissions, which in turn yield more neutrons, and so forth. Such a continuous self-sustaining series of fissions constitutes a fission chain reaction. A significant portion of the energy of fission is converted to heat the instant that the fission reaction splits the initial target nucleus into fission fragments. The bulk of this energy is deposited in the fuel, and a coolant is needed to remove the heat to maintain a balanced system (and to transfer the heat energy to the power generating plant). The most common coolant is water. If there was a defect in the cooling system of the reactor that allowed one or more of the nuclear fuel elements to exceed its melting point, meltdown could occur, i.e. a critical accident in which severe overheating of the nuclear reactor results in the melting of the reactor's core.

Nuclear fusion is a process by which nuclear reactions between light elements form heavier elements (up to iron). In cases where the interacting nuclei belong to elements with low atomic numbers (e.g. hydrogen or its isotopes deuterium and tritium), energy is released. The idea of using these reactions is that if the products of the reaction can be made to slow down in the fusing medium (plasma), they can be used to help maintain the reaction temperature. In addition, if neutrons are produced they can escape the medium to heat up the surrounding materials, where coolants can then be used at high temperature to generate electricity using a conventional steam or gas turbine.

Fusion reactors get heated to temperatures much higher than the core of the sun which converts hydrogen gas into a hydrogen plasma. Strong magnetic fields or high-powered lasers then confine the plasma into a small controllable region where fusion can happen. Efforts towards the realization of fusion energy have so far focused on mainly two schemes: the magnetic confinement approach and the inertial confinement approach. Magnetic confinement uses strong magnetic field to confine the fuel particles in gas form whereas inertial confinement uses fuel particles compressed to very high density. In both schemes the fuel consists of a mixture of deuterium and tritium.

The more advance technology is now present in the magnetic confinement approach, particularly in using the so-called tokamak concept for a fusion reactor. The current largest tokamak facility is the Joint European Torus (JET), a multinational European venture operated in England. In 1997, JET generated 16 megawatts of peak power with a fusion gain (the ratio of fusion power produced to the net input power) of 0.6. The next major step aimed at the demonstration of fusion power generation is the International Thermonuclear Experimental Reactor (ITER) currently under construction in Cadarache, France. This is a very large experiment that will investigate both the plasma physics and reactor technology. The current participants to the project are the European Union (represented by EURATOM), Japan, the People´s Republic of China, India, the Republic of Korea, the Russian Federation and the United States of America.

The main ITER Technical Objectives are:

— Achieve inductive plasma burn with power amplification, *Q* (ratio of fusion power to auxiliary heating power), of at least 10, under stationary conditions on the timescales of plasma processes;
— Aim at demonstrating steady-state operation with $Q > 5$;
— Do not preclude the possibility of controlled ignition;
— Integrate the technologies essential for a fusion reactor (e.g. superconducting magnets, remote maintenance);
— Test components for a future reactor (e.g. divertor and torus vacuum pumps);
— Test tritium breeding module concepts for DEMO (demonstration fusion power plant).

The next step in making fusion a commercial available source of energy is DEMO. It will be an electricity producing fusion reactor prototype, also based on a tokamak concept, which will deliver about 1 GW electric power steady-state or quasi-steady state. It will be slightly larger than ITER. Specifically, it will operate at higher density and will require somewhat higher confinement and stability margins.

Safety classification of a structure, system, or component (SSC[1]) in the fusion projects is important as it determines the design quality and manufacturing requirements. Ranking SSCs according to their significance to safety helps to determine the design, quality and manufacturing requirements to be applied to individual SSCs and helps to follow a more graduated approach. Classification is a top down process that begins with a basic understanding of the plant design, its safety analysis and how the main safety functions will be achieved. Based on the classification, a complete set of engineering rules needs to be specified which then dictate the codes and standards that are used by the designers. Classification is an important element in any design process and has to be developed at the earliest stage of the design development because without classification the detailed design cannot be substantiated.

1.2. Objective

Safety classification of SSCs used in nuclear power plants [1–3] is provided in the IAEA Safety Guide SSG-30 [2] and other international standards but they are mostly aimed at the fission applications. There are some notable differences between the fission and fusion applications when safety issues are considered. For example, in the fusion applications, there is no reactivity control or emergency cooling requirements and no core melt conditions to be addressed. Prevention of core meltdown is not a safety related function for fusion reactors. Most of the main parameters like the safety functions, consequences of failure, fault frequencies, contributing to the safety classification process in fission applications are different for the fusion applications. However, in the fusion nuclear facilities there are other kind of accidents that can be postulated due the fact that a tokamak is a complex and dense zone with many different energy source terms (plasma, coils, cooling systems, helium). In addition, the first confinement barrier, which is surrounded by those energy source terms, has a very complex boundary.

This publication is a compilation of the work carried out by selected experts in the field of fusion research to provide guidance on how to use the knowledge from the safety

[1] See List of Abbreviations.

classification process to help design a component by selecting appropriate design codes, and, to help substantiate the design by knowing the failure modes and the allowable damage limits. Methods currently used in the two main fusion projects (ITER and DEMO) to classify components important to safety are described and finally guidelines for engineering design rules are discussed. The approach presented herein integrates design and safety processes.

This TECDOC considers the current practice, highlights the differences in the approaches used to identify and classify SSCs that are important to safety and offers guidance for fusion applications. In addition to considering the variations from the fission applications, the report also provides guidance on inclusion of the new Design Extension Conditions (DEC) which have been added after the review of IAEA Safety Guides following the Fukushima Daiichi NPP accident. In addition, the recently amended EU Directive on Nuclear Safety [4] is considered.

1.3. Scope

The scope of this TECDOC is limited to magnetic confinement applications tokamaks. However, the principles of safety classification of mechanical components are basically the same for inertial and magnetic fusion devices.

The current scope of this publication excludes any Remote Handling (RH) equipment and Electrical Controls and Instrumentation (EC&I) components. The need for this guidance was highlighted in Refs [1,2] where the expert opinion was to give urgent attention to this topic.

1.4. Structure

The report is structured as follow (see also Fig. 1):

(a) Section 2 gives an outline of a general approach to safety classification and highlights the safety requirements for fusion applications.
(b) Section 3 presents safety classification process and provides more details on plant states including typical normal, abnormal, fault and accident conditions considered for fusion reactors and classifies them based on the frequency of occurrence. It also includes more details on safety functions and requirements for fusion reactors and the various plant states.
(c) Section 4 provides guidance on allowable damage limits and Safety Important Components (SIC) grading, and shows how the safety classification impacts the other classifications related with quality, seismic and design.
(d) Section 5 discusses various failure modes of mechanical components.
(e) Section 6 links the Failure Modes and Effects Analysis (FMEA) with the allowable damage limits that will help in design substantiation.
(f) Section 7 provides guidance on engineering design rules and links SIC grading with design codes. Recommendations on design codes are made that can be used to meet the design requirements
(g) Section 8 describes the link between safety analysis and structural assessment.
(h) Sections 9 and 10 deal with quality and nuclear design qualification and verification issues.
(i) Section 11 summarises the conclusions and remarks.
(j) Annex I considers environmental effects.
(k) Annex II covers the material data.

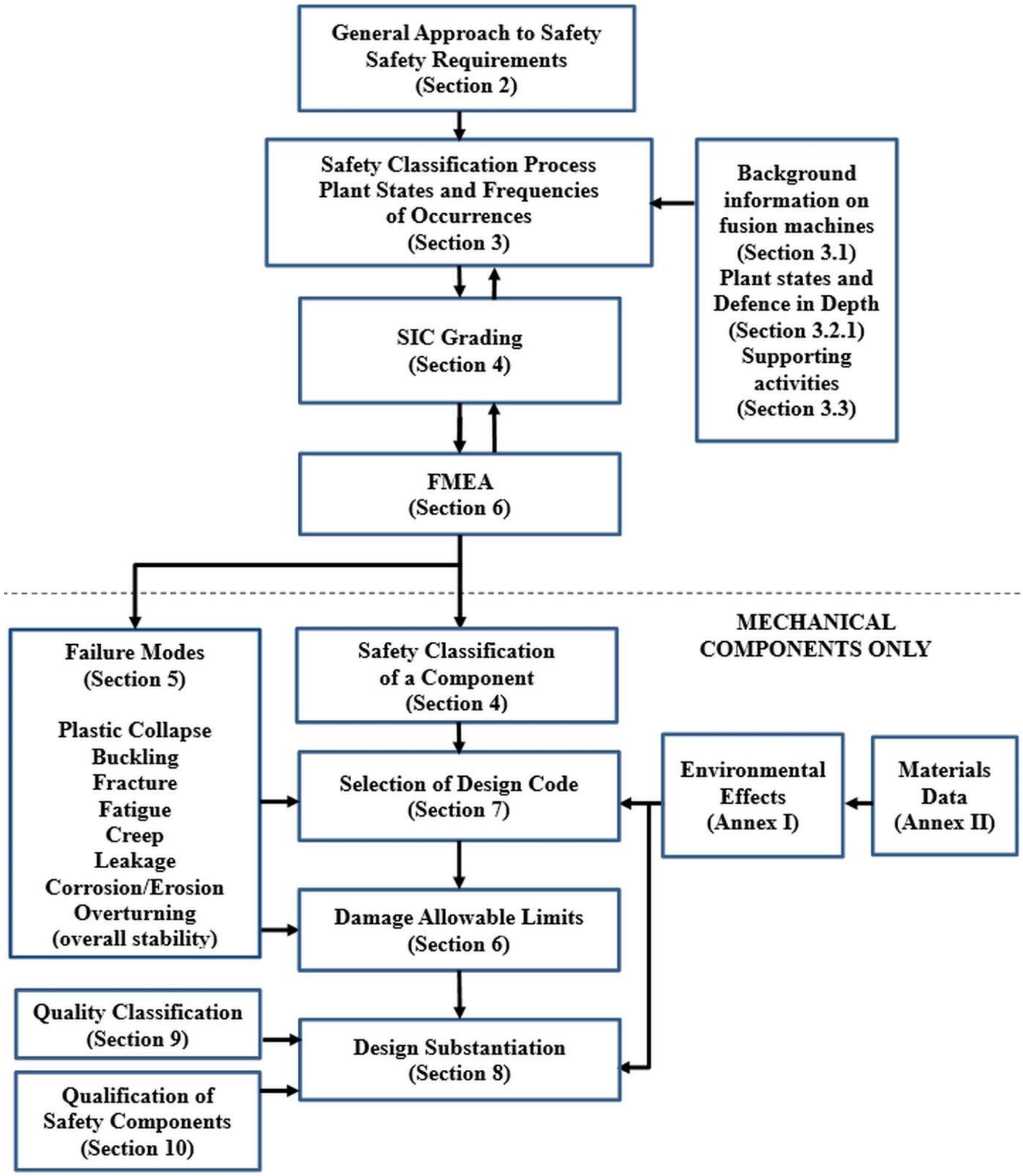

FIG. 1. TECDOC's structure based on an integrated approach.

2. SAFETY FUNCTIONS

The IAEA has shown particular interest in the area of fusion safety since the 1970s by organizing several technical meetings and workshops. As a result, two TECDOCs were published in 1983 [5] and 1986 [6] which constitute an early start in this endeavour. Some of the past IAEA activities on fusion safety are summarized in a brief overview in Ref. [5]. However, the previous publications did not address safety classification of fusion components.

Contributing to this end, a safety function is a specific purpose that is accomplished for safety of a facility or activity to prevent or to mitigate radiological consequences of normal operation, anticipated operational occurrences and accident conditions[2]. Before identifying and classifying the structures, systems and components (SSCs) that are important for safety, it is necessary to be clear about what are the necessary safety functions.

For example, some safety functions that are defined for a fission reactor are absent in a fusion plant. Reactivity control, needed to avoid a criticality event in a fission reactor, and emergency cooling needed to avoid a core melt event, are not relevant in a fusion plant. The principal safety functions in a fusion system are

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

The confinement function is generally provided by a succession of strong physical barriers, together with some active systems such as isolation valves and ventilation systems. To mitigate potential challenges to the barriers that provide the confinement function, many supporting functions are defined. These include the control of energies that could create such a challenge. For example, if there is a potential for a rupture of a coolant pipe inside a confinement volume that could lead to a pressure rise exceeding the design pressure of the confinement barrier, then control of this coolant energy is a supporting function. Listing the supporting functions needed for a plant design is an important step in preparation for the safety classification of its SSCs.

The second safety function, limiting exposure to ionizing radiation, is generally provided by a combination of source control, dedicated radiation shielding and collective protection equipment, access control and optimisation of human work effort. It is implemented after efforts have been made to maintain potential occupational doses as low as reasonably achievable (ALARA), for example by the choice of materials to reduce the level of neutron activation and by the utilization of RH techniques to reduce the need for maintenance operations involving human intervention. For this safety function, too, there may be supporting functions that need to be identified.

As an example of a list of safety functions and their supporting functions, Table 1 includes those being adopted in the conceptual design activities of ITER and a European demonstration fusion power plant (DEMO). These include additional functions in recognition of the need to control non- radiological hazards in the large industrial plant that comprises a DEMO facility, as well as the limitation of long-term environmental impact by the minimization of the quantity and hazard level of radioactive waste.

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[2] IAEA Safety Glossary 2016 revision (http://www-ns.iaea.org/standards/safety-glossary.asp).

TABLE 1. EXAMPLES OF LISTS OF SAFETY FUNCTIONS AND SUPPORTING FUNCTIONS, FOR ITER AND DEMO

<table>
<tr><th colspan="4">ITER</th><th colspan="2">DEMO</th></tr>
<tr><th colspan="2">Safety Function</th><th colspan="2">Detailed Safety Functions</th><th rowspan="5">Fundamental Safety Functions</th><td rowspan="5">Confinement of radioactive and hazardous materials.
Limitation of exposure to ionizing and electromagnetic radiation.
Limitation of the non-radiological consequences of conventional hazards.
Limitation of environmental legacy.</td></tr>
<tr><td rowspan="2">1</td><td rowspan="2">Confinement of radioactivity</td><td>1a)</td><td>Process confinement barries.</td></tr>
<tr><td>1b)</td><td>Building confinement barriers including systems for maintaining depression and filtering/detritiating effluents.</td></tr>
<tr><td rowspan="2">2</td><td rowspan="2">Limitation of exposure</td><td>2a)</td><td>Shielding to limit exposure and ALARA principle.</td></tr>
<tr><td>2b)</td><td>Access control.</td></tr>
<tr><th colspan="2">Supporting Functions</th><th colspan="2">Detailed Supporting Functions</th><th rowspan="15">Supporting Functions</th><th>Functions in support of confinement:</th></tr>
<tr><td rowspan="9">3</td><td rowspan="9">Protection of systems for confinement and limiting exposure</td><td>3a)</td><td>Management of pressure.</td><td rowspan="5">Control of plasma energy.
Control of thermal energy.
Control of confinement pressure.
Control of chemical energy.
Control of magnetic energy.
Control of coolant energy.</td></tr>
<tr><td>3b)</td><td>Management of chemical energy.</td></tr>
<tr><td>3c)</td><td>Management of magnetic energy.</td></tr>
<tr><td>3d)</td><td>Management of heat removal and long-term temperatures.</td></tr>
<tr><td>3e)</td><td>Fire detection/mitigation.</td></tr>
<tr><td>3f)</td><td>Mechanical impact (including seismic, dropped load, etc.)</td><th>Functions to support personnel and the environmental protection:</th></tr>
<tr><td>3g)</td><td>Management of mobilizable radioactive inventory.</td><td rowspan="3">Limitation of radioactive and toxic material exposure to workers.
Limitation of airborne and liquid operating releases to the environment.
Limitation of electromagnetic field exposure to workers.
Limitation of other industrial hazards.</td></tr>
<tr><td>3h)</td><td>Management of activated and contaminated material.</td></tr>
<tr><td>3i)</td><td>Control of safety protection and mitigation systems.</td></tr>
<tr><td rowspan="5">4</td><td rowspan="5">Supporting functions</td><td>4a)</td><td>Providing auxiliaries essential for implementing safety functions (electrical power supply, I&C, compressed air, etc.).</td><th>Supporting functions to limit environmental legacy:</th></tr>
<tr><td>4b)</td><td>Monitoring plant status: safety functions, radiation monitoring, etc.</td><td rowspan="4">Limitation of waste volume and hazard level.
Facilitation of clean-up and the removal of components.</td></tr>
<tr><td>4c)</td><td>Providing protection of important to safety systems (e.g. earthing, lightning, etc.)</td></tr>
<tr><td>4d)</td><td>Provide transport/lifting of radioactive components/materials.</td></tr>
<tr><td>4e)</td><td>Providing support to operator intervention (lighting, communications, etc.)</td></tr>
</table>

Safety analyses may determine the need for additional safety functions depending on the specific design of a fusion plant. For example, whereas for an experimental plant (such as ITER) the safety analyses show that a loss of cooling event does not directly result in any unacceptable safety consequence, it is possible that in a commercial fusion power plant a prolonged cooling loss could lead to a temperature rise resulting in structural degradation at the level that mobilisation of a radioactive inventory may result. In such a case the removal of decay heat could become a safety function.

In addition, future fusion power plants will have to deal with waste management (including environmental protection) for which the existing safety standards in this area would be applicable.

2.1. Confinement of radioactive material

The inventories to be confined in a fusion plant comprise radioactive solid, liquid and gaseous materials. They are principally tritium and the products of neutron activation. The tritium is mainly in gaseous form (T_2 or HT), liquid as tritiated water (HTO), or absorbed within solid materials by permeation. It may be present in many of the following locations in the facility:

— 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 RH 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 products of neutron activation are typically:

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

The confinement function is generally provided by a succession of physical barriers, which may be purely passive or may include active features such as isolation valves that are needed to close in certain off-normal situations. Typically, two independent confinement systems are provided with one or more physical barrier in each system. Such confinement barriers are typically supported by ventilation systems to maintain a pressure differential between the confined volumes so that any leakage is always in the direction towards the more contaminated volumes. Where these ventilation systems are vented to the atmosphere, it is necessary to provide filtering, including detritiation systems, to minimize the quantity of radioactive material that could reach the environment. Therefore, these ventilation, filtering and detritiation systems are also providing a confinement function.

2.2. Limitation of exposure to ionizing radiation

It is a fundamental principle of radiological protection that personnel exposure to ionizing radiation has to be limited in all normal and off-normal plant conditions, so that doses are ALARA. This limitation may be achieved by:

- The minimization of radiation source terms;
- The provision of radiation shielding;
- Protection against internal dose uptake (e.g. by inhalation);
- Reduction of human work effort by the optimization of maintenance procedures and the use of remote maintenance where feasible.

The minimization of radiation source terms implies the optimization of the design to reduce inventories, in particular those of neutron activation products in or near components that will require maintenance by human intervention. This may be done by careful location of components to minimize their exposure to neutron flux during operation, the provision of neutron shielding to reduce this flux, and the use of low activation materials wherever beneficial.

Shielding requirements need to be considered right from the beginning of the plant design process and reviewed at each step of the machine integration. They include the provisions to limit exposure to direct radiation from the plasma during operation and those to minimize exposure from secondary sources due to activation. In designs with water-cooled components experiencing a significant neutron flux, attention needs to be paid to the short-lived activation products of water such as ^{16}N, which may produce a significant gamma-ray source in parts of the cooling circuit outside of the main bio shield.

Such shielding provisions, as well as ventilation systems that limit airborne contamination, are coupled with access control systems that ensure personnel are excluded from areas where radiation dose rates are significant. The use of a zoning scheme to identify areas of different radiation levels is recommended, together with strict controls over the duration and circumstances in which access is permitted in the various zones. The zoning scheme used at ITER, and employed in other studies including the European DEMO project, is based on French regulations and is summarized in Table 2.

TABLE 2. SUMMARY OF RADIATION ZONING SCHEME USED FOR ITER AND OTHER FACILITIES

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

Some general recommendations[3] for limitation of personnel exposure include:

- Component design: simplicity and modularity, reducing time for disassembly and re-assembly.
- Component location: critical components preferably in low-dose areas to facilitate maintenance.
- Materials: selection of low activation materials in the design process.
- Maintenance processes and procedures: defined to minimise time for maintenance and intervention.
- Tooling: Preferred semi-automated or fully automated/RH tools.

It may be noted that radiation shielding implemented to limit occupational doses may also serve to limit radiation doses to materials and the consequent effects on materials properties (see Section 8). Minimizing materials ageing effects is also of benefit to safety where it reduces the frequency of repair or replacement operations that may involve human exposure to radiation. Additionally, neutron shielding may limit the activation of components that could represent a radiation source term during maintenance.

3. SAFETY CLASSIFICATION PROCESS

This section provides a description of the process proposed for the safety classification of mechanical SSCs of fusion installations. Background information is provided about fusion machines and SSCs classification and specifically about its hierarchical breakdown. The main

[3] The opinions expressed in this paper — and any recommendations made — are those of the international experts and do not necessarily represent the views of the IAEA, its Member States or the other cooperating organizations.

phases of the process are described in terms of objectives and supporting methods and techniques. The essentially four steps in the safety classification process are illustrated in Fig. 2.

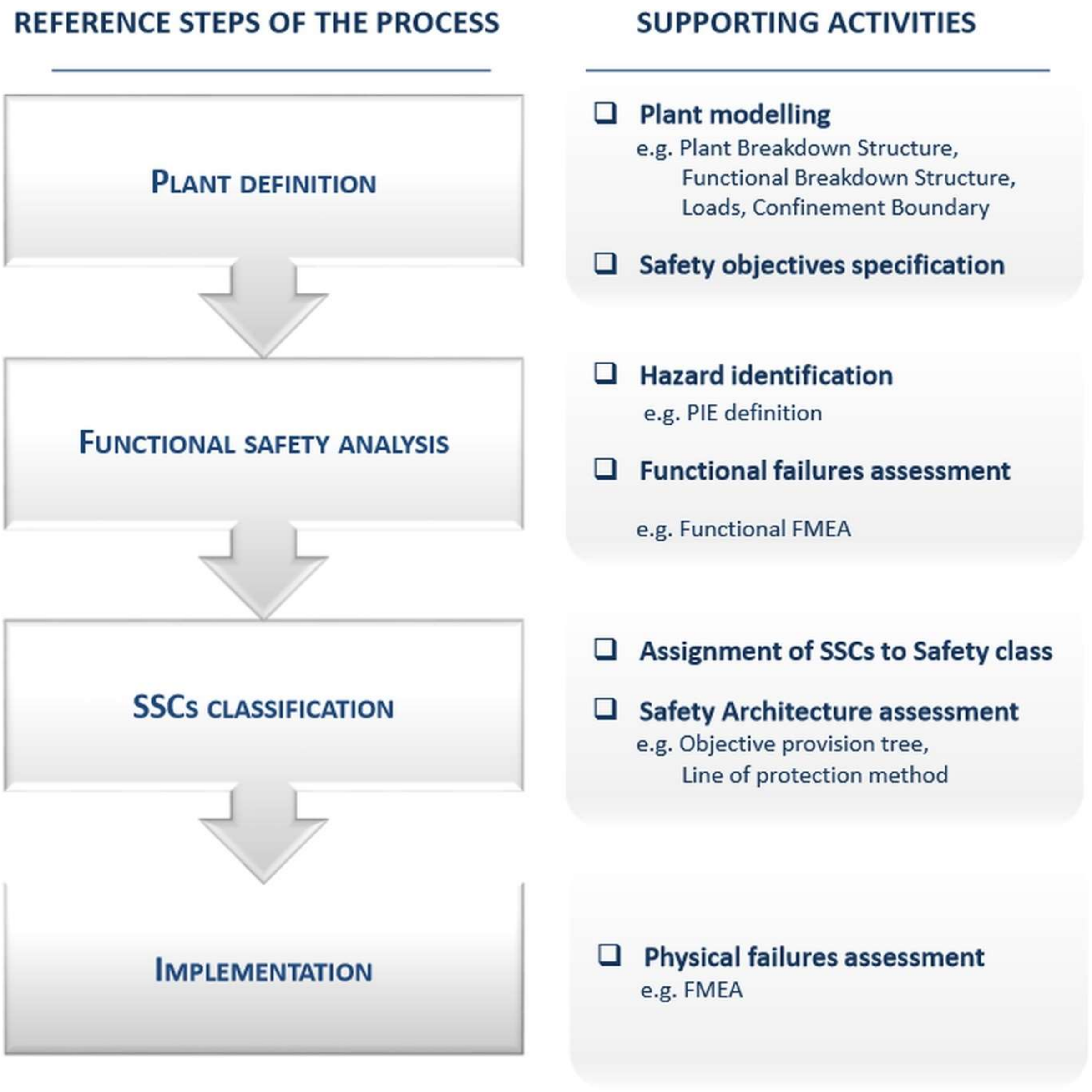

FIG. 2. Four steps process, from the plant design concept to the fabrication of SSCs.

The first step (Plant definition) aims at providing the information needed by the subsequent categorization of safety functions and safety classification of SSCs. It includes the definition of the safety objectives and the specification of the plant design concept and (functional and physical) breakdown(s).

The second step (Functional Safety Analysis) aims at categorizing the safety functions implemented by the plant. This activity needs to be supported by the assessment of the plant's functional failures.

The third step (SSCs classification) aims at classifying the SSCs according to the allocated safety functions and to their categorization. It requires a representation of the relations between the SSCs and the implemented (safety) function, the unambiguous definition of the safety classes to be assigned, the results coming from the categorising of safety functions, and the assessment of the plant's SSCs failure. At this stage, SSCs have to be defined (at least) in terms of functions implemented and external (mechanical, electrical, hydraulic, pneumatic) interfaces. This activity needs to be supported by the assessment of the SSCs failure from a functional perspective.

A functional failures assessment is needed by the second and third step of the process and can be developed through the FMEA methodology (see Sections 3 and 6).

The fourth step (Implementation) includes the design, the design justification, the prototype qualification and the fabrication of the SSCs. All these activities need to be supported by the detailed assessment of the failure modes of the SSC and its subparts. A physical failures assessment can be developed through the FMEA methodology.

The functional and physical failures assessments can be performed by the proper application of the FMEA methodology. Different approaches need to be adopted, different results need to be searched and different recommendations can be specified.

The proper application of methods and techniques, as well as the full understanding of the process and criteria proposed for the safety classification of SSCs, require a close collaboration between the design team and the safety team. The design team needs to provide the knowledge of the plant and its SSCs, under normal and accidental conditions. The safety team needs to provide the expertise needed by the deterministic and probabilistic safety demonstration. Specific expertise could be needed about external hazards (e.g. seismic, flood). This integrated design and safety process is to be preferred over an approach where safety assessment is carried out after design process.

The general classification process is same as given in SSG-30 [2]. Details of each activity are provided in the following sections.

3.1. Background information on fusion machines

3.1.1. Plant breakdown structure

The complete identification of SSCs in a plant is usually done in a hierarchical breakdown, the top-level listing systems, each divided into sub-systems, parts and components at successive lower levels, until individual detailed components are identified at the lowest level. It is normal for this Plant Breakdown Structure (PBS) to be defined only at the highest levels at the start of a conceptual design process, and become defined at lower level as the design matures and becomes detailed. In the same way, safety classification may be done first at the top level, and later in progressively more detail at lower levels.

The top level of a typical PBS for a tokamak plant is listed in Table 3. This example is based on the PBS for the conceptual design of a European Demonstration power plant (DEMO). In general, not all of these systems are necessarily present in a given plant design, for example only one or two of the heating and current drive systems at PBS 10, 11 and 12 might be included. For other magnetic confinement systems, such as a stellarator, there may be some variations, and for an inertial confinement plant there would be many differences. In this table, PBS 19, balance of plant, includes secondary (and tertiary, if any) cooling circuits and heat rejection systems as well as the power conversion systems (turbines etc.). PBS 25, auxiliaries, includes supplies of gasses and fluids to other plant systems. An example of how breakdown into sub-systems at level 2 may be affected is shown in Table 4, where the main sub-systems of the fuel cycle are elaborated.

TABLE 3. EXAMPLE OF PLANT BREAKDOWN STRUCTURE LEVEL 1 FOR A DEMO POWER PLANT

PBS level 1	Description
01	Magnet System
02	Vacuum Vessel
03	Divertor
04	Blanket and first wall
05	Limiter
06	Cryostat
07	Thermal Shields
08	Fuel cycle
09	Tritium Extraction from blanket
10	Electron Cyclotron (EC) System
11	Neutral Beam Injection (NBI) System
12	Ion Cyclotron (IC) System
13	Plasma Diagnostic and Control System
14	Primary Heat Transfer System
15	Vacuum vessel pressure suppression
16	Remote Maintenance System
17	Assembly
18	Radioactive Waste Treatment and Storage
19	Balance of Plant
20	Site Utilities
21	Cryoplant and Cryodistribution
22	Electrical Power Supply Systems
23	Buildings
24	Plant Control and Monitoring System
25	Auxiliaries

TABLE 4. EXAMPLE OF PBS LEVEL 2 FOR FUEL CYCLE SYSTEMS

PBS level 1	PBS level 2	Description
08	01	Vacuum systems and exhaust processing
08	02	Fuelling systems
08	03	Deuterium and tritium processing systems
08	04	Deuterium and tritium storage
08	05	Fuel cycle control and monitoring system
08	06	Fluid detritiation systems
08	07	Atmosphere detritiation system

The example PBS presented in Table 3 is not definitive. There are varieties of ways in which plant systems can be distributed between PBS elements. Whatever choices are made in the breakdown of systems, it is of course necessary that all SSCs in the plant are included, completeness being essential for a PBS.

3.1.2. Functional breakdown structure

Whereas the PBS represents a breakdown of the physical SSCs in the plant, the functions that they provide is the subject of a Functional Breakdown Structure (FBS). A function is a statement of a specific purpose or objective to be accomplished, without a description of how it is achieved. Like the PBS, a FBS is constructed in a hierarchical manner, with each top-level function broken down into the purposes that can be fulfilled to achieve this top-level objective. And like the PBS, it is only useful if it is comprehensive and complete. Then the FBS can be used to check that all needed functions are provided, by assigning each function at the lowest level to one or more items in the PBS.

Functions in the FBS are generally related to the processes needed to achieve the overall aim of the fusion power plant: to produce electrical power. An example of a portion of an FBS is given in Table 5, related to the functions needed to provide fuel for the plasma. This example is also taken from studies for the European DEMO design [7]. These functions would mainly be assigned to items in the fuel cycle SSCs in the PBS, Table 4.

Whereas the FBS provides a complete list of the functions carried out by the plant, it is also necessary to develop a full list of safety functions, together with all needed supporting functions. These are the actions or provisions that have to be made by SSCs to fulfil the safety requirements in all the plant states and conditions, both normal and off-normal.

As previously noted, some safety functions defined for fission reactors are not relevant in fusion machines; in particular, the reactivity control (needed to avoid a criticality event in a fission reactor) and the emergency cooling (needed to avoid a core melt event).

TABLE 5. AN EXAMPLE OF A PORTION OF A FUNCTIONAL BREAKDOWN STRUCTURE

FBS level 1	**FBS level 2**	**FBS level 3**	**Description**
1			To manage fuel.
1	1		To supply fuel to the plant through external supplies.
1	2		To recover tritium from breeding and multiplier materials.
1	3		To recover unspent D-T from the tokamak exhaust.
1	4		To recover unspent D-T from the tritiated process fluids.
1	4	*1*	*To recover tritium from coolants.*
1	4	*2*	*To recover tritium from cryogenic fluids.*
1	4	*3*	*To recover tritium from inert gases.*
1	5		To recover unspent D-T from the tritiated wastes.
1	6		To store fuel gas (Hydrogen isotopes).
1	6	*1*	*To provide long-term storage of hydrogen isotopes.*
1	6	*2*	*To provide short-term storage of hydrogen isotopes.*
1	7		To supply fuel to fuel injection systems in plasma.

The principal safety function in a fusion system is the confinement of radioactive material. This means the prevention of the mobilisation and dispersal of radioactive material — tritium and the products of neutron activation — within the plant, and the avoidance of the leakage of any part of this radioactive inventory to the environment. The term ‘confinement’ is typically used to refer to the safety functions preventing the release of radioactive material, whereas ‘containment’ refers to the means for achieving that safety function, i.e. to methods or physical structures designed to prevent the radioactive release.

A further fundamental safety function is the limitation of exposure to ionizing radiation, to prevent plant personnel from occupational radiation exposure arising from direct radiation from the plasma, from neutron activated material or by the inhalation of tritium or mobile activation products.

Further fundamental safety functions concern the need to control non-radiological hazards in large industrial plants (which include the DEMO facility), and the limitation of long-term environmental impact by the minimization of quantity and hazard level of radioactive waste.

In addition to these fundamental safety functions, supporting functions are listed to identify any provisions needed to enable the fundamental functions to be achieved in all plant states, including normal operation, maintenance, abnormal conditions, and accidents. Listing the supporting function needed for a particular plant design is an important step in preparation for the safety classification of SSCs.

An example of a list of safety functions and their supporting functions is provided in Table 1 of Section 2.

Safety analyses may determine the need for additional safety functions depending on the specific design of a fusion plant. For example, whereas for experimental and demonstration plant (e.g. ITER and DEMO) the safety analyses may show that a loss of cooling event does not directly result in any unacceptable safety consequence, it is possible that in a commercial fusion power plant a prolonged cooling loss could lead to a decay-heat driven temperature rise resulting in structural degradation at the level that mobilisation of a radioactive inventory may not be excluded. In such a case the removal of decay heat could become a safety function. It will be determined only by analysis and will depend on specific aspects of the design including materials selection.

3.1.3. Typical components in the confinement boundary

The confinement of the radioactive fuel in a fusion device is provided by the main vacuum vessel together with its port extensions and primary vacuum envelopes. The primary cooling system confines the activated corrosion products generated by the corrosion and erosion of in-vessel actively cooled components.

The plant involves a variety of passive and active mechanical components. A general list is provided hereafter, highlighting special components that are typically present in the first confinement system:

(a) Main reactor vessel, including large assemblies of shells and box structures connected with welds and bolted joints.
(b) Vacuum ports and extensions in the reactor vessel to provide access for RH operations, diagnostics, heating, and vacuum systems. The vacuum ports may contain a variety of SIC, including:
    (i) Vacuum flanges with single or double seals, monitored vacuum interspace and bolted connections.
    (ii) Isolation valves to isolate the vacuum vessel environment from upstream components maintaining the specified rates of vacuum and tritium leak tightness.
    (iii) Pressure relief systems to limit the pressure of the vacuum chamber during off-normal pressurisation events.
    (iv) Bellows to absorb the thermal expansion and contraction of components, absorb vibrations, to hold parts together and to allow movements due to specific operational modes and off-normal events (e.g. seismic).
    (v) Penetrations, these components can be of different nature (e.g. vacuum, cooling, electrical) and represent a discontinuity in some or all the features of the safety barrier itself; a penetration through a safety barrier is composed by the penetrating element (e.g. pipes, cables) and the interface with the safety barrier; when electrical isolation is needed, the feedthrough design might include ceramic parts, as well as ceramic-to-metal connections.
    (vi) Transmission lines, fusion devices require complex systems providing external heating to the plasma and monitoring the plasma conditions; such systems provide the needed power and/or signals through assemblies of transmission lines extending from the main reactor chamber.
    (vii) Vacuum windows are often used in the diagnostic and heating systems of fusion machines and includes ceramic-to-metal joints; these windows form part of the

pressure boundary of the vessel, and as such need to be able to withstand all normal operating and fault conditions, pressure loadings and thermal loadings, and to contain the reactant materials (as tritium in the vessel); the structural failure of any such window would lead to a loss of containment, and so it needs to be demonstrated that its integrity will not be compromised during its design life; in addition, each window needs also to maintain its optical transmission function;

(c) Primary cooling system provides cooling water to client systems for heat removal during plasma operations and for decay heat removal after operations, they ensure additional operations such as baking, draining and drying of components, and confine the activated corrosion products and tritium present in the cooling water; the primary cooling system may include many components including piping, flanges, pumps, valves, bellows and structural supports.

3.1.4. Typical load types

3.1.4.1. Operational loads

The loads acting on the mechanical SSCs during operations, incidents and accidents events in a fusion machine can be categorized into four different types:

(a) Inertial loads:
   - (i) Self-weight;
   - (ii) Loads resulting from seismic accelerations, including forces generated by fluid-to-structure interactions.

(b) Pressure loads:
   - (i) Differential pressure on the vacuum boundary;
   - (ii) Coolant pressure;
   - (iii) Over-pressurisation of plasma chamber during accidents (e.g. loss of coolant/vacuum accidents).

(c) Electromagnetic loads: Eddy and halo currents are both sources of significant loads in tokamaks).
   - (i) Eddy currents: Electromagnetic Loads (EMLs) due to Lorentz forces when electric current crosses magnetic field lines. Large currents may be circulating through the conductive structures during fast transient events, and in combination with the background fields this could generate significant internal and mutual loads. The forces and moments generated during electromagnetic events are typically strong design drivers for fusion components.
   - (ii) Halo currents: Current which flows outside the confined plasma region, in the scrape-off layer. During a Vertical Displacement Event, the plasma makes contact with a material surface and starts to be scraped off, causing a fraction of the plasma current to flow along the field lines. The current path intercepting a material surface will take the path of least resistance, closing in a halo pattern. This generates forces, moments and heat loads.

(d) Thermal and nuclear loads from plasma: Plasma operation is a major source of loads on in-vessel components, vacuum vessel and the tokamak structure in general. Normal plasma operation includes all phases of a plasma pulse (initiation, current ramp-up, steady state phase and ramp-down/termination) as well as fast transient events, including core/edge instabilities, disruptions and vertical displacement events (the two latter events leading to a fast and abrupt termination of the plasma pulse).

(i) Nuclear loads: neutrons generated by the D–T fusion reaction (14 MeV at birth) and prompt X rays. In addition, Alpha particle bursts (caused by plasma instabilities, transient load). In addition, radiation from activation products during operations and maintenance periods, radiation from the neutron test area, and tritium has to be included as necessary.

(ii) Thermal loads: thermal loads originate from radiation from the plasma as well as from conduction and convection of plasma particles, for steady state and transient events. In addition, gas and pellet injection used for fuelling, divertor heat load control and plasma disruption mitigation also contribute to thermal loads. Transient loads include energy bursts from plasma instabilities (such as Edge Localized Modes), power loads due to unwanted plasma-wall contact and thermal loads due to disruptions (controlled or not). Finally, partial neutral beam power absorption in the plasma (Neutral Beam shine-through) also generated local heat loads on the inner wall of the tokamak.

(iii) Physical impact of particles: Physical impact of particles is caused by conductive and convective (steady state) transport as well as being associated to (bursty) plasma transients. In addition to thermal loads, these particles cause physical erosion and, in principle, may cause morphological changes in materials. Neutron impact and slow-down in materials is a major ‘particle impact’ load in tokamaks, resulting in transmutation and internal structural damage (displacement events).

3.1.4.2. Non-operational loads

Loads associated to non-operational phases may be strong design drivers for fusion components and include:

— Manufacturing;
— Transportation;
— Installation and/or assembly (e.g. preload);
— Testing and maintenance (e.g. pressure testing, leak testing, draining and drying, venting).

3.2. Background information on Structures, Systems, or Components classification

3.2.1. Plant states and Defence in Depth

Fig. 3 provides the definitions of the plant states according to the IAEA SSR-2/1 [1]. The original design basis has been extended to include the DEC. The paragraph below summarizes the concept of design basis of a structure, system or component as reported in Refs [1,4].

The design basis of a structure, system or component is the set of information that identifies conditions, needs and requirements necessary for the design, including the:

- Functions to be performed by a structure, system or component of a facility;
- Conditions generated by operational states and accident conditions that the structure, system or component needs to withstand;
- Conditions generated by internal and external hazards that the structure, system or component needs to withstand;
- Acceptance criteria for the necessary capability, reliability, availability and functionality;
- Specific assumptions and design rules.

The design basis of a structure, systems or component is completed and supplemented by specification sheets and by detailed design calculations.

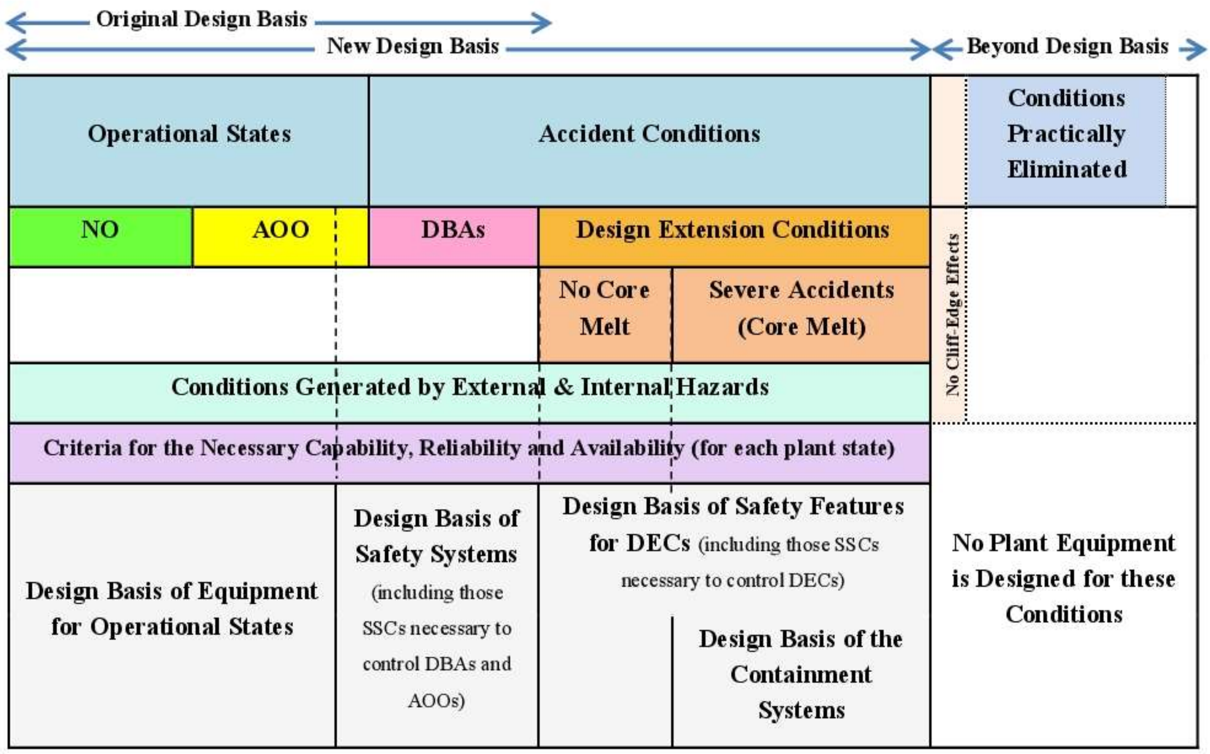

FIG. 3. Plant states definition.

The DEC definitions that exist in IAEA guides may not be directly applicable to fusion applications. However, DEC are also included here in the range of event categories for fusion, as a reminder that safety features introduced in the design to control accidents need to consider common cause and common mode failures, more generally multiple failure events that may be categorised as DEC.

Similarly, new requirements to consider extreme accidents in design specified by the newly amended EU Directive on Nuclear Safety [8] are introduced. These changes have included some extreme events (with low probability but high load) in the design basis. This has posed new challenges for the designers to design and verify their designs to withstand extreme events. The 2014 EU Directive on Nuclear Safety is an amendment to the 2009 EU Directive on Nuclear Safety and Member States were needed to implement it by law by 15 August 2017.

For the design of new reactors, the structure of the levels of Defence in Depth (DiD) shown in Fig. 4 is proposed by Western European Nuclear Regulators Association (WENRA) [9] which considers the lessons learnt from the Fukushima Daiichi accident[4]. It introduces relations

[4] WENRA recommends reinforcing and strengthening the DiD approach (compared to previous realizations). Most prominently, DiD Level 3 is subdivided into sub-level 3a, which entails postulated single failure events, and sub-level 3b, which covers multiple failure events not leading to a postulated severe accident. DiD Level 4 addresses postulate severe accident scenarios.

between the levels of DiD and the associated plant condition categories (note that it includes reference to the ‘core melt’ scenario, which is not of relevance for fusion machines).

Levels of defence in depth	Objective	Essential means	Radiological consequences	Associated plant condition categories
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation, control of main plant parameters inside defined limits	No off-site radiological impact (bounded by regulatory operating limits for discharge)	Normal operation
Level 2	Control of abnormal operation and failures	Control and limiting systems and other surveillance features		Anticipated operational occurrences
Level 3 (1) 3.a	Control of accident to limit radiological releases and prevent escalation to core melt conditions (2)	Reactor protection system, safety systems, accident procedures	No off-site radiological impact or only minor radiological impact (4)	Postulated single initiating events
3.b		Additional safety features(3), accident procedures		Postulated multiple failure events
Level 4	Control of accidents with core melt to limit off-site releases	Complementary safety features(3) to mitigate core melt, Management of accidents with core melt (severe accidents)	Off-site radiological impact may imply limited protective measures in area and time	Postulated core melt accidents (short and long term)
Level 5	Mitigation of radiological consequences of significant releases of radioactive material	Off-site emergency response Intervention levels	Off site radiological impact necessitating protective measures(5)	-

FIG. 4. Defence in Depth definition (for footnotes within the figure, please see Ref. [9], 13).

It is worth noting that no SSC is designed for conditions beyond DEC as they are practically eliminated but the boundary between the new design basis and the Beyond Design Basis Accident (BDBA) is not distinct. The key notions of ‘cliff edge effects’ and ‘practical elimination’ assume particular relevance in this context.

IAEA SSR 2/1 and WENRA consider that the possibility of a certain condition occurring is ‘practically eliminated’ if it is physically impossible or if it can be considered with a high degree of confidence to be extremely unlikely to arise. The robustness and the reliability of safety-important SSCs need to support the demonstration that postulated events with consequences overcoming the design limits are practically eliminated.

According to the IAEA SSR 2/1, "the design shall be conservative and the construction shall be of high quality to provide assurance that failures and deviations from normal operation are minimized, that accidents are prevented as far as is practicable and that a small deviation in a plant parameter does not lead to a cliff edge effect". A cliff edge effect is "an instance of severely abnormal plant behaviour caused by an abrupt transition from one status to another following a small deviation in a plant parameter, and thus a sudden large variation in plant conditions in response to a small variation in an input."

Another term widely used in recent years, even if no world-wide recognized document provides its definition, as 'Hardened Safety Core'. It indicates a limited number of material, organisational, human systems providing essential safety functions even in extreme circumstances, i.e. exceeding those adopted for the general design of the facility[5].

The identification of SSCs materializing the Hardened Safety Core (i.e. Hard-Core Components (HCCs)) is the part of the classification process specifically focused on the role of SSCs against Design Extension Condition and Beyond Design Basis conditions.

3.2.2. SSG-30 Classification process

According to the IAEA SSG-30 [2], "the classification[6] of the Structures, Systems and Components (SSCs) of a (fusion) nuclear installation shall be derived from the categorization of the safety functions, according to the risk-reduction needed."

The objective of the SSG-30 Safety Guide is "to provide recommendations and guidance on how to meet the requirements established in the IAEA SSR-2/1 [1] and GSR Part 4 [3] for the identification of SSC important to safety and for their classification based on their function and safety significance". It states, "All items important to safety shall be identified and classified based on their function and their safety significance."

The classification process recommended by the SSG-30 is "consistent with the concept of Defence in Depth set out in the IAEA SSR-2/1". The functions to be addressed are "primarily those that are credited in the safety assessment and should include functions performed at all five levels of DiD".

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[5] The term 'Hardened Safety Core' has been used by the French Nuclear Safety Authority (ASN), according to the indications provided by the European Nuclear Safety Regulators Group, among the measures imposed after the accident at Fukushima Daiichi to reinforce the safety requirements for the prevention of natural risks, the management of loss of electrical power and cooling systems situations and for management of severe accidents. Stress Tests (or 'Complementary Safety Assessments') were defined and performed in a number of countries as a targeted reassessment of the safety margins of Nuclear Power Plants (NPP) in the light of the lessons learnt from the Fukushima accident about extreme natural events challenging the plants safety functions.
In this context, the use of 'bunkered' or 'hardened core' of safety-related systems has been indicated among the means to increase the robustness of NPPs against external hazards. Bunkered system ensures an additional level of protection after the external events, able to cope with a variety of initiating events, including those beyond the design basis, providing back-up to ordinary stand-by system and provisions for emergency management. This concept evolved in the form of the Hardened Safety Core considering, in addition to equipment, trained staff and procedures intended to cope with a wide variety of extreme events.

[6] According to SSG-30, the term 'categorization' is reserved for functions, the term 'classification' for SSCs.

The classification follows a top down process. It begins with a basic understanding of the plant design and safety features, its safety analysis and how the main safety functions are achieved. This information is used for the identification of functions and design provisions needed to fulfil the main safety functions, which need to be done systematically for all plant states.

The method for classifying items important to safety needs to be based primarily on deterministic methods, and complemented by probabilistic methods where appropriate. It has to consider:

— The safety function(s) to be performed by the item;
— The consequences of failure to perform a safety function;
— The frequency with which the item will be called upon to perform a safety function;
— The time following a postulated initiating event at which the item will be called upon to perform a safety function.

Table 6 provides the definition of the (three) severity levels proposed by SSG-30. Their assignment is based on the worst consequences that could arise if the function is not performed.

TABLE 6. SEVERITY LEVELS AND CRITERIA FOR ASSIGNMENT DEFINED BY THE SSG-30

Severity levels	**Criteria for assignment**
High	If failure of the function could, at worst: • Lead to a release of radioactive material that exceeds the limits accepted by the regulatory body for design basis accidents; • Cause the values of key physical parameters to exceed acceptance criteria for design basis accidents.
Medium	If failure of the function could, at worst: • Lead to a release of radioactive material that exceeds limits established for anticipated operational occurrences; • Cause the values of key physical parameters to exceed the design limits for anticipated operational occurrences.
Low	If failure of the function could, at worst: • Lead to doses to workers above authorized limits.

To account for the implementation of the defence in depth principle in the assessment of consequences, the IAEA TECDOC on “Application of the Safety Classification of Structures, Systems and Components in Nuclear Power Plants” [10] proposes that:

— “The assessment of the safety significance of Anticipated Operational Occurrence (AOO) related functions should be performed assuming that other functions for AOOs (i.e. reactor trip) or for Design Basis Accidents (DBAs) (functions accomplished by the safety systems) will respond as expected, provided that the associated systems are not affected by the initiating event.”
— “The significance of an AOO related safety function should not be lowered since an independent DEC safety function would also be available to control the event.”

— “The assessment of the safety significance of functions used to mitigate the consequences of design basis accidents or design extension conditions should be performed ignoring the role of other functions allocated to other defence in depth levels.”

3.2.3. Conceptual framework for risk acceptance

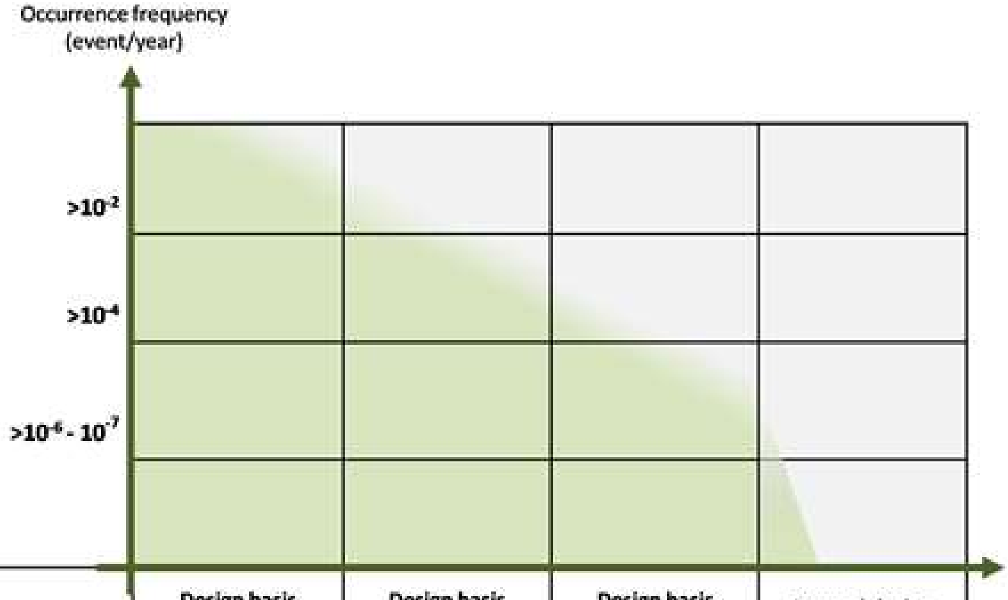
The process and criteria proposed for the safety classification of mechanical SSCs of fusion machines are based on the conceptual framework defined by the Risk Domain in Fig. 5, which combines:

— The definition of the plant states provided by the SSR-2/1.
— The structure of DiD levels proposed by WENRA and their references to the plant states.
— The key notions of ‘cliff edge effects’, ‘practical elimination’ and ‘Hardened Safety Core’.
— The levels of severity defined by the IAEA SSG-30 within the process for the classification of SSC.
— The safety objectives (without numerical values) and the events frequency categories adopted for ITER and defined for DEMO [7].

The risk matrix in Fig. 5 represents roughly the criteria for risk acceptance, by combining the categorization of the off-normal states of the plant according to their frequency of occurrence and the safety objectives to be met for each category.

Events with a frequency > 10^{-2}/year (higher row in Fig. 5) are acceptable only if their consequences do not overcome the limits specified for AOO (i.e. the relevant safety objective are met). The provisions implemented by the plant to control AOOs refer to DiD level 2.

Events with a frequency < 10^{-2} but > 10^{-6}/year (two central rows in Fig. 5 on the next page) are acceptable only if their consequences do not overcome the limits specified for the Design Basis Accident (i.e. the relevant safety objective are met). Note that an intermediate safety objective can be defined for (unlikely) events with a frequency < 10^{-2} but > 10^{-4}/year. This group of events covers DEC and extreme events in the EU Directive. The provisions implemented by the plant to mitigate DBAs refer to DiD levels 3 and 4.



PLANT STATES	Design basis AOO	Design basis DBA	Design basis DBA / DEC	Beyond design basis
DiD LEVEL	DiD Level 2 No off-site radiological impact	DiD Level 3.a No or only minor off-site radiological impact	DiD Level 3.b No or only minor off-site radiological impact DiD Level 4 Limited protective measures in area and time	No cliff-edge effect & Pratically elimination
SEVERITY LEVEL SSG 30	Low & Medium	Medium	High	
EVENT CLASSIFICATION (ITER / DEMO reference)	Anticipated events (Ctg. II)	Unlikely events (Ctg. III)	Extremely unlikely events (Ctg. IV)	Hypothetical events
SITUATIONS (ITER / DEMO reference)	Incidental situation	Accidental situation		Hypothetical accident
OFF-SITE DOSE (ITER reference)	≤0.1 mSv/year		≤10 mSv/event	-
(DEMO reference)	Chronic Dose: 1mSv/year	Chronic Dose: 5mSv/event	Chronic Dose: 50mSv/event Early Dose: 10mSv/event	Early Dose: 50mSv/event

FIG. 5. Conceptual framework of the Risk Domain.

Events with consequences overcoming the limits for the Design Basis Accident (right hand column on Fig. 5) are BDBAs which can be mitigated or not by the plant. In the first case, the event is included in the list of DECs; the sequence including the initiating event and the failure of the related HCC need to be practically eliminated. In the second case, the event itself has to be practically eliminated.

Fig. 6 provides the representations of the criteria specified by the SSG-30 for the classification of functions to reach a controlled state after an AOO or DBA, and to mitigate the consequences of DEC. Colour coding is used to indicate the three safety categories.

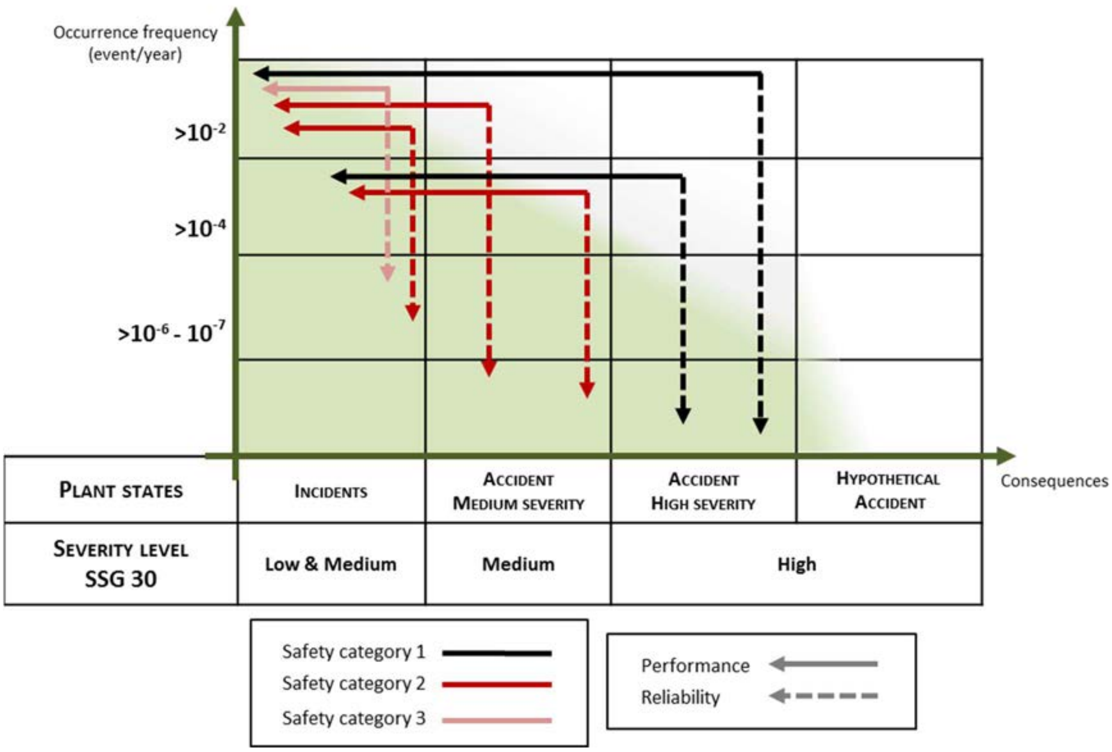


FIG. 6. Risk domain and SSG-30 classification criteria.

In general, there are two ways to bring an undesirable event that puts a plant into an uncontrolled state back into a controlled or safe state (shaded zone). A safety related SSC (more generally, a 'layer of provisions') can be introduced, against a (initiating) event with unacceptable consequence, to either reduce the severity of consequence or reduce the frequency of occurrence or both. Consistently, the SSC is characterized by:

— Its physical performance, i.e. its ability to perform the safety function reducing the consequences of the undesired event to an acceptable level regarding its probability.
— Its robustness and reliability, i.e. its ability to perform the safety function with a sufficiently low failure rate and to ensure that no operational loads or loads caused by Postulated Initiating Events (PIEs) will adversely affect this ability.

With reference to Fig. 6:

— A solid horizontal line represents the SSC physical performance, i.e. the reduction of consequence of the undesired event;
— A vertical dotted line represents the SSC robustness and reliability, i.e. the reduction of the frequency of occurrence of the undesired event (initiating event and loss of the safety function due to the SSC failure).

The representation through a risk matrix allows underlining the need and the opportunity to specify reliability targets for the SSCs implementing safety functions. The opportunity is to classify further the functions and related SSCs implemented by the plant, even if in the same safety category (from a performance perspective), with reference to the necessary reliability. The need to consider the reliability of safety important SSC comes from the assessment of complex accidental sequences, including the (internal or external) initiating event and the failures of SSCs implementing the necessary safety function(s).

As represented in Fig. 7[7], a safety function implemented against a given event (with unacceptable consequences related to its frequency of occurrence) can fail. The sequence including the initiating event and the loss of the safety function could remain unacceptable, according to the reliability of the SSC(s) carrying out the function. Deterministically, multiple independent layers of provisions need to be implemented according to the DiD concept. Probabilistically, the sequence including the initiating event and the loss of the safety functions implemented at different DiD levels need to be practically eliminated. This assessment requires a suitable representation and assessment of the "safety architecture" of the plant (see Section 3.3.6).

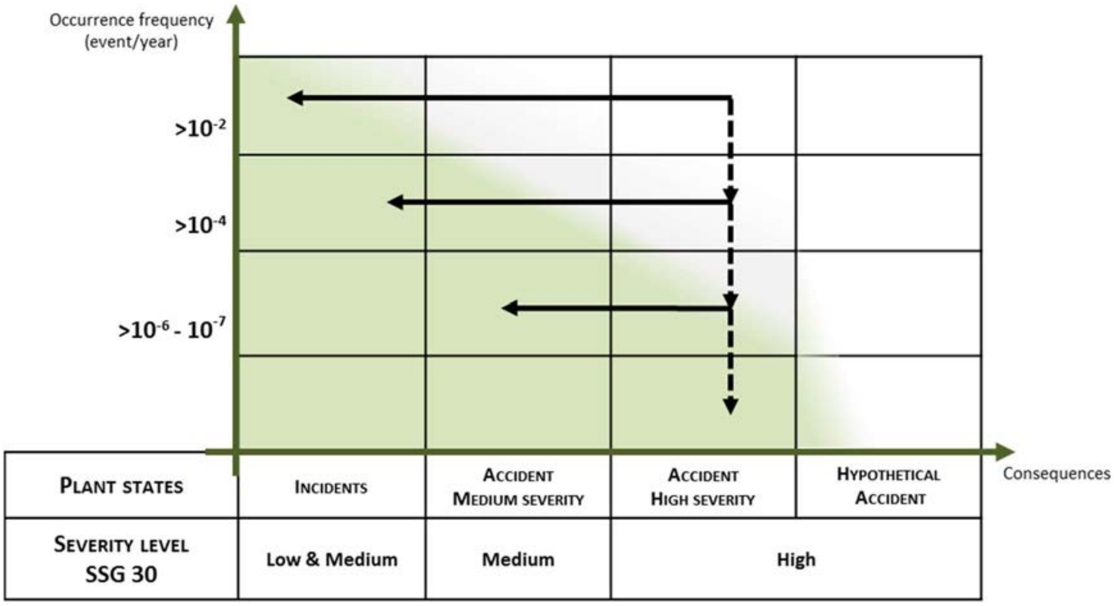


FIG. 7. Risk domain and sequence of events.

The 'safety architecture' of the plant is the set of provisions set up by the designer in order:

— To ensure that the mission is carried out by the plant while the required safety objectives are met;

[7] For instance, Fig. 7 refers to an in-vessel overpressure (initiating) event, which can be controlled by the pressure suppression system(s), and mitigated by the 1st confinement system (to be designed for the environmental and loading conditions due to the failure of the pressure suppression systems), and by the second confinement system (to be design for the environmental and loading conditions due to the failure of the 1st confinement).

- To prevent the degradation of the facility, i.e. to avoid that operational limits are exceeded;
- To bring to and to maintain the facility in a safe state, from the off-normal condition due to an internal (e.g. SSCs failure or internal hazard) or external (hazard) initiating event.

3.3. Supporting activities

3.3.1. Plant modelling

Preliminary activities concern the identification of all the plant's SSCs, the specification of their functions, of the necessary (capability, reliability and robustness) performance and of the key influencing parameters.

The (safety) functions implemented by the plant can be broken-down into an increasingly detailed set of subsidiary functions (FBS in Section 3.1.2). Similarly, the SSCs can be decomposed into an increasingly detailed set of sub-systems and components (PBS in Section 3.1.1). The categorization of functions and the classification of SSCs can be applied at different levels within these hierarchies. As general guidance, the breakdown processes has to continue to (at least) the level at which the role of each physical item in the delivery of single safety function become clear and unambiguous.

The Master Logic Diagram (MLD) is one of the earliest techniques used for the functional analysis of fusion machine. It provides an easy-to-read representation of the relations between functions (i.e. functional breakdown), SSCs (i.e. physical breakdown), and between functions and SSCs (i.e. functions allocation). MLDs are useful input for the functional failure assessment, as they support the assessment of the consequences of failures from a functional perspective. In more evolved applications, the safety architecture of the plant could be represented in terms of SSCs implementing safety functions at the different levels of DiD. Practical techniques could support this representation (see Section 3.3.6).

3.3.2. Safety objectives and risk acceptance framework

The categorization of safety functions and initiating events, as well as the classification of SSCs and their parts, require the definition of safety objectives and risk acceptance criteria.

A general framework for the assessment of hazardous events (including the failures of SSCs) is proposed by summarizing the indications provided by the IAEA requirements (SSR-2/1) [1] and guidelines (SGG-30) [2], the recommendations provided by WENRA about the design of new nuclear power plants [9], and the insights coming from the ITER [11] and DEMO experiences [7].

Table 7 provides the criteria for the categorization of events, based on their expected frequency of occurrence. Table 8 defines the safety objectives to be met for each event category. Risk acceptance criteria come from their joint application.

TABLE 7 CRITERIA FOR THE CATEGORIZATION OF EVENTS BASED ON THE EXPECTED FREQUENCY OF OCCURENCE

Events	**Incidents**	**Accidents**		**Hypothetical Accident**
Plant state	Anticipated Operational Occurrence (AOO)	Design Basis Accident (DBA)	Design Basis Accident (DBA) and Design Extension conditions (DEC)	Beyond Design Basis Accident
Reference DiD Level	DiD Level 2	DiD Level 3.a	DiD Level 3.b DiD Level 4	-
Frequency of occurrence (f) [$year^{-1}$]	$f > 10^{-2}$	$10^{-2} > f > 10^{-4}$ (Unlikely events)	$10^{-4} > f > 10^{-6}$ (Extremely unlikely events)	$f < 10^{-6}$ (Hypothetical event)

Within this framework, the subdivision of the level 3 of DiD introduced by WENRA[8] can be applied and generalized. In both cases, a further layer of protection mitigates the scenarios due to their failure, implementing the DiD level 4. The difference between the sub-level 3a and sub-level 3b of DiD is not only related to the mitigation of single and multiple postulated initiating events (as in WENRA) but more generally related to the different severity of consequence of their failure.

The safety objectives in Table 8 are based on ITER [11] and DEMO [7] experiences and include the acceptable limits for the on-site, off-site early and off-site chronical doses. The numerical values specifying the quantitative criteria need to be defined for each specific installation.

A simplified classification scheme can be defined with reference to two levels of severity of consequence (medium and high). The threshold between these levels is defined by the maximum on-site dose (20 mSv/event, on-site dose should be occupational exposure or dose of workers) and off-site chronic dose (5 mSv/event, off-site chronic dose should be public exposure or dose of public). Note that no off-site early dose is acceptable for medium severity consequence; only medium severity consequence is acceptable for Incidents; only High severity consequence is expected for hypothetical accidents; design basis accidents may have medium or high severity consequence.

[8] WENRA subdivides the level 3 of DiD into a sub-level 3a, which covers the design-basis single failure events, and a sub-level 3b, which covers the design-basis multiple failure events and the design extension conditions.

TABLE 8 SAFETY OBJECTIVES TO BE MET FOR EACH EVENT CATEGORY

Events category	Incidents	Accident with Medium severity	Accident with High severity	Hypothetical Accident
Key physical parameters	Key parameters within acceptance criteria for AOO	Key parameters within acceptance criteria for DBA (single PIE)	Key parameters within acceptance criteria for DBA (multiple PIE) and DEC	-
On-site Dose	≤ 5 mSv / year	≤ 20 mSv / event	Considering the time constraints related to the management of the accident and post-accident situation	
Off-site Early Dose	No off-site radiological impact		≤ 10 mSv / event (to the most exposed individual member of the public).	≤ 50 mSv / event (to the most exposed individual member). No cliff-edge effects. Countermeasures limited in time and space.
Off-site Chronic Dose	≤ 1 mSv / event	≤ 5 mSv / event	≤ 50 mSv / event	No cliff-edge effects. Countermeasures limited in time and space.
Consequence severity	Medium severity		High severity	

3.3.3. Hazard identification studies

A thorough identification of all hazards in a plant is, of course, an essential step in a complete safety analysis. Only by being certain that all hazards are known can it be asserted that the plant contains all necessary safety provisions. Every identified hazard needs to be eliminated or reduced in frequency of occurrence, and its consequences need to be minimized. This is done in the design of SSCs providing a safety function, and therefore classified as safety important.

For hazard identification to be fully comprehensive it needs to be performed by a systematic method. The aim is to investigate every conceivable deviation from normal behaviour of items in the plant, discern all possible sequences of events that may follow, and note all potential consequences of these events. This is a very large task that can only be managed by using an established systematic method. If a Probabilistic Safety Assessment (PSA) is being performed, then the first stage is devoted to the comprehensive identification of events that could trigger an accident situation. But even if there is no PSA, the deterministic analyses need to be chosen to ensure that all possible consequences of failures or abnormal behaviour have been enveloped.

There are fundamentally two approaches to hazard identification: bottom-up and top-down methods of analysis. In bottom-up analyses, for every item at the lowest (most detailed) level of the PBS, the feasible types of failure or abnormal operation are identified, and the possible consequences noted. FMEA is the most common method applied to this approach. In top-down analysis, an undesirable event is selected (e.g. release of tritium to the environment), and at the first level down the combination of abnormal conditions that would be having to occur to enable it are listed. Subsequent levels show progressively more detail of the fault conditions involved until, at the lowest level, all component failures that contribute in some way to the outcome are identified. A global fault tree is an example of this approach in use. The bottom-up and top-down approaches are complementary; in a full safety analysis both may be applied in separate hazard identification studies as a check of consistency.

Hazard identification studies necessarily require a detailed design if they are to reveal the component-level failures that may lead to a safety consequence. Before the design is fully developed, or at the conceptual design stage, this is not possible.

Functional FMEA is an example of technique that can be applied at this design stage (see Section 6.2). Another technique that may be useful, particularly before the full detailed design is complete, is that of Hazard and Operability studies (HAZOP). In a HAZOP study, the plant is described as a set of nodes whose behaviours are characterized by many parameters. The effects of deviations from normal values of each parameter are systematically catalogued; the more significant ones are flagged for investigation.

One of the outcome of the hazard identification studies (including FMEA and HAZOP) is a list of elementary initiating events. This may be a very long list, and it is impractical to study event sequences that may develop for each one. But since many of them will have a similar effect on the plant, events may be grouped together, with all the initiating events expected to have a qualitatively similar impact in a group characterized by one PIE. The PIE chosen to represent each group of initiating events is the one expected to have the greatest impact. In this way a long list of identified initiating events can be reduced to a manageable list of PIEs.

3.3.4. Failures assessment

Failures assessments can support at various steps the process needed for the safety classification (and subsequent fabrication) of SSCs.

The safety classification of SSCs has to be derived from a ‘functional’ safety analysis developed for the identification of the (fundamental and supporting) safety functions implemented by the plant and its SSCs.

The failures assessment needs to be performed iteratively throughout the plant design. At least, a preliminary assessment focused on the plant’s and SSC functions and covering the objectives related to the hazards identification, needs to be followed by a re-assessment focused on the physical implementation of SSC and on their failures under the defined environmental and loading conditions. The failure assessment needs to be iterative also because the implementation of provisions for prevention and/or mitigation of abnormal situations can itself generate potential hazardous situations.

The operating and environmental conditions experienced by the SSC need to be identified and considered during the failures assessment. Indeed, the intervention of SSC not normally operating can be needed during a different phase, some failures can be only possible in these phases, and some failures can have different consequences and be more challenging for safety.

Scenario to be addressed comprise normal conditions (e.g. operation states, standby states, shutdown states, outage and maintenance states), off-normal conditions (as represented by the PIEs), and other lifecycle states (e.g. construction, commissioning and decommissioning).

The addressing of SSC failure modes with reference to each off-normal state allows assessing their role within the plant's 'safety architecture' and verifying its adequacy (e.g. the fulfilment of the single failure criterion, the presence and independence of subsequent DiD levels).

Moreover, the results provided by the failure assessment have to support the application of the construction code(s) selected for the SSC fabrication; specifically, the analysis has to provide the main failure modes to be taken as reference in the SSC design, design justification and qualification, including any relevant degradation and aging phenomena.

The FMEA is one of the earliest systematic methodologies for failures assessment. Different FMEA approaches are used in different engineering fields and applications. Their common objective is the identification of the relevant failure modes and the evaluation of their effects. FMEA typically includes the assessment of the failures causes (or mechanisms) and can be complemented by FMECA by the semi-quantitative appraisal (i.e. through classes) of the dimensions of risk (frequency/probability of failure, severity of consequences, and degree of inspectability/detectability in some cases). Details on the FMEA development are provided in Section 6.2.

3.3.5. Assignment of Structures, Systems, or Components to safety class

The safety classification of SSCs is performed through the application of SIC grading criteria, based on the information coming from failures assessment. Criteria are defined in Section 4.2, within the conceptual framework for risk acceptance introduced in Section 3.3.2.

The following general rules need to be fulfilled, despite the specific criteria adopted:

(a) SSC important for personnel or public safety and environment needs to be assigned to a safety class and credited in the safety assessment.
(b) Design solutions, quality assurance and demonstration of the capability, reliability and robustness needed to the SSC needs to be adequate to their safety classification.
(c) SSCs whose failure may challenge the assumptions made in the plant safety assessment need to be safety important and assigned to the most severe safety class of the SSCs whose capability, reliability or robustness could be degraded beyond the necessary limits.
(d) The failure of Non-SIC needs not to lead to the failure of a safety-important SSC (i.e. SIC).
(e) The failure of a SSC assigned to a certain safety class needs not to lead to the failure of another SSC assigned to a more stringent safety class (i.e. no failure of SIC-3/SIC-2 needs to lead to the failure of a SIC-2/SIC-1 respectively).

The fulfilment of the last two rules allows that:

— A safety important system/component may contain components/subparts non-safety important or belonging to a less stringent safety class (e.g. SIC-1 systems including Non-SIC or SIC-3 or SIC-2);
— No further recommendation is specified for SSC involved in the (physical) interface between systems with different safety classifications.

3.3.6. Safety architecture assessment

The response of the plant to each off-normal event is achieved by a 'layer of provisions' implemented to manage the 'mechanisms' challenging the safety function(s). For a given off-normal event, in the logic of DiD, the safety architecture has to address any possible deficiencies/failures, through an independent 'layer of provisions', functionally redundant, allowing the achievement of the safety objectives.

The representation and assessment of the safety architecture implemented by the plant has the objective to identify, for each plausible plant condition, i.e. for each initiating event and for each sequence generated by any plausible failures, the 'layers' of provisions that embody the different levels of DiD. They can be supported by the Objective Provision Tree (OPT)[9] and by the Line of Protection (LOP) methodology[10], consistently with the safety assessment process defined by the IAEA GSR Part 4.

OPT concerns the systematic identification, for a given level of DiD and for a given 'Safety Function' (SF) and corresponding safety objectives, of the 'challenges' to the SF and relevant 'mechanisms and phenomena' to be prevented or controlled by a proper set of 'provisions'.

The development of OPTs is possible from the early stages of the plant design and allows the identification of the SSCs that materialize the different levels of DiD, to be classified independently from the safety perspective. Fig. 8 represents the general structure of an OPT.

The development of a PSA for the plant, and more generally the use of probabilistic methods, could provide complementary insights for the assessment of the safety architecture implemented by the plant.

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[9] OPT method has been proposed within the Integrated Safety Assessment Methodology (ISAM) for Generation IV Nuclear Systems [12].

[10] LOP method has been used recently for the assessment of the Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID reactor).

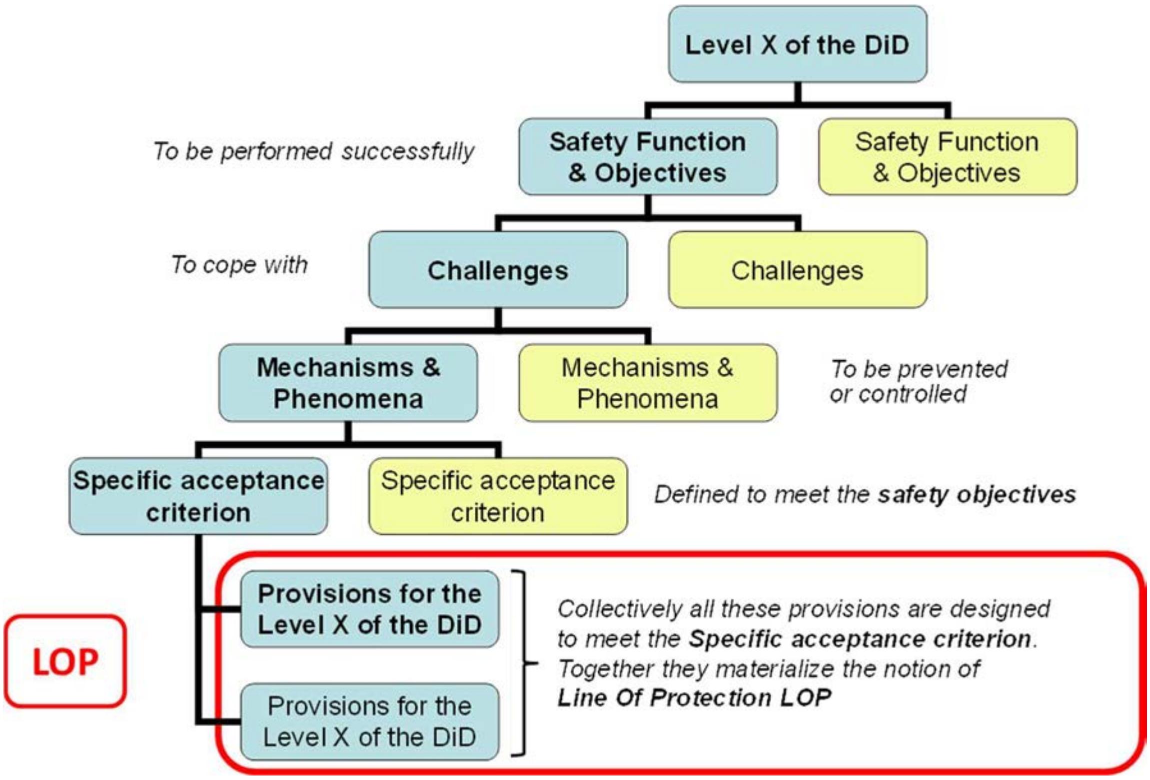

FIG. 8. OPT general structure and LOP identification.

In this regard, the subdivision between the PSA level 1[11] and level 2[12] evolved in the experience of fission nuclear power plants is not of relevance for fusion machines. The safety assessment of the plant is focused on the determination of the magnitudes and frequencies of radioactive releases. The assessment of the off-site consequences, i.e. the PSA level 3[13], could be a subsequent step.

Primarily, a probabilistic approach is embedded into the SSCs safety classification process being used for the categorization of events (including SSCs failure).

The PSA needs to be developed to assess the confinement response of the plant up to the determination of magnitudes and frequencies of radioactive release. It could support the assessment of the reliability and robustness of SSCs, the verification of the independence among the SSCs materializing different DiD levels (for a given event), and the optimization of the safety architecture implemented by the plant in the view of the ALARA principle.

Once risk acceptance criteria are defined, before the formal assessment through PSA, it is possible to roughly sketch the safety architecture of the plant from a probabilistic perspective, through the LOP methodology.

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[11] Level 1 PSA concerns the assessment of plant failures leading to core damage and the determination of core damage frequency (CDF).
[12] Level 2 PSA concerns the assessment of containment response leading, together with the results of Level 1 analysis, to the determination of release magnitudes and frequencies.
[13] Level 3 PSA concerns the assessment of off-site consequences leading, together with the results of Level 2 analysis, to estimates of risk to the public.

Consistently with the OPT representation, the LOPs are the sets of provisions against specific mechanisms and phenomena challenging a safety function, characterized through their performances, reliability and mutual independence.

Typically, ‘strong’ (10^{-3}–10^{-4} per year or per demand) and ‘medium’ (10^{-1}–10^{-2} per year or per demand) LOPs are defined, with reference to different reliability targets.

4. SAFETY IMPORTANT COMPONENT GRADING

This section provides the criteria proposed for the safety classification of the mechanical SSCs of fusion installations, within the context defined by the SSCs classification process described in Section 3 and specifically by the safety objectives and risk acceptance framework introduced in Section 3.3.2.

A graded approach to safety requires that SSCs with higher safety importance needs to assure the necessary performance (structural integrity and functionality) with higher reliability and robustness against internal and external hazards. Thus, safety important SSCs needs to be identified and classified. This classification has direct impact on the design and manufacturing of SSCs. The complex physical and functional architecture of fusion machines suggest the definition of detailed criteria for the identification and classification of SIC. Conversely, a simplified graduation is justified by the low hazard potential and positive safety characteristics of fusion machines compared with fission power plants[14].

4.1. Structures, Systems, or Components ‘roles’ in the safety architecture

The safety classification of SSCs need to consider their ‘role’ within the safety architecture of the plant and the consequence of their failure during normal and off-normal operation. Table 9 provides the list of the possible roles of a given SSC and the criteria to verify their applicability. Possible roles are defined with reference to the potential contribution of the SSC in the evolution of incidental and accidental scenarios, and specifically to its involvement in the implementation of the (fundamental and supporting) safety functions and to its capability to generate hazardous conditions.

[14] Specifically: plasma burn is terminated inherently when fuelling is stopped, is self-limiting about power excursions, and is passively terminated by the ingress of impurities; the energy and power densities are low, the energy inventories are relatively low; the releasable radioactive inventories are limited; physical barriers exist inherent to the tokamak concept (embedded rather than added).

TABLE 9. POSSIBLE ROLES OF SSC AND RELATED CRITERIA

Possible SSC roles	Criterion	Remark
Protection or mitigation SSC	SSC is needed to limit the consequences of an incident or accident that would lead to significant risks of exposure or contamination	It includes: • Provisions for the confinement of radioactive and hazardous materials; • Provisions for the limitation of non-radiological consequences of conventional hazards.
Passive system SSC for shielding	SSC is needed to limit of exposure to ionizing and electromagnetic radiation, under normal condition and during and after off-normal events	-
Supporting SSC	SSC is needed to ensure the functioning of Safety-Important SSCs	It includes auxiliaries for SSCs implementing functions for the prevention, protection or mitigation of accidents.
Hazardous SSC	SSC failure can directly initiate an incident or accident, leading to significant risks of exposure or contamination.	It includes: • Initiating event of incidental and accidental scenario; • Aggravating failures during an incidental and accidental scenario due to a different initiator; • Provisions for accidents prevention (detection and control of deviations from normal behaviour).

4.2. Criteria for the assignment of Structure, System, or Component to a safety class

The criteria for the safety classification of the mechanical SSC are consistent with the safety objectives and the risk acceptance framework defined in Section 3.3.2. They are defined considering the specific features of fusion machines, being consistent with IAEA requirements and specifically with the DiD concept, as well as with the SSG-30 [2] and the ITER [11] and DEMO [7] experiences. They refer to:

— The safety function(s) to be performed by the SSC;
— The consequences of SSC failure to perform the needed safety function;
— The classification of the event requiring the safety function (i.e. the frequency with which the SSC will be called upon to perform the function);
— The role of the component in the achievement of a controlled or safe state of the plant, based on the scenario evolution.

Two different classifications are proposed with reference to the capability of SSCs[15]:

— To lead to or mitigate AOOs and DBAs;
— To mitigate DEC and BDBAs not-practically eliminated.

Three safety classes are proposed for the SIC, i.e. SSCs leading to or mitigating AOO and DBA: ‘SIC-1’, ‘SIC-2’, ‘SIC-3’[16]. A complementary class includes all the SSCs without any safety relevance: ‘Non-SIC’.

A fourth safety class is proposed for the classification of the SSC mitigating DEC and BDBA not-practically eliminated: ‘HCC’. A complementary class includes all the SSCs without any relevance: ‘Non-HCC’. All the general rules provided in Section 3 need to be fulfilled.

One or more safety classification criteria are defined for each safety class. The fulfilment of at least one criterion leads to the assignment of the SSC to the related safety class.

For a given SSC, the classification criteria need to be applied for each one of its possible roles. If different classifications are defined for the different roles of the same SSC, the most severe one has to be assumed.

Criteria for SIC-1 have to be applied first; if (at least) one criterion is met, the SSC is a SIC-1 (i.e. the SSC is assigned to SIC-1 class); otherwise, criteria for SIC-2 are applied, and so on. If no criterion is met for any safety class and possible role, the SSC is Non-SIC / Non-HCC.

The following Tables 10–13 provide the criteria defined for the assignment of SSCs to the different safety classes, and some examples related to a Tokamak machine. Each table refers to a single safety class and provides the criteria to be applied to a given SSC for each one of its role.

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[15] This dual classification allows managing the SSC assigned to different classes against AOO/DBA and against DEC/BDBA. E.g. SIC-2 with respect to DBA (which generally requires a less stringent qualification than for SIC-1) and HCC with respect to DEC/DBDA (which could require stringent qualification than for SIC-1, e.g. against seismic event).

[16] Note that the SIC-3 include the SSCs classified in Category 3 according to the SSG-30 (i.e. low consequences).

TABLE 10. CRITERIA FOR THE ASSIGNMENT TO SIC-1 CLASS

SIC-1 class	
SSC role	**Criteria**
Hazardous SSC	• SSCs whose failure can result in high severity consequences, including: ➢ Failures without effective and/or reliable protection (to be mitigated). E.g. SSCs materializing the first confinement system (e.g. vacuum vessel and its extensions, e.g. isolation valves), Tritium process safety-important SSCs, Cooling circuit with significant inventories of tritium and activated corrosion products. ➢ Failures without effective and/or reliable mitigation (to be practically eliminated). E.g. catastrophic failure of vacuum vessel.
Protection or mitigation SSC	• SSCs implementing safety functions needed to bring to and to maintain the plant in a controlled* or safe state after an incident or a design-basis accident (AOO or DBA), and whose failure (when challenged) can result in high severity consequences. (DiD L3). E.g. vacuum vessel, pressure suppression system, detritiation system. *A SIC-2 is needed to bring to and to maintain the plant in a safe state.
Supporting SSC	• SSCs ensuring the capability, reliability and robustness needed to (other) SIC-1. E.g. VV support, emergency electrical power supplying for active SIC-1, safety instrumentation and control for SIC-1.
Passive SSC for shielding	• Passive SSCs protecting ➢ Workers and public from harmful effects of radiation; ➢ Safety-Important SSCs from damages due to internal or external hazards and whose failure can result in high severity consequences.

TABLE 11. CRITERIA FOR THE ASSIGNMENT TO SIC-2 CLASS

SIC-2 class	
SSC role	**Criteria**
Hazardous SSC	• SSCs whose failure can result in medium severity consequences. E.g. SSCs providing confinement of limited radioactive inventories, e.g. some cooling circuits, tritium components, detritiation system.
Protection or mitigation SSC	• SSCs implementing safety functions needed to bring to and to maintain the plant in a controlled or safe state after an incident or a design-basis accident (AOO or DBA), and whose failure (when challenged) can result in medium severity consequences. (DiD L3). E.g. protection and mitigation systems as some fire extinguishing systems in area with tritium inventory, Local air cooler to facilitate the intervention of critical mitigation system. • SSCs implementing mitigation for the failure of SIC-1 after an incident or a design-basis accident (AOO or DBA), and whose failure (when challenged) can result in high severity consequences. (DiD L4). E.g. SSCs materializing the second confinement system (e.g. building, shutter valves). • SSCs implementing safety functions needed to bring to and to maintain the plant (already in a controlled state) in a safe state after an incident or a design-basis accident (AOO or DBA), and whose failure (when challenged) can result in high severity consequences. E.g. detritiation system as mitigation of given initiators.
Supporting SSC	• SSCs ensuring the capability, reliability and robustness needed to (other) SIC-2. E.g. area radiological monitoring, Fire detectors, waste management facilities.
Passive SSC for shielding	• Passive SSCs protecting ➢ Workers and public from harmful effects of radiation; ➢ Safety-Important SSCs from damages due to internal or external hazards and whose failure can result in medium severity consequences.

TABLE 12. CRITERIA FOR THE ASSIGNMENT TO SIC-3 CLASS

SIC-3 class	
SSC role	**Criteria**
Hazardous SSC	• SSCs designed to detect and control deviations from normal operation, to reduce the frequency of an accident. (DiD L1/L2).
Protection or mitigation SSC	• SSCs implementing safety functions needed to bring to and to maintain the plant (already in a controlled state) in a safe state after an incident or a design-basis accident (AOO or DBA), and whose failure (when challenged) can result in medium severity consequences. • SSCs having some safety relevance / implications that are not credited in the safety assessment. E.g. emergency lighting, some ventilation systems, some cooling water subsystems, RH equipment with no confinement function.
Supporting SSC	• SSCs ensuring the capability, reliability and robustness needed to (other) SIC-3. E.g. emergency electrical power supplying for emergency lighting, ventilation systems.
Passive SSC for shielding	• Passive SSCs protecting workers, whose failure can result in consequences (dose) above the authorized limits*. * if less than the limits for medium severity.

TABLE 13. CRITERIA FOR THE ASSIGNMENT TO HCC CLASS

HCC class	
SSC role	**Criteria**
Hazardous SSC	-
Protection or mitigation SSC	• SSCs implementing safety functions needed to mitigate the consequences of Design Extension Conditions (DEC) and Beyond Design Basis Accidents (BDBAs) not-practically eliminated, including external hazard events, and whose failure (when challenged) can result in high severity consequences. E.g. shutter valves to complete the confinement boundary materialized by the Tokamak building at wall penetrations.
Supporting SSC	-
Passive SSC for shielding	-

4.3. Further recommendations and remarks

Regarding the implementation of the DiD concept, the following further recommendations are specified according to IAEA SSG-30 [2] and TECDOC-1787 [10]:

- “The assessment of the safety significance of SSC used to control AOO should be performed assuming that other functions (for AOO or DBA) will respond as expected (provided that the associated SSC are not affected by the same initiating event).”
- “The safety significance of SSC used to control AOO cannot be lowered because an independent safety function mitigating DEC is implemented.”
- “The assessment of the safety importance of SSC used to mitigate the consequences of DBA or DEC should be performed ignoring the other functions allocated to other DiD levels.”

The fulfilment of the above recommendations assures a conservative classification of SSC with respect to the results of a detailed representation of the safety-architecture of the plant, i.e. by the analysis of the possible sequences of events (initiating events and loss of safety functions at different levels of DiD). However, the safety classification of SSC does not assure that the implemented measures are sufficient to meet the safety objectives nor that the residual risk has been reduced ALARP.

With respect to using probabilistic methods, according to the SSG-30 [2], “deterministic methodologies should be applied, complemented where appropriate by probabilistic safety assessment and engineering judgement to achieve an appropriate risk profile”. At this regard, the integrated use of probabilistic and deterministic assessments is not further developed by the SSG-30 [2] and TECDOC-1787 [10]. IAEA GSR Part 4 [3] defines the global framework for the safety assessment “by means of deterministic and probabilistic methods”, recognizing that “probabilistic approaches may provide insights on system performance, reliability, interactions and weaknesses in the design, the application of DiD, and risks, that it may not be possible to derive from a deterministic analysis[17].”

Within a probabilistic context, for instance, the reliability targets to be achieved by SSC need to be defined with reference to the frequency of occurrence of the initiating event(s) to be managed, to the subsequent layers of provisions (including SSC) implementing redundant safety functions according to the DiD principle, and to the safety objectives to be met (i.e. to the maximum frequency of occurrence related to the event category).

Even if methods and criteria for the integrated use of deterministic and probabilistic safety analysis are not well established and subject of on-going discussions among experts, the assessment of the safety architecture implemented by the plant seems essential to verify the safety classification of SSC (see Section 3). This assessment needs to include the verification of:

- The global capability, reliability and robustness of the SSC materializing the different levels of DiD needed to achieve the safety objective (for a given initiating event);
- The independency between the SSCs materializing the different levels of DiD (for a given initiating event);

[17] With reference to DiD concept, if it can be demonstrated (by Deterministic Safety Assessment) that the installation follows all the applicable DiD principles, and if an independent PSA confirms a low risk for the plant, there would be a well-founded confidence in an adequate level of safety; on the other hand, if PSA identifies a high or unbalanced risk profile for the plant, there are doubts on the adequacy of the current application of the DiD concept and additional safety provisions are expected.

— The implementation of specific provision to manage single and multiple design-basis accidents (through the level 3a and level 3b of DiD).

The representation of the 'safety architecture', e.g. through OPT (see Section 3.3.6), supports this assessment giving information about the 'layers of provisions' (SSC) that implement the (challenged) safety function at the different levels of DiD, for each given initiating event.

Fig. 9 provides examples of safety important SSC through the conceptual framework for risk acceptance introduced in Section 3.3.2. Colour coding is used to indicate the safety classes HCC, SIC-1, SIC-2, SIC-3 as defined in Section 4.2.

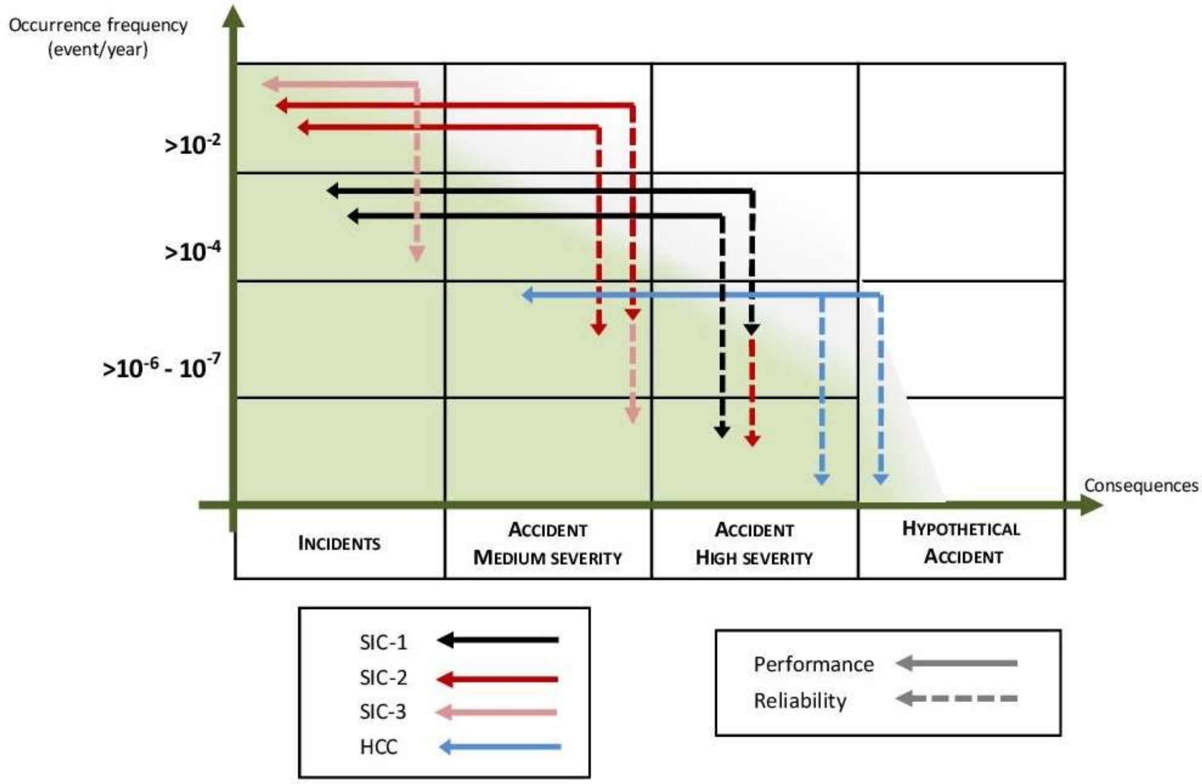

FIG. 9. Risk domain and examples of SSCs classification.

5. FAILURE MODES

Potential modes of structural failure to be considered are as follows:

(a) Ductile modes: plastic collapse, plastic instability, fast fracture;
(b) Non-ductile modes: plastic flow localization, local fracture due to exhaustion of ductility, brittle fracture;
(c) Buckling;
(d) Ratcheting;
(e) Fatigue;
(f) Creep;
(g) Overturning (overall stability);
(h) Corrosion/erosion.

For safety consideration non-structural failure modes need to be considered such as operational malfunctions leading to leakages. Most of the design codes provide rules for prevention of structural failure modes, which are discussed more in detail in this section. Most of existing structural codes do provide corrosion allowances, but these may not cover some specific causes and mechanisms such as flow-assisted corrosion, plasma erosion, etc. which may have to be considered at start of life and at end of life.

A design substantiation process then involves carrying out stress analysis of the component under needed design loading conditions (normal as well as fault conditions) to assess the load effects to preclude all the potential modes of failure. There can be several issues facing a designer e.g. strength, reliability, thermal considerations, corrosion, friction, safety, weight, size, stiffness. It is important to know which issue is being considered for a given design condition. To ensure adequate strength, stiffness and hence, safety and reliability, a typical design process will follow the following steps:

— Understanding the structural behaviour: whilst it is easier to understand deformation behaviour of a simple system, computational simulation techniques like the Finite Element Method (FEM) can be of great benefit in ascertaining the structural behaviour of a complex system and its response under complex loads. For example, behaviour of a structure under dynamic or transient load may be totally different from that under static load. Care needs to be taken when representing the real structure by a mathematical model.
— Identifying the failure modes: analysis can identify the weakness in the structure/component and identify vulnerability to failure mainly due to plasticity, fracture, fatigue or buckling.
— Allowable limits for failure modes: having established the failure mode of a structure/component, the maximum stress or load causing the failure is determined. This defines the load carrying capacity of the structure, which is used to determine the safety margin by comparing against the allowable limit. The capacity may vary depending on the material ductility or lack of ductility, and other effects like ageing, corrosion and irradiation embrittlement.

The main objective of a structural design is to ensure that a structure remains fit for purpose, and can sustain all loadings without failure. The type of loadings depends on the type of structure and its operational requirements.

Several types of damage are considered for fusion components, comprising ductile damage modes, which include immediate plastic collapse, immediate plastic instability, and time-dependent plastic instability, as well as non-ductile damage modes resulting mainly from the loss of ductility and strain-hardening capability of materials when subjected to a neutron irradiation environment (these include immediate plastic flow localisation, immediate local fracture due to exhaustion of ductility, fast fracture). The design rules do not cover measures to prevent damage resulting from erosion or corrosion [13] and to ensure the correct operation of components with mechanisms or moving parts. Fatigue is a dominant failure mode in fusion because plasma disruptions produce transient electromagnetic and thermal stresses which are short in time, restricted to a thin skin in bare material and vary cyclically causing fatigue. Neutron irradiation can cause embrittlement, swelling, irradiation-induced creep, time-dependent material properties changes.

5.1. Typical failure modes for Safety Important Components

The following Tables 14–15 summarise general failure modes and mechanisms for typical components forming part of a confinement system. The information provided are only indicative since the precise characterisation of the failure modes requires the knowledge of the actual components, manufacturing and assembly processes, operational modes and off-normal conditions, as well as environmental and ageing effects.

TABLE 14. FAILURE MODES AND MECHANISM FOR TYPICAL COMPONENTS

	Passive metallic components (e.g. structures, parts, mechanical feedthroughs)	**Welded assemblies**	**Bolted flange assemblies**
Failure modes	Ruptures (and leakages) caused by: • Plastic collapse; • Fracture; • Creep; • Ratcheting; • Fatigue; • Buckling. Deformations exceeding functional limits.	Ruptures (and leakages) caused by: • Plastic collapse; • Fracture; • Creep; • Ratcheting; • Fatigue.	Ruptures (and leakages) caused by: • Plastic collapse; • Fracture; • Creep; • Ratcheting; • Fatigue; • Loss of bolts preload.
Failure effects	• Damage to SIC; • Loss of confinement.	• Damage to SIC; • Loss of confinement.	• Damage to SIC; • Loss of confinement.
Failure mechanisms	Typically associated with presence of: • Geometrical and physical discontinuities; • Defects and variations in the microstructure; • Residual stresses;	Typically associated with presence of: • Geometrical and physical discontinuities; • Defects and variations in the microstructure; • Residual stresses;	• Thread stripping of the internal or external threads; • Tensile or fatigue failure of bolt shank; • Overloading and excessive bearing stress under nut, bolt head or within joint; • Insufficient contact pressure between flanges and seal.
Ageing and environmental conditions	• Irradiation-induced effects (creep, swelling and loss of ductility); • Corrosion and erosion; • Temperatures effects.	• Irradiation-induced effects (creep, swelling and loss of ductility); • Corrosion and erosion; • Temperatures effects.	• Irradiation-induced effects (creep, swelling and loss of ductility); • Corrosion and erosion • Temperatures effects (e.g. differential thermal expansion).
Manufacturing and assembly considerations	• Residual stresses and strains from manufacturing processes.	• Residual stresses and stress concentrations from fabrication; • Changes of properties in the joint zone (hardness, strength, toughness, ductility, corrosion resistance); • Weld imperfections (slag inclusions, porosity, undercut etc.), shape defects and initial cracks (e.g. solidification cracking); • Weld distortions and changes to nominal stress distribution.	• Installation and tolerances between seal and seat might cause load asymmetry and uneven plastic deformation of seal.

TABLE 15. FAILURE MODES AND MECHANISM FOR TYPICAL COMPONENTS

	Expansion bellows	**Ceramic assemblies (e.g. windows, electrical Feedthrough)**	**Active mechanical components (e.g. vacuum/fluid valves and pumps)**
Failure modes	Ruptures (and leakages) caused by: • Plastic collapse; • Fracture; • Creep; • Ratcheting; • Fatigue; • Buckling. Deformations exceeding functional limits.	For ceramic parts: • Brittle fracture; • Fatigue. For ceramic-to-metal joints: • Plastic collapse; • Fracture; • Creep; • Ratcheting; • Fatigue.	Ruptures (and leakages) caused by: • Plastic collapse; • Fracture; • Creep; • Ratcheting; • Fatigue; • Buckling. Failure of actuation (operability issues). Deformations exceeding functional limits.
Failure effects	• Damage to SIC; • Loss of confinement.	• Loss of confinement.	• Loss of confinement (passive function) • Loss of isolation (active function)
Failure mechanisms	• Column instability (column squirm); • In-plane deformation of the convolution side wall (in-plane squirm); • Creep-fatigue; • Burst and collapse over stretching.	• Exceeded strength due to thermal expansion mismatch between ceramics and metallic mounts; • Crack formation and propagation from microscopic flaws; • Fast fracture.	Valves: • Thermal binding of gate; • Defects/damages of seals and seats; • Loss of actuation due to damage of actuator system. Pumps: • Shaft fracture; • Cavitation damage.
Ageing and environmental conditions	• Irradiation-induced effects (creep, swelling and loss of ductility); • Corrosion and erosion; • Temperatures effects.	• Irradiation-induced effects (dimensional change, changes of mechanical/thermal/optical properties); • Corrosion and erosion of ceramic-to-metal joints; • Temperatures effects.	• Irradiation-induced effects (dimensional change, changes of mechanical/thermal/electrical properties); • Temperatures effects; • Presence of dust particles in the environment; • Erosion/corrosion (for fluid equipment).
Manufacturing and assembly considerations	• Failure modes strongly depends on bellow construction (e.g. edge welded or hydro-formed).	-	-

6. FAILURE MODES AND EFFECTS ANALYSIS AND DAMAGE LIMITS

6.1. Introduction

It is always desirable to discover any design deficiencies at early stages of a design to ensure safety of a system. In case of a complex engineering system with several components, the failure may not always be due to a failure of a component but can be due to human factors in operation or maintenance. Since the scope of this document is to give guidance on safety classification of mechanical components, only hardware failures will be considered here as procedural failures require human factors analysis which is out of scope of this document. There are design and safety review techniques like the FMEA, Failure Tree Analysis (FTA) and Event Tree Analysis (ETA) that help to understand how failures can occur and estimate their probability of occurrence.

FTA and ETA are deductive techniques in which a logical procedure is used to identify all possible causes/events that may lead to total failure of the system resulting in a top event like a fire, explosion or release of radioactivity. FMEA (discussed earlier in Section 3.3) is an inductive technique which is most widely used in initial stages of system design and therefore very relevant to designers. FMEA can help identify failure modes and failure frequencies at component or system level. Even more important, in this context, is to know how to reduce the probability of their occurrences. As has already been explained in Section 3, undesirable events can be brought under control by reducing either their probability of occurrence and/or their impact on safety.

In this section, a brief description is given of the design and safety review techniques like FMEA which can help identify the safety role of the mechanical components in a system and evaluate consequences of their failure. Failure modes of mechanical components are then defined and guidance is given on the acceptable damage limits depending on the importance of the safety role assigned to the component.

6.2. Failure Modes and Effects Analysis

Different FMEA approaches are used in different engineering fields and applications. Their common objective is the identification of the relevant failure modes and the evaluation of their effects. FMEA typically includes the assessment of the failures causes (or mechanisms) and can be completed by the semi-quantitative appraisal (i.e. through classes) of the dimensions of risk (frequency/probability of failure, severity of consequences, and degree of inspectability/detectability in some cases).

Functional as well as physical failure assessments need to be carried out.

FMEA is usually performed during the conceptual design stage with the aim of identifying the different failures and failure modes that can occur at the component, sub-system or system level. It helps to determine the effect of the component failure on the following:

— Overall performance of the system;
— Ability to meet the performance requirements.

6.2.1. Functional failures assessment

In general, SSC needs to be credited with some function to prevent or reduce the consequences of an incident or accident, then that function needs to be included in the safety or supporting functions list and the SSC needs to be classified as safety important.

At the concept design stage, the assessment of the plant's failures needs to be developed from a functional perspective. The analysis has to be focused on the functions carried out by the plant and its SSC, rather than their physical implementation.

The functional failures assessment need to be developed with the following objectives:

(a) To investigate systematically the effects of SSC failures in the missed or incorrect execution of each implemented function;
(b) To identify the (fundamental and supporting) safety functions implemented by the plant and its SSC;
(c) To support the definition of a preliminary list of the PIEs and to support their categorization;
(d) To provide information for the representation of the plant's safety architecture.

The main results expected from the functional failures assessment need to include:

— The detailed lists of safety functions (confirming or integrating the list provided in Section 2);
— The preliminary list of PIEs (for each plant operating condition) and related categorization;
— The evaluation of consequences of the (functional) failure modes (for each plant operating condition);
— The categorization of the safety functions and a preliminary classification of SSCs, i.e. assignment of each SSC to the relevant safety class, according to predefined criteria (SIC grading).

A functional FMEA can be developed to obtain the above results. The analysis has to consider the effects of loss of functions from the FBS (see Section 3.1.2), instead of analysing the consequences of failures of components from the PBS (see Section 3.1.1). SSCs need to be defined (at least) in terms of functions implemented and external (mechanical, electrical, hydraulic, pneumatic) interfaces.

The functional FMEA needs to be based on a comprehensive representation of the functions implemented by the plant and its SSC. For each loss of function identified, the potential consequences are noted and, if the design information allows it, possible event sequences are elaborated. If the MLD representation of the plant is available, the possible causes of each loss of function are identified in terms of failure of the SSC in the execution of the necessary function. A preliminary version of OPTs of the plant, if available, can support the functional FMEA and can be refined through the analysis.

The availability of the preliminary list of PIEs (for each plant operating condition) and the related categorization allow defining a set of loads and environmental conditions for the subsequent iteration of the failures assessment.

6.2.2. Physical failures assessment

When the design of the plant is advanced, the failure of the SSCs can be analysed considering their physical implementation. A physical FMEA needs to be developed with the following objectives:

(a) To identify the failure modes to be considered during the design, design justification and qualification of the SSC, consistently (as much as possible) with the selected construction code(s);
(b) To verify the completeness of the list of PIEs, and to integrate and/or to modify this list as needed;
(c) To assess the physical interface of SSCs as additional causes of SSCs failures and/or pathways for the propagation of their effects among systems;
(d) To investigate systematically the effects of all the credible failure modes of the SSCs, during each reference scenario, verifying (i.e. confirming or updating) their safety classification;
(e) To support the definition of the acceptable damage limits of SSCs during each reference scenario, consistently with the safety class assigned.

The main results expected from the physical failures assessment need to include:

— The potential failure modes of each SSC, defined as much as possible in compliance with the selected construction code(s), and referred to the identified PIEs;
— The consolidated assessment of the external and internal causes and consequences of each credible failure modes of each SSC, for each defined scenario;
— The consolidated list of PIEs (for each plant operating condition) and the related categorization;
— The consolidate categorization of safety functions and classification of SSCs.

A physical FMEA can be developed to obtain the above results. The analysis has to be focused on the single components of the different systems (such as pipes, manifolds, pumps, heat exchangers, coolers, electric heaters, isolation valves, control valves, relief valves).

To be fully systematic, the physical FMEA needs to be developed for each defined scenario, including the normal operation of the plant, and the loads and environmental conditions due to each identified PIEs.

The 'external' causes of failure need to be identified explicitly in terms of environmental and loading conditions, in addition to the 'internal' causes (random or systematic mechanisms) of failure of the SSC.

Each failure condition needs to be referred to one of the PIE identified (preliminary by the functional FMEA), according to the expected consequences. If the event is not included or enveloped by the accidents in the list, a new PIE is introduced. Moreover, PIEs can be modified by changing their representative event.

Combination of SSC needed to deliver a safety function (in a particular condition) needs to be considered and classified as a single group (e.g. single SSCs mechanically, electrically, hydraulically or pneumatically connected). This has to include, for instance, the Instrumentation and Control (I&C) needed for the SSC operation and/or monitoring and the items providing cooling and power supply.

A final remark concerns I&C needed by many mechanical systems in a fusion installation. Among them, there are active components involved in the execution of the main safety functions, e.g. which complete the (dynamic) first confinement of the Vacuum Vessel (VV) at the different penetrations (typically protecting more vulnerable components that act as passive barriers during operation, i.e. windows). The functionality of I&C is a safety issue to be assessed in their design and qualification as well as their structural integrity. Being performed through a functional perspective, the classification of the SSCs encompasses its mechanical parts as well as I&C needed for its functionality. Even if some inconsistencies between international standards and local regulations exist, the approach (proposed by IEC 61226 [16]) for the design of the I&C architecture is consistent with a safety architecture implementing the DiD concept. Indeed, it is based on the independence among functions operating as different 'lines of defence' and on the fulfilment of constraints about the physical separation and electrical isolation of systems, the environmental conditions, the plant layout, the testing and maintenance operations.

6.2.3. Further recommendations for Failure Modes and Effects Analysis

Some further recommendations[3] about the use of the FMEA to support the SSCs classification are provided in this Section.

If two or more SSCs operate closely together and are in the same environment or share other similarities introducing common causes of failure, they need to be singularly analysed in the FMEA but considered as a single group in the safety classification.

If two or more SSCs providing the same safety function are identified, the FMEA could provide useful information about the effectiveness of this redundancy in any postulated conditions. Preferably, one of them needs to be identified as the principal means (and the other as backup) and designed with the higher safety classification. In relation to the level of independence that can be demonstrated for these SSCs (for a given PIE), they could materialize different levels of DiD within the safety architecture (e.g. isolation valves through the VV extension and shutter valves at Port cell penetrations) or just a redundancy at the same level of Did (e.g. isolation valves and diamond windows through the VV extensions)

Sometimes, criticality or severity of failure is also included in the analysis and in that case, it is called Failure Modes, Effects and Criticality Analysis (FMECA). Following is a summary of typical steps in an FMECA:

(a) Identify all the potential failure modes of the system.
(b) Relate the causes, effects and hazards of each mode of failure.
(c) Prioritise the failure modes relative to their probability of occurrence, severity and detection capability.
(d) Provide suitable follow-up or corrective actions for each type of failure mode.

The above information plays crucial role in safety classification of mechanical components and helps a designer to select appropriate design codes and the allowable damage limits. This allows the conceptual design to progress towards functional design and ultimately to substantiation of the final design. Therefore, the strategy and rationalisation of the applicable codes and standards need to be confirmed during the final design. Any justification for use of the selected codes and standards need to take due account of the legislation, regulations, construction codes, structural integrity assessment criteria, irradiation effects, manufacturability including welding, NDE/NDT and QA.

6.3. Role of design codes and standards

The overall objective of a designer is to ensure that a component can safely withstand all loading conditions and remain fit for purpose throughout its design life. This requires assessing load effects. All loadings are either resisted internally or transmitted to other components in which case safe load paths and connections also need to be assessed. Designers generally rely on standards or codes of practice to design.

Due to complex features of fusion facility, many different codes may have to be used because the fusion reactor components need to deal with specific environmental conditions (high temperatures combined with electromagnetic fields and nuclear radiation), complexity of loads (nuclear heating and electromagnetic forces), special materials (i.e. non-metallic or plasma facing materials) and the lack of in-service inspection.

It is necessary to establish the safe loads that have sufficient margin of safety relevant to the consequence of failure. Codes and standards consider these issues and specify safe allowable limits for each type of load effect.

6.3.1. Industry codes and standards

Currently, in the fusion industry, there are two existing industrial codes for nuclear safety related components: RCC-MR Edition 2007 [14] and ASME III [15]. RCC-MR has been updated to RCC-MRx 2015 which includes Eurofer grade and future edition 2018 will include irradiated material data. Another fusion code being developed for fusion is ASME III Division 4. ASME III is the result of several decades of nuclear and industrial experience and has a long history of safe use for design, construction and operating of nuclear power plants. The code strategy reflects a continuous attempt to understand all failure modes, and provide rational margins of safety against each type of failure. Some of the significant features of the latest version of the ASME III are: the explicit consideration of thermal stress, the recognition of fatigue as a possible mode of failure, the use of plastic limit analysis and prediction of ductile failure after some plastic action.

Furthermore, both codes (RCC-MR and ASME III) define limitations to protect components from failure due to the application of mechanical and/or thermal loads. The addressed failure mechanisms are substantially the same: excessive deformation (plastic collapse), plastic instability, elastic or elastic-plastic instability (buckling), and degradation mechanisms associated to cyclic loading (i.e. progressive deformation induced by repeated loads ratcheting/shakedown and fatigue).

However, some general differences between the codes (RCC-MR and ASME III) can be identified:

(a) RCC-MR rules are organized according to the damages that are possible at high temperature, which is different from how ASME code is organized.

(b) RCC-MR specifies the Level A service limits for normal operating conditions and current incidents; thus, the ASME level B does not exist in RCC-MR (the design pressure cannot be exceeded during incidents).
(c) RCC-MR specifies more stringent Level C service limits than in ASME code.
(d) RCC-MR tends to specify higher load factor (allowable/actual) but there are differences in the way design allowable limits are calculated from the material data (i.e. values for a given material and temperature may not be the same in the two codes); the comparison is misleading.
(e) The analysis methods for class 1 and class 2 vessels are the same in RCC-MR, differently from ASME.
(f) There is no fatigue rule exemption for class 1 components, but it is not of concern for ITER.
(g) They differ significantly about the treatment of creep damage, but it is not of concern for ITER.

Additionally, other codes derived from ASME III and used for design of class 1 components are: French AFCEN RCC-M, Japanese JSME S NC-1, Korean KEA KEPIC-MN and Canadian CSA N-285. ASME BPVC is an international design and construction code covering the full life-cycle of pressure equipment, including the design and the construction phases. It is a single code covering the design by test, by analysis, by experiment, and by previous experience, the prescriptions on materials and on testing procedure, the links between design code and testing standards (ASTM), and wide and established database of test procedures.

The RCC-MR (RCC-MRx now) is a nuclear code that was originally developed for the French Fast Breeder Reactor programme; therefore, rules were focused on low primary stress but large and rapid thermal gradients. The RCC-MR code is being enlarged with the integration of more European harmonized standards and includes a specific part for ITER Vacuum Vessel. Now RCC-MRx includes also rules for irradiated components (research reactors).

6.3.2. Design code principles

Design codes are continuously being improved to include new knowledge and experiences. It is conceivable that new design codes may be written for fusion components for reasons which have been explained earlier. Past, present and future design codes used in a design need to specify acceptable damage limits to ensure safety. Before defining the damage limits, it is important to understand the concept of two alternative design principles.

— Allowable Stress Design: these codes specify a safe allowable stress limit which is usually a fraction of the material yield strength or the buckling strength. The maximum value of the actual load is taken as the design load and the stresses under the design conditions need to be less than the specified allowable stresses. Structural codes like AISC (American) and pressure vessel codes like ASME III (American), RCC-M and RCC-MRx (French) are all allowable stress codes, which are commonly used in industry.
— Limit State Design: in these codes a structure is designed to reach a limiting state in which the maximum design loads are multiplied by a factor and the strength (yielding or buckling) is reduced by dividing by a factor. These factors, known as partial safety factors, depend on the uncertainty (scatter) in the load and material data. These factors are derived from statistical analysis and are calibrated to ensure certain level of reliability.

The new Eurocodes are limit state design codes. Probabilistically calibrated partial safety factor approach is being considered for the DEMO Design Criteria [7].

It is thus essential, before using a code dedicated to the design of fusion reactor equipment, to show the adequacy of the code with the safety requirements of the equipment to be designed. If need be, the code can be revised, to bring the elements of demonstration necessary for that (studies, tests). Thus, for the design of ITER vacuum vessel, code RCC-MR, initially developed for fast breeder and high-temperature reactors, was supplemented by a specific appendix dedicated to ITER vacuum vessel.

6.4. Damage limits

The linear design rules according to ASME III are purely based on the concept of design stress on the component and stress limits of the material. This historical approach was based on simple engineering beam/shell theory. Stresses calculated on the component are first divided into membrane, bending and localized stresses, then categorized into primary, secondary and peak stresses. The primary stresses results from the load applied to the structure, i.e. dead-weight, pressure, external forces, are not self-limiting, and need to be limited to avoid catastrophic failure and to control plastic deformations. The secondary stresses are self-limiting stresses due to thermal effect or constraint effect at material and geometrical discontinuity that need to be limited to avoid progressive/incremental deformation. The peak stresses are the combined stress concentration and some thermal stresses which are used to control fatigue failure.

For each stress category a design limit is defined according to design criteria for assessment. The design verification is performed through comparing the design stress for the stress categories with the related value of the material stress limits.

The design process accounts for different type of load with different occurrence probability and groups loads in different load sets (design conditions): normal operation (Level A), upset (Level B), emergency (Level C), faults (Level D). The design evaluation needs to be conducted for each load set with appropriate criteria and requirements, progressively less restrictive, but adequate to maintain in all cases the integrity of the pressure boundary.

The safety is assured through the consideration of the failure and damage, defined by various design criteria and rules for each of these load sets. In such a way, stress states for loads in design and service limit Level A are under the lowest allowable limits, thus ensuring the best normal operation condition. Meanwhile, stress states for loads in service limit level D can be controlled in such a way that the highest level of material damages is allowed, but a catastrophic damage can be prevented.

In accordance with ASME III Division 1, a non-linear design evaluation is an alternative to the linear design evaluation. Depending on which stress intensity limit is violated in the linear design evaluation, there are (two) types of non-linear analysis needed in ASME III for the alternative non-linear design evaluations:

- Collapse-load analysis;
- Non-linear transient analysis.

Generally, such analyses can be conducted effectively using general-purpose finite element software. When inelastic analysis is used, gross plastic deformation is prevented by restricting the allowable load with respect to either the limit load or the plastic load of the vessel. The limit load is the maximum load satisfying equilibrium between external and internal forces when an elastic perfectly-plastic material model and small deformation theory are assumed. The plastic load is based on a more complex analysis, which may include large deformation effects and/or material strain hardening. When large deformations are significant, the material may exhibit geometrical strengthening or weakening. When strain hardening is included, plastically deformed material can support stresses greater than yield, enhancing the strength of the material.

The prevention by design of structural failure due to mechanical/thermal loadings is well addressed by the French codes, including load specification, design, analysis, manufacturing quality. For both the ASME codes and the RCC-MR, failure modes due to corrosion and irradiation of materials have to be covered by additional requirements. RCC-MRx covers irradiation of materials.

RCC-MR distinguishes between two broad types of possible damages, P type and S type. The P type damages result from the application of a steadily increasing load or constant load. The S type damages occur due to repeated application of loading. The P type damages include immediate excessive deformation, immediate plastic Instability, time-dependent excessive deformation, time-dependent plastic instability, time-dependent fracture, and elastic or elastoplastic instability. The S type damages include progressive deformation and fatigue or progressive cracking. Most of the design rules contained in RCC-MR are very similar to those in the ASME Code. The classification of stresses into primary and secondary, and into membrane, bending, and peak is identical to the ASME Code. To handle multiaxial stresses, RCC-MR allows the use of either the maximum shear theory (Guest-Tresca) or octahedral shear theory (von Mises) to compute stress intensities or stress range intensities.

Tables 16–17 [17] provides a comparison among the codes (RCC-MR and ASME III Division 1) about the different non-linear analysis. The ASME section VIII Division 2, which is the reference code for the non-PIC component, is also considered. According to Tables 16–17, both Codes RCC-MR and ASME III Division 1 have to be significantly improved to assure clear and precise use of different non-linear analysis.

TABLE 16. OVERVIEW OF THE NON-LINEAR ANALYSIS METHODOLOGIES COVERED IN THE COMPARED CODES FOR MONOTONIC LOADING

	Plastic collapse				Plastic instability				Stress triaxiality	
	Limit analysis		Direct elastic-plastic FEA		Limit analysis		Direct elastic-plastic FEA		Direct elastic-plastic FEA	
	Material properties	Criteria	Material properties	Criteria	Material properties	Criteria	Material properties	Criteria	Material properties	Criteria
RCCM	Y	Y	N	N	N	N	N	N	N	N
ASME III	Y	Y	N	N	Y	Y	N	N	N	N
JSME	Y	Y	N	N	N	N	N	N	N	N
RCC-MRx	Y	Y	Y	Y	Y	Y	Y	P	N	N
KEPIC	Y	Y	N	N	Y	Y	N	N	N	N
PNAEG	N	N	N	N	N	N	N	N	N	N
KTA	N	N	N	N	N	N	N	N	N	N
R5	Y	Y	N	N	N	N	N	N	N	N
ASME VIII	Y	Y	Y	Y	Y	Y	P	Y	P	Y
EN 13445	Y	Y	N	N	N	N	N	N	N	N

Y = covered; N = Not covered; P = Partially covered

TABLE 17. OVERVIEW OF THE NON-LINEAR ANALYSIS METHODOLOGIES COVERED IN THE COMPARED CODES FOR CYCLIC LOADING

	Plastic shakedown				Fatigue Ke		
	Direct elastic -plastic analysis using FEA				Direct elastic-plastic analysis using FEA		
	Material properties	Material constitutive equation	Criteria	Extrapolation rules	Material properties	Material constitutive equation	Method
RCCM	N	N	N	N	N	N	N
ASME III	N	N	N	N	N	N	N
JSME	Y	P	Y	N	Y	N	Y
RCC-MRx	P	P	N	Y	Y	P	N
KEPIC	N	N	N	N	N	N	N
PNAEG	N	N	N	N	Y	Y	Y
KTA	N	N	N	N	N	N	N
R5	N	N	Y	N	N	N	N
ASME VIII	Y	N	Y	N	Y	N	Y
EN 13445	N	N	N	N	N	N	N

Y = covered; N = Not covered; P = Partially covered

6.4.1. Buckling

Overall buckling of a structural member under compression depends primarily on its slenderness ratio, which is the ratio of effective length to radius of gyration. Effective length depends on whether the end conditions of the member are free or restrained. The radius of gyration is a property of the cross-section, and is the square root of the ratio moment of inertia to cross-sectional area. There is an interaction between buckling and plastic collapse at low slenderness ratios and stresses approaching yield. Local buckling may occur in the wall of shell structures or vessels under compression, or in flat parts of plate.

Typically, the allowable compressive loads/stresses are limited to 2/3rd of the critical buckling. Analytical solutions or computational analysis can be relied on to estimate the critical buckling mode.

6.4.2. Fatigue assessment

Article NB-3216 of ASME III provides a step-by-step procedure for the evaluation of fatigue at any point within a pressure vessel. The alternating stress intensity (Sa), one-half the total principal stress difference (stress intensity) range, is used with an appropriate design fatigue curve to obtain the allowable (N) number of cycles. An adjustment for the effect of elastic modulus is needed by multiplying Sa by the ratio of the modulus of elasticity given in the design fatigue curve to that used in the analysis. From the specified number of design cycles or transient pairs, for which the Sa occurs, the number of actual cycles (n) is obtained. Fatigue damage is then calculated as n/N. The acceptance criterion is that n/N is less than unity.

In practice there is likely to be more than one Sa value. The cumulative effect is evaluated by means of Miner's rule. The partial usage factors (n_1/N_1) are summed to obtain a total fatigue cumulative usage factor (CUF) value. The acceptance criterion is a CUF of less than unity. In determining the number of occurrences of a Sa value, a simple peak-to-peak event pairing counting method algorithm is proposed within article NB-3222.4 of the Code.

The same basic approach to the fatigue assessment can be found in both codes, but RCC-MR provides more precise indications than ASME III and new detailed conditions for the use of fatigue curves. Design fatigue curves for a material will be constructed according to the criteria of the ASME (section III, Division 1 – Appendices, III-2200) and the RCC-MR (A3.GEN.23) codes in a similar way. They are obtained from fatigue lifetime (number of cycles to rupture) data of uniaxial strain-controlled fatigue (low cycle fatigue, LCF) tests performed with a strain rate in the order of 10^{-3} sec^{-1}. A best fit to experimental data is obtained by applying the method of least squares to the logarithms of the strain range values. The design fatigue curves are then deduced from the best fit curve by applying a factor of 2 on strain range or a factor of 20 on cycles, whichever is the more conservative at each point. These factors are intended to cover effects such as those of the environment, the scale (between the material and the test specimen), surface finish and data scatter. They in no case constitute a safety margin.

6.4.2.1 Thermal mechanical fatigue

Thermal Mechanical Fatigue (TMF) is caused by combined thermal and mechanical loading where both the stresses and temperatures vary with time. This type of loading can be more damaging compared with isothermal fatigue at constant operating temperature. Material properties, mechanical strain range, strain rate, temperature, and the phasing between temperature and mechanical strain all play a role in the type of damage formed in the material. These types of loadings usually occur during start-up and shut-down cycles. If significant plastic strains occur then the design life may be limited to just a few hundred cycles. One of the main causes of the damage is the prevention of thermal expansion or contraction due to the constraint imposed by the surrounding material. In this case thermal strain is converted into mechanical strain which causes fatigue damage in the structure. Total constraint exists when all the thermal strain is converted into mechanical strain. Over constrain can occur in a stress concentration where the mechanical strain is greater than the thermal strain. One measure of the degree of constraint is the ratio of the thermal and mechanical strain rates. TMF loading is often described to be in-phase (IP) or out-of-phase (OP). In IP loading, the maximum temperature and strain occur at the same time. In OP loading, the material experiences compression at highest temperature and tension at lower temperatures. OP loading is more likely to cause oxidation damage because an oxide film can form in compression at the higher temperature and then rupture during the subsequent low temperature tensile portion of the loading cycle where the oxide film is more brittle.

Cyclic thermal loading in presence of a constant mechanical loading can also cause ‘ratchetting’. This event occurs under cyclic stress. Even when cycling between prescribed stress limits, the hysteresis loop becomes unsymmetrical and does not end at the same point. This cause progressive ‘creep’ or ‘ratchetting’ which is incremental cyclic growth of component or strain. Failure is due to instability caused by dimensional growth causing thinning of the section. This damage can be predicted and prevented by designers by following one of the four approaches:

(a) Code rules limiting the sum of the primary stress and secondary stress range to be less than the 3Sm limit which is usually twice the yield stress of the material.
(b) Follow the limits set by the Bree diagram which give the safe combination of primary and secondary stresses for which ratchetting will not occur.
(c) Follow the limits set by the ‘Efficiency Diagrams’ based on material tests given in RCC-MRX code.
(d) Perform finite element analysis using a nonlinear kinematic hardening material whose material properties have been obtained from stabilised cyclic stress tests covering different strain ranges.

In Fig. 10 on the following page, the FMEA and damage limits diagram for design substantiation of mechanical components is shown.

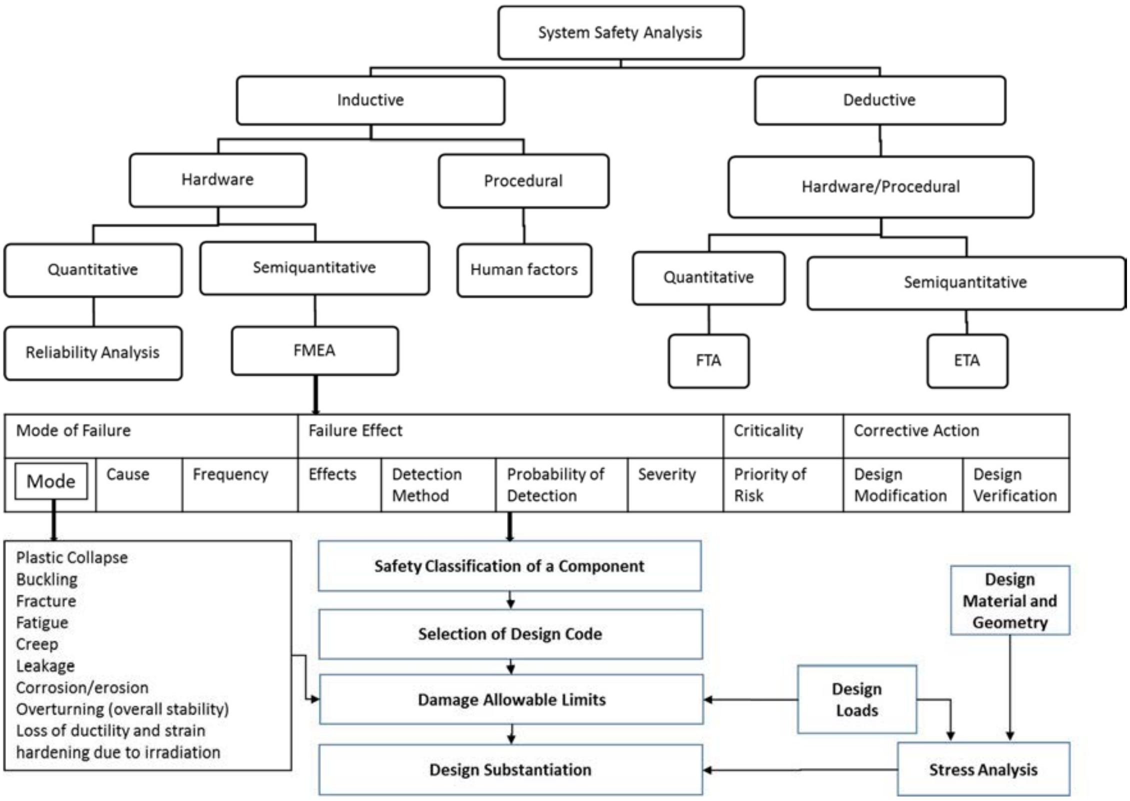

FIG. 10. FMEA and damage limits diagram for design substantiation of mechanical components

6.4.3. Fracture assessment

To ensure the necessary safety margin against an unstable fracture, Reactor Pressure Vessels (RPV) are operated with the use of a rigorous fracture mechanics analysis according to Appendix G of the ASME Code sections III and XI. The analysis is based on Charpy impact data, which were collected through surveillance tests. ASME Code section III, NB-2300 describes a procedure to determine the reference temperatures of the fracture toughness curve from conventional notch impact test data. The US federal regulations 10 CFR 50.61 and the ASME III Appendix G (Fracture Toughness Criteria for Protection Against Failure) prescribe the screening values of the Charpy V-notch impact data for an operation of commercial pressurized water reactors. These criteria have been considered too conservative.

More recently, a new fracture mechanics-based transition parameter has been investigated by using Charpy-sized pre-cracked, PCVN specimens. The 'master curve' method is considered as a more promising and logical tool for the evaluation of an irradiation embrittlement of RPV steels. The test method was standardized by ASTM designation number E1921. ASME Code has adopted its application to commercial reactors through Code Case Numbers N-629 and N-631.

The method of ASME III Appendix G is retained by the RCC-M Appendix ZG. The RCC-MRx also recommends an alternative method of analysis in Appendix A16. This alternative method, used extensively in France, is more refined and takes into consideration the fatigue crack growth of the reference flaw which is defined from considerations on manufacturing and inspection.

To assess structures containing flaws, there are also the R6 procedure low-temperature fracture assessment procedure which uses a Failure Assessment Diagram approach and the R5 procedure for high temperature applications. Both treat the effects of secondary stresses on structural integrity. Recently, there have been many developments, which identify the way forward for these procedures. A modified J-integral definition has been derived, which is path independent for cases of proportional and non-proportional loading and is ideal for evaluating the crack driving force for defects in secondary and residual stress fields.

7. GUIDELINES FOR ENGINEERING RULES

After having fixed general engineering rules to each class of safety SSC, the last step is to correctly specify performance and reliability requirement for each safety SSC. For that, it needs to be defined for all the events or event combinations that a given safety SSC needs to withstand, classified by category, and what are, for each event or event combination, the safety requirement. All this information is the input for the designers, who needs to identify the more adequate code to meet these safety requirements or to set up specific studies for that.

Additionally, the capability of the safety SSC to exclude accident needs to be considered. For example, the access control system needs to be robust enough to avoid exposure of operator during a transfer cask. The support of the tokamak needs to be robust to exclude its failure under severe impact on the main basement of the tokamak building, which is part of the second barrier.

Safety classification has an impact on other classifications like quality classification and seismic qualification.

7.1. Link Safety Important Components grading to code class

Once a SSC has been identified as a SIC, it is necessary to design it in order that it fulfils the missions which are allotted to it so that the plant's safety objectives are achieved. In practice, a component is likely to be involved in multiple normal or accidental events during the plant's operation. For each of these events, a component copes with operating conditions and needs to have a given behaviour (Leak tight, leakage rate, operable). It is then about what is called a 'safety requirement' for this component. By way of an example, a tank needs to remain tight with an internal pressure of X bars and a temperature of Y°C in the event of accidental steam inside this tank. Each component then needs to be designed to meet several safety requirements.

A safety requirement can be satisfied by designing a component so that it resists the event met without suffering damage or by admitting the component can undergo a certain level of damage while guaranteeing the safety requirement.

This is translated in calculations of component's resistance to the event met (thus in the design studies) by the introduction of a safety margin (or in other words of a safety margin). This safety margin is important if it is wished the component would meet the safety requirement without suffering damage. This safety margin is lower if one admits a certain level of damage for the component, damage which will not prevent from meeting the safety requirement.

It is thus necessary for each safety requirement to define the acceptable damage limit, which will be translated in design studies by considering a given safety margin or safety margin.

To have a coherent comprehensive approach on a plant, it is first advisable to define a classification of damage limits which will be used to design components. As example, Table 18 presents a classification usually used.

TABLE 18. DAMAGE LIMIT CLASSIFICATION

Damage limit	General objective
A	No damage. Component remains functional.
B	Negligible damage. Component remains functional but can require an anticipated maintenance or a minor adjustment.
C	Tolerated local deformations but being able to require an inspection, a repair or a replacement.
D	Tolerated general deformations for which a repair is not economically realizable.

It is thus advisable then to define for each safety requirement of component the acceptable damage limit. To choose the acceptable damage limit, two main criteria are to be considered: event frequency and safety margin.

7.1.1. Event frequency

One generally asserts low damage limits for safety requirement which the component will have to often satisfy during operation, therefore for the most frequent events, and the more important damage limits for safety requirement that the component need to meet only occasionally, therefore for the events of low probability. In practice, damage limits are associated with each category of events classification (see section dedicated to events classification).

This general rule suffers however from exceptions such as for example: in certain cases, incompatibilities appear between the acceptable damage limit needed by the classification and the safety requirement. For example, one cannot impose important deformations for component with ceramic to metal brazed or on a diamond window when it is expected that the component needs to maintain leak tightness regarding tritium.

SSC important to safety needs to be designed considering appropriate combinations of loadings associated with normal operations, incidents and accidents events. The operational load conditions can be classified into various categories based on the probability of occurrence. A general example for fusion machines is described below:

— Load Category 1 (LC1) − operational loads: load conditions associated with normal operations, including start-up, operations with plasma-on and plasma-off cycles, stand-by and shutdown.

— Load Category 2 (LC2) – moderate frequency loads: load conditions associated with incident and events with moderate probability of occurrence. For a fusion device such conditions might include electromagnetic loads induced by plasma disruption events and fast magnet discharge, as well as pressurisation of the vacuum chamber due to internal coolant leaks.
— Load Category 3 (LC3) – very low frequency loads: load conditions associated with accidents and events with very low probability of occurrence. For a fusion device such conditions might include large plasma disruptions, vacuum vessel pressurisations due to large in-vessel leaks.
— Load Category 4 (LC4) – extremely low frequency loads: load conditions associated with extremely low probability postulated events.

Additional categories have to be introduced to cover the following conditions:

— Design conditions: this category has to be derived from the most severe conditions (design pressure, design temperature and any other design driver load) that Semiparametric Skew-Symmetric Shape Model (SSSM) needs to withstand ensuring their structural integrity, as well as the needed operability for active components. They are not intended as a combination of the highest single loadings unless such conditions occur at the same time.
— Test and maintenance conditions: this category includes load conditions associated with pressure testing, vacuum leak testing, baking, venting, draining, drying and other tests or maintenance operations. In some cases, these conditions are very demanding and may be considered as design drivers for SSSM.

7.1.2. Safety margin

Since a damage limit was retained to meet a safety requirement for an events category, it is necessary, for the design studies, to define a value for the safety margin. But this value is not a fixed value but a value which can vary in a range of values. Indeed, if, for example, it is wished an absence of damage for component in each event (damage limit A), this can be obtained by taking a safety margin which has a minimum but which, in theory, does not have maximum limit.

The safety classification of the component will assist in selecting of the value of the safety margin to be retained. Thus, for the same damage limit, plus the component is important for safety, plus the safety margin will be important. This is illustrated in Table 19.

TABLE 19. SAFETY MARGIN CLASSIFICATION

Event category	**Event category I**	**Event category II**	**Event category III**	**Event category IV**
Damage limit	A	B	C	D
Safety Margin for SIC 1	SM (A; 1)	SM (B;1)	SM (C; 1)	SM (D; 1)
Safety Margin for SIC 2	SM (A; 2)	SM (B; 2)	SM (C; 2)	SM (D; 2)
Safety Margin for SIC 3	SM (A; 3)	SM (B; 3)	SM (C; 3)	SM (D; 3)

This general rule also suffers however from exceptions such as for example:

- It is advisable to pay attention to the component which plays a main function in an accidental event or which intervenes in a significant number of accidental events. For this component, one will be able to retain a more important safety margin.
- It is necessary to aim at important safety margin to design component which has a safety requirement which allows the exclusion of an accident whose consequences are unacceptable. As example, ITER tokamak support was designed with a large margin to avoid the collapse of the machine on the main basement.

Thus, design studies of a component are made by associating with the safety requirements which arise from the safety analysis. On one hand acceptable damage limits are important and on the other hand the safety margins have to be tailored considering the frequency of the event and the safety function. That is knowing the limits and knowing how close one can get to it. Table 20 on the following page illustrates the whole of the approach.

TABLE 20. SAFETY REQUIREMENT CLASSIFICATION EXAMPLE

	Category	Event	Safety requirement	Design criteria (structural, operational, safety)	Safety Margin	Code, Standard
SSC safety SSC « X » (safety class 1)	Cat. I Normal operation	Water ingress level I	Leak tight at pressure X	A	SM (A; 1)	Code X, Criteria X1
		Air ingress level I	Integrity at pressure Y	A	SM (A; 1)	Code Y, Criteria Y1
	Cat. II $> 10^{-2}$	Water ingress level II	Leakage rate 1 at pressure Z	B	SM (B; 1)	Code X, Criteria X2
		Air ingress level II	Operability of isolation valves at pressure Y	A	SM (A; 1)	Standard Z, Criteria Z1
		Water ingress level I and air ingress level I	Leakage rate 1 at pressure T	B	SM (B; 1)	Specific study
	Cat. III $10^{-2} >< 10^{-4}$	Water ingress level III	Leakage rate 2 at pressure U	C	SM (C; 1)	-
		Air ingress level III	Operability of isolation valves at pressure Y	C	SM (C; 1)	-
		Water ingress level II and air ingress level II	Leakage rate 2 at pressure V	C	SM (C; 1)	-
		Dust explosion level I	Integrity	C	SM (C; 1)	-
		Earthquake Level I	Integrity	C	SM (C; 1)	-
	Cat. IV $10^{-4} >< 10^{-6}$	Water ingress level IV	Leakage rate 2 at pressure U	D	SM (D; 1)	-
		Air ingress level IV	Operability of isolation valves at pressure Y	D	SM (D; 1)	-
		Water ingress level III and air ingress level III	Leakage rate 2 at pressure V	D	SM (D; 1)	-
		Dust explosion level II	Leakage rate 1 at pressure W	D	SM (D; 1)	-
		Earthquake Level II	Integrity	D	SM (D; 1)	-
	Excluded accident	Support failure	Integrity	A	SM (A; 1)	-

7.2. Plant conditions and acceptable damages

The description of the various plant conditions needs to include the definition of:

- Necessary plant safety functions during and after the event;
- Necessary inspection, maintenance and repair operations (impact on doses to personnel);
- Extent of plant contamination;
- Applicable limits for release to environment (higher load categories).

The acceptable damage limits of SIC has to be established for each plant condition, considering the following aspects:

(a) Component safety functions and grading: nuclear confinement, safe isolation, protection of safety important components.
(b) Component type (active/passive): the damage limit for active components (e.g. pumps/valves) during or after incidents and accidents may be more stringent to ensure the necessary operability. This is important to prevent loss of dimension stability that could affect the component functions when higher stress limits are allowed (e.g. LC2, LC3).
(c) Material characteristic: the specific behaviour of materials needs to be considered (e.g. ductile vs brittle).
(d) Consequence of failure: the failure of any system or component needs to be evaluated for its effects on all other systems, components and ultimately on the entire plant.

7.3. Loads, assessment criteria and limits

The loads, assessment criteria and limits are reported in Table 21. The assessment criteria for SIC need to be defined to ensure the following:

- Structural integrity of ‘standard’ and ‘special’ components;
- Operability of active mechanical components;
- Protection of SIC.

TABLE 21. LOADS, ASSESSMENT CRITERIA AND LIMITS

Plant Condition	Load category (frequency of occurrence)	Loads	Design Criteria	Design Limits
Normal	LC1 (operational)	Service Loadings A • Normal operational (individual) loads and load combinations; • Ageing: component to be considered at end-of-life.	Design criteria A include: • All relevant structural criteria, including fatigue; • Criteria to verify operability of active components; • Criteria to verify deformation; • Criteria for verifications of special components; • Criteria for verification of Leak tightness.	Design limits A
Upset	LC2 (moderate frequency)	Service Loadings B • Individual loads and load combinations with moderate frequency of occurrence; • Ageing: component to be considered at end-of-life.	Design criteria B • The criteria may be equivalent to the previous case.	Design Limits B
Emergency	LC3 (very low frequency)	Service Loadings C • Individual loads and load combinations with very low frequency of occurrence; • Ageing: component to be considered at end-of-life.	Design criteria C • Structural criteria might not include ratcheting and fatigue verifications.	Design Limits C
Faulted	LC4 (extremely low frequency)	Service Loadings D • Individual loads and load combinations with extremely low frequency of occurrence; • Ageing: component to be considered at end-of-life.	Design criteria D • Structural criteria might not include ratcheting and fatigue. • Excessive deformation is allowed.	Design Limits D
-	Design conditions	Design loadings	Defined based on worst case loading	Design Limits
-	Test and maintenance conditions	Test and maintenance Loadings	Defined based on Test & Maintenance conditions	Test and maintenance Limits

Design criteria A, B, C and D are not just limited to structural integrity but may also include safety requirement such as leakage rate and operability.

8. STRUCTURAL INTEGRITY ASSESSMENT

The diagram below in Fig. 11 shows the links between the safety analysis of SSSM and the structural integrity assessment. This approach needs to be complemented with the assessment of the operability of active mechanical components that are involved in the safety functions of the machine.

The main links are introduced hereafter:

— The selection of appropriate design rules and material properties needs to be consistent with the assessment of ageing mechanisms.
— The SIC grading has to be considered as a basis for the selection of the most appropriate code classes
— The applicable design criteria have to be established starting from the analysis of the necessary plant conditions (overall damage limits).

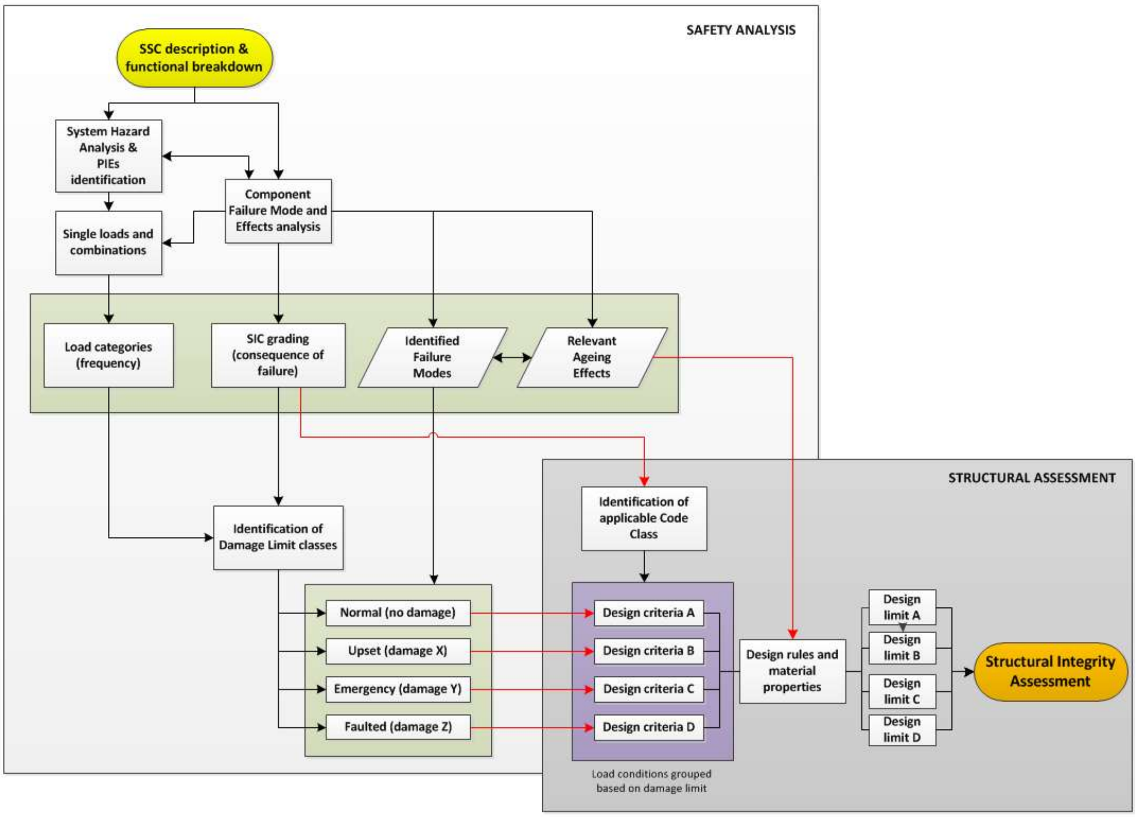

FIG. 11. Fusion devices: from safety analysis to structural assessment.

8.1. Recommendations[3]

(a) Existing codes and standards may introduce an excessive separation between structural integrity assessment and verification of operability of active mechanical components. A more integrated approach needs to be followed.
(b) Existing codes and standards introduce the concept of design conditions, as ‘design pressure’ and ‘design temperature’. For fusion components this category may be extended covering additional design driver loadings.

(c) The load category for test conditions needs to be extended to maintenance conditions (e.g. hot baking, draining and drying, venting). Such operations are often design drivers for fusion components.
(d) Codes and standards for fusion applications need to include rules and best practices for the design, analysis and qualification of 'special components' such as:
   (i) Shell elements and box structures;
   (ii) Bellows;
   (iii) Ultra-High Vacuum (UHV) flanges;
   (iv) Ceramic assemblies (windows, electrical penetrations).
(e) Codes and standards for fusion applications need to include rules and best practices for the evaluation of relevant ageing and environmental effects such as:
   (i) Neutron irradiation;
   (ii) Corrosion and erosion;
   (iii) Thermal effects.
(f) The full life-cycle needs to be considered to characterize the structural performance of SSSM and determine their potential failure modes.

9. QUALITY CLASSIFICATION

Quality classification of an SSC depends on the following:

(a) SIC assigned to the item: once the safety classes of the SSSM have been established, corresponding general engineering design rules need to be specified and applied. Quality is one of these rules. As an example, Table 22 on the next page presents the general engineering design rules used for ITER.
(b) Anticipated impact of item failure or malfunction on machine availability. Additional useful considerations in quality classification apart from the functional and confinement barriers are related with operational matters such as machine availability rather safety. For example:
   (i) Ease of replacement/repair;
   (ii) Ease of fault/malfunction detection;
   (iii) Ease of identification of defective part;
   (iv) Availability of spare part;
   (v) Availability of qualified personnel.
(c) Maturity and complexity related to a risk of failure or malfunction. Factors to be considered when assessing the risk of failure or malfunction would include:
   (i) Degree of design innovation;
   (ii) Complexity or uniqueness of the item;
   (iii) Design, performance and manufacturing margins;
   (iv) Involvement of innovative processes;
   (v) Need for special controls and surveillance over processes and equipment;
   (vi) Involvement of processes which cannot be fully verified by inspection or test;
   (vii) Degree to which functional compliance can be demonstrated by inspection or test; quality history and degree of standardization of the item.

TABLE 22. ITER DESIGN RULES

SSC classification	Redundancy	Quality	Environmental qualification
SIC 1	Yes, for active component	Q1	Yes
SIC 2	Case by case	Q2	Yes
SR	No	Q3	Yes

10. QUALIFICATION AND VERIFICATION OF SAFETY IMPORTANT COMPONENT

The main purpose of the qualification is to demonstrate that the design and construction of a SIC complies with its safety requirements through its lifetime. An example of qualification process is reported in Fig. 12. The safety requirements are very often related to functions of confinement of radioactive material or limitation of exposure to ionizing radiation. For confinement of radioactive material, classical safety requirements are:

— Structural integrity;
— Leak tightness (e.g. UHV flanges);
— Safe isolation (e.g. UHV valves).

For nuclear shielding, classical safety requirements are:

— Nuclear damage;
— Limitation of exposure to ionizing radiation.

Verification is demonstration of compliance with design codes and standards. In case of absence of a suitable code or standard it is acceptability of design based on known test data or experience.

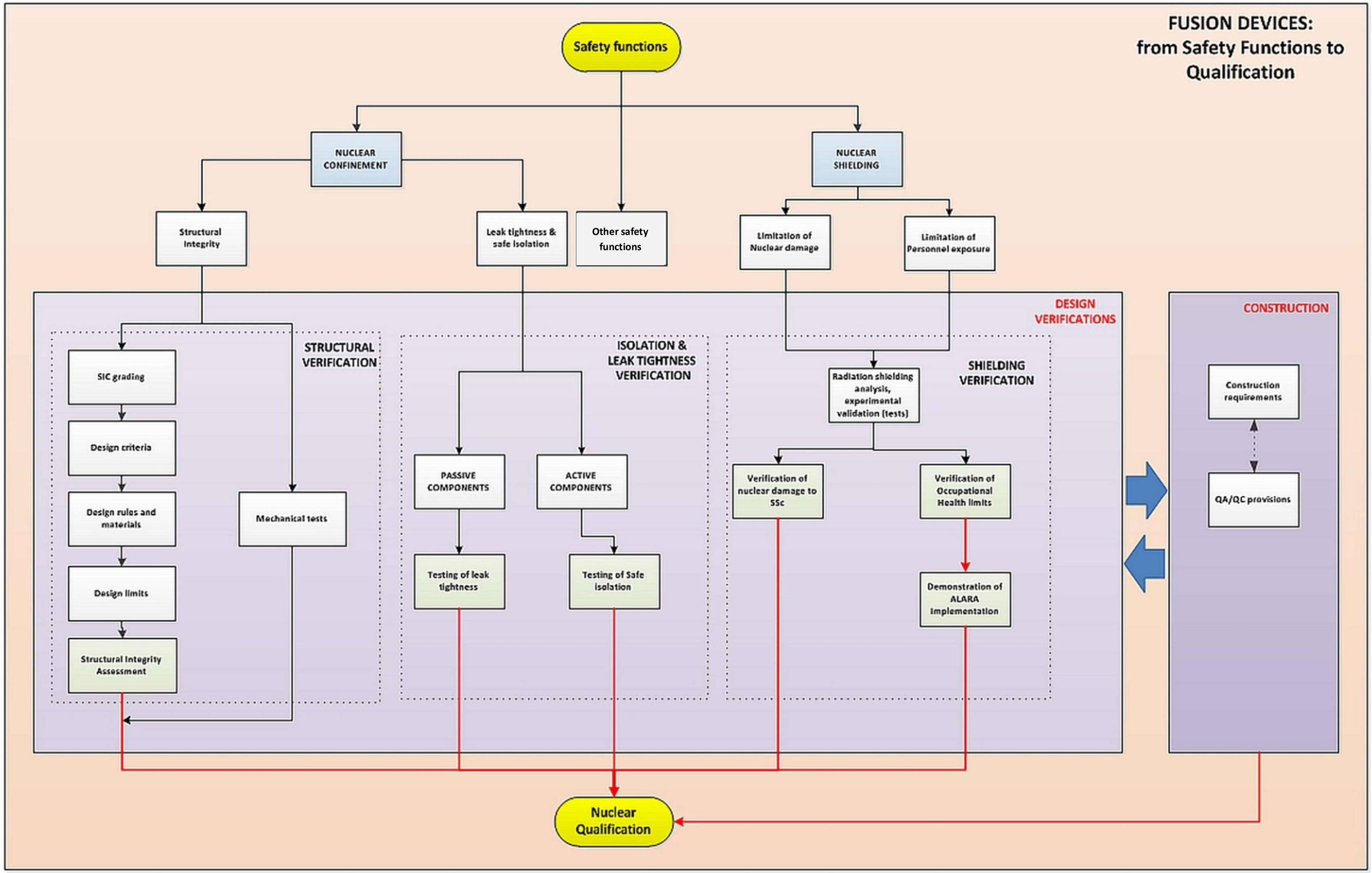

FIG. 12. Example of qualification process.

Existing nuclear codes and standards provide design and verification criteria related to the structural integrity assessment but these may not be available for some SSC.

In case of active components like valve and pumps which have vacuum isolation as safety requirement. ASME QME provides rules for qualification, but there are no design guidelines.

Another example is the leak tightness of nuclear UHV components (e.g. metallic flanges, feedthroughs). At present there are no industrial codes and standards providing rules and guidelines for nuclear damage. General guidelines need therefore to be produced defining the typical parameters involved in nuclear damage analysis:

- Particle flux density: neutrons and gammas penetrating surface area per time unit and energy, $cm^{-2} \cdot s^{-1}$.
- Particle fluence: integrated neutron/gamma flux, cm^{-2}.
- Displacement damage rate: displaced lattice atoms due to collisional damage (dpa/s), relevant for damage effects (dynamical effects due to atomic displacements), relevant for material selection.
- Integrated displacement damage: accumulated atomic displacements (dpa).
- Gas production: hydrogen/helium emission from nuclear interactions, (appm) relevant for swelling and readability.
- Solid transmutation – nuclear transmutations due to non-elastic interactions (appm), irradiation parameter relevant for damage effects (impact on microstructure).
- Absorbed dose rate: kinetic energy release rate in material due to ionizing radiation (Gy/s) relevant for damage effects (dynamical effects due to ionizing energy).
- Absorbed dose: integrated kinetic energy release in material due to ionizing radiation (Gy) relevant for damage effects (accumulated effect of ionizing energy).

Additionally, guidelines need to be produced describing the typical steps for the nuclear analysis to verify the compliance with the limits of exposure to personnel.

In such cases, a design can be substantiated through analysis, test or experience. A set of guidelines needs therefore to be developed to guide the process of design, starting from standard industrial practices and established analysis methodologies as suggested in Table 23. It is also possible to use experience based data bases, if available, to substantiate a design. For example, use of the Seismic Qualification Utilities Group (SQUG) database for seismic qualification.

TABLE 23. DESIGN VERIFICATION AND DESIGN CRITERIA TO ACHIEVE SAFETY FUNCTION

Safety function	**Safety Requirements**	**Verification method**	**Design Criteria**
Confinement of radioactive material	Structural integrity	Analysis / tests / experience based databases	Codes and standards for design and qualification
	Safe isolation	Tests (analysis)	Codes and standards for qualification (e.g. ASME QME). Guidelines needed for design
	Leak tightness	Tests (analysis)	Guidelines needed for design (e.g. vacuum flanges)
Limitation of exposure to ionizing radiation	Nuclear damage	Analysis, tests	Guidelines needed
	Personnel exposure	Analysis	Guidelines needed

But there are also special fusion components which might require specific qualification process. For example, in the case of the scrubber column of the detritiation system, for which the safety requirement is its efficiency to detritiate.

11. CONCLUSIONS AND REMARKS

Various issues related with safety classification of mechanical components for fusion applications have been identified and discussed. Whilst the basic principles described in SSG-30 remain applicable there are some important differences between the fission and fusion applications. These differences have been highlighted and relevant information has been presented to help complete safety classification of mechanical components for fusion applications.

The TECDOC has also integrated safety with design and presented the whole process from system safety analysis to design substantiation covering intermediate steps involving FMEA to identify failure modes, safety classification, selection of design codes and allowable limits.

In preparation of this TECDOC, it was realised that there are still several areas where further work needs to be done for the fusion components. Some of the important areas requiring future work where there is still lack of information and guidance are as follows:

(a) Lack of processes and criteria for:
   - (i) Classification of shielding function;
   - (ii) Definition of design pressure for vessel.
   - (iii) Lack of data for:
     - — Material properties for structural materials under fusion irradiation conditions (14 MeV neutrons);

— Material properties for ceramic to metal joints in irradiated environment (with right spectrum);
— Lack of reliability data for components;
— Uncertainty related with disruption loads and plasma stability.

(c) Future area of developments:
   (i) Classification and analysis of tritium breeder components.

(d) Outside scope:
   (i) Consideration about RH to reduce doses to personnel;
   (ii) Consideration about control and instrumentation under irradiation.

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Annex I

ENVIRONMENTAL EFFECTS

I–1. NEUTRON IRRADITION

The high energy neutrons produced by the plasma reaction can have many effects on materials. They can modify the crystal structure by various mechanisms, such as transmutations, ion implants, atomic displacements, and lattice defects. They can enhance the precipitation of impurities and phase modifications, often inducing detrimental effects on material properties. Although a large fraction of the produced neutrons would be absorbed by the in-vessel neutron shields (e.g. front blankets and neutron shields in the vessel ports) the components forming part of the first confinement system could still be subject to non-negligible neutron fluxes, which could degrade their structural performance. The detailed ageing mechanisms induced by neutron irradiation are very complex, but, from a macroscopic point of view, three phenomena are important:

(a) Creep: irradiation creep strains affect the distribution of stresses, which in turn can influence other damage mechanisms such as fatigue and excessive deformation of the structure. Irradiation-induced creep may have a beneficial effect of relaxing residual or strain-controlled stresses. On the other hand, this relaxation may result in a stress reversal when thermal loadings are removed.
(b) Swelling: irradiation can induce gross structural distortions by swelling in isotropic materials or by growth in anisotropic materials. This phenomenon is irreversible and may lead to high stresses when the swelling is constrained or spatially varying. Such stresses are strain controlled, and the beneficial relaxation effects of irradiation-induced creep have to be considered in the calculation of the constrained swelling stresses.
(c) Time dependent material properties: large changes in materials properties due to irradiation can be induced by atomic displacements, nuclear transmutation, and gas formation (He). Helium is insoluble in steel and therefore helium will be present as defects starting with small He-bubbles that can grow into pores and eventually promote formation of cracks. At elevated temperatures (such as during welding) helium will be re-distributed and may migrate to grain boundaries forming larger helium filled pores that can result in degradation (cracks) at the welded joints. Irradiation effects in copper alloys and austenitic stainless steels include significant irradiation-induced hardening, loss of ductility, loss of strain-hardening capability, and reduction in fracture toughness at a relatively low neutron damages. For high performance materials such as precipitation hardened materials the effect at low doses is the opposite with decomposition of precipitates and loss of mechanical strength at low doses, but with higher doses the irradiation hardening effect will result in regained strength due to irradiation hardening. Thus, the influence of irradiation on materials properties is a dynamic process strongly depending on alloying compositions and present phases in the microstructure.

Swelling, creep, and time dependent properties cause time-dependent stresses, but these in themselves do not require fundamental changes in the design rules when compared with existing codes. With reduced ductility, however, secondary and peak stresses become more important. Existing codes rely on sufficient ductility to simplify the analysis, ignoring secondary and peak stresses apart from their effect on strain ratcheting and fatigue. Limits on secondary and peak stresses need therefore to be considered to account for both the stress and strain limits of the material.

Fusion plants require a complex system of integrated neutron shields to achieve two main functions:

— Limitation of nuclear damage to structures and components;
— Minimization of personnel exposure to ionizing radiations.

I–1.1. Limitation of damage to structures and components

Neutron shields are designed to absorb high energy neutrons from the plasma to reduce the damage to structures and components. Neutrons from the fusion plasma can modify a material's crystal structure by various mechanisms, such as transmutations, ion implants, atomic displacements, and lattice defects. It enhances the precipitation of impurities and phase modifications, often inducing detrimental effects on material properties.

Radiation induced property changes are highly material dependant (e.g. type, purity, treatment, micro-structure) and are sensitive to the irradiation environment, ambient temperature, dose rate and integrated dose.

Mechanical property changes include effects like irradiation-induced hardening, loss of ductility, loss of strain-hardening capability, and reduction in fracture toughness, which can occur at relatively low neutron damage (typically 0.1–0.5 displacements per atoms (dpa)). Changes in physical properties are due to radiation-induced electrical degradation, radiation enhanced diffusion, radiation induced electromotive force, increase of dielectric loss, radiation-induced absorption and others. Some of those effects might be significant at very low neutron dose ($<10^{-2}$ dpa).

Large changes in materials properties due to irradiation can be induced by atomic displacements, nuclear transmutation, and gas formation (H, He). Helium is insoluble in steel and therefore helium will be present as defects starting with small He-bubbles that can grow into pores and eventually promote formation of cracks. At elevated temperatures, such as those seen during welding, helium will be re-distributed and may migrate to grain boundaries forming larger helium filled pores that can result in degradation in form of cracks at the welded joints.

Irradiation effects in copper alloys and austenitic stainless steels include significant irradiation induced hardening, loss of ductility, loss of strain hardening capability, and reduction in fracture toughness at a relatively low neutron damages (for steel at doses higher than 0.5 dpa, for copper alloy higher than 0.1 dpa).

For high performance materials such as precipitation hardened materials the effect at low doses below 0.2 dpa is the opposite with decomposition of precipitates and loss of mechanical strength at low doses, but higher doses will result in regained strength due to irradiation hardening.

Thus, the influence of irradiation on materials properties is a dynamic process strongly depending on alloying compositions and present phases in the microstructure. Gamma irradiation is due to prompt photons from nuclear interactions of neutrons but also from activated materials, from activated cooling water.

Due to the limited energy the primary damage to materials is due to (indirect) ionisation, particularly leading to excitations of electrons. This gives rise to dynamical effects, such as radiation-induced conductivity and radio-luminescence, and permanent effects like optical transmission and reflectivity losses.

I–2. PLASMA EROSION

Plasma facing components are subjected to heat and particle fluxes released by the plasma either continuously or in bursts, which can cause surface and structural damage due to the intense power deposition in the materials. These loads can occur during normal operations or because of plasma instabilities such as disruptions, vertical displacements events and edge localized modes. The thermal energy dumped on the components can induce large temperature gradients, thermal stresses, melting and evaporation of the surfaces, resulting in surface erosion and component damage.

I–3. CORROSION AND EROSION

Primary cooling water systems in fusion devices may be needed to cool client systems, such as first wall, blanket modules, vacuum vessel, and other assemblies. Additional operations may include baking of in-vessel components, chemical control of water provided to client systems, draining and drying for maintenance, leak detection and leak localization. Although cooling water systems are typically not needed to ensure safe shutdown, corrosion and erosion phenomena still play an important role in mobilizing activated materials in fusion machines. The correct characterization of corrosion and erosion mechanisms is essential to prevent and mitigate the risk of failure of components connected to the primary cooling water system and that form part of the confinement boundary. In addition, the characterization and limitation of the inventory of radioactive material, in the form of activated corrosion products, is essential with regards to occupational radiation exposure during maintenance scenarios and for severe accidents, such as a loss of coolant outside the main reactor vessel. The relevant corrosion and erosion mechanisms are described hereafter:

- Galvanic corrosion: this mechanism occurs when two metals are in mechanical or electrical contact. In a corrosive enviornment one of the metals acts as an anode and undergoes corrosion, while the second metal acts as a cathod and remains unattacked. In fusion devices the primary heat transfer system may be connected to components made of different materials (e.g. stainless steel, copper) hence increasing the potential for galvanic corrosion at connections.
- Crevice corrosion: localized form of corrosion that occurs within narrow clearances or under shielded metal surfaces. It could be relevant for the primary heat transfer system (and the attached components), for example under gaskets, seals or welding roots.
- Pitting: localized form of corrosion by which cavities are produced in the material. The process takes place at points where the protective oxide film might be weakened, for example by slag inclusions, or in case of a damaged surface or imperfections in the passive layer.
- Stress corrosion cracking (SCC): cracking of a metal or alloy by the combined action of (tensile) stress and a corrosive environment. The tensile stress may be induced by external loads or could also be the residual stress from metal working processes such as machining or welding.
- Corrosion fatigue: cracking of a metal or alloy by the combined action of a repeated cyclic stress and a corrosive environment. The same mechanism which applies to SCC also applies to corrosion fatigue, with the added complexity of the cyclic loads.

- Cavitation corrosion: combined mechanical and corrosion attack caused by the collapse and impingement of vapour bubbles in a liquid near a metal surface. It can occur where there is a high velocity fluid flow and where pressure changes are encountered (e.g. across control valves).
- Erosion corrosion: mechanical effect induced by the movement of a corrosive liquid (or also with solid impurities) against the metal surface, without the need for cavitating bubbles. It can occur in various types of equipment exposed to fast moving liquids, such as piping systems, bends, elbows, valves and pumps.

I–4. THERMOMECHANICAL FATIGUE

Thermal mechanical fatigue is caused by combined thermal and mechanical loading where both the stresses and temperatures vary with time. Technical details are given in Section 6.4.4.

Annex II

MATERIALS FOR FUSION

II–1. INTRODUCTION

The structure of the Annex is depicted in Table II–1.

TABLE II–1. STRUCTURE OF THE ANNEX

II–1.1. Introduction to mechanical behaviour of the materials

The successful employment of materials (metals and/or alloys) in engineering applications relies on the ability of the material to meet design and service requirements and to be fabricated to the proper dimensions. The capability of a metal to meet these requirements is determined by the mechanical and physical properties of the metal. Physical properties are those typically measured by methods not requiring the application of an external mechanical force (or load). Typical examples of physical properties are density, magnetic properties (e.g. permeability), thermal conductivity and thermal diffusivity, electrical properties (e.g. resistivity), specific heat, and coefficient of thermal expansion. Mechanical properties are described as the relationship between forces (or stresses) acting on a material and the resistance of the material to deformation (i.e. strains) and fracture. This deformation, however, may or may not be evident in the metal after the applied load is removed.

The mechanical properties of the materials are highly dependent on microstructure (e.g. grain size, phase distribution, second phase content), crystal structure type (i.e. the arrangement of atoms), and elemental composition (e.g. alloying element content, impurity level). A common illustration of the relationship between microstructure and mechanical performance is the often observed increase in yield stress with a decrease in grain size. Relationships like these between metal structure and performance make mechanical property determination important for a wide variety of structural applications in metal working, in failure analysis and prevention, and in materials development for advanced applications.

In addition, to alloy composition, microstructure, and mechanical properties, there are two other important components that determine the properties of the materials; processing and performance. About the relationships of these five components, the microstructure of a material depends on of its alloy composition and how this material has been processed. Furthermore, a material's performance will be a function of its properties. Thus, the interrelationships between the five components are linear, as depicted in the schematic illustration shown in Fig. II–1.

Alloy Composition ⇒ Processing ⇒ Microstructure ⇒ Properties ⇒ Performance

FIG. II–1. The linear interrelationships between the five components of the materials that determine their properties.

II–1.2. Nature of metals and alloys

A variety of metal properties are unique among materials and of importance technologically. These properties are conferred by metallic bonding, in which the 'extra' outer valence electrons are 'shared' among all metal ion cores. This bonding is different from other types of solids in that the electrons are free to acquire energy, and the metallic ions are relatively mobile and quite interchangeable regarding their positions in the crystal lattice, the three-dimensional repeating arrangement of atoms in a solid.

Metals are almost always crystalline solids with a regular repeating pattern of ions. Many atomic level defects occur in this periodic array. Many atomic sites are 'vacancies' (point defects) not occupied by atoms. The number and mobility of vacant sites increase rapidly with temperature. The number and mobility of vacancies in metals are quite high compared with other materials because there is no charge balance or local electron bond considerations. This means that solid metal can undergo significant changes with only moderate thermal excitation as vacancy motion (diffusion) provides atom by atom reconstruction of the material. Vacancies allow solid metals to homogenize in a 'soaking pit' after casting and permit dissimilar metals to diffusion bond at moderate temperatures and within short times. In the process, substitutional metallic atoms (ions) move via vacancy jumps while small interstitial atoms such as carbon move from interstice to interstice. Vacancy mobility gives rise to major changes in mechanical properties during and is an important mechanism in creep deformation under load at elevated temperature.

At a slightly larger level, linear atomic packing defects known as dislocations, give rise to the ability of metallic materials to deform substantially under load. When a plane of atoms in the lattice ends, it gives rise to an edge 'dislocation'. Such a dislocation can break and remake bonds relatively easily in a metal and thereby shift an atomic distance. The process can continue until a surface step results. Many dislocations moving in this fashion can give rise to significant shape change in the material at moderate stresses. The onset of such massive dislocation motion in a metal is termed yield and occurs at the 'yield stress' or 'elastic limit'. Dislocations explain how a fine grained polycrystalline metal containing many microstructural features which interfere with dislocation motion may have a yield stress as great as 10 GPa. Dislocations interact with each other in three dimensions and multiply.

Structural components require the use of alloys because alloying elements addition for enhancing the mechanical properties or other material characteristic (e.g. corrosion resistance). The alloys may consist of over ten different elements in specific concentrations with the purpose to optimize a variety of properties. Minor alloying additions typically do not alter the basic crystal structure if the elements remain in solid solution. At sufficiently high concentrations, other phases (either with the same or different crystallographic forms) may precipitate within the base metal (at grain boundaries or within the grains). Solid solution elements and precipitates/particles are used during alloy design to improve the strength of a metal.

II–1.3. Overview of mechanical properties for component design

Many materials, when in service, are subjected to force and loads. The response will depend on many factors. The type of loading (e.g. tension, compression, shear, or combinations thereof) is one key factor. The strain rate, temperature, nature of loading (monotonic versus alternating fatigue stresses), and the presence of notches will also affect the deformation response of the metal. Chemical influences, such as those associated with stress corrosion cracking (SCC) and hydrogen embrittlement, as well as physical alterations, such as those resulting from radiation damage, may affect the deformation behaviour. In such situations, it is necessary to know the characteristic of the material and to design the member from which it is made such that any resulting deformation will not be excessive and fracture will not occur.

The design is the ultimate function of engineering in the development of products and processes, and an integral aspect of design is the use of mechanical properties derived from mechanical testing.

II–1.3.1. Deformation

To understand the different deformation modes (elastic and plastic), the structure of a metal needs to be considered. Elastic deformation can be conceptualized by considering the bonds between individual atoms as springs. As mentioned above, a metal will stretch under the application of a load, but will return to its original shape after the removal of that load if only elastic deformation occurs. Just as a spring constant relates the force to the applied displacement (i.e. $F = kx$), the elastic modulus (E) relates the tensile stress to the applied tensile strain (i.e. $\sigma = E\varepsilon$) and is simply the slope of the linear portion of the tensile stress-versus-tensile strain curve produced in the tension test.

Plastic deformation results in a permanent change of shape, meaning that after the load is removed, the metal will not return to its original dimensions. This implies a permanent displacement of atoms within the crystal lattice. If a perfect crystal is assumed, this deformation could only occur by breaking all the bonds at once between two planes of atoms and then sliding one row (or plane) of atoms over another (exhibiting perfect plasticity). However, as it was described in the section before, metals and alloys are not crystallographically perfect. Instead, the lattice contains many imperfections. One such imperfection is dislocations (edge, screw or mixed of both), which, for simple cubic structures, can be the extra half plane of atoms. While it may appear that this structure is unfavourable, dislocations are necessary for alloys for deformation and strength.

II–1.3.2. Strength

The strength of a metal is related to the ease, or conversely the difficulty, of dislocation motion. If dislocation motion is uninhibited (i.e. the motion is initiated easily and continues without hindrance), the strength will be low and relatively little work hardening will occur. In contrast, the presence of obstacles, or barriers, within the microstructure, slow dislocation motion, resulting in an increase in strength and hence hardening.

Grain boundaries provide an obstacle to dislocation motion. As the grain size is decreased, the strength (σ) of the metal typically increases according to the Hall-Petch relationship [II–1,II–2]:

$$\sigma = \sigma^0 + kd^{-\frac{1}{2}} \tag{II–1}$$

where σ^0 is the intrinsic strength of the metal, k is a coefficient, and d is the grain diameter. At small grain sizes, there is a larger probability of dislocation-dislocation interactions (e.g. dislocation 'pile-up' at the grain boundaries), leading to a larger resistance to dislocation motion. As the grain size increases, the opposition to dislocation motion, due to back stresses associated with dislocation tangles at grain boundaries, lessens due to the larger distances between grain boundaries.

The strength of a metal is also related to the alloying elements content. There are two scenarios for incorporating atoms into a metallic matrix; substitutional and interstitial atoms. Substitutional atoms take the place of matrix atoms. Because of the mismatch in atomic size between the substitutional atom and the matrix atom, the lattice may become locally strained. This lattice strain may impede dislocation motion and is conventionally considered to be the source of solid solution strengthening in metals.

Interstitial atoms can also be present within the metal. In this case, the atom is much smaller than the matrix atoms and is in the gaps (or interstices) in the crystal lattice. Most often, interstitial atoms can diffuse to the dislocation core due to the most open structure and the local tensile stresses in this region of the crystal lattice. The presence of the interstitial can inhibit dislocation motion, leading to dislocation 'locking'. This locking necessitates larger applied stresses to produce dislocation motion and further plastic deformation.

Alloying element (substitutional and interstitial) additions can often cause second phase particles or precipitates to be present in the structure. A fine dispersion of small particles generally produces a higher strength than a coarse dispersion of large particles. At each volume fraction, small particles produce a higher strength than large particles. The strengthening increase is related to two factors:

- A higher probability of the mobile dislocation intersecting the particles due to the smaller interparticle spacing;
- The higher fracture resistance of smaller particles.

Conversely, as the size of the particles increases at a constant volume fraction, the interparticle spacing increases, causing the particles to become less effective strengthens (i.e. barriers to dislocation motion) [II–3].

II–1.3.3. Fracture mechanism

The fracture may be defined as the mechanical separation of a solid owing to the application of stress. Fractures of engineering materials are broadly categorized as ductile or brittle, and fracture toughness is related to the amount of energy needed to create fracture surfaces.

Fracture mechanism technology has significantly improved the ability to design safe and reliable structures. The application of fracture-mechanism concepts has identified and quantified the primary parameters that affect structural integrity. These parameters include the magnitude and range of the applied stresses; the size, shape orientation, and rate of propagation of the existing crack; and the fracture toughness of the material.

Two categories of fracture mechanism are Linear-Elastic Fracture Mechanism (LEFM) and Elastic-Plastic Fracture Mechanism (EPFM). Linear-elastic fracture mechanism is used if the crack tip in a body is sharp and there is only a small amount of plastic deformation at or near the crack tip. Some materials that are designed using LEFM concepts are high strength steels, titanium, and aluminium alloys. Elastic-plastic fracture mechanism is used when the crack tip is not sharp and there is some crack-tip plasticity (blunting). Elastic-plastic fracture mechanism is used to design materials such as lower strength, higher toughness steels. Elastic-plastic fracture mechanism is also used in the evaluation of ceramic matrix composites.

The LEFM approach to fracture analysis assumes a part or specimen contains a crack or other flaw, the crack is a flat surface in a linear elastic stress field, and the energy released during rapid crack propagation is a basic material property and is not influenced by part size.

Linear elastic fracture mechanism technology is based on an analytical procedure that relates the stress field magnitude and distribution near a crack tip to the nominal stress applied to the structure: to the size, shape, and orientation of the crack or crack-like imperfection; and to the material properties. A crack in a loaded part or specimen generates its own stress field ahead of a sharp crack, which can be characterized by a single parameter called stress intensity (K). Relations between the stress intensity factors and various body configurations; crack sizes, shapes, and orientations; and loading conditions are available in the published literature (e.g. Ref. [II–3]). K represents a single parameter that includes both the effect of the stress applied to a sample and the effect of a crack of given size in the sample. It can have a simple relation to applied stress and crack length, or the relation can involve complex geometry factors for complex loading, various configurations of real structural components, and variations in crack shapes. In case of EPFM, parameters like J-Integral and Crack Tip Opening Displacement (CTOD) can be considered.

In summary, the role of structural engineers is to determine stresses and stress distributions within members that are subjected to well-defined loads. This may be accomplished by experimental testing techniques and/or by theoretical and mathematical stress analyses.

The design of components requires an understanding of the materials properties and how they will be used by the component. The manufacturing process that is used to produce the part also needs to be considered during the design process because manufacturing methods influence materials properties and the selection of appropriate mechanical testing methods to ensure that the component will meet its needed life cycle.

II–2. OBJECTIVE

The objective of this annex is to give an overview of the key mechanical properties and key data analysis for design structural components for fusion.

The sections describe the fundamental principles of the structural materials from the point of view of materials science (microstructure and mechanical properties) to materials engineering (materials limit data and design rules).

In addition, key factor analyses on mechanical properties and design rules on structural materials are presented.

II–3. RADIATION-INDUCED DEFECTS ON MICROSTRUCTURE

The principal reason of mechanical degradation of a component during operation is due to the changes that experienced its microstructure because of loads, temperature, and irradiation.

Irradiation damage caused by high-energy particles (electrons, ions, protons, or neutrons) occurs when the particles displace atoms from their normal lattice positions to form Frenkel defects (vacancies and interstitials) [II–4]. The atom displaced by the high-energy particle transfers energy to surrounding atoms, often displacing some of them, which, in turn, may also cause displacements, resulting in a displacement cascade. The extent of the displacement damage is expressed in terms of how often an atom is displaced from its normal lattice position during the irradiation as displacements per atom, dpa.

In addition to displacement damage, neutrons cause transmutations reactions with atoms of the irradiated steel that produce solid and gaseous reaction products. The solid products are usually another metal atom, which it is not expected to produce detrimental effects on properties, with a few exceptions. The gases produced are helium and hydrogen.

The consequence of each displacement event is the production of a vacancy (a vacant lattice site left by the displacement) and an interstitial (a displaced atom that came to rest in an interstitial position). It is the deposition of the vacancies and interstitials that is the primary cause of the irradiation effects on properties. At temperatures of reactor operation, interstitials and vacancies are mobile, and most are eliminated by a one to one recombination and have not affect on properties. The defects that do not recombine migrate to sinks, where they are absorbed. Sinks include surfaces, grain boundaries, precipitate matrix interfaces, dislocations, and existing cavities. If vacancies and interstitials are accepted equally at the sinks, they also annihilate. It is when the vacancies or interstitials are accepted preferentially at sinks that damage accumulates and properties are affected. Mechanical and physical properties are affected by the defect clusters that can form. Clusters consisting of interstitials can evolve into dislocations loops. Vacancy clusters can develop into vacancy loops, micro voids, or cavities. Solute clusters and precipitates can also form under certain conditions.

The type of defect cluster that forms depends on irradiation temperature [II–5–II–7]. At temperatures below ~ 0.3 T_m (T_m is the melting point of the material), interstitials are mobile relative to vacancies, and interstitials combine to form dislocations loops that increase strength and decrease ductility (more details in the next section).

Vacancies become increasingly mobile for irradiation above ~ 0.3 T_m, producing a dislocation and cavity structure (Fig. II–2). In the absence of dissolved gasses can collapse into loops. Cavities form in the presence of dissolved gases and can cause an increase in volume (swelling). This phenomenon occurs because certain sinks have a bias and do not accept vacancies and interstitials equally [II–5]. Two types of cavities can from bubbles and voids. Bubbles contain gas atoms at a pressure in equilibrium with the surface tension. Voids can contain gas atoms, but the pressure is less than the equilibrium pressure. For void swelling to occur, the temperature needs to be high enough for the vacancies to be mobile and low enough for vacancy supersaturation to occur.

Finally, at high irradiation temperatures, higher than about 0.35–0.4 T_m, the defect clusters are unstable. The vacancy concentration is high and the diffusion is rapid. These two factors lead to vacancy interstitial annihilation, and displacement damage (dpa) has little effect on properties. However, the helium transmuted at high temperatures can give problems in embrittlement and a loss in ductility, mainly if the bubbles nucleate and growth at the grain boundaries.

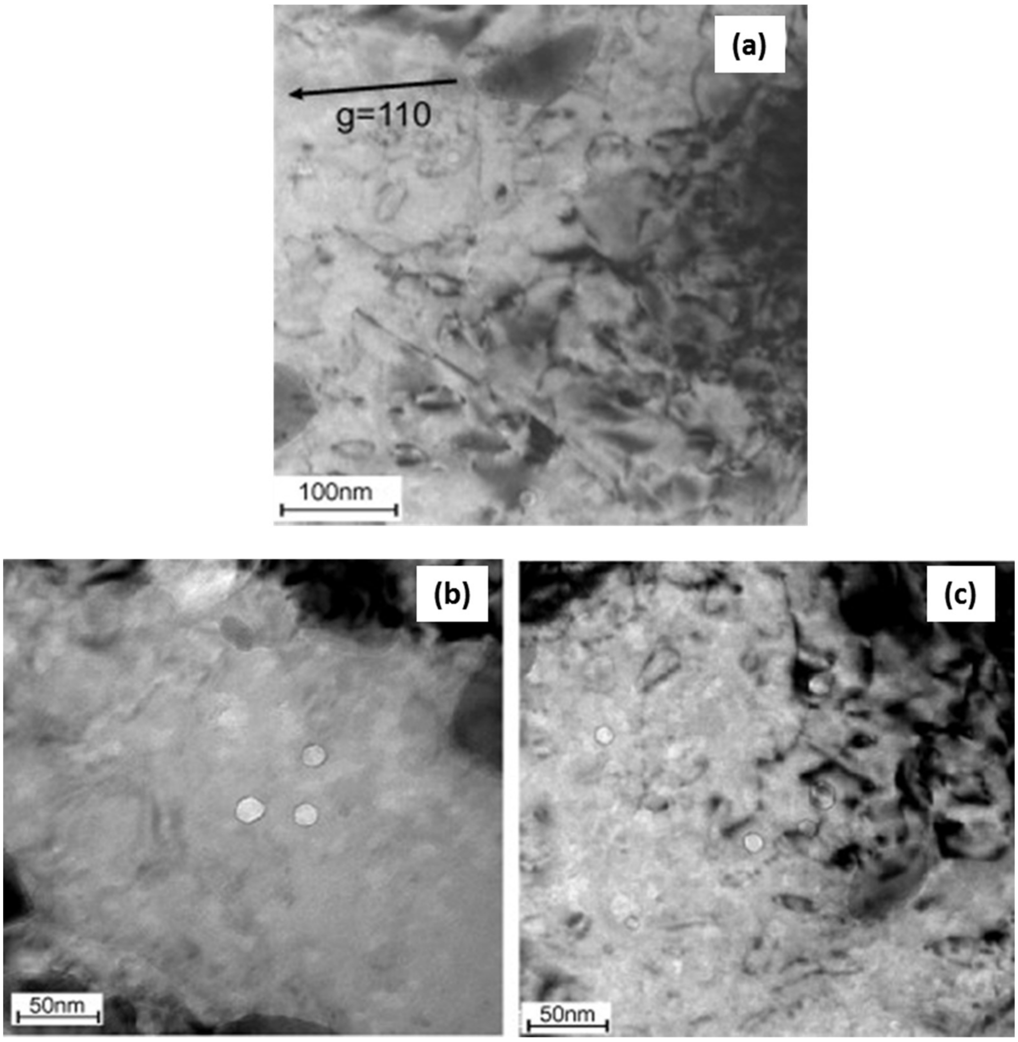


FIG. II–2. (a) TEM images of EUROFER97 neutron irradiated at 350ºC and 16.3 dpa showing dislocation loops; (b) TEM images showing cavity formation; (c) same as (b) [II–8].

During irradiation at elevated temperatures the microstructure can also experience, a non-equilibrium segregation process called Radiation Induced Segregation (RIS). This phenomenon can occur as consequence of two processes:

- The strong interaction between solutes and the point defects (vacancies and interstitial atoms) generated during irradiation, resulting in coupled transport of the solute atoms by the point defects fluxes to and away from sinks, such as grain boundaries, free surfaces, dislocation loops, and voids surfaces.
- The inverse Kirkendall Effect, whereby the faster-diffusing species exchange more often with the irradiation induced vacancies migrating to sinks than slow diffusing species.

In addition to RIS, secondary phases can nucleate and/or transform during irradiation. The fundamental processes which affect phase formation and stability include

(a) Dissolution, disordering and mixing, leading to phase decomposition, transformation and new formation;
(b) Diffusion enhanced by irradiation;
(c) RIS.

II–4. MECHANICAL PROPERTIES FOR FUSION STRUCTURAL COMPONENTS

The structural materials will be exposed to high temperatures and high levels of irradiation (14 MeV of neutrons), as well as to high mechanical and thermo-mechanical stresses. The fusion neutron spectra will produce atomic displacement cascade and nuclear transmutation reactions (He and H) within the irradiated materials. The final microstructure of the irradiated materials results from a balance between environmental conditions, especially radiation damage and temperature, and stress/strain histories.

The microstructure evolution in a fusion reactor environment may engender degradation of the mechanical properties, leading to strong hardening and/or embrittlement effects.

Candidate structural materials for blanket components have a chemical composition based on reduced activation chemical elements (Fe, C, Cr, W, V, Ta, Ti). They are mainly Reduced Activation Ferritic/Martensitic (RAFM) steels and ODS Ferritic Steels (ODS-FS). Their development in the middle 1980s emerges from the limitations of austenitic steels applications under neutron irradiations and the good performance of ferritic/martensitic steels as fuel claddings in liquid metal cooled fast reactor. Development of RAFM steels has been the object of a wide cooperation between the different fusion parties. The steel denominated EUROFER97 (9Cr-1W VTa) was developed within the European Union as reference structural material [II–9,II–10] and it is the steel proposed as structural material for the Test Blanket Modules (TBM) of ITER [II–11].

F-82H (8Cr-2WVTa) and JLF-1 (9Cr-2WVTa) were introduced as the major candidates from Japan [II–12,II–13]. After the development of European and Japanese RAFM steels, other variants were developed in China, China Low Activation Martensitic (CLAM) [II–14,II–15], INRAFM of India [II–16], and Advanced Reduced-Activation Alloy (ARAA) of Korea [II–17], with compositions based on the EUROFER97.

Tungsten based alloys are also candidate materials for structural applications in the high-temperature region of plasma facing components, such as high heat flux and high-temperature heat removal units of DEMO-relevant helium cooled divertor concepts.

The structure and the environment of the structural components for fusion application have many unique features that require special attention on the mechanical properties of the structural materials.

The more usual considered mechanical properties of metals and alloys include strength, ductility, fatigue, fatigue crack growth, thermal and irradiation creep, and fracture toughness. All these properties are important in the design of a structure that is to experience an irradiation environment.

II–4.1. Tensile properties: strength and ductility

While determining the mechanical properties of irradiated materials, tensile properties, typically yield strength, ultimate tensile strength, uniform elongation, total elongation, and reduction of the area are the most commonly considered because they are usually the simplest and the least costly to measure. In addition, the tensile properties can be used as an indicator of the other mechanical properties.

Ductility is a more vulnerable parameter than strength to radiation effects since it tends to be very high in unirradiated and is often reduced to quite low levels by irradiation. Like strength, ductility exhibits saturation with increasing fluence, although the behaviour is significantly more complex than that of strength.

The effect of neutron irradiation on the strength of ferritic/martensitic steels depends on the irradiation temperature. If the irradiation temperature is below the range of 400–500ºC, irradiation-induced microstructural changes lead to lattice hardening, which causes an increase in the yield stress and ultimate tensile strength and a decrease in the uniform and total elongation (Fig. II–3).

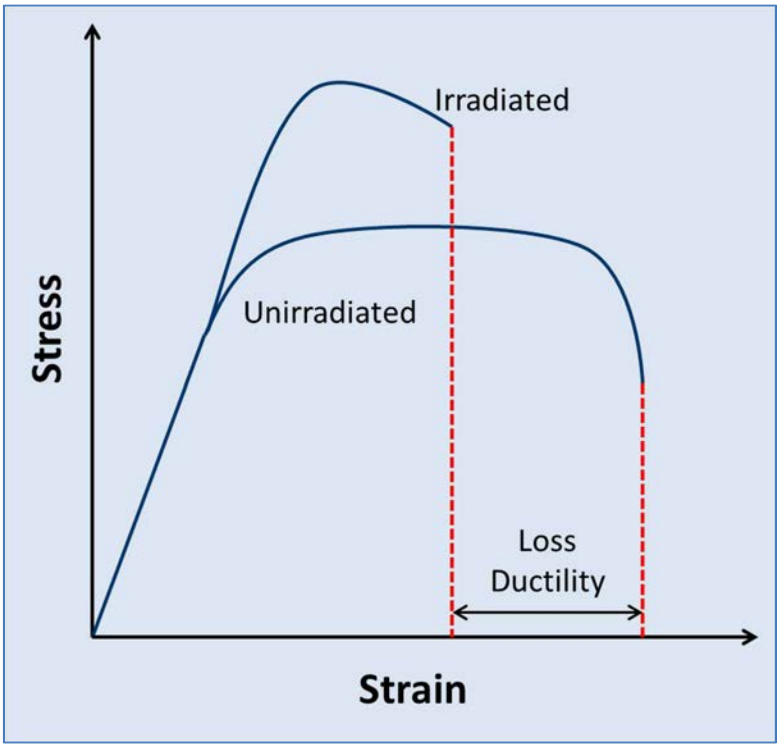


FIG. II–3. Schematic diagram of stress-strain curves showing the tensile properties variation after irradiation in the temperature range where irradiation hardening occurs.

The grade of the hardening (increase strength) decreases as the irradiation temperature increase until it disappears between 400ºC and 500ºC. Hardening is caused by irradiation-induced dislocation loops and precipitation. Irradiation produces dislocations loops have their greatest effect for low-temperature irradiation.

Most of the work performed to determine and evaluate the effects of irradiation on strength and ductility on structural materials for fusion has been done on steels irradiated in fast and mixed-spectrum reactors, with some other studies in test reactors where only low fluence irradiations are possible.

An example of yield stress variation in function of the damage is shown in Fig. II–4 for low activation steels irradiated in fast reactors [II–18].

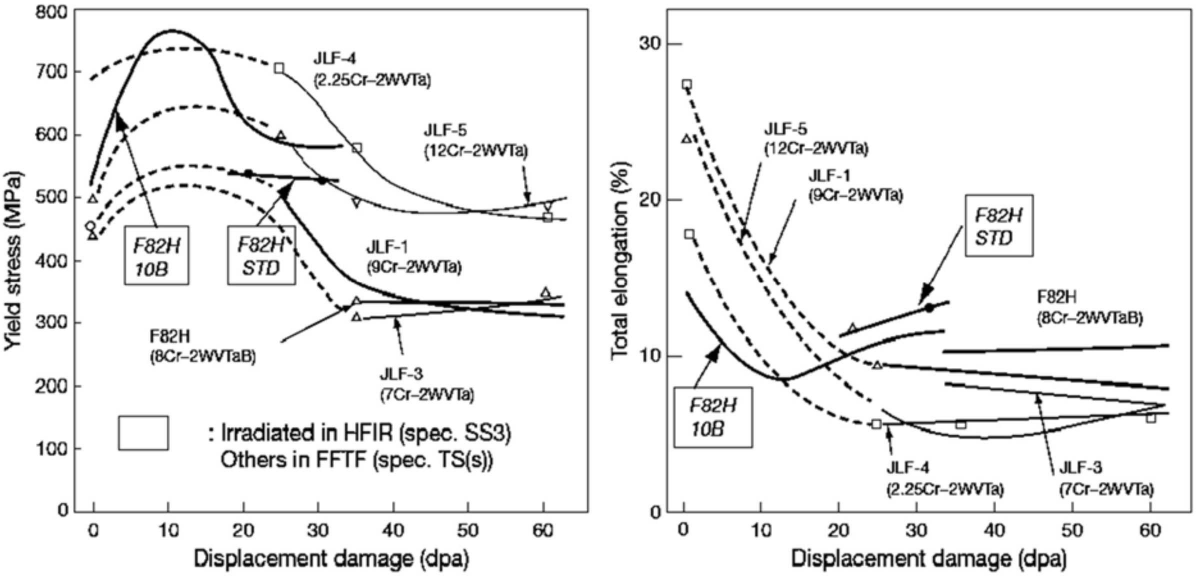


FIG. II–4. Yield strength of several martensitic alloys following irradiation at 400ºC in FFTF or high flux isotope reactor [II–18].

At lower damage levels, ferritic/martensitic alloys show a pronounced peak in strength as result from rapid hardening due to irradiation-produced defects, but the effect of irradiation hardening is offset by irradiation-enhanced recovery, resulting in a decrease in strength and hence a peak in strength [II–19]. Strength and ductility exhibit saturation with increasing the damage level beginning at about 30 dpa.

Similar observations on strength and ductility have been made on EUROFER97 steel. Their properties were characterized in several irradiation programmes up to 80 dpa. The irradiations were carried out in different test reactors [II–20–II–27] at irradiation temperatures between 300ºC and 350ºC. The recompilation and analysis of the results have been done by Gaganidze et al. [II–28]. As can be seen in Fig. II–5, neutron irradiation leads a significant increase of ultimate tensile strength and yield stress (hardening). Close values of the ultimate tensile strength (R_m) and yield stress ($R_{p0.2}$) in the irradiated conditions indicates a strong suppression of the strain hardening capability under neutron irradiation. Hardening (increase of yield strength) on EUROFER97 is very sensitive to the irradiation parameters (dose and temperature). However, yield stress values after irradiation at 350ºC/15 dpa seem to indicate softening attributable to the reduction of dislocation density (partial recovery). Irradiation accelerates thermal ageing because the irradiation produced vacancies enhance diffusion.

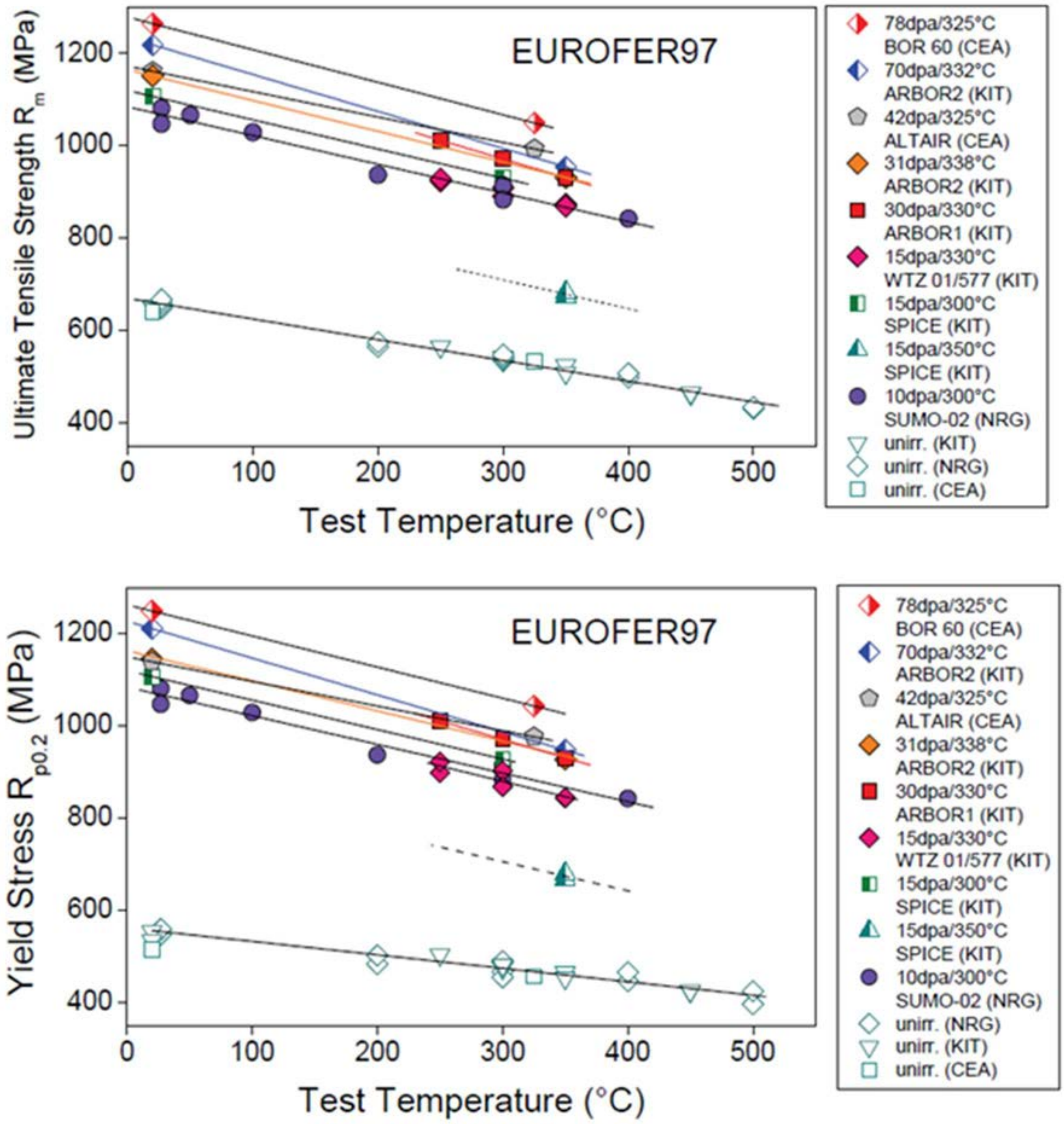
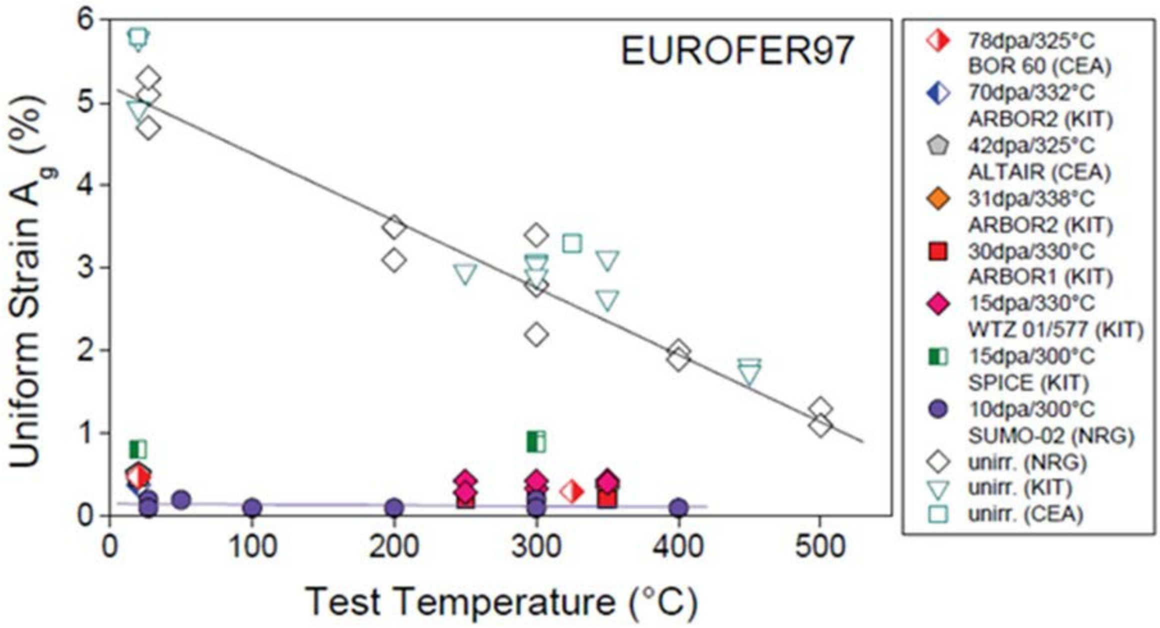


FIG. II–5. Ultimate tensile strength and yield stress vs. test temperature for EUROFER97 in the unirradiated condition and after neutron irradiations in different European irradiation programmes [II–28].

Tensile ductility is a more vulnerable parameter than strength to radiation effects since it tends to reduce to quite low levels by irradiation. As can be seen in Fig. II–6, the ferritic/martensitic steel EUROFER97 irradiated at low temperature (300–350ºC) show very little uniform elongation with values mostly below 0.5%. Total elongation also decreases significantly, although the values most are above 10%. Total strain values higher than 10% are the great relevance for structural materials applications.

All the RAMF steels designed and development for their application as structural components, present low strain hardening capacity [II–28,II–29] at low irradiation temperatures.



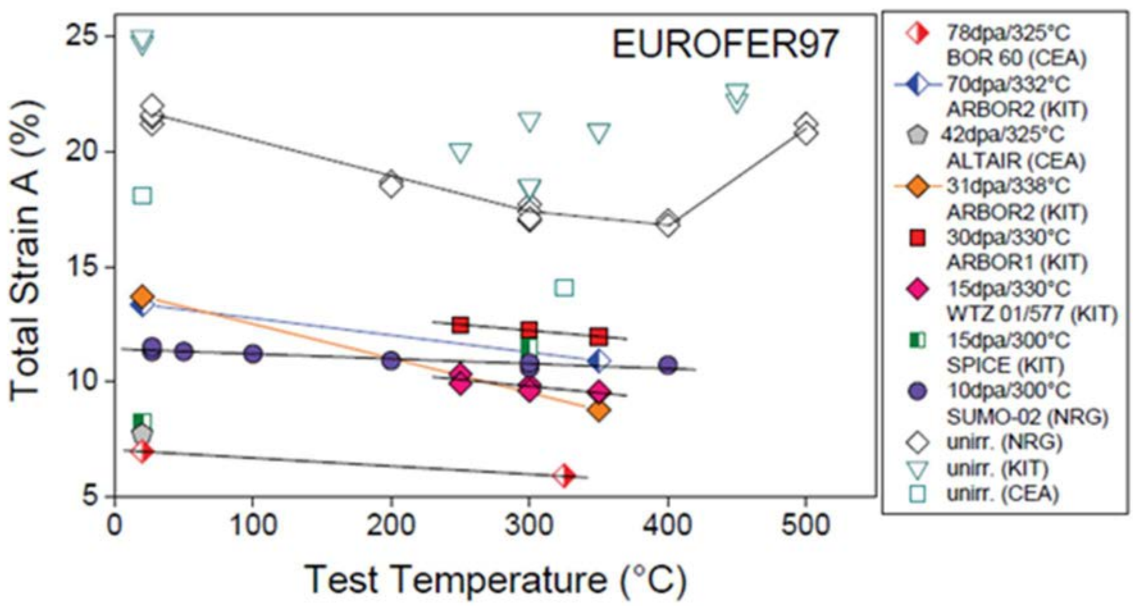


FIG. II–6. Uniform and total strain vs. test temperature for EUROFER97 in the unirradiated condition and after neutron irradiations in different European irradiation programmes [II–28].

From the point of view of DEMO design, design rules, and the corresponding stress limits need to be established according to selected code and standard in order to prevent failures (see Section 5). For that, a key parameter to analyze is the evolution of the hardening with dose.

For the structural material EUROFER97 and other RAFM steels, the evolution of the hardening, as an increase of the yield strength (Fig. II–7) has been evaluated by Gaganidze and Aktaa [II–28] up to 80 dpa. At 70 dpa the irradiation hardening is very similar for all the materials investigated, but the number of the data is very limited to perform detailed statistical analysis of the results.

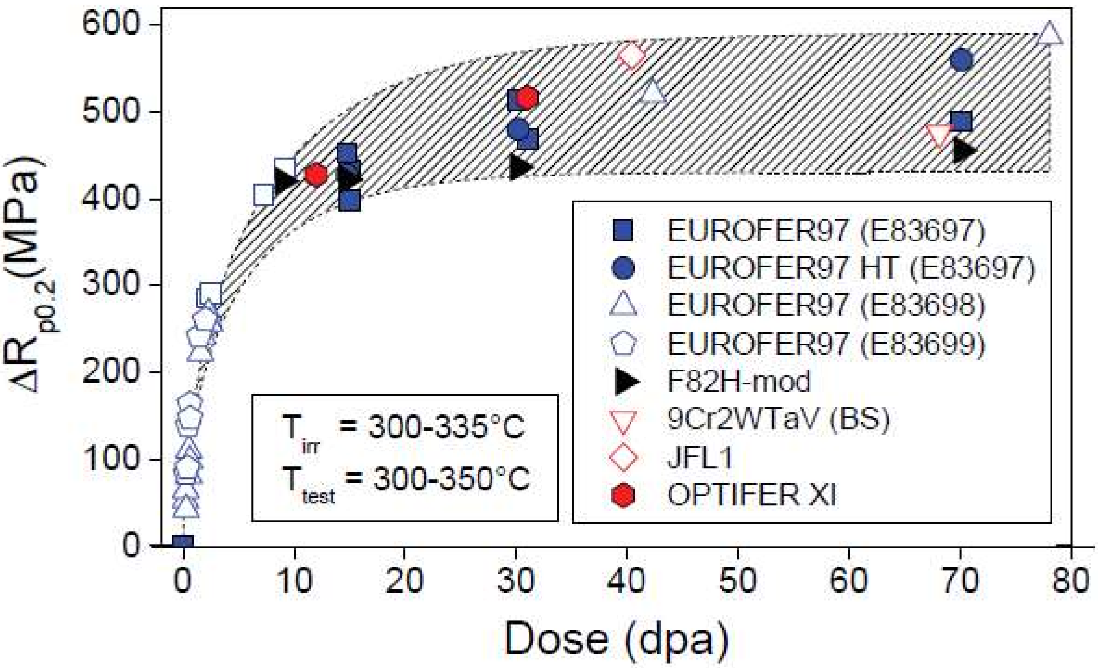


FIG. II–7. Irradiation hardening versus irradiation dose for EUROFER97 and other RAFM steels for T_{irr} =300–335 °C and T_{est} =300–350°C [II–28].

II–4.2. Fatigue properties

High-temperature components in power plants (boilers, nuclear reactors) are subject during service to fatigue straining due to thermal cycling or a combination of thermal and mechanical deformation in which the stain cycle includes a hold period. The structural materials in a fusion system will undergo thermomechanical fatigue because of the mechanical and electromagnetic loadings and the cyclic strain induced by the temperature changes during the plasma burn and off-burn periods [II–30,II–31].

Fatigue is the progressive, localized, and permanent structural damage that occurs when a material is subjected to cyclic or fluctuating strains at nominal stresses that have maximum values less than (and often much less than) the static yield strength of the material [II–32]. This process of fatigue failure can be divided into different stages, which, from the standpoint of metallurgical processes, can be divided into five stages [II–32]:

(a) Cyclic plastic deformation prior to fatigue crack initiation;
(b) Initiation of one or more microcracks;
(c) Propagation or coalescence of microcracks to form one or more microcracks;
(d) Propagation of one or more macrocracks;
(e) Final failure.

This division is defined by the characterization of the underlying fatigue damage of a material. It also clearly defines the requirement of plastic deformation for the onset of crack initiation. In general, three simultaneous conditions are needed for the occurrence of fatigue damage: cyclic stress, tensile stress, and plastic strain. If any one of these three conditions are not present, a fatigue crack will not initiate and propagate. The plastic strain resulting from cyclic stress initiates the crack, and the tensile stress (which may be localized tensile stresses caused by compressive loads) promotes crack propagation [II–32]. In general, the fatigue process consists of a crack initiation and a crack propagation phase.

Another important engineering advance is the transfer of the multistage fatigue process from the field to the laboratory. To study, explain, and qualify component designs, or to conduct failure analyses, a key engineering step is often the simulation of the problem in the laboratory.

Fatigue tests may be either stress (or load) or strain (or displacement) controlled. Stress control testing can be used for design situation in which the applied stress is primarily within the elastic range and the resulting endurance is high (High Cycle Fatigue, HCF); the material strength controls the behaviour and crack initiation is the dominant event under these conditions. However, the strain controlled method is applied in the design and evaluation of components subjected to secondary stresses (first wall and breeding blanket in a fusion reactor). The load is high as near the notches, and the total strain range ($\Delta\varepsilon_t$) has a significant plastic component ($\Delta\varepsilon_p$) relative to the elastic strain ($\Delta\varepsilon_e$). In this case, the response of the material is deformation dependent, the ductility being the prime factor governing the fatigue resistance, and the number of cycles to failure is low (Low Cycle Fatigue, LCF). Cracks initiate relatively early in life and crack growth is the dominant failure criterion. However, the fatigue process may be modified because of stress relaxation by thermal creep deformation and cracking during the tension and/or compression hold periods in creep-fatigue tests.

From fatigue tests three basic types of properties can be obtained depending on the fatigue design philosophy:

— Stress-life (*S*-N).
— Strain-life (ε-N).
— Fracture mechanic crack growth (d*a*/dN-ΔK).

The reduction in fatigue endurance with increasing strain range may be described by the Coffin-Manson relation [II–33–II–35]:

$$\Delta\varepsilon_p \cdot N_f^{\beta} = C_p \tag{II–2}$$

The test temperature and strain-range dependences of the LCF lives of the steels may be modified by introduction of hold periods at the peak tension and/or compression strains of the cycles. In general, hold times reduce the high-temperature endurance compared to that in continuous-cycling tests, the magnitudes of the effects being more pronounced at lower total strain range. In general, compressive hold period is slightly more damaging than tensile holds for 9-12% martensitic steels [II–36]. For the EUROFER97, the fatigue behaviour becomes much more complicated when introducing hold times in LCF tests (Fig. II–8), showing different characteristic with tension, compression or symmetrical hold times [II–37].

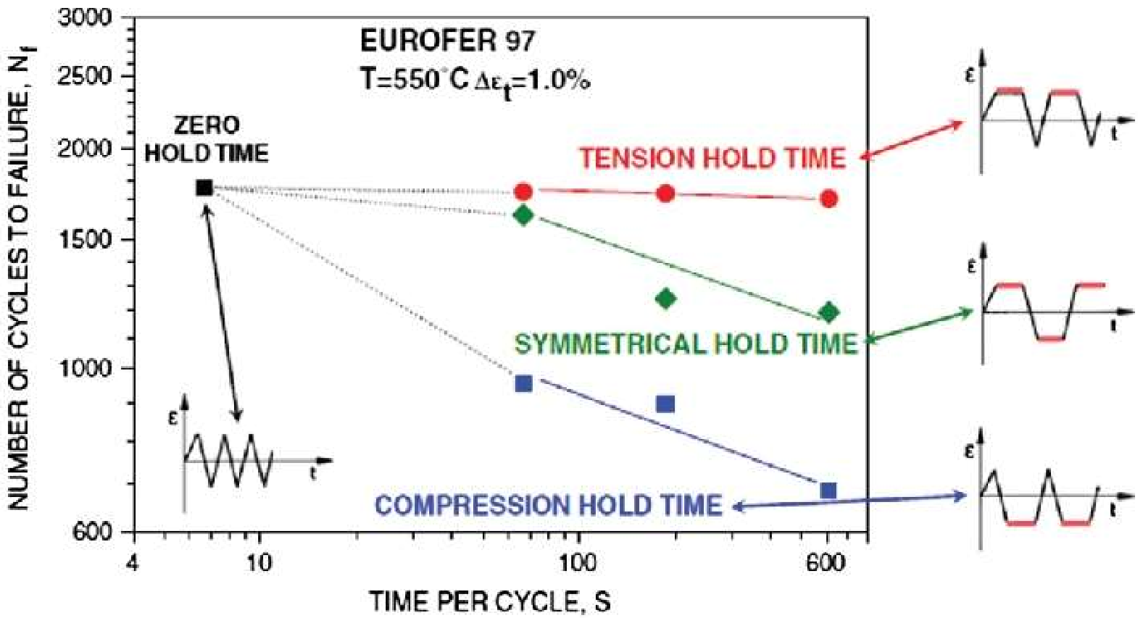


FIG. II–8. Influence of hold times on fatigue life of EUROFER97 [II–37].

This difference observed could be attributed to the tests environment (air), because limited tests performed on vacuum have shown no difference between hold times in compression and in tension.

The ferritic/martensitic steels exhibit very pronounced softening under LCF tests (Figs. II–9 and II–10). The cyclic stress amplitude decreases rapidly after a few cycles and then stabilizes at a value which decreases slowly as a function of the number of cycles, dropping sharply just before failure of the specimen [II–38].

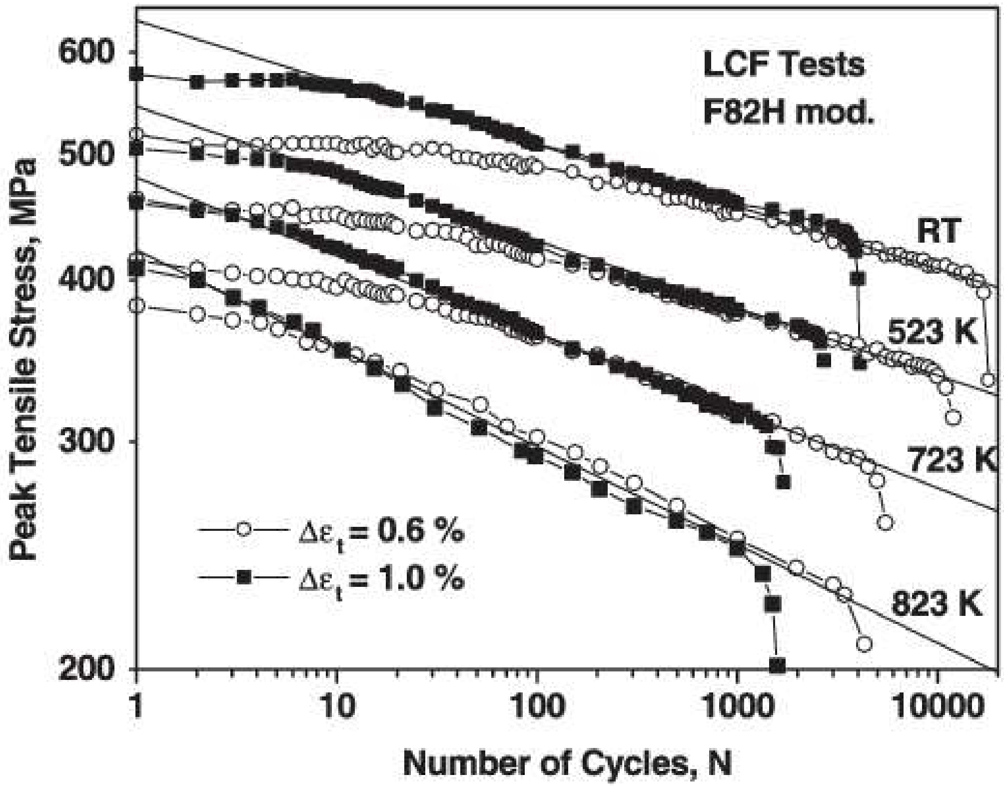


FIG. II–9. Evolution of the peak tensile stress versus number of cycles during LCF testing of F82H mod. [II–38].

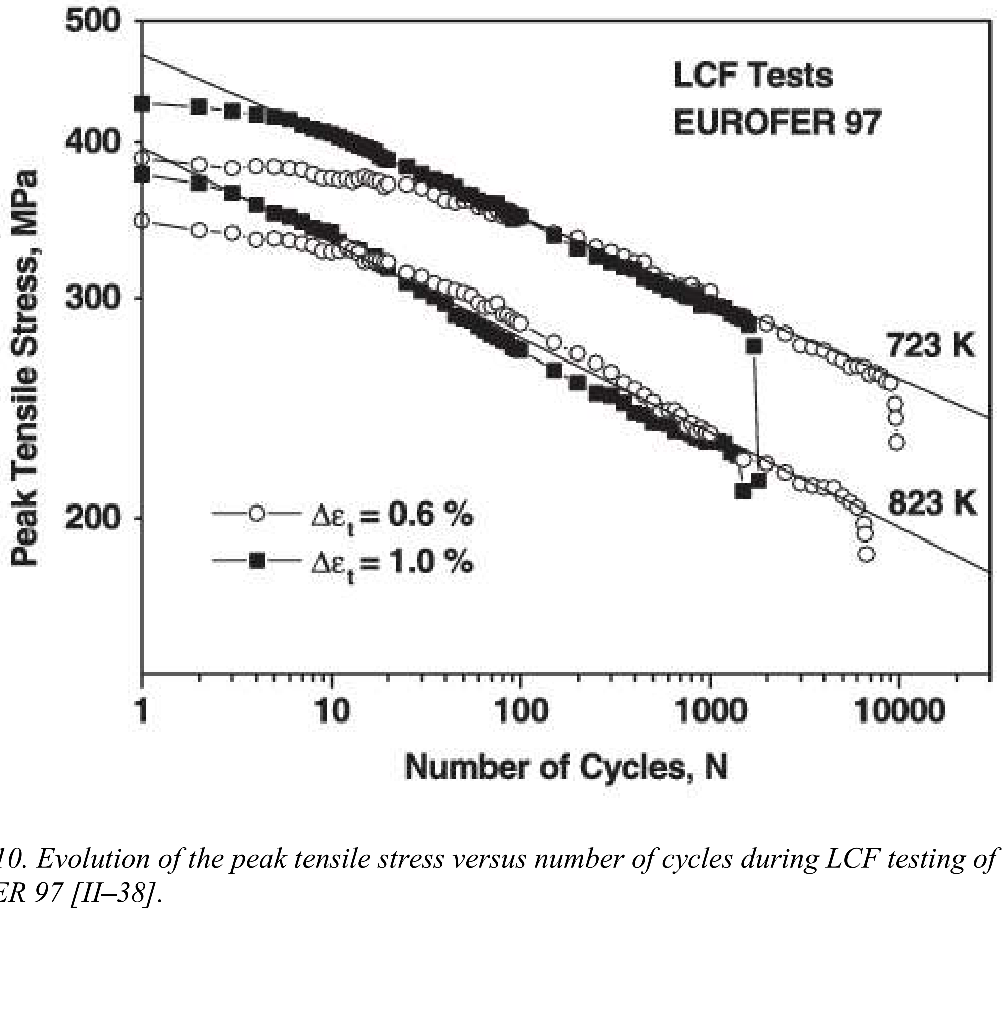


FIG. II–10. Evolution of the peak tensile stress versus number of cycles during LCF testing of EUROFER 97 [II–38].

There are few data on the effects of irradiation (neutron irradiation) on the fatigue behaviour of EUROFER97 and other RAFM steels [II–39–II–41]. As a rule, the fatigue lifetime of the irradiated material has to be shorter due to loss of ductility. However, post irradiation tests performed on EUROFER97, which is summarized in Ref. [II–28], reveal that the majority of neutron irradiated specimens has only a minor influence on fatigue behaviour (Fig. II–11). Irradiation induced hardening may differently affect LCF behaviour and can yield:

- Enhanced lifetime in comparison to the unirradiated state because of the reduction of the inelastic strain amplitude especially at low strain ranges.
- Reduced lifetime because of accelerated fatigue damage accumulation due to enhanced stress levels especially at high strain ranges.

The fatigue life could be considerably shorter when the tests are performed concurrently with irradiation [II–42–II–45]. Post-irradiation experiments do not consider the fact that the microstructure of the material is changing during fatigue as well as during irradiation. Simultaneous irradiation and fatigue can develop a different microstructure and lead to a different material response. This phenomenon is however very dependent on the ratio between the dose rate and strain rate.

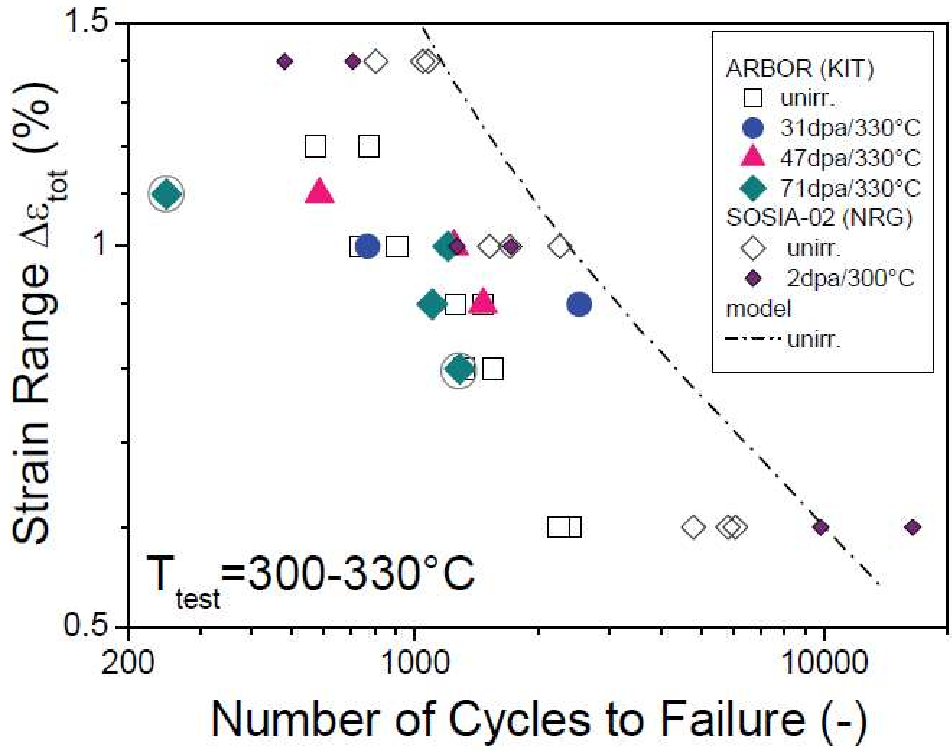


FIG. II–11. Fatigue lifetime for unirradiated and up to 71 dpa irradiated (T_{irr} = 300-337 °C) EUROFER97 vs. total strain range. The dashed line represents the model description of the unirradiated data [II–28].

II–4.3. Creep properties

Creep is defined as the time-dependent plastic deformation that occurs under the influence of a constant applied stress. At elevated temperatures, it is referred to as thermal creep. The rate at which this deformation occurs depends not only on the magnitude of the applied stress, but also on time and temperature. Thus, it is appropriate to consider creep to be a kinetic process and to write an appropriate rate law. In addition, the rate at which a material creeps depends on the size, spacing, and distribution of relevant microstructural features such as a fine dispersion of thermally stable secondary phases.

Deformation processes that operate during thermal creep include:

(a) Dislocation climb in combination with dislocation glide that leads to slip;
(b) Dislocation climb that leads to sub-grain formation;
(c) Grain boundary sliding;
(d) Grain size shape change by diffusional processes.

The first two processes are the most important for the thermal creep of RAFM steels at temperatures ≤ 600ºC.

Creep can also occur under the influence of the simultaneous application of stress and irradiation. Thermal creep becomes significant for irradiation at temperatures ≥ 0.5 T_m, being T_m the absolute melting temperature. However, irradiation creep can be significant at much lower temperatures. As in the case of thermal creep, dislocation climb and glide play a prominent role in the deformation processes that occur during irradiation creep.

Thermal creep (Fig. II–12) has been extensively investigated on two reference structural materials; F-82H and EUROFER97 [II–46–II–49]. Both alloys exhibit creep rupture strength levels comparable to other RAFM steels. Nevertheless, the RAFM steels exhibit a significant decrease of creep strength at temperatures ≥ 550ºC. This behaviour is a limiting factor to its application at higher temperatures than 550ºC.

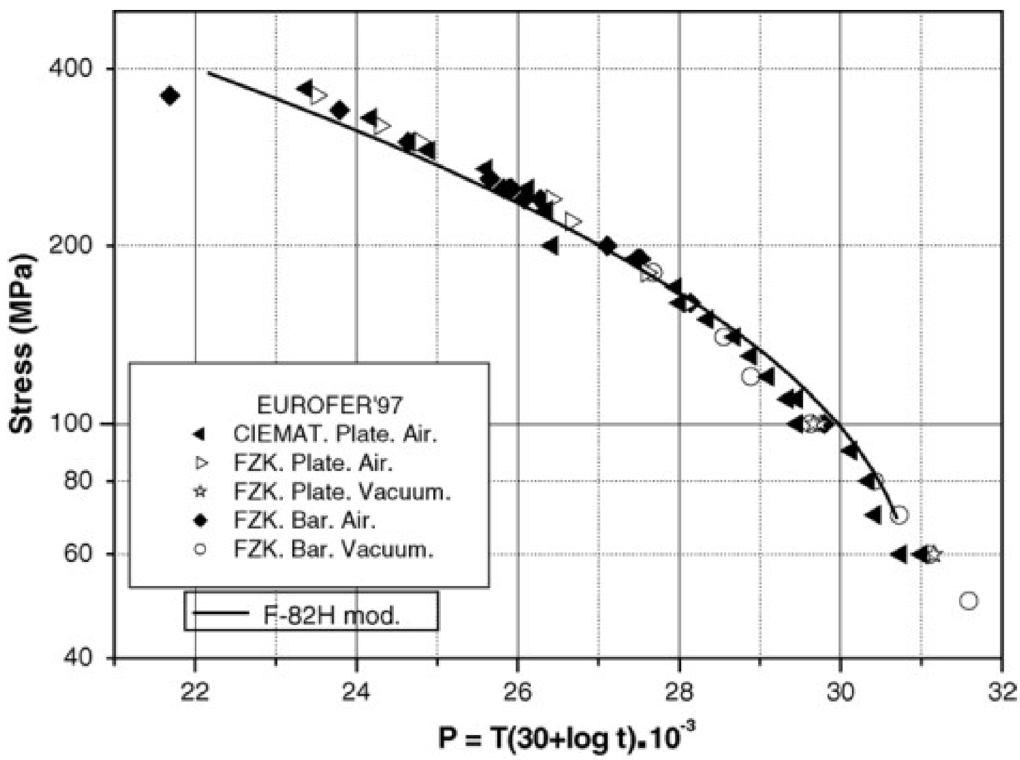


FIG. II–12. Comparative Larson–Miller parameter of the EUROFER97 steel and the F-82H mod. steel in the as-received condition (normalized plus tempered) [II–48].

The irradiation creep resistance is strongly dependent on the microstructural evolution of the defects generated during irradiation under stress. In general, deformation processes involve the stress induced absorption of irradiation produced point defects on dislocations that cause the dislocations climb, which can subsequently lead to glide of the dislocations.

There are limited data to determine and evaluate the irradiation creep behaviour of RAFM steels. Some investigations have been performed on F-82H and several variants of JLF-1 at 300ºC and 500ºC up to 5 dpa using helium-pressurized creep tubes irradiated in HFIR. These tubes were pressurized with helium to hoop stress levels of 0–400 MPa at the irradiation temperature [II–50]. The authors stated that the F82H and JLF-1 with a 400 MPa hoop stress show small creep strains (<0.25%) after irradiation at 300ºC. The irradiation creep strain at 300ºC in these steels is linearly dependent on the applied stress at stress levels below 250 MPa. However, at higher hoop stress levels, the creep strain becomes nonlinear. At 500ºC, the irradiation creep strain of F82H is linearly dependent on the applied stress level below 100 MPa. At higher stress levels, the creep strain increased strongly because had also occurred during irradiation. The lack of irradiation data on irradiation creep indicates the need to perform this kind of experiments.

II–4.4. Fracture toughness and fracture

Fracture toughness is defined as a generic term for measures of resistance to extension of a crack. The term fracture toughness is usually associated with the fracture mechanism methods that deal with the effect of defects on the load-bearing capacity of structural components. Fracture toughness is an empirical material property that is determined by one or more of many standard fracture toughness test methods.

On the other hand, charpy impact tests are frequently used as a screening test on structural materials to evaluate the relative effects of operating conditions (temperature, irradiation) on producing embrittlement in structural alloys. However, charpy data cannot be used directly for design. Design has to be based on a defect tolerant approach [II–51]. Whereas Charpy test measures the total energy to initiate a crack from the notch and propagate the crack cross the material to produce a complete fracture, fracture toughness tests measure just the critical load to extend a pre-existing crack.

Fracture mechanical properties of EUROFER97 neutron irradiated have been assessed by Gaganidze and Aktaa [II–28]. As can be seen in Fig. II–13, neutron irradiation produces shift in Fracture Toughness Transition Temperature (FTTT), regardless of the specimen geometry.

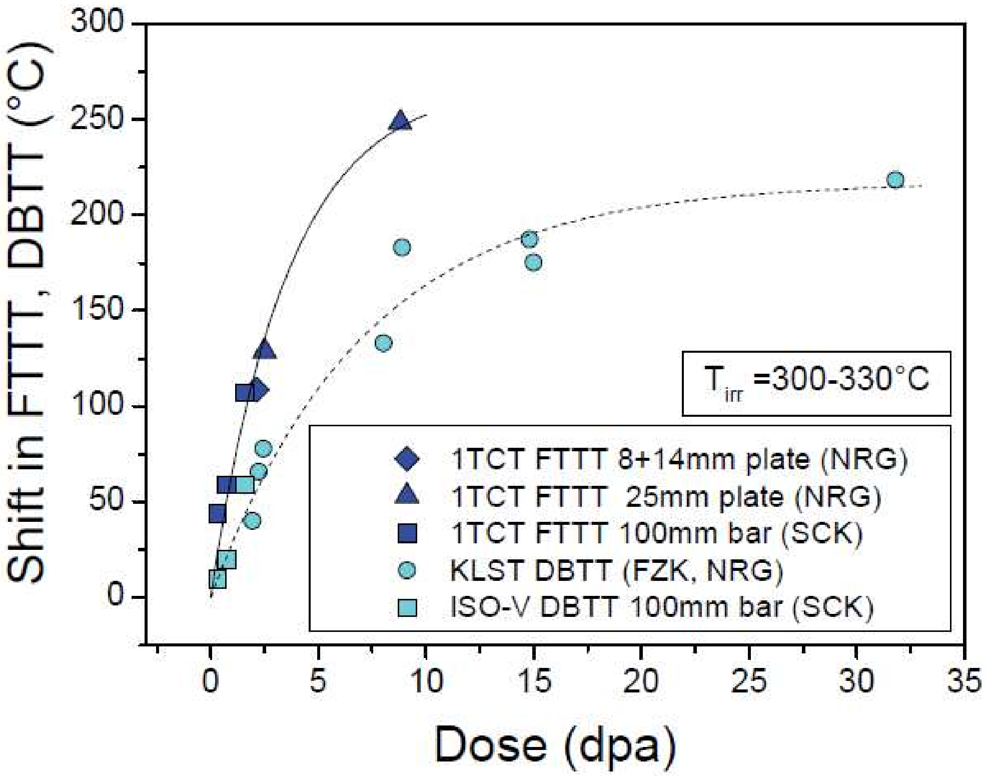


FIG. II–13. Irradiation-induced shift in FTTT and KLST and ISO-V Ductile to Brittle Transition Temperature (DBTT) for EUROFER97 versus irradiation dose [II–28].

Progressive material embrittlement has been observed for EUROFER97 indicating no saturation of FTTT for the achieved damage doses. Irradiation-induced shifts in reference FTTT are significantly larger than shifts in Charpy Ductile to Brittle Transition Temperature (DBTT) [II–28] which indicates non-conservative estimations of the embrittlement by Charpy tests. A critical need exists for fracture toughness data on the irradiated material.

II–5. DESIGN RULES

The design of structural components requires performing design activities according to codes and standards. These codes contain design rules to prevent failure during operation because of imposed loadings. The purpose of the design rules is to ensure that the necessary safety margins are maintained relative to the types of mechanical damage which might occur.

The structure and the environment of the structural components of fusion reactors have unique features that require special consideration; plasma disruptions producing transient of dynamical and thermal stresses, mechanical properties degradation due to the irradiation and the use of new materials (RAFM steels). All these factors require the use a multi-code approach based on ASME, SDC-IC, RCC-MRx, because the existing industrial codes can not cover all the fusion features by only one. The SDC-IC is the only code that contains specific rules taking into account the effects of irradiation on structural materials because was developed under the umbrella of ITER, but its scope is limited to design criteria. As a possible alternative to SDC-IC is the RCC-MRx, which includes specific rules for irradiated materials in order to have a single code suited for the design of all nuclear components to be operated in next generation reactors (fusion and fission).

The purpose of the rules is to ensure by analysis (elastic, inelastic, or experimental), that, if the rules are satisfied, then a component does not undergo any damage. These rules are mathematical expressions, as well as the corresponding limits, depend generally on the considered operating conditions. These conditions joining to the corresponding loadings are classified into several categories based on the probability of occurrence and consequence of failure. To each category, a different 'level' of criteria is then associated.

The rules to be satisfied differ according to:

(a) The level of criteria;
(b) The method of analysis;
(c) The damage type;
(d) The temperature experienced by the component (low or high temperature).

In the case of the structural components for fusion, these rules or their limits also depend on three factors: temperature, neutron damages, and possibly neutron flux, because properties experienced severe degradation due to these three factors, as can be see described in Section 3.

To establish the design rule (criteria to prevent failure), it is necessary to identify the possible failure modes taking into account the operation conditions and the mechanical properties of the material selected. In the context of DEMO in-vessel components, the following failures modes have been identified [II–52]:

(a) Priority failure modes:
    (i) Immediate plastic collapse, immediate plastic instability, immediate plastic flow localisation;
    (ii) Immediate local fracture due to exhaustion of ductility;
    (iii) Fast fracture;
    (iv) Thermal creep;
    (v) Ratchetting;
    (vi) Fatigue;
    (vii) Creep-fatigue.
(b) Secondary Failure Modes:
    (i) Buckling;
    (ii) Excessive deformation.

(c) Modifying Environmental Effects:
   (i) Excessive corrosion;
   (ii) Excessive plasma erosion;
   (iii) Irradiation-induced hardening and embrittlement;
   (iv) Irradiation-induced swelling;
   (v) Stress-corrosion cracking;
   (vi) Impure helium impact on fatigue and creep-fatigue.

II–5.1. Damages

II–5.1.1. Immediate plastic collapse

It is a ductile damage load that occurs when a structure is subjected to a proportional and steadily increasing loading, initially, the structure behaves elastically but at higher loading, irreversible plastic deformation can occur although the load was removed. If the loading is continually increased, all the structure would ultimately reach plastic collapse. Consequently, the structure would not return to its original shape.

II–5.1.2. Immediate plastic instability

This failure mode is also a ductile damage mode. This failure could occur when a structure is loaded well into the plastic regime. The response in a local region of the structure depends on its change in geometry and the strain hardening behaviour of the material.

II–5.1.3. Immediate plastic flow localization

It is a non-ductile failure mode. In a material with very low strain hardening capability and/or with the loss of ductility due to neutron irradiation, the plastic strain may not be readily homogenized, and the structure may fail by the localization of plastic flow. Plastic flow localization appears as a large strain within a narrow band, inclined at an angle to the load. This type of failure could appear in an irradiated material with low uniform elongation.

II–5.1.4. Immediate local fracture due to exhaustion

Reduced ductility is associated with a low elongation (strain) at rupture. This can lead cracking in small regions with high- stress concentration. It is also a non-ductile failure mode.

II–5.1.5. Fast fracture

The term ‘fast fracture’ is used to denote any fracture which initiates from an existing defect or defects under monotonic loading and it is not preceded by an appreciable plastic deformation of the material. Fast fracture is generally caused by unstable propagation of a crack. This type of failure is a damage that cannot be predicted in a deterministic stress analysis.

II–5.1.6. Thermal creep

This type of damage is called time-dependent plastic instability. It is analogous to the immediate plastic instability described before, except that it is time dependent.

II–5.1.7. Ratchetting

If a structure is subjected to cyclic loading, the structure may show signs of permanent deformation at the end of the first cycle. During subsequent cycles, the overall permanent deformation continues to increase with every loading cycle, and the structure gradually changes from its original shape. This is called progressive deformation or ratchetting. The code rules are based on Bree diagram.

II–5.1.8. Fatigue

When the loading applied to a structure varies in a cyclic fashion, the material is subjected to cyclic deformation. If the number of cycles and their amplitudes are sufficiently large, they can cause the material to crack. The damage is initiated by small microscopic cracks or structural imperfections that may growth with repeated cycles, eventually leading to fracture. The code rules are usually based on S-N curves.

II–5.1.9. Creep-fatigue

If the temperature is sufficiently high, creep deformation may occur during each cycle, accelerating the appearance of cracks by the process of creep-fatigue interaction. This phenomenon is associated with thermal creep.

II–5.1.10. Buckling

Buckling is a phenomenon associated with compressive or shear loading of the structures. It consists of the development of deformation modes or patterns which are different in shape from those that manifest themselves at low loading levels. Typical buckling patterns include bows, bulges, or wrinkles. Buckling is a form of instability that, depending on the geometry, may result in immediate collapse or may result in a new, stable configuration. If the latter occurs, then additional loading beyond the point of buckling can cause general instability as well as large deformation or large variations in local deformation. The code rules usually limit applied load to 2/3 of buckling capacity.

II–5.1.11. Excessive deformation

If a structure undergoes large deformation due to elastic, plastic, thermal creep, or irradiation-induced creep strain during operation, the functional adequacy of the component may be compromised.

II–5.1.12. Modifying environmental effects

The modifying environmental effects are not considered as damage mode, but they require of the special attention because they may have a strong influence on the mechanical properties of the materials described in Section 3, leading to accelerating the failure.

II–5.2. Material design limits

Materials design limits can be considered as data collection analysis and curves from the results obtained on experimental tests (physical and mechanical). For each structural material, a set of recommended properties data is needed for structural analysis, which definitions and analysis of the data are described in the codes. Taking as reference the SDC-IC code,

Appendix A, Materials Design Limit Data [II–53], the properties of each structural material need to be the following:

(a) Physical properties:
  (i) Coefficient of thermal expansion;
  (ii) Young's Modulus;
  (iii) Poisson's ratio;
  (iv) Mass density;
  (v) Thermal conductivity;
  (vi) Specific heat.
(b) Tensile strength properties:
  (i) Monotonic stress-strain curves;
  (ii) Minimum and average yield strength at 0.2% offset. ($S_{y,\,min}$, S_y);
  (iii) Minimum and average ultimate tensile strength: ($S_{u,\,min}$, S_u);
  (iv) Minimum and average uniform elongation: ($\varepsilon_{u,\,min}$, ε_u);
  (v) Minimum and average total elongation: ($\varepsilon_{t,\,min}$, ε_t);
  (vi) Minimum and average true strain at rupture ($\varepsilon_{tr,\,min}$, ε_{tr});
  (vii) Tensile creep curves;
  (viii) Minimum and average time to stress rupture ($t_{r,\,min}$, t_r);
  (ix) Minimum and average creep ductility: ($\varepsilon_{c,\,min}$, ε_c);
  (x) Minimum and average true strain at rupture for creep: ($\varepsilon_{ctr,\,min}$, ε_{ctr}).
(c) Curves for tests on creep or swelling:
  (i) Negligible thermal creep curve (t_c);
  (ii) Negligible swelling curve (ϕt_s);
  (iii) Negligible irradiation-creep curve (ϕt_{c2});
  (iv) Curves for the test to determine if the nonlinear analysis is needed (ϕt_{c1} and ϕt_{s1}).
(d) Allowable stress intensity values, which values are calculated through mathematical formulae in function of the mechanical properties variation by the temperature and neutron flux:
  (i) S_m, S_e, S_d, S_t, $K_{eff,rect.}$
(e) Fatigue curves for unirradiated and irradiated materials.
(f) Cyclic stress-strain curves. Values of K_ε and K_υ;
(g) Fracture toughness;
(h) Isochronous stress -strain curves;
(i) Determination of the swelling law.

As example, Fig. II–14 [II–37] shows the variation in the S_m values as a function of the temperature according to SDC-IC and RCC-MRx codes. From this figure a structural designer can obtain the allowable S_m value for each temperature and then, introduce this value in the criteria (design rule) of a code to prevent the failure.

Nowadays, the material design limit data of structural materials for fusion is lack. Most of the data is unirradiated condition. A summary of the materials limits data status is described in the following section.

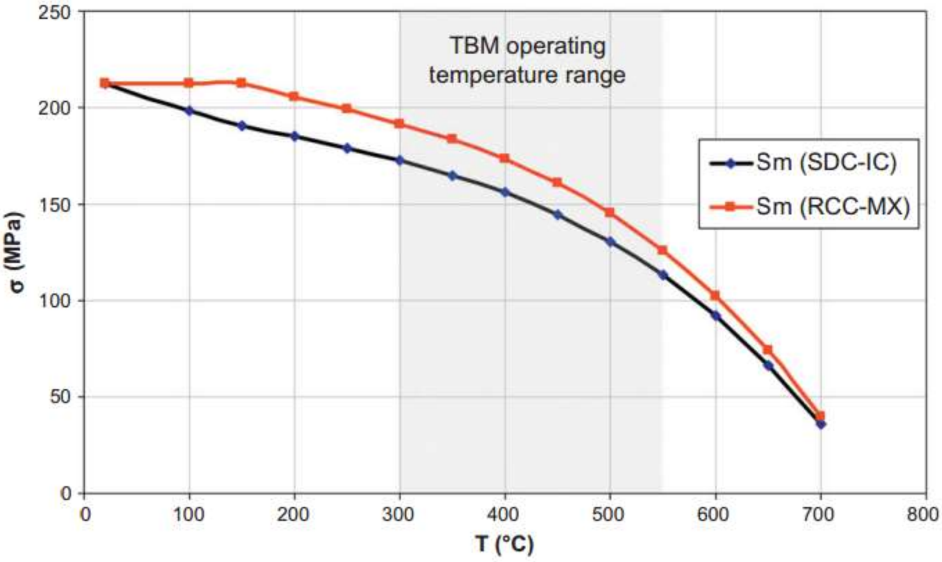


FIG. II–14. Values of S_m calculated according to SDC-IC and RCC-MX for Eurofer97 [II–37].

II–6. STATUS OF DESIGN-RULES AND MATERIALS DESIGN LIMITS DATA

During 2011, a depth analysis under the framework of European Fusion Development Agreement (EFDA) was performed on industrial code and standards. The objective was to identify the gaps concerning to design rules and materials limits data for DEMO components [II–52]. The analysis was carried out in the context of DEMO: Water Cooled Divertor (WCD), Helium Cooled Divertor (HCD) and Helium Cooled Blanket (HCB). Three codes were analysed: ASME-BPVC, RCC-MRx and SDC-IC.

For each of the primary materials (copper alloys, EUROFER, tungsten and tungsten alloys), the body of design criteria (ASME-BPVC, RCC-MRx, SDC-IC) was examined to assess their treatment in the code. Gaps in design allowable and design criteria for each of the failure modes were sought. Where design criteria were not available, the authors considered whether the existing design criteria could be applied to the materials of interest. The treatment of relevant joining techniques within each design criteria was also examined. The materials and joining techniques considered for gap analysis are summarized in Table II–2 [II–52].

TABLE II–2. MATERIALS AND JOINING TECHNIQUES CONSIDERED IN THE GAP ANALYSIS

Component	WCD	HCD	HCB
Materials (Temperatures)	W*, 800–1300°C CuCrZr, 150–350ºC CuAl25, 150–400ºC Stainless steel, 50–650°C (ODS steel) (Vanadium alloys) (Zr-alloys) (Al-alloys) (W-Cu composites)	EUROFER97, 350–550°C, Tungsten / Tungsten-ODS, 750–1200°C (EUROFER-ODS / Ferritic ODS, 450–750°C)	EUROFER97, 350 – 550°C.
Joining Techniques	Welding - Electron beam - Tungsten inert gas - Laser and hybrid laser MAG Brazing Hot Isostatic Pressing, diffusion bonding Plasma spraying (Braze welding, direct casting, active metal casting) (Pulsed electric current sintering) (High velocity oxyfuel spraying, detonation – D gun) (Powder injection moulding)	Diffusion bonding, Welding: - Electron beam; - Tungsten inert gas; - Laser.	Diffusion bonding, Welding: - Electron beam; - Tungsten inert gas; - Laser.

The analysis of the three codes ASME, RCC-MRx and SDC-IC revealed that none of the three codes respond completely to the needs (materials data, design rules, joining techniques, etc) identified for the DEMO blanket modules and divertor [II–52].

As was mentioned before, the analysis was focused essentially on materials and design rules. The three codes partially cover the grades preliminary identified for divertor and blanket module. In the following Tables II–3–II–7 are summarized the main key points missing in the codes examined.

* Consideration only as an armor material.

TABLE II–3. STRUCTURAL MATERIALS CONSIDERED FOR DEMO AND THEIR COVERED IN THE CODES

Materials	ASME-BPVC	RCC-MRx	SDC-IC
CuCrZr, CuAl25	Not included	Not included	Partially Covered
Eurofer	Not included	Partially covered	Not included
ODS steels	Not included	Not included	Not included
V-Alloys	Not included	Not included	Not included
Zr-Alloys	Not included	Few grades included	Not included
Al-Alloys	Not included	Few grades included	Not included
W and W-alloys	Not included	Not included	Partially covered
W-Cu composites	Not included	Not included	Not included

TABLE II–4. DAMAGE MODES NOT COVERED IN THE CODES EXAMINED

Damage Mechanisms	ASME-BPVC	RCC-MRx	SDC-IC
Immediate local fracture due to exhaustion of ductility	Not included		Partially Covered
Immediate plastic flow localization	Not included		Partially Covered
Ratchetting	Rules exist but not applicable to cyclic softening or non-ductile materials		
Creep-fatigue	Rules exist but not applicable to cyclic softening or non-ductile materials		

TABLE II–5. MODIFYING EFFECTS FOR DEMO NOT COVERED IN THE CODES EXAMINED

Modifying Effects	**ASME-BPVC**	**RCC-MRx**	**SDC-IC**
Irradiation induced hardening	Not included	Partially covered	
Irradiation induced embrittlement	Partially covered		
Irradiation induced swelling	Not included	Not included	Not included
Environmental effects (e.g. corrosion, erosion)	Only recommendations for corrosion		
Impure helium on fatigue and creep-fatigue	Not included	Not included	Not included
Stress-corrosion cracking	Only recommendations		Not included

TABLE II–6. JOINING TECHNIQUES FOR DEMO NOT COVERED IN THE CODES EXAMINED

Joining Techniques	**ASME-BPVC**	**RCC-MRx**	**SDC-IC**
Irradiation induced hardening	Not included	Partially covered	
Irradiation induced embrittlement	Partially covered		
Irradiation induced swelling	Not included	Not included	Not included
Environmental effects (e.g. corrosion, erosion)	Only recommendations for corrosion		
Impure helium on fatigue and creep-fatigue	Not included	Not included	Not included
Stress-corrosion cracking	Only recommendations		Not included

TABLE II–7. JOINING TECHNIQUES FOR DEMO NOT COVERED IN THE CODES EXAMINED

Joining Techniques	ASME-BPVC	RCC-MRx	SDC-IC
Welding (EB, TIG, laser)	Not examined	Partially covered	Partially covered
Brazing		Partially covered	Partially covered
HIP and Diffusion Bonding		Partially covered	Not included
Plasma spraying		Not included	Not included
Active metal casting		Not included	Not included
Pulsed electric current sintering		Not included	Not included
High velocity oxyfuel spraying, detonation – D gun		Not included	Not included
Powder injection moulding		Not included	Not included

In EUROFER, the design rules contained in the actual codes can be applicable [II–52], but the degree of applicability or conservatism need to be confirmed by more experimental tests. Especially, design rules concerning to immediate local fracture due to exhaustion of ductility, immediate plastic flow localization, ratchetting, creep-fatigue and irradiation damage.

The analysis of the data (physical and mechanical properties) for the EUROFER are collected on Demo Interim Structural Design Criteria (DISDC), Appendix A, Material Design Limit Data [II–54], reveal also that the mechanical properties of EUROFER need to be further investigated because a great part of the recommended materials properties data (Section 4.2) are missing.

II–7. KEY FACTOR

One of the principal technological barriers to design structural components for fusion applications are the structural materials because its behaviour under thermomechanical loadings and high energy neutron irradiation is partially known.

The main issues identified for structural materials concerning mechanical properties and design rules are:

(a) Limited strength at high temperatures of the RAFM steels development at industrial scale up to now. They show a drop in tensile strength and creep strength at temperatures ≥ 550ºC. In addition, RAFM steels experiment softening during cyclic loading, which may lead to maximum allowable loads much smaller than the limits predicted by the current design rules.

(b) Hardening and embrittlement effects for RAMF steels at temperatures < 400ºC.
(c) Fabrication techniques, joining, not well characterize and qualified.
(d) Limited database (physical and mechanical) on both irradiated and un-irradiated conditions.
(e) Weak interaction between materials scientists and designers.
(f) Weak interaction between designers and regulators.
(g) Slow development of the new generation of RAFM/ODS/FS steels for low and high temperature applications due to a poor involvement of the steel industry.
(h) The development of W-based alloys for divertor applications is at the beginning.

II–8. SUMMARY

This annex gives an overview of the key mechanical properties of structural materials for component design. A comprehensive description of them has been presented, including the status of design rules and materials limit data. As result of the mechanical properties revision, identification of the key factors has also been described.

Nowadays, the most investigated structural materials are the RAMF steels. They show a drop in tensile strength and creep strength at temperatures ≥ 550ºC. In addition, RAFM steels experience softening during cyclic loading, which may lead to maximum allowable loads much smaller than the limits predicted by the current design rules. Consequently, their mechanical properties need to be investigated by further experimental tests, especially on irradiated state, to complete the materials limit database, to check the applicability of the design rules in the present-day codes. For this purpose, many areas of research remain in the qualification of structural materials and components, and a detailed summary of the characteristics of many research reactors and devices that can accomplish an important portion of these necessary studies can be found in Ref. [II–55].

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GLOSSARY

accident. (a) Any unintended event, including operating errors, equipment failures and other mishaps, the consequences or potential consequences of which are not negligible from the point of view of protection or safety.

(b) Postulated event sequences or conditions not likely to occur during the life of the plant.

accident conditions. Deviations from normal operation that are less frequent and more severe than anticipated operational occurrences.

For information: Accident conditions comprise design basis accidents and design extension conditions.

ageing. General process in which characteristics of a structure, system or component gradually change with time or use

anticipated operational occurrence. An operational process deviating from normal operation which is expected to occur at least once during the operating lifetime of a facility but which, in view of appropriate design provisions, does not cause any significant damage to items important to safety or lead to accident conditions.

cliff edge effect. In a nuclear power plant, an instance of severely abnormal plant behaviour caused by an abrupt transition from one plant status to another following a small deviation in a plant parameter, and thus a sudden large variation in plant conditions in response to a small variation in an input.

confinement. Prevention or control of releases of radioactive material to the environment in operation or in accidents.

controlled state. Plant state, following an anticipated operational occurrence or accident conditions, in which the fundamental safety functions can be ensured and which can be maintained for a time sufficient to effect provisions to reach a safe state.

defence in depth. A hierarchical deployment of different levels of diverse equipment and procedures to prevent the escalation of anticipated operational occurrences and to maintain the effectiveness of physical barriers placed between a radiation source or radioactive material and workers, members of the public or the environment, in operational states and, for some barriers, in accident conditions.

design basis. The range of conditions and events taken explicitly into account in the design of a facility, according to established criteria, such that the facility can withstand them without exceeding authorized limits by the planned operation of safety systems.

design basis accidents. A postulated accident leading to accident conditions for which a facility is designed in accordance with established design criteria and conservative methodology, and for which releases of radioactive material are kept within acceptable limits.

design basis external events.	The external event(s) or combination(s) of external events considered in the design basis of all or any part of a facility.
design extension conditions.	Postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process for the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits.
external event.	Events unconnected with the operation of a facility or the conduct of an activity that could influence the safety of the facility or activity.
incidents.	Deviations from normal operation, comprising event sequences or plant conditions not planned but likely to occur due to failures one or more times during the life of the plant, but not including Normal Operation.
hardened safety core.	It indicates a limited number of material, organisational, human systems providing essential safety functions even in extreme circumstances, i.e. exceeding those adopted for the general design of the facility.
initiating event.	An identified event that leads to anticipated operational occurrences or accident conditions.
master logic diagram.	One of the earliest techniques used for the functional analysis of fusion machine. It provides an easy-to-read representation of the relations among functions (i.e. functional breakdown), among SSCs (i.e. physical breakdown), and among functions and SSCs (i.e. functions allocation). MLDs are useful input for the functional failure assessment, as they support the assessment of the consequences of failures from a functional perspective.
safe state.	Plant state, following an anticipated operational occurrence or accident conditions, in which the reactor is subcritical and the fundamental safety functions can be ensured and maintained stable for a long time.
safety architecture.	The set of provisions set in the design: • To ensure that the mission is carried out while the required safety objectives are achieved, i.e. significant parameters are maintained within the allowable operational limits; • To prevent the degradation of the facility, i.e. to avoid that operational limits are exceeded; • To bring to and to maintain the facility in a safe state, in case of failures.
safety feature for design extension conditions.	Item designed to perform a safety function or which has a safety function in design extension conditions.
semiparametric skew-symmetric shape model	A model capable of capturing inherent variability of shapes provided the realization contours remain within a certain neighbourhood range around a “mean” with high probability.
stress corrosion cracking.	Cracking of a metal or alloy by the combined action of (tensile) stress and a corrosive environment. The tensile stress may be induced by external loads or could also be the residual stress from metal working processes such as machining or welding.

structures, systems and components.	A general term encompassing all the elements (items) of a facility or activity which contribute to protection and safety, except human factors.
thermal mechanical fatigue.	It is caused by combined thermal and mechanical loading where both the stresses and temperatures vary with time. This type of loading can be more damaging compared with isothermal fatigue at constant operating temperature.

LIST OF ABBREVIATIONS

AOO	anticipated operational occurrence
ALARA	as low as reasonable achievable
BDBA	beyond design basis accident
DiD	defence in depth
DISDC	demo interim structural design criteria
DBA	design basis accidents
DEC	design extension conditions
DBTT	ductile to brittle transition temperature
EPFM	elastic-plastic fracture Mechanism
EC&I	electrical controls and instrumentation
EML	electromagnetic loads
ETA	event tree analysis
FMEA	failure mode and effects analysis
FMECA	failure mode effect and criticality analysis
FTA	failure tree analysis
FEM	finite element method
FTTT	fracture toughness transition temperature
FBS	functional breakdown structure
HCC	hard core component
HAZOP	hazard and operability
HCB	helium cooled blanket

HCD	helium cooled divertor
HCF	high cycle fatigue
I&C	instrumentation and control
LEFM	linear-elastic fracture mechanism
LOP	line of protection
LCF	low cycle fatigue
MLD	master logic diagram
NDE	non-destructive examination
NDT	non-destructive test
OPT	objective provision tree
ODS-FS	oxide dispersion strengthened ferritic steels
PBS	plant breakdown structure
PIE	postulated initiating event
PSA	probabilistic safety assessment
QA	quality assurance
RAFM	reduced activation ferritic/martensitic
RIS	radiation induced segregation
RH	remote handling
RPV	reactor pressure vessels
SIC	safety important components
SCC	stress corrosion cracking
SF	safety function

SM	safety margin
SQUG	seismic qualification utilities group
SSC	structures, systems and component
SSSM	semiparametric skew-symmetric shape model
TBM	test blanket modules
TMF	thermal mechanical fatigue
UHV	ultra-high vacuum
WCD	water cooled divertor
WENRA	western European nuclear regulators association

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