DEMO divertor cassette body:design procedure with RCC-MRx code

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DEMO DIVERTOR CASSETTE BODY: DESIGN PROCEDURE WITH RCC-MRx CODE

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INDEX

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Abstract

With the construction of ITER well underway, attention is now posed to the design of a successor device: a future fusion energy generation Demonstration Fusion Power Plant (DEMO). DEMO will be the nearest-term fusion reactor design capable of producing net electricity power of 500 MW, operating with a closed fuel-cycle and being a single step between ITER and a commercial reactor foreseen by 2050 (1, 2, 3).
For the main ITER in-vessel components (e.g. divertor, shielding blanket, port plug, etc.) the austenitic stainless steel AISI 316 L(N) IG has been used as structural material. Good experience has been done using design code for nuclear components like SDC-IC or RCC-MR. When the nuclear damage increases, as in

ITER TBM (Test Blanket Module) or in Demo in-vessel components, it is not possible to use AISI 316 but material with low activation properties and good resistance to neutron damage as Eurofer mast be foreseen. To this scope it is under development a design code that will be used for structural design under nuclear regulation. A first tentative is the French code RCC-MRx.
The principal objective of the present thesis will be to study both the mechanical behavior of all possible structural materials that can be used for in-vessel components (like divertor cassette) and existing design codes to prevent the failure of such components with respect to possible mechanical damage. In particular focus will be putted on Eurofer steel and RCC-MRx code. For this objective, all the significant loads and relative combinations for the Divertor Cassette Body will be identified and accounted for verifying the structural continuity and integrity.

Keywords: Fusion reactor, DEMO, divertor cassette body, design rules, Eurofer, AISI 316, RCC-MRx code

1. Introduction

Demo fusion reactor has to demonstrate the technological feasibility for civil electrical power generation at commercial power scale by 2050. The development of a divertor concept for post ITER fusion power plants is deemed to be an urgent task (4). The divertor is one of the high heat flux components of the fusion reactor. Nearly 15 % of the total fusion thermal power has to be removed by the divertor which has to withstand a high surface heat flux (up to 15 MWm^{-2}). Divertor developing involves many uncertainties including physical boundaries and candidate materials.

The principal objectives of the present thesis are:

- describe DEMO plant and particularly its divertor cassette body;
- identify all the significant loads (and their combinations) to be accounted for verifying the structural integrity of the divertor cassette body. The analysis has been determined mainly from ITER information related to the same topics;
- investigate on possible structural materials for in-vessel components;
- describe mechanical properties of Eurofer97, with or without neutron irradiation, as principal candidate for in-vessel structural material;
- compare mechanical properties of Eurofer97 with other structural material (e.g. AISI 316);
- describe all design codes for nuclear components and particularly RCC-MRx;
- apply RCC-MRx design rules to the DEMO divertor cassette body.

2. DEMO power plant

DEMO is the last step before commercialization of fusion (fig. 1). It will begin the transition from science and technology research facilities to a field-operated commercial system.
Demo must demonstrate that fusion is affordable, reliable, profitable, and meets public acceptance. Demo must also convince public and government agencies that fusion is secure, safe, has a low environmental impact, and does not deplete limited natural resources. In particular DEMO will (1):

- demonstrate tritium self-sufficiency through efficient breeding and extraction technologies;
- demonstrate the intrinsic safety of a fusion power plant through regulatory licensing;
- develop nuclear design codes and standards as necessary mechanisms to aid the demonstration of compliance of fusion nuclear systems to regulatory requirements;
- demonstrate the potential to achieve power-plant relevant levels of system reliability, availability and maintainability;
- reveal the potential for a competitive cost of electricity from fusion power;

- engage and secure industrial involvement throughout the DEMO Power Plant design, R&D, construction and operation programme.

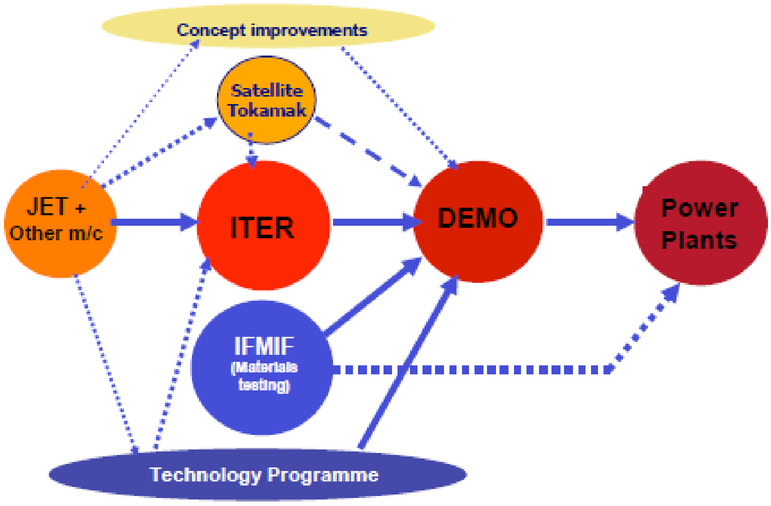


Fig. 1 Roadmap to fusion power

Two different DEMO design options are currently investigated (3):

– a near-term DEMO (DEMO1) is a rather "conservative baseline design", i.e. a DEMO concept deliverable in the short to medium term (e.g., construction possibly starting ~20 years from now), based on the expected performance of ITER (Q = 10) with reasonable improvements in science and technology; i.e., a large, modest power density, long-pulse inductively supported plasma in a conventional plasma scenario. The Balance of Plant (BoP) design must also make use of mature and reliable technology;

– a more advanced DEMO design concept (DEMO2) based around more optimistic (but "less mature") physics assumptions, which are at the upper limit of what may be achieved in ITER phase 2, i.e., an advanced higher power density, high current drive, steady state plasma scenario. It is clear that this can only be delivered on a longer term (e.g. construction to be started on a much longer time scale assuming that the required significant advances in the physics basis will be demonstrated using ITER and the limited number of satellite fusion devices available in the next 10–20 years).

2.1 DEMO Plant Build-up

The DEMO plant breakdown structure is defined in (5).

2.1.1 Tokamak

The tokamak structure is composed of three main systems, each providing support to other systems:

1. The **vacuum vessel** (VV) is a torus-shaped double-walled pressure vessel. It provides the primary vacuum and shields the magnet system from neutrons. Being one of the two backbone structures of the tokamak it supports the in-vessel components and other systems:
   - Breeding blanket with first wall (FW). The actively cooled FW withstands the plasma heat radiation and additional heat loads due to impacting particles. The actively cooled breeder units contain lithium that is transmuted into tritium and helium-4 due to neutron radiation. The tritium is removed from the breeder unit in a closed loop and extracted from that loop by the tritium extraction system. The heat from both FW and breeder unit is exhausted by

the primary coolant and transferred to the secondary cooling loop via the heat exchanger. The DEMO blanket concept has not yet been chosen.

- Divertor. It is primarily a high heat flux component. The magnetic topology is chosen in a way that the field lines outside the last closed flux surface intersect the divertor targets, which collect most of the particles and energy exhausted by the plasma. In the intended operation scenario the divertor target is subject to the highest heat loads; the particle impact causes the most severe erosion in the tokamak.
- Plasma start-up and shut-down limiters. These are primarily high heat flux components and are part of the plasma-facing wall integrated in the assembly of blanket segments. They protrude however the blanket FW slightly (stick out). They intersect the scrape-off layer (SOL) in plasma scenarios where the plasma becomes in contact with the plasma-facing wall and hence avoid to a large extent direct particle impact onto the blanket FW.
- FW protection limiters. As above these are primarily high heat flux components and are part of the plasma-facing wall integrated in the assembly of blanket segments. They protrude however the blanket FW slightly (stick out). They protect the FW from high heat loads during flat top due to off-normal events.
- Auxiliary heating systems in the vessel port structures. These are ECRH, ICRH and Lower Hybrid systems whose front parts (launchers/antennas or ducts/liners) are integrated in the assembly of blanket segments. These systems face the plasma and radiate electromagnetic waves into the plasma transferring energy to certain particles (e.g. electrons for ECRH) at certain velocities but also provide additional functions to support e.g. plasma break-down, heating to H-mode, MHD-control, current drive and others. The electromagnetic waves (ECRH and ICRH) are delivered through wave guides from outside the tokamak building.
- Diagnostics. There is variety of different sensors that are installed on the vessel, on the in-vessel components, or in VV port structures. These measure plasma and magnetic field parameters mostly in the framework of the plasma control system.
- Fuelling system. It is located outside tokamak, forms pellets of frozen D, T, or D-T, accelerates these pellets, and guides them through a pipe up to the level of the FW. The pipe runs through a vessel port and penetrates the in-vessel components. Alternatively a gas fuelling system could be considered.
- Impurity seeding system: One or more types of impurity gases are injected in several locations in the divertor area and the main chamber.

2. The **magnet system** is an assembly of planar superconducting (SC) coils which provide the magnetic field required to break-down and confine the plasma, to drive its current and to define its poloidal structure. It is actively cooled by liquid helium at ~4K. Both electric current and liquid helium are supplied by ex-tokamak systems through magnet feeders to the SC coils. The TF coil assembly is the 2nd backbone structure of the tokamak (the vacuum vessel being the 1st) supporting the following systems:
   - PF coils. Six individual coils are mounted to the outboard leg of the TF coils providing the vertical and radial magnetic field. Whereas the TF coil current remains constant during plasma operation the currents in the PF coils are altered over the duration of the plasma pulse contributing to driving the plasma current but also to actively stabilize the plasma.
   - Central Solenoid (CS). The CS is a stack of coils in the bore of the torus. Before plasma operation it is charged. Its current is then partly discharged providing the voltage required to break-down the hydrogen gas into a plasma. Over the duration of the plasma pulse it is further discharged driving the plasma current until it reaches the opposite peak current, which marks the end of the plasma pulse.
   - Vacuum vessel thermal shield. It is a shell-like structure supported by the TF coils that encloses the vacuum vessel and its port structures. It is actively cooled by helium at ~80K and shields the cryogenic magnet system from radiation heat from the vessel.
   - Cryostat thermal shield (outboard part). It is designed and operated based on the same principles as the vacuum vessel thermal shield. It shields the magnet system from radiation

loads the cryostat and in-cryostat components. The top and bottom part of the cryostat thermal shield are supported off the cryostat, see below.

3. The **Cryostat** is a large, single-walled, passively cooled vacuum vessel at room temperature. It provides the vacuum required to operate the magnet system in cryogenic condition and supports the two backbone structures of the tokamak: the vacuum vessel and the TF coil system. It is itself supported by the tokamak building. The cryostat has a large removable top lid through which the tokamak is assembled and through which it could be maintained in exceptional situations. The cryostat has a large number of openings, in particular to allow access into the vessel ports but also to allow in-cryostat maintenance as well as penetration of cooling pipes and magnet feeders.

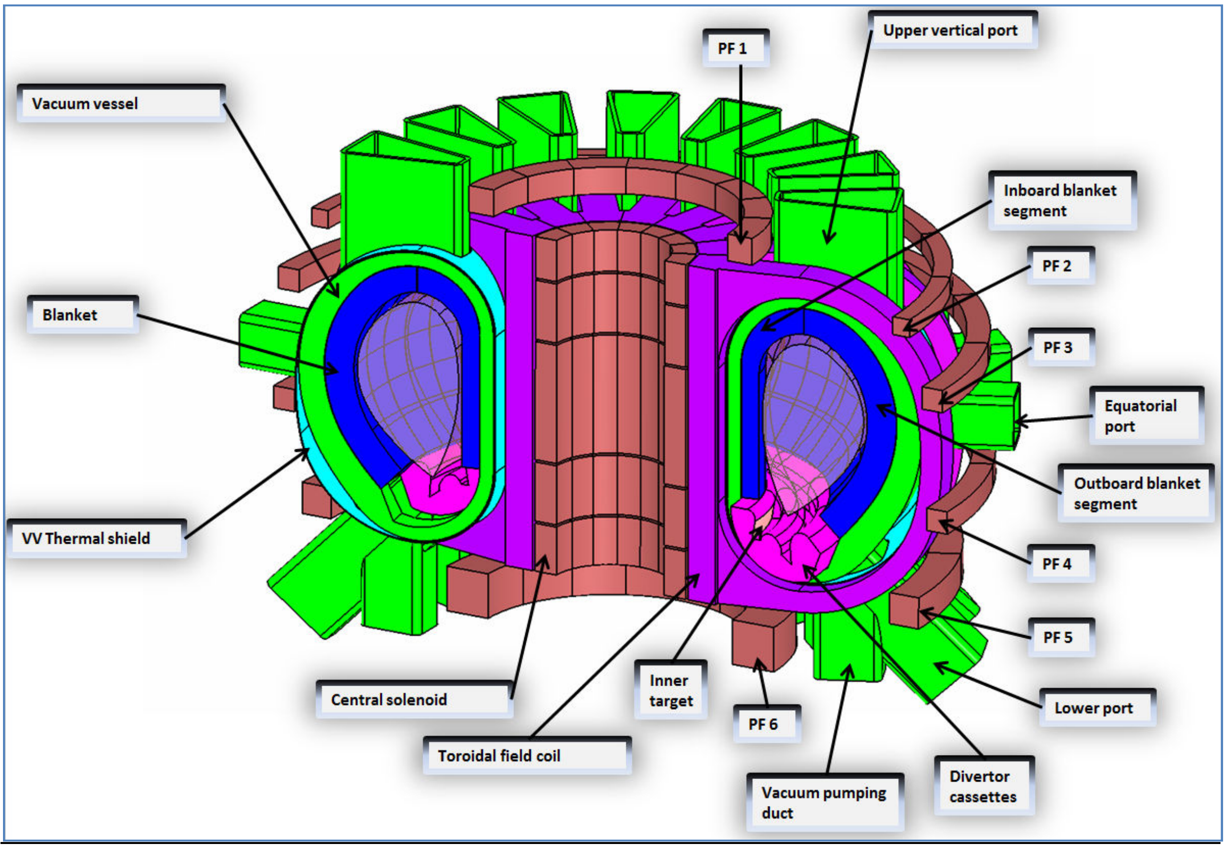


Figure 2) Main DEMO Tokamak Systems

2.1.2 Basic tokamak parameters for DEMO 2014

Parameter	Unit	Quantity
Plasma power	MW	1572
Thermal power including n-multiplication in blanket	MW	1972
Plant electricity output capability	MW	500
Lifetime neutron damage in steel in the FW	dpa	20+50
Major radius, R_0	m	9.0
Minor radius, a	m	2.25
Plasma current	MA	14
Toroidal field, B_0 at R_0	T	6.8
Elongation, κ_{95}		1.56
Triangularity, δ_{95}		0.33
Plasma volume	m^3	1453
Plasma surface area	m^2	1084
Auxiliary heating power, P_{inj}	MW	50
Auxiliary ramp-up power, $P_{ramp\text{-}up}$	MW	>60
Average neutron wall load	MW/m^2	1.067
Nuclear heating in blanket	MW	1380
Power to divertor	MW	180

Table 1- Basic Tokamak Parameters **(**2)

Component	Number	Weight [tons]	
		Single	Total
Cryostat			**15000**
TF coils	16	1000	**16000**
PF coils	6		**7960**
– PF1	1	680	
– PF2	1	680	
– PF3	1	1200	
– PF4	1	1200	
– PF5	1	2100	
– PF6	1	2100	
CS	5		**2800**
Vessel			**13680**
– Main vessel			6000
– Lower port extension and duct	16	80	1280
– Equatorial port extension and duct (standard/NB)	13/3	50/50	800
– Upper port extension and duct	16	100	1600
– In-wall shielding			4000
Plug structures			**1600**
– Upper port plug	16	50	800
– Equatorial port plug (standard/NB)	13/3	50	800
– Lower port plug	-		
Blanket			**5440**
– Inboard segment	32	50	1600
– Outboard segment	48	80	3840
Divertor	48	17	**816**

Table 2- Weight of Major Core Components (6)

2.1.3 DEMO design options

The DEMO aspect ratio has been defined preliminarily as R/a = 4. It is currently being studied in the range R/a = 2.6 - 3.6. Due to this range the following tokamak parameters vary as indicated below (6):

- Plasma cross-sectional size (and hence 2D size of tokamak), minor radius a = 3.4-2.5m
- Plasma current (17.7-24 MA)
- Total TF coil current (range: 190-320 MA)
- Peak field at TF conductor: 9.9-13T
- B_0 (range: 4.25-7.0T)
- Major radius (8.8-9.1m)
- Number of TF coils: (16, 18, or 20)
- Average neutron wall load (range: 0.91-1.22 MW/m²)

Given the early state of the concept phase a number of design configuration options are still open.

2.2 Major design choices

In the following table 3) the major design choices are reported (6):

Plasma operation	Pulsed operation, long-pulses (≥1 h) mainly inductively driven
Power	A plasma power of about 1.5 GW was chosen in order to generate 500 MW of net electricity.
Fuel	A T-breeding blanket is implemented in the design to achieve tritium self-sufficiency
Magnet technology	Low-temperature super conducting coils (Nb_3Sn/ NbTi) were chosen given their superior technology readiness and the low power consumption
Divertor configuration	A single-null divertor configuration was chosen
Divertor target	A water-cooled Cu-based target concept was chosen for DEMO given the superior power handling capability and superior technology readiness, developed primarily within the ITER programme.
In-vessel components structural material	Reduced activation ferritic-martensitic (RAFM) steel (EUROFER) was chosen as structural material for all internal components mainly to minimize radioactive waste.
Armour material	Tungsten was chosen as armour material for all plasma facing components due to its refractory properties and low erosion rate.
IVC lifetime	<u>Blanket:</u> 20 dpa is currently defined as irradiation damage limit in the front-wall steel of the starter blanket;50 dpa chosen as irradiation damage limit in the front-wall steel of the 2nd blanket. <u>Divertor:</u> 5 dpa is currently defined as irradiation damage limit in copper.
Vacuum vessel	A water-cooled ITER-like vessel was chosen for DEMO made of austenitic steel. Steel plates for neutron shielding are stacked into the inter-space between the two shells. ITER-like ports are integrated at the divertor and the equatorial level; large upper vertical ports extending up to the cryostat top lid are integrated at the upper level.
Remote maintenance strategy	<u>Divertor:</u> The divertor is segmented into 3 cassettes per lower port and maintained through the lower port using the ITER divertor remote maintenance approach. <u>Blanket:</u> The blanket is segmented into 3 outboard and 2 inboard segments per upper port. These segments can be removed and re-installed remotely through the upper ports, provided that the divertor cassettes are removed. Their weight is carried from the top. A mover, temporarily installed in the divertor area, supports the kinematic sequence and reacts the unbalance due to the weight being carried off the center of gravity.

Table 3- DEMO deign choices

2.3 Main differences between ITER and DEMO

The main differences between ITER and DEMO are summarised in table 4. A variety of fusion power plant system designs have been studied across the world. In many cases promising performances are unfortunately opposed by rather immature underlying technologies and optimistic physics assumptions. The recent EU fusion roadmap (7) advocates a pragmatic approach and considers a pulsed "low-extrapolation" DEMO to be constructed in the short to medium term. It shall be based on mature technologies and reliable regimes of operation to be, as much as possible, extrapolated from the ITER experience, and on the use of materials adequate for the expected level of neutron fluence. The mission requirements of such a near- to medium-term DEMO put more emphasis on solutions with high technical readiness levels and reliable performance and component reliability, rather than on high-efficiency.

	ITER	DEMO
Overall mission	Experimental device with physics and technology development missions.	Demonstrate capability for net electricity production
Pulse length	400 s pulses.	Long pulses (≥ 1h)
Availability	Experimental campaigns. Long dwell time, outages for maintenance, component replacements.	High availability.
Complexity	Diagnostics with large variety of purposes	Diagnostics primarily required for operation.
	NB, EC, and IC	EC, NB, IC (current design configuration). LH (alternative option)
Plasma	Large uncertainties regarding plasma performance and behaviour, large variety of plasma regimes.	With ITER (and other) experience smaller uncertainties regarding plasma performance and behaviour, concisely defined plasma regime.
IVC cooling	IVC cooling systems optimized for minimum stresses and sized for modest heat rejection.	IVC cooling systems optimized for heat conversion efficiency (high temperature operation).
Tritium	Tritium required for plasma operation provided from external sources	Tritium self-sufficient, only start-up supply required.
Materials	Copper-based FW for improved heat flux capability	Eurofer-based FW (due to irradiation damage, upper temperature limit (~400°C) and activation of copper).
	Conventional 316 stainless steel structure for in-vessel components.	Nuclear hardened, reduced activation steel for in-vessel components.
n-fluence	Very modest lifetime n-fluence, low dpa and He production.	High fluence, significant in-vessel materials damage.
Licensing	Licensed as nuclear facility, but like a laboratory, not a reactor.	Licensing as nuclear reactor more likely.
	"Progressive start-up" permits staged approach to licensing.	Similar "Progressive start-up" as in ITER. In addition initial operational phase with a "starter" blanket with moderate lifetime, then switch to blankets qualified for a longer lifetime.
	During design, licensing in any ITER party had to be possible.	Unknown.

Table 4 - Main differences between ITER and DEMO (3, 6)

3. DEMO divertor

3.1 Design description of Divertor System

The structure of the DEMO divertor derives from the one developed for ITER: as the divertor geometry takes into account the physical behaviour of plasma and the way in which all the loads are withstood, the geometry itself is the direct consequence of the way in which those loads really act.
In this way we can say that the divertor assembly installed in the VV consists of 48 divertor cassettes. Each divertor cassette consists of the cassette body (CB) and three PFCs, namely the Inner Vertical Target (IVT), Outer Vertical Target (OVT), and the Dome (DO).
Each CB is supported on the inner and outer region by a concentric structure similar to the ITER toroidal rails.

The PFCs are mounted on the CB. For cooling scheme two options are under investigation:
a) Only one cooling circuit with the PFCs cooled in series and the water coolant is routed via the CB.
b) Two cooling circuit, one for PFC and one for CB.

Each PFC consists of a number of PFUs, cooled in parallel and assembled onto a steel supporting structure. The CB and PFCs must be designed to withstand the various loads including coolant water pressure, electromagnetic (EM) and thermal loads, etc. The CB provides the neutron shielding for the VV and magnetic coils, and by incorporating internal cooling channels, acts as the manifold for the PFCs.
Every component of the divertor will have a geometry that shall be studied properly: the design will begin with ITER choices, but during the process of “detailed” evaluation of loads adjustments in geometry or in material specification can surely be proposed and provided.
The angle of targets with respect to the toroidal magnetic field lines must be chosen and optimized;
The position of the toroidal rails strongly affects the eddy current path and the value of preload: then also the cassette body attachments to the VV must be analysed and optimized as well.
The following figure 3 shows the CAD model of this first study of DEMO Divertor in ITER-like configuration (8).

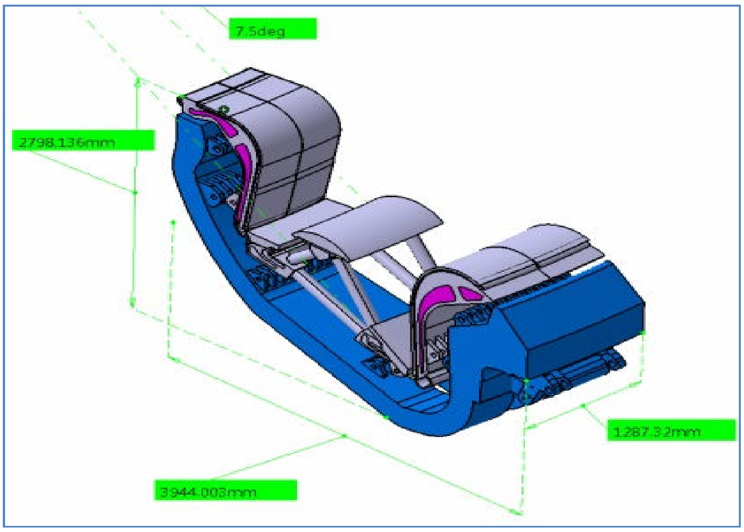


Fig. 3 Cad of Demo Divertor (8)

3.2 Loads classification

The loads applied on the divertor are (8):

•<u>Assembly loads:</u> these loads are applied to the components of the Divertor System during the assembly phase (e.g. radial preload of the divertor cassette body, preload of the bolts, interface loads transmitted to the divertor radial rails).
•<u>Pressure loads</u>: these loads include coolant pressure at operating conditions, coolant pressure at testing and baking regimes.
•<u>Inertial loads</u>: these are caused by accelerations due to gravity and seismic events.
•<u>Electromagnetic loads</u>: these loads are an important design issue during transient electro-magnetic events (plasma disruptions, VDEs and magnet current fast discharges).
•<u>Thermal loads</u>: these loads are surface loads induced by the heat fluxes from plasma and body loads caused by the neutron heat generation. They are a strong design driver and produce temperature gradients in the Divertor System components and also a temperature difference between interfacing components of Divertor System and Vacuum Vessel (VV).

The loads mentioned in this thesis always referred to a global Cylindrical Coordinate System in which X states for radial direction, Y states for toroidal (circumferential) direction and Z for the vertical one. But also the Global Cartesian Coordinate System centred in the machine centre will be adopted (Fig. 4).

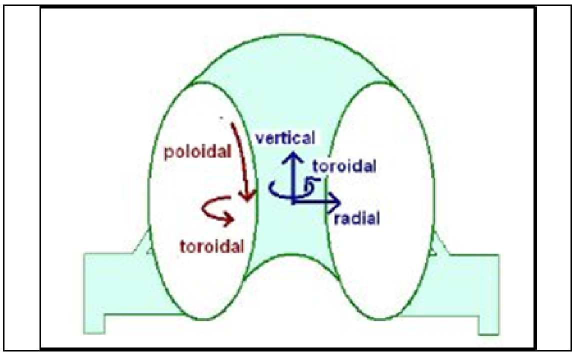


Fig. 4 Cartesian and cylindrical coordinate system

The analysis relative to the Divertor System shall regard the following physical environment:

-Thermal transient and steady-state analysis;
- EM transient analysis;
- Structural stress analysis for the individual single loads, including:
- Dynamic and static analysis at the EM loads and at seismic accelerations;
- Static analysis at thermal loads, at hydraulic pressure, at dead weight (DW), at preload of the CB and preload of bolts;
- Structural stress analysis at loads combinations;
- Evaluation of cyclic strength of the divertor components and parts.

The main purpose of the thermal transient analysis is the calculation of temperature fields in the different components of the Divertor System at normal operating conditions.

The main goal of the EM analysis is the calculation of induced currents and EM loads (including integral forces F and integral moments M on each divertor component) due to halo and eddy currents, taking into account both the thermal and current quenches.

The main purpose of the structural analysis for the individual loads is to evaluate the reaction forces at the constraints, the field of stress and strains, the field of displacements in the Divertor components.

The main purpose of the mechanical analysis at the load combinations is to obtain the results (mechanical reactions, stresses, strains and displacements in the parts of Divertor System) related to the combined influence of the individual single loads, defined at ITER various operational events/states.

The stress analysis for the EM loads is performed in static and dynamic approaches, allowing also the evaluation of Dynamic Amplification Factors (DAFs) for the stresses, strain and displacements. It is performed in linear and non-linear options (e.g. assuming the elastic or elastic-plastic behaviour of materials), depending on the requirements of the specified design criteria and availability of the stress categorization.
The evaluation of cyclic strength of the Divertor components allows the verification of the obtained stresses, strains and displacements, related to each load combination, against the specified design criteria.

3.3 Loads categories

This paragraph specifies the primary loading conditions which affect tokamak components and their load path through the mechanical connections; more it classifies the loads and their combinations into categories.

The categorization reported in the following table are the ones defined for ITER (9, 10) that is:
- Category I: Operational Loading Conditions
- Category II: Likely Loading Conditions
- Category III: Unlikely Loading Conditions
- Category IV: Extremely Unlikely Loading Conditions

Other events or events combinations that are beyond design basis may be classified in an additional category (category V). Unless specifically required by safety or project decision, the events in this category do not need to be considered for the design of the mechanical components, but they may be required for the secondary safety barrier.

The typical process of component design involves the following steps:
1) Identify and specify all conditions (e.g., Seismic, Gravity, Disruptions, etc.);
2) Identify and categorize condition combinations (e.g., Seismic + Disruption);
- Evaluate loads arising from each condition combination;
- Calculate the stresses due to each load combination;

3) Establish the acceptable damage limit for each condition combination (e.g., Normal, Upset, Emergency, or Faulted) also based on whether the component has any safety importance.
4) Establish the structural service criteria for each damage limit (e.g., ASME Level A, B, C, and D). The structural service criteria are defined in the design code selected for the component (e.g. RCC-MRx (13)).
5) Compare the evaluated stresses against the established criteria.
The meaning of different types of damage limits for components and plant are reported in the following table 5.

Condition	Component level damage limits	Plant Level damage limits and Recovery (Plant Operational Condition)
Normal/ Design/ test	The component should maintain specified service function.	Within specified operational limit. No special inspection will be required other than routine maintenance and minor adjustment. For test condition inspections are required as specified in the selected C&S
Upset	The component must withstand these loadings without significant damage requiring special inspection or repair.	After minor adjustment, or replacement of the faulty component, the plant can be brought back to normal operation. No effect on other components that may call for special inspection or repair.
Emergency	Large deformations in areas of structural discontinuity, such as at nozzles, which may necessitate removal of the component from service for inspection or repair. Insignificant general permanent deformation that may affect safety function of the component concerned. General strains should be within elastic limits. Active components should be functional at least after transient.	The plant may require decontamination, major replacement of damaged component or major repair work. In addition to the damaged component, inspection may reveal localized large deformation in other components, which may call for the repair of the affected components. Nevertheless, the plant maintains the specified minimum safety function during and after the events.
Faulted	Gross general deformations with some consequent loss of dimensional stability and damage requiring repair, which may require removal of component from service. Nevertheless deformation should not lead to structural collapse which could damage other components. The fluid boundary is maintained but degraded; however the safety function is maintained. The confinement of radioactive material is maintained. Active components may not be functional after transient.	Gross damage to the affected system or component. No loss of safety function which could lead to releases in excess of the guidelines established for Accidents. No design consideration will be given for recovery. The recovery of the plant may be judged from the severity of damage. This level of state is not expected to occur, but is postulated for safety assessment because its consequences would include the potential for the release of significant amounts of radioactive material.

Tab. 5 - Plant and Component Level Damage Limits

The following table 6 indicates the relationship between Load Combination Category (loads and likelihood categories) and acceptable damage limit as a function of the component safety class (SIC or non-SIC) (10).

Loading Category		Category I: Operational/ Design Loading	Category II: Likely Loading	Category III: Unlikely Loading	Category IV: Extremely Unlikely Loading	Test Loading
Plant Level		Normal	Normal	Emergency	Faulted	Normal/test
Component	SIC-1	Normal	Normal (4)	Emergency (3)	Faulted (1), (3)	Normal/test
	SIC-2 (5)	Normal	Normal (4)	Emergency (3)	Faulted (3)	Normal/test
	Non SIC	Normal	Upset (3)	Emergency (2), (3)	Faulted (2), (3)	Normal/test
Notes (1) Faulted for passive components with no deformation limits. Emergency for active SIC-1 components and some passive components in which general deformations should be limited. (2) Events need not to be considered from the safety point of view, but only for investment protection if required. (3) Damage limits can be made more stringent either for investment protection reasons or to reduce delay/cost of post incident inspections, for components where the design is sufficiently robust. Normal for some SIC-1						

active components and some passive ones which are required to function following Category III and IV accident (e.g. fire dampers and detritiation system) (4) Normal damage limits are assumed to have a robust design for SIC components not only for category I event, but also for category II. (5) Damage limits can be modified based on case by case considerations (systems classified as SIC-2 are not credited for bringing the plant into a safe status)

Tab. 6 - Damage Limits for Loading Conditions Categories

The safety class of DEMO divertor components are under discussion.

3.4 Load Combination

The same set of rules used for the ITER design (9,10) to establish event combinations and to classify them will be adopted here. A fundamental question is to establish the probability of one condition triggering another loading event. In fact, after an initiating condition, other additional conditions may occur.

- Conditions are here called "probable" to occur if their conditional probability is higher than 1%.
- Conditions are here called "conceivable" to occur if their conditional probability is smaller than 1% but still plausible with some physical basis.

The 1% value is chosen based on the normal practise applied in nuclear power plants where the ratio between the probabilities of occurrence of two subsequent categories is of the order10^{-2}.

Whenever lacking a more comprehensive probabilistic analysis, conditions are categorized as follows:

- **Category I**, for a combination of:

All Category I conditions when occurring at the same time or "Probable" to be triggered by the initiating condition.

- **Category II**, for a combination of:

The above Category I combinations with other Category I conditions also when they are "Conceivable" to be triggered by the initiating condition.
A Category II condition with other Category I and II conditions which are present or "Probable" to be triggered by the initiating condition.

- **Category III**, for a combination of:

- The above Category II combinations with other Category II or I conditions also when they are "Conceivable" to be triggered by the initiating condition.
- A Category III condition with other Category I, II, and III conditions which are present or "Probable" to be triggered by the initiating condition.

- **Category IV**, for a combination of:

- The above Category III combinations with other Category III, or II or I conditions also when they are "Conceivable" to be triggered by the initiating condition.
- A Category IV condition with other Category I, II, III, and IV conditions which are present or "Probable" to be triggered by the initiating condition.

A Category IV condition with any other event that is considered "Conceivable" to be triggered by the initiating condition is not considered in the design basis.

When multiple events are combined the overall likelihood of the combination has to be assessed to properly categorise it. The component designers have to identify the additional loads (gravity, pressure, thermal, etc.) that need to be considered for the related component.
In order to minimize the number of load combinations to be used for the purposes of the model and structural verification of the present design of the Divertor System, so-called "enveloped" load combinations are selected, based on:
- The recommendations used for ITER documentation, including the categorization of the loads combinations (10);
- The experience gained from previous supporting analysis of the Divertor Systems.

The “enveloped” load combinations, essential for the Divertor System, and their categories are given in the table 7. The following assumptions have been used for the data given in this table:
1) Single loads always to be considered for the Divertor System:
The following single loads are always to be included in the stress analysis of Divertor System at plasma operating conditions:
- Preload of the CB;
- Preload of the bolts;
- Internal hydraulic coolant pressure (CP);
- Dead weight (DW);
- Thermal loads (TH).
2) “Enveloped” load combinations to be considered for the Divertor System:
The “single” events have a certain specified number of occurrences.

	Single events in DEMO			Additional loads specific to divertor	Category	Number of events
	Seism	Plasma	Magnet			
1		Testing regime		DW, Hydr. Test pressure	Test	1
2		Baking regime		Preload, DW, CP, TH	I	Tbd
3		Normal operation		Preload, DW, CP, TH	I	Tbd
4		SDVDE-II		Preload, DW, CP, TH	II	Tbd
5		FDVDE-II		Preload, DW, CP, TH	II	Tbd
6		MD-I	MFD-II	Preload, DW, CP, TH	II	Tbd
7	SL-I		MFD-II	Preload, DW, CP, TH	II	1
8		SDVDE-III		Preload, DW, CP, TH	III	1
9		FDVDE-III		Preload, DW, CP, TH	III	1
10	SL-I	SDVDE-II	MFD-II	Preload, DW, CP, TH	III	1
11	SL-I	FDVDE-II	MFD-II	Preload, DW, CP, TH	III	1
12	SL-2			Preload, DW, CP, TH	IV	1
13		SDVDE-IV		Preload, DW, CP, TH	IV	1
14		Rotating Asymmetric VDEs		Preload, DW, CP, TH	IV	1
15		Baking regime		Preload, DW, CP, TH	IV	1
16		Maintenance		DW	IV	1

Tab. 7 - Enveloped Load combination for Divertor System and their categories (8)

3.5 Assembly loads

During the assembly of Divertor System, each divertor cassette is transported by Remote Handling (RH) equipment to a specified position inside of the VV. For the correct installation and for a proper functionality during operations, each divertor cassette is positioned in the proper position of the VV with a radial pre-compression load at room temperature (RT). The ITER value of such pre-compression is about 100 kN (internal compressive forces in the CB: fig. 5). For DEMO a first evaluation of this loads is 150÷200 kN with an internal displacement of (TBD).

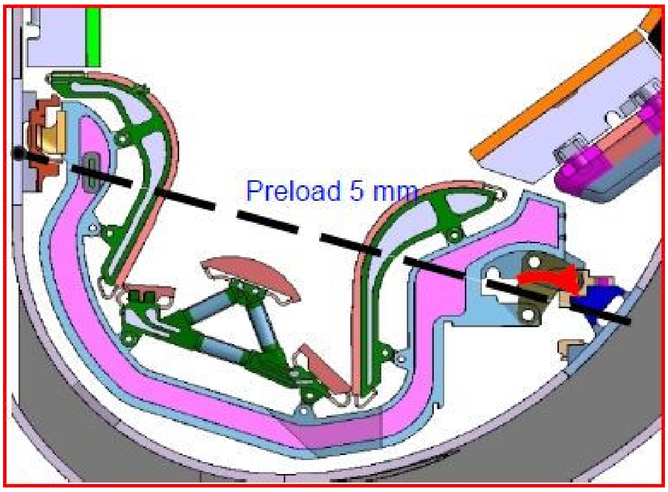


Fig. 5 - Divertor assembly pre-compression load

All the bolts joining the various Divertor components (e.g. bolts fixing the parts of the divertor rails to the VV, or those composing the attachments of the PFC to the CB) in accordance with terminology of RCC-MRx are classified as preloaded structural joints (8).

3.6 Pressure loads

The DEMO divertor cassette components are loaded by the following pressure loads:
1) Hydrostatic pressure test (see ITER procedure (9, 10));
2) Baking: can be done with water or gas (pressure and temperature TBD).
3) Normal operating conditions (by water pressure).

The possible water cooling layout is under investigation:
- Option A: – One single water cooling circuit for divertor cassette and PFC with inlet temperature about 150-200°C and pressure of about 5 MPa;
- option B:– Two water cooling circuit, one for divertor cassette (with PWR condition 300°C/15÷16 MPa) and one for PFCs (with inlet temperature about 150-200°C and pressure 5 MPa.

The inlet water temperature and water pressure have been preliminary deduced from the DBTT threshold and creep limit of expected materials to be chosen. After that, other choices must be taken, that is the pressure drops between the main PFC placed on the CB.

3.7 Inertial Load

3.7.1 Dead Weight

The total dead weight (DW) or mass of the whole 48 full-W divertor cassettes have been estimated (8) to be about 825 Ton.
The water content of (tbd) ton will be evaluated.

Component	Mass of metallic structure (kg)	Mass of contained water kg	Radial coordinate of CoG (mm)	Vertical coordinate of CoG (mm)
Cassette Body	9000	tbd	8266	-5483
Inner Vertical Target	1443	tbd	6663	-4196
Dome	2133	tbd	7792	-5260
Outer Vertical Target	3049	tbd	8984	-4909
Total	15625	tbd	8195	-5221

Tab. 8 - Mass and coordinates of centre of gravity of full W Divertor cassette components

3.7.2 Seismic loads

The ground seismic acceleration can be both in the horizontal and in the vertical direction and typically has a spectral content which leads to some level of support reaction load amplification.
In ITER three seismic events SL-1, SMHV and SL-2 are considered. The SL-1 and SL-2 are request from international standards as IAEA-SAFETY STANDARDS SERIES No. NS-G-1.6 SEISMIC DESIGN AND QUALIFICATION FOR NUCLEAR POWER PLANTS. SMHV is a request from the France regulator (8).
SL-1, and SL-2 are levels of ground motion regarding buildings and equipment that are classified in seismic class 1 and 2 (SC1 and SC2).

-SL-1 corresponds to an event with a probability in the order of 10-2 per year and represents an investment protection earthquake level (following the Nuclear Pressure Equipment regulation it corresponds to a foreseeable event). The facility has to be designed to restart and operate after an SL-1 event without special maintenance or test. As this event has a return period of more than 100 years it is expected to occur only once in the machine life. If required by investment protections and the sake of project risk reduction, SL-1 event shall be assumed to occur a maximum number of 5 times because, otherwise, no further operation may be allowed after its occurrence without the additional demonstration that the plan has not degraded the safety level of operation.

-SMHV (Séismes Maximaux Historiquement Vraisemblables = Maximum Historically Probable Earthquakes) is the most penalising earthquakes liable to occur over a period of about 1000 years (8).

-SL-2 corresponds to the seismic level required by French nuclear practise. In this event it shall be demonstrated that all safety functions are maintained. For ITER site the SL-2 (also called SSE – Safe Shutdown Earthquake) Design Response Spectra (DRS) is defined by two spectra: SMS and PALEO spectra.

Either types of spectra (or the envelope) need to be analysed.
If not combined with other loads, then:

- SL-1 event is classified as a Category-II event;
- SL-2 event is classified as a Category-IV event.

3.8 Electromagnetic loads

Maximum EM loads on the Divertor System components are caused by:

- Two kinds of abnormal terminations of plasma pulse: MDs and VDEs;
- Toroidal and poloidal magnetic field coils fast discharges: MFDs.

A MD consists of an abnormal termination of the plasma pulse consisting of two phases, the thermal quench with a fast loss of the plasma thermal energy and the current quench with a fast drop in plasma current often accompanied by a vertical drift and compression of the plasma core.

A VDE consists of an abnormal termination of the plasma pulse initiated by a failure of vertical position control, followed by an irreversible plasma vertical drift, a transition to limiter configuration, the thermal

quench, plasma current decay and by increased compression of the plasma core.
In both VDEs and MDs the halo current can reach very large fractions of the total plasma current once the plasma touches the wall. In both cases the final stages include a plasma drift to either upper or lower part of the VV.

A magnet fast discharge (MFD) is an event in which the current flowing in the superconducting Magnet System (TF, PF, CS, CC or all) is rapidly brought to zero by means of discharge resistors which dissipate the large stored magnetic energy. Such an event is usually triggered by the magnet quench detection systems which intentionally do so to protect the conductor from overheating.

Two types of MFD are defined:

- MFD-I which corresponds to a fast discharge of the CS and PF coils only;
- MFD-II which correspond to an event in which all the superconducting coil systems are discharged.

At least three kinds of plasma disruptions (MD, VDE and MFD) need to be simulated in DEMO to provide the EM and thermal loads. For the divertor design are important the Major Disruption (MD) and Downward Vertical Displacement Event (D-VDE).

In a VDE large vertical forces are exerted mainly on the Vacuum Vessel (VV) due to the vertical asymmetry of the event, then this event will be used to obtain the maximum vertical loads that the VV must withstand. The rather large upper vertical ports foreseen in DEMO are particularly concerning from the vertical stability point of view, due to the reduced stabilizing effect in this region, then the U-VDE simulation is a priority for the design of the VV. To accomplish this goal is planned to use the MAXFEA code, a reliable and long term used electromagnetic code

The fast VDEs are characterized by great eddy currents, while slow VDEs have great halo currents. In pure VDE less than 10% of the plasma thermal energy is dissipated on the whole FW during the slow, pre disruption TQ (τ1-2 of fig. 6) and the remainder of the energy is lost in the fast TQ (τ2 of fig. 6) when the plasma touch the divertor PFCs. In the fig. 6 is highlighted the time length of the delay of thermal quench for several well-known fusion machines.

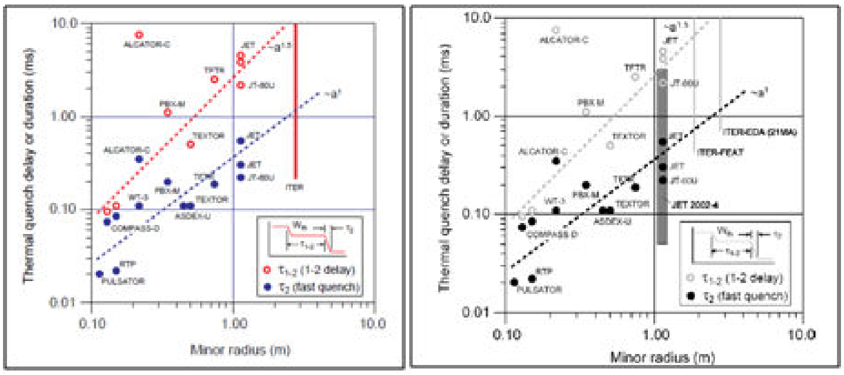


Fig. 6 -Thermal quench duration values for various fusion machine

3.9 Thermal loads

3.9.1 Nuclear loads

The nuclear heating induced by neutrons emitted from DT plasma and gamma rays could be critical in

particular in zones which are insufficiently cooled. The damage (dpa) induced by neutron irradiation might cause degradation of the physical and mechanical properties of the components and the helium production in the pipes that could impair their re-weldability.
Adequate shielding capabilities must be also demonstrated to avoid high nuclear loads on the vacuum vessel and superconducting magnets. Furthermore, neutron irradiation causes activation of divertor components that might impact on remote handling, waste and occupational dose strategy.

The nuclear heating, dpa and gas production can be calculated with MCNP5 Monte Carlo code using a 3-D neutronics model of DEMO. Nuclear loads depend on the machine configuration, fusion power, plasma source, irradiation scenario, materials as well as on the components design.
Preliminary estimations of divertor loads have been performed by extrapolating the neutronics results obtained from the 3-D detailed calculations on ITER- full W divertor carried-out in 2012, which has been used as reference data for the final design review.
The input conditions considered for a conservative scaling of the ITER results are as follows (8):

- DEMO scenario with Pfus 1572 MW ;
- DEMO neutronics calculations carried-out in 2013;
- ITER divertor neutronics results;
- Neutron wall loading distribution of ITER;
- In progress DEMO divertor cassette design.

Maximum damage in dpa, total nuclear heating on inner vertical target (IVT) , outer vertical target (OVT), dome and cassette body (CB) and power breakdown, as well as He-production in the pipes are reported.
In general the displacement per atom (dpa) quantifies the radiation damage and it may be related to the changes in the macroscopic properties of the materials such as mechanical, physical, thermal, electrical, etc. (i.e. radiation effects).
The estimation of the expected maximum dpa on DEMO divertor components are summarised in table 9 . These correspond to 1 full power year (FPY) operations at 1572 MW of fusion power and these have been obtained by scaling ITER results at FPY using the ratio of the neutron wall loading in divertor OVT. The ratio of the neutron wall loading for 1572 MW and ITER 500 MW on the divertor OVT peak can be conservatively assumed as 2.2.
The damage on W, Cu/CuCrZr and steel were calculated in ITER with MCNP5 considering the convolution of the damage cross section, neutron spectra and proper damage function with energy threshold of the different materials. In DEMO OVT 10 dpa on Cu can be reached in ~1 FPY operations, on Eurofer in ~1.6 FPY and on W in ~1.6 FPY.
Assuming 2 FPY operations the maximum level of dpa in plasma facing components (PFC) is ~20 dpa. For the cassette body the damage level is 10 times lower than in the PFC and it is ~1.2 dpa for 2 FPY operations. The four positions of the maximum dpa in IVT, OVT, DOME and CB are shown in figure 7.

Component	**W**	**Cu/CuCrZr**	**SS**
1) IVT PFU	2.1	9.7	5.2
2) OVT PFU	2.2	10.2	6.2
3) Dome Umbrella PFU	2.0	8.6	5.6
4) CB plate below Dome			0.6

Tab. 9 - Demo divertor components maximum dpa/FPY at 1572 MW

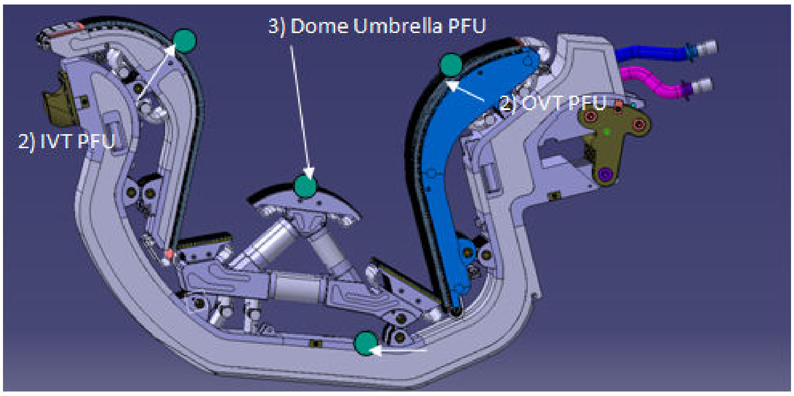


Fig. 7 Regions with highest dpa values in ITER Divertor

The nuclear heating (NH), i.e. the energy deposited by neutrons and secondary gamma rays, is important to correctly design the cooling system to limit the thermal stresses.
The extrapolated data from ITER for the nuclear heating on the various subcomponents of DEMO are reported in table 10. The estimation has been obtained by multiplying the ITER calculations on one cassette by volume (1.5) and NWL (2.2) correction factors, consistently with present design of DEMO cassette. Total deposited nuclear power is 161 MW on 48 cassettes (3.35 MW for 1 cassette).
Most of the power is deposited on OVT and IVT and DOME (including plasma reflector plates and pipes), only 17 % is the NH on the cassette body (27 MW). The present estimation is reasonable considering that the ratio of P_{div}/P_{fus} is 10.2 %, close to ITER value of 10.9% and consistent with a typical expected value on tokamak machines of ~10%.

Component	**1 cassette**	**48 cassettes**	**%**
IVT	0.81	39.0	24.3
OVT	1.16	55.7	34.7
Dome	0.81	39.0	24.3
CB	0.56	27.0	16.8
Total	**3.35**	**160.7**	**100**

Tab. 10 Total nuclear heating in Divertor (MW)

Another important quantity is the He production that might impact the material re-weldability. For ITER divertor the assumed design target of the He production was 1 appm.
From scaling Iter calculation it is expected that the He-production in Demo divertor pipe can reach the value of 2 appm in 2 FPY operations (exiting the limit of 1 appm for material re-weldability).

3.9.2 Surface heat loads

The surface heat loads are the other type of thermal loads and they are related to photonic radiation, neutral particles sputtering due to charge exchange and backscattering and heat fluxes parallel to magnetic field lines.

Assuming the total fusion power is Pfus=1600Mw we have 65 Mw as total radiation power to all divertor PFCs (8). A first subdivision of this power on different regions can be the following:

– 33% to the inner strike area;

– 67% to the outer strike area.

During burning plasma operation on ITER the full set of thermal load combinations is possible. They are equally probable and thus they are classified as Category-I loads.

The design surface heat fluxes on the divertor cassette components during baseline burning plasma operation are categorized according to the following time envelopes:

1) Design time–averaged surface heat fluxes (duration 2 h full power flat-top during normal operation (1):
tbd seconds ramp-up to the full power;
tbd s full power flat top;
tbd s ramp-down to zero power.

2) Design slow transient surface heat fluxes (duration tbd sec):
tbd sec for the IVT and OVT lower straight parts;
tbd sec for the IVT and OVT upper curved parts and for the components of the Dome.

The distribution of the design time-averaged front surface heat fluxes for full-W divertor cassette at normal operation is shown in Figure 8 for the ITER case.

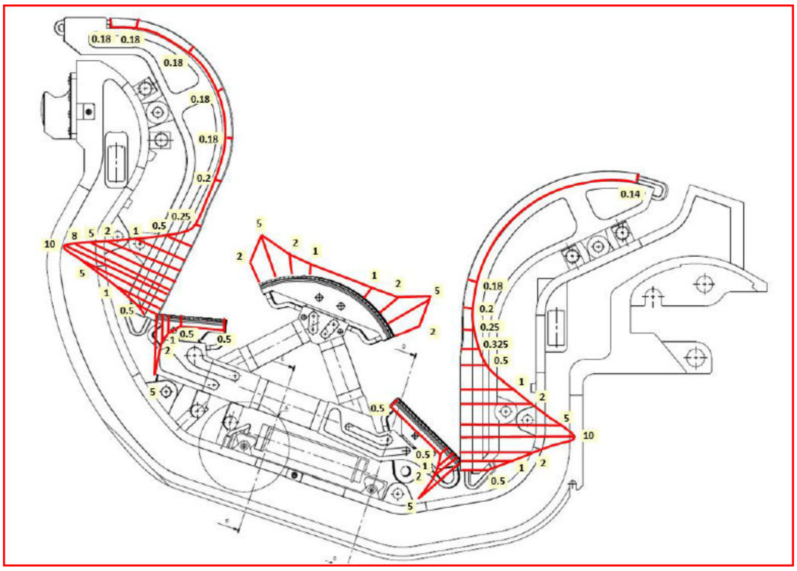


Fig. 8 – ITER design time-averaged surface heat flux (MW/m^2) on front surfaces of PFUs

The distribution of the design slow transient front surface heat fluxes for full-W divertor cassette must be evaluated and in the following figure 9 there is, for example, the distribution relative to that one of ITER.

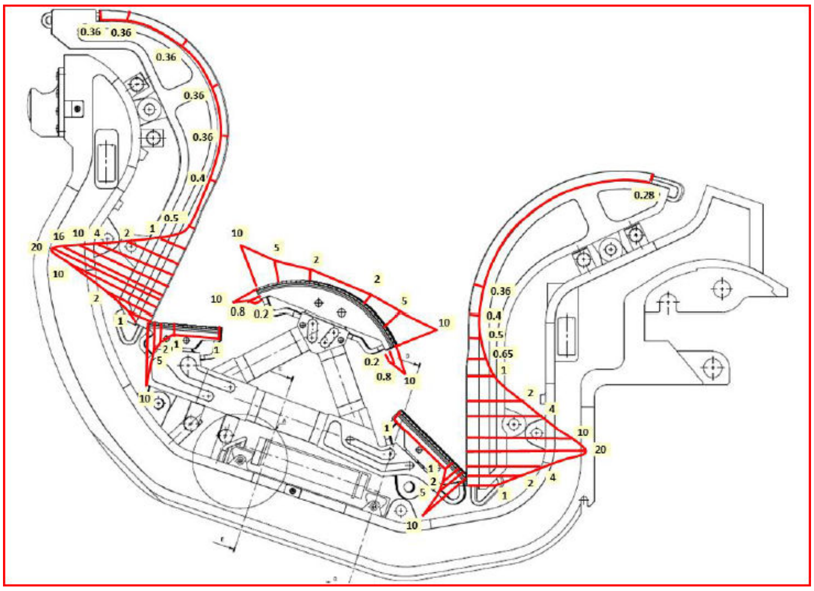


Fig. 9 – ITER design slow transient heat flux (MW/m2) on front surfaces of PFUs

3.10 Loads summary

Demo divertor cassette loads summary is given below.

Mechanical loads

1) dead Weight of whole divertor cassette (without water) is: 17,2 Ton

2) radial pre-compression load in the divertor CB at room temperature is ~150-200 kN with an internal displacement to be determinate (TBD);

   Till today no decision has been taken if the radial pre-compression load is necessary or not. If necessary the radial pre-compression load in the divertor CB at room temperature is ~150-200 kN with an internal radial displacement to be determinate (TBD);

3) maximum seismic accelerations: TBD

4) maximum modules of the integral EM loads at various EM events (11) are:

   -within |F| ~ 0.02 …1.1 MN for the integral EM forces applied to the whole divertor cassette;
   -within |M| ~ 0.06…1.0 MN·m for the integral EM moments applied to the whole divertor cassette;

5) hydraulic pressure:

   a) The maximum pressure during normal operating is or 5 MPa or 16 MPa (see par. 3.6);
   b) During testing regime: see RB 3000 of RCC-MRx code and par. 6.7;
   c) During baking regimes: TBD

Thermal loads

a) Surface heat fluxes:

The objectives of Demo divertor project are (12):

1. to develop the "near-term" water-cooled divertor target solution (i.e. ITER-like target) together with the required technologies for application under 'DEMO operation conditions' (steady state heat flux: >10 MW/m², neutron dose:<10 dpa). This solution shall be evaluated in terms of power exhaust capability;
2. to develop "advanced" water-cooled divertor PFC solutions for application in DEMO based on advanced materials (both for vertical target and dome) together with the required technologies. These solutions shall be evaluated in terms of power exhaust capability under 'DEMO operation conditions' (steady state heat flux: >10 MW/m², neutron dose:<10 dpa for the vertical target and steady state heat flux: 5 MW/m², neutron dose:<20 dpa for the dome target, respectively).

With a Pfus=1572 MW the total radiation power to all divertor PFCs is 65 MW (8).

b) Nuclear heating:

According to (4) the deposited nuclear heating (defined as the energy deposited by neutrons and secondary gamma rays) for one cassette in total is 3.35 MW with Pfus= 1572 MW. Only 17% of the total 3.35 MW is the nuclear heating on the cassette body.

With Pfus=1572 MW total nuclear heating on Divertor (48 cassette) is 160.7 MW (8) (see table 10 par. 3.9.1).

4. Design codes for nuclear components

The main reference C&S (Code and Standard) for a fusion in-vessel component are: RCC-MRx (13, 14), ASME III, SDC-IC (15, 16). These codes have rules that are in general very similar, but present some little differences.

4.1 Choice of a reference design code for DEMO divertor cassette

To choice a design code for DEMO divertor cassette several factors need to be considered:
– Safety classification;
– Investment protection;
– Regulation requirements: if Demo will be built in France, divertor cassette must fulfil French and European regulations on (nuclear) pressure vessel equipment.

Among the existing French and European regulatory documents, those identified as establishing specific requirements for the design and manufacture of the divertor cassette are:
– The Pressure Equipment Directive (PED – 97/23/EC) adopted by the European Parliament and European Council in May 1997 (17).
– The French Order dated 12th December 2005 concerning nuclear pressure equipment (Equipements Sous Pression Nucléaires – ESPN (18))
– The French Order concerning Quality of Design, Construction and Operation of Basis Nuclear Installations (19).

These above legislation define the principle of equipment classification; formulate the general Essential Safety Requirements (ESR) and the ways to fulfil these requirements from the legal (and sometimes technical) point of view.

According to ESPN, Nuclear Pressure Equipment (NPE) are classified into three levels, from N1 to N3, in relation to the significance of the radioactive emissions possibly resulting from their failure, and in 5 categories according to the pressure hazard: Category 0, I, II, III and IV.

The philosophy of the regulatory documents is to give the manufacturer of given equipment the choice of the design and manufacturing code for the component but he has to demonstrate the conformity of the selected Code and Standards (C&S) with the ESR. In many cases, the conformity shall be assessed by a special third party organisation, so called Notified Body.
It is important to note that even when a code contains specific provisions to meet the ESR, its use does not automatically assure conformity with regulations: other factors, like the materials used and the component's operating conditions need to be taken into account.

Design rules contained in the codes are intended to prevent failure resulting from specific damage modes, and the selected C&S shall cover all possible damage modes specific to the component (divertor cassette). Each code has a domain of applicability, and it is to the designer to verify that a given component falls within this domain.

In case of divertor cassette, the selected C&S should at least cover the following features:
– Guidelines for conformity with PED and ESPN ESR;
– Design rules for high temperature operation;
– Design rules for irradiated materials,
– Consistency of design rules with Eurofer97 mechanical properties data and divertor cassette specific manufacturing. Reference Procurement Specifications (RPS) providing the requirements for procurement of Eurofer97 parts and products shall also be included.

4.2 Available C&S for ITER in vessel components

A multi-code approach has been applied for the selection of C&S for ITER in vessel components (table 11). This is consistent with the approach used for ITER mechanical components (15).

C&S	Conformity with PED	Conformity with ESPN	High Temp. Design rules	Irradiated materials design rules	Eurofer97 properties	Comments
ASME	yes	No	yes	no	no	
SDC-IC	no	no	yes	yes	no	Scope limit to design rules
RCC-MR	yes	yes	yes	no	no	
RCC-MX	yes	yes	yes	yes	no	Developed for internal use by CEA only
RCC-MRx	yes	yes	yes	yes	possibly	

Table 11 Assessment of selected C&S for ITER TBMs design

For the design of in-vessel components, ITER recommended the use of the ITER Structural Design Criteria for in-vessel Components (SDC-IC) (15, 20). SDC-IC has been developed in order to have a specific code covering all the features of plasma-facing components. In particular, SDC-IC contains specific rules to take into account the effects of irradiation on structural materials not covered by the ASME BPVC. On the other hand, the scope of SDC-IC is limited to criteria related to design. Rules for manufacturing, quality assurance and examination are instead covered by ASME BPVC.
ASME has published a guide explaining how to meet ESR defined by PED using BPVC Section VIII, but no guidelines were given to meet conformity with the ESPN order (16).

Moreover Eurofer97 was included neither in ASME nor in SDC-IC material data sections.

For these reasons, in past EU TBMs design activities, RCC-MR (21) has been considered as the reference design code for the TBM assembly instead of ASME.

RCC-MR has been the reference French code for high temperature reactors and the 2007 edition contained a specific appendix (A18) to cover PED and ESPN requirements.

Since RCC-MR didn't cover measures to be taken to prevent damages resulting from irradiation, the use of SDC-IC was still necessary for TBMs. This approach however was still not satisfactory because:
– design rules addressing the effects of irradiation in SDC-IC have been developed mainly for the 316L(N)-IG steel grade on which the design of the ITER blanket is based (16). The mechanical properties of Eurofer97 are very different from those of austenitic steels. The validity/degree of conservatism of the rules defined in SDC-IC needs therefore to be verified for RAFM steels;
– Eurofer97 was included neither in RCC-MR nor in SDC-IC material data sections.

Concerning neutron-irradiation effects, a possible alternative to SDC-IC has been RCC-MX (22), an extension of the RCC-MR code developed for internal use by CEA. Indeed, the 2012 edition of RCC-MR called RCC-MRx (13) include RCC-MX 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.
Eurofer97 is included in the code, allowing the use of RCC-MRx as the sole code for the design and manufacturing of TBMs (16).

In fig. 10 is shown a schematic diagram illustrating the selection of the specific C&S for the ITER mechanical components (15). Mechanical components include vessels, piping, tanks, pumps, valves, heat exchanges etc. and supports.

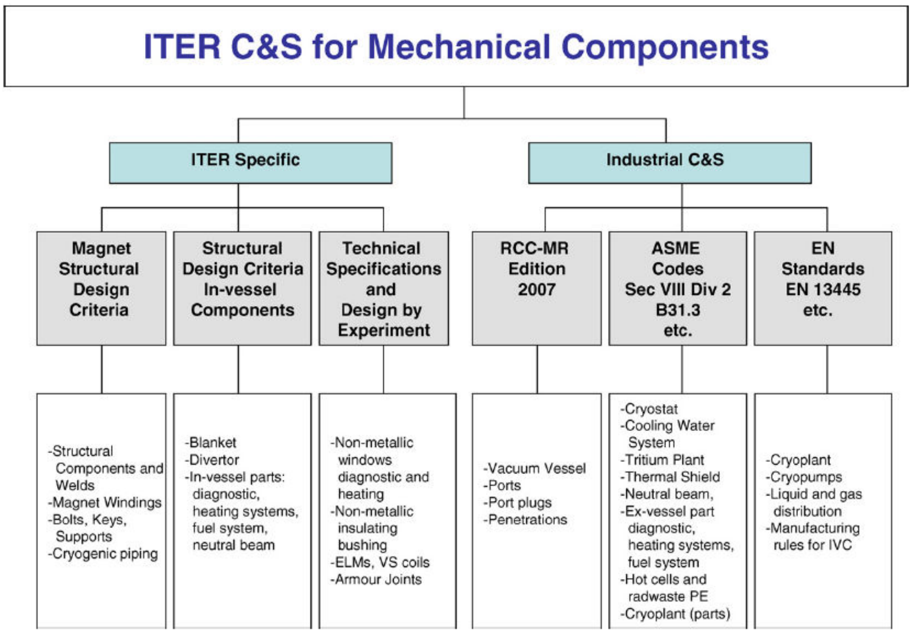


Fig 10 Schematic diagram of C&S for the ITER mechanical components

The following table 12 describes which materials are taken into account in ASME-BPVC, RCC-MRx and SDC-IC codes.

Materials	ASME-BPVC	RCC-MRx	SDC-IC
CuCrZr	Other Cu alloys covered (e.g.70Cu-30Ni) no irradiation data	Not covered	Partially covered
Eurofer97	Other 9Cr steels covered (e.g.9Cr-1Mo) no irradiation data	Covered in probatory rules (creep data limited) No irradiation data	Not covered
ODS steel	Not covered	Not covered	Not covered
W-Cu Laminates	Not covered	Not covered	Not covered
W&W-alloys	Not covered	Not covered	Partially covered
Al-alloys	Several grades covered (e.g.95154) no irradiation data	Several grades covered (e.g.6061-T6) no irradiation data	Not covered
Zr-alloys	Several grades covered (e.g. UNS R60702) no irradiation data	Several grades covered (e.g.Zircaloy 2) FM data limited	Not covered

Table 12 - Codes and material allowable (23)

4.3 Structural Design Criteria for In-vessel Components (SDC-IC)

The ITER Structural Design Criteria for In-vessel Components (SDC-IC) (15) contains rules for structural design of the blanket, divertor and in-vessel parts of various ITER systems (Fig. 10).

The SDC-IC are based on the RCC-MR code and extensive modifications have been provided to include unique structure of ITER components, useful features of other codes (in particular ASME), and national requirements to address the unique features of these components.
These criteria were developed because existing industrial codes are not applicable to the ITER in-vessel components. The structure and the environment of the in-vessel components of ITER have a number of unique features. In particular, the neutron irradiation has a number of effects on material properties, including:
– Time dependent material properties;
– Embrittlement of the material (reduced ductility and fracture toughness),
– Irradiation-induced creep;
– Swelling (at specific conditions).

Moreover the geometry of the various components is non-axisymmetric (for example the modules and the arrangement of the cooling channels is irregular, thermal and electromagnetic loads are applied in various directions).

SDC-IC consists of the main Design Criteria document and three appendices. The main document includes definitions and classifications of different damage and failure modes, type of stresses, joints, thermal creep phenomena, buckling, etc.
The other parts of document include design rules for general single layer homogeneous structures at low and elevated temperatures, rules for welded joints and rules for bolts. Design rules for multilayer heterogeneous structures are also included, but they are limited to only low temperature application. Rules for irradiated materials are described. For materials with reduced ductility, secondary and peak stresses become more important. The SDC-IC includes direct limits on secondary and peak stresses that account for both the stress and strain limits of the material. SDC-IC incorporates rules to prevent failure by plastic flow localization, to limit the elastically calculated primary plus secondary membrane stress intensity. In a similar fashion, to prevent local failure due to exhaustion of ductility, the primary membrane plus bending plus secondary stress is limited. Both limits depend on uniform elongation of material after irradiation.
Appendix A, Materials design limit data, currently includes the data for the following materials–austenitic

steel 316L(N)-IG (ITER Grade), CuCrZr alloy, dispersion strengthened Cu alloy, Alloy 718, Ti-6Al-4V alloy, precipitation hardened steel 660, beryllium, tungsten, carbon fibre composites, NiAl bronze, austenitic steel XM-19, standard austenitic 316L steel, pure Cu.
Appendix B, Guideline for analysis, provides analysis methods and guidelines of the design rules of SDC-IC.
Appendix C, Rationale or justification of the rules, includes the background information including references to literature that provided the foundation for the rules proposed in the SDC-IC.

Currently the scope of SDC-IC is limited to criteria related to design. Standard manufacturing rules (e.g. welding, forming and non-destructive examination) are based on EU harmonized standards.

4.4 ASME III for nuclear pressure components

At the beginning of the XXst century, after several steam boiler explosions in the United States, the American Society of Mechanical Engineers (ASME) appointed a committee to formulate a standard specification for the construction of steam boilers and other pressure vessels and for the care of same in service.

With time, these rules evolved into the Boiler and Pressure Vessel Code (BPVC) including section III dedicated to nuclear vessels (rules for construction of nuclear power plant components). The first edition (1915) was 114 pages long while the 2001 edition is about 16000 pages long. This to give an idea of large amount of work and experimentation it contains.

Following a brief history of its evolution.

1915: first publishing of "Rules for the construction of stationary Boiler and For Allowable Working pressures", design factor =5 (allowable stress = specified tensile /design factor). Section I for fired vessel, Section VIII pressure vessels (mainly for the petrochemical industry). 1942-45 World War II due to steel shortages the design factor is reduce to 4. This was justified (a posteriori) by the use of better material and improved non-destructive examination method.

1955: review aiming to reduce fabrication costs (design factor 3 or 3.5) mainly for the chemical industry (Section VIII). The safety was maintained by (what appears to be now the cornerstones of the code):
– Material selection;
– Fracture toughness;
– Rules for fatigue;
– Extensive non-destructive;

1963: Section III dedicated to the Nuclear industry is published with the Pressurized Water Reactor and Boiling water reactor in mind (T<800°F<430°C). It is basically a simplified version of Section VIII (only steam and water).

1976: issue of Criteria for the Design of Elevated temperature Class 1 components (effect of thermal creep) which will become subsections NH. Oak Ridge N.L. was developing breeder reactor at the time and was supporting this development.

1983: ASME Code is published in both metric units and US customary units (inch, psi) as previously.

ASME III was accepted by the Atomic Energy Commission (in 1963). It is now legally incorporated by reference by the United States Nuclear Regulatory Commission see: Title 10, Code of Federal Regulations § 50.55a Codes and standards. Alternatives are accepted.

The last version was released in 2010+2011 addendum (24).

In 2001, after the introduction of the European Equipment Directive (PED), ASME published a guide to ASME stamp holders explaining how to meet ESR of the directive using ASME Section VIII Division 1.

4.4.1 ASME III Division 4 for fusion energy devices

ASME III is composed of 5 divisions:

- Div1: Class 1, 2, 3 & MC Components; Supports; Core Support Structures; and Class 1 Components in Elevated Temperature Service;
- Div2: Concrete Containments;
- Div3: Containments for Transportation and Storage of Spent Nuclear Fuel and High Level Radioactive Material and Waste;
- Div4: Fusion Energy Devices (in development);
- Div5: High Temperature Reactors.

Division 4 is developing rules for components that are considered part of the pressure boundary and/or structural integrity of a fusion system.
The target is to issuance of first draft of fusion component rules by the 2015 BPV Edition (William Sowder, personal communication).

4.5 RCC-MR

The RCC-MR code (21, 25) was originally developed and published in 1985 as a complete set of design and construction rules for mechanical equipment of Fast Breeder Reactors Nuclear Islands including high temperature applications. On request of the ITER the scope of the code was enlarged to include the ITER vacuum vessel (VV).

The fourth edition of the RCC-MR code has been issued in October 2007 by AFCEN. In addition to including in the scope the ITER VV, the issue of the new 2007 edition was driven by needs of:
- Application in the code of European harmonized standards instead of French AFNOR standards;
- Introduction of requirements of the ESP and ESPN;
- Improvement of the design rules based on R&D results.

For application of the RCC-MR code to the ITER vacuum vessel, the following features shall be noticed:

– materials: from the beginning of the VV design development, austenitic steel type 316L(N)-IG has been selected based on steel with controlled nitrogen from the RCC-MR code, edition 1985.
The main driving force for the selection of this material from similar austenitic steels (316L, 316LN, etc.) is its high minimum tensile mechanical properties (combined with good toughness), that results in high stress intensity. In RCC-MR edition 2007 the steel grade is X2CrNiMo17-12-2 with controlled nitrogen content. The 316L(N)-IG has specific requirements for Co, Nb and Ta and lower limit for S and P;

– The design of the ITER VV is a box type of structure. Special rules for box types of structure have been developed in previous editions of RCC-MR for Class 1 components. In edition 2007 similar rules for Class 2 components have been introduced,

– based on features of box type structure, four categories of welded joints for VV have been proposed,

– RCC-MR edition 2007 includes rules for different type of bolted Joints.

Appendix 18 of RCC-MR introduces the requirements of the Pressure Equipment Directive and the ESPN. Appendix A19 gives complementary requirements specific for the design of the ITER vacuum vessel. The vacuum vessel is classified as a Class 2 welded box structure which allows to reflect the double shell assembly and to categorize the type of authorized welded joints. In accordance with ESPN, the ITER vacuum vessel is classified as multi-chamber equipment with pressure Category IV and nuclear level N2.

4.6 RCC-MX

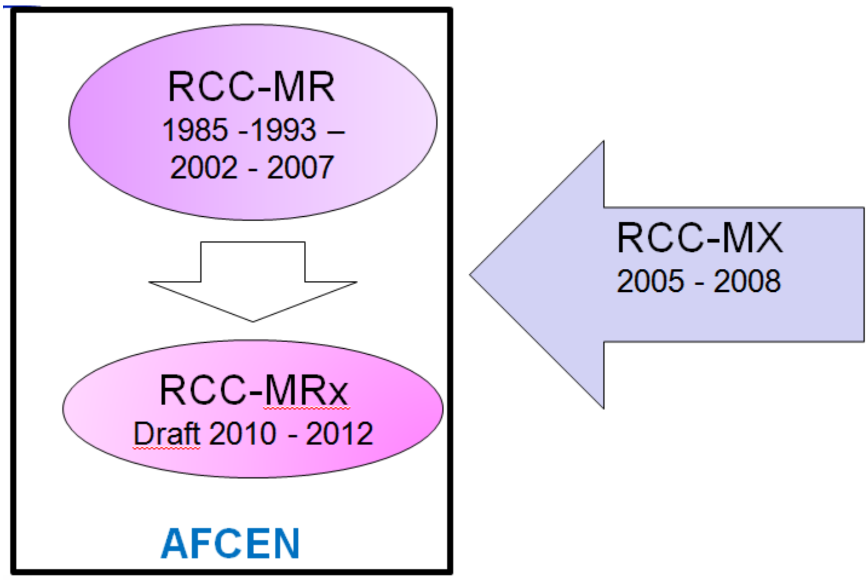
The RCC-MX is a design and construction code developed in the context of the JHR (Jules Horowitz reactor) project, by CEA and AREVA (AREVA TA and AREVA NP) (22, 25). This code dedicated to research reactor components, their auxiliaries and irradiation devices integrated the lessons learned of several tenths (60 years) of French research reactor design and operation and was founded on up-to-date standards (ISO International and or EN European standards). A huge effort has been done by collecting the irradiated material data and is continued through an important characterization program of material especially focused on irradiated material properties and aluminium and zirconium alloys.

4.7 RCC-MRx

The RCC-MRx code (13, 14, 26 and 27) constitutes a single document that covers the design and construction of components for high temperature reactors and research reactors and the associated auxiliaries, examination and handling mechanisms and irradiation devices.

The design rules were adapted to cover the mechanical resistance of structures close to neutron sources that can also operate in significant thermal creep conditions.

The RCC-MRx code was first issued in 2009 from the merging of the RCC-MR code, 2007 edition, devoted to high temperature reactor and ITER vacuum vessel, and the RCC-MX code, edition 2008, dedicated to research reactors and related devices (table 13). A new version of the RCC-MRx code has been published at the end of 2012.



Tab. 13 RCC-MRx genesis

The scope of application of the RCC-MRx design and construction rules (13) is limited to mechanical components:

• considered to be important in terms of nuclear safety and operability,
• playing a role in ensuring leaktightness, partitioning, guiding, securing and supporting,
• containing fluids such as pumps, valves, pipes, bellows, compartmentalized structures, heat exchangers and their supports.

To enable the introduction of complementary design and construction rule sets such as those contained in Standards NF EN 13445 (pressure vessels) and NF EN 13480 (pipes), the usual RCC code format has been modified with the creation of three sections (Figure 14):
- Section I: general provisions common to the entire code;
- Section II: additional requirements for the alternative use of other applicable rule sets (Standards NF EN 13445 and NF EN 13480 for instance) for Class 3 components, and special instructions for components subject to local regulations such as pressure equipment or nuclear pressure equipment;
- Section III: set of applicable rules with six Tomes:

Tome 1 being sub-divided into subsections: it contains the design and construction rules and comprises subsections which are alphanumerically numbered:

- Volume A: general provisions for Section III,
- Volume B (C and D): design rules for class N1Rx (N2Rx and N3Rx) components and supports of the nuclear reactor and its auxiliary systems,
- Volume K: design rules for examination, handling or drive mechanisms,
- Volume L: design rules for irradiation devices
- Volume Z: technical appendixes associated to Tome 1

Tomes 2 to 5 contain the rules corresponding to various technical areas:
Tome 2: part and product procurement specifications,
Tome 3: destructive tests and non-destructive examination methods,
Tome 4: qualifications for welding operations and welding procedures and their application,
Tome 5: manufacturing operations other than welding,
Tome 6: rules in probationary phase

The three classes of design and construction proposed (N1Rx, N2Rx and N3Rx) correspond to decreasing levels of assurance of ability to withstand different types of mechanical damages to which the component might be exposed as result of loading corresponding to specific operating conditions.

Figure 1.2: general plan of the RCC-MRx code

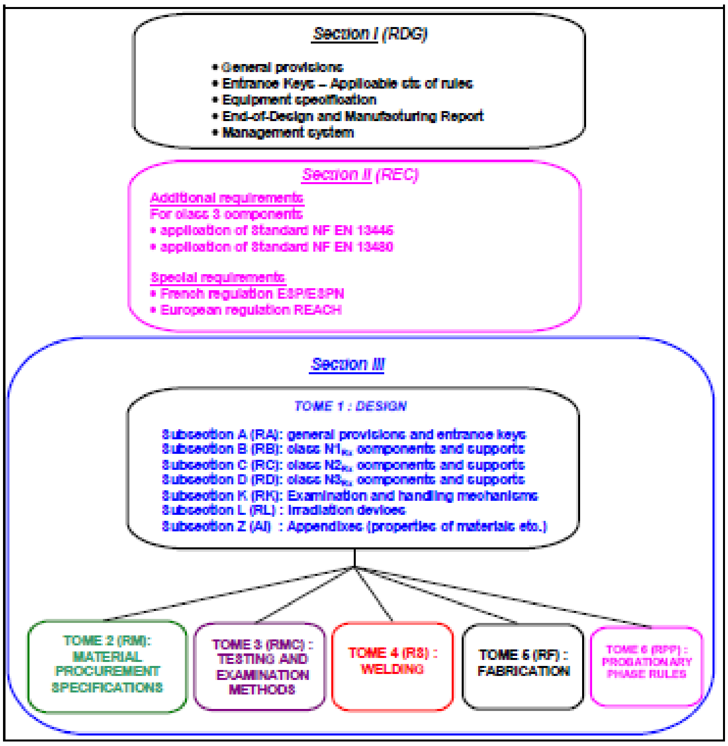


Fig. 11 General Plan of RCC-MRx code

New materials, different from the steels used in Pressurized Water Reactors (PWR) and Sodium Fast breeder Reactors (SFR), such as alloys of aluminium and zirconium can meet requirements for neutron transparent materials of research or irradiation reactors (RR) (table 14).
For instance, for reflectors, the substitution of 10 mm of steel tube for 10 mm of zirconium or aluminium alloy tubes makes it possible to potentially obtain 50% more thermal flux. It is therefore necessary to make sure that suitable compromises are made between mechanical strength, thermal control (gamma heating of 10 to 20W/g in core) and activation effect control (e.g. cobalt content).

Although the initial editions of the RCC-M and RCC-MR rely greatly on the French NF series standards at the time, the rapid development of European standards (EN and EN ISO) covering all subjects such as procurement, destructive and non-destructive testing, welding and even design, leaded to carry out a considerable amount of work to integrate these new standards into the RCC-MR 2007 and RCC-MX 2008. This update of the standards was found in the RCC-MRx code.

RCC-MRx subsection Z - Appendix A3 or Probationary Phase Rules	*A3.4 : Basic data*	*A3.5 : Creep data*	*A3.6 : Irradiation data*	*A3.8 : Fracture Mech. data*
Non Alloy Steels (13 RPS in Tome 2)				
A3.10NAS : P235GH	X			
A3.11NAS : P265GH	X			X
A3.12NAS : P295GH	X			X
Alloy Steels (16 RPS in Tome 2, 4 RPS in RPP)				
A3.11AS : 25CrMo4, 42CrMo4, 30CrNiMo8	X and Bolts			
A3.13AS : 16MND5	X			
A3.14AS : 10CrMo9-10 fully annealed	X	X		
A3.15AS : 13CrMo4-5 quenched and tempered	X	X		
A3.16AS : 2.25% Cr, 1% Mo normalised tempered or quenched tempered	X	X		
A3.17AS : X10CrMoVNb9-2 quenched tempered	X	X		
A3.18AS : X10CrMoVNb9-1 normalised tempered or quenched tempered	X	X		X
A3.19AS : Eurofer X10CrWVTa9-1 normalised tempered	X			
Stainless Steels (25 RPS in Tome 2)				
A3.1S : X2CrNiMo17-12-2(N) solution annealed	X	X	X	X
A3.2S : X6CrNi18-10 et X5CrNi18-10 solution annealed	X	X		
A3.3S : X2CrNiMo17-12-2, 17-12-3, X2CrNiMo18-14-3	X	X	X	
A3.4S : X2CrNi18-9, X2CrNi19-11	X	X		
A3.6S : X15CrNiW22-12 solution annealed followed by aging	X	X		
A3.7S : X2CrNiMo17-12-2 around 20% work hardening	X and Bolts	X	X	
A3.8S : X4CrNiMo16-05-01 quenched and annealed	X and Bolts			
A3.10S : X6NiCrTiMoVB25-15-2 heat treated structural hardening	X and Bolts	X		
Special Alloys Ni-Cr-Fe (5 RPS in Tome 2)				
A3.5SA : X5NiCrTiAl33-21 after annealing heat treatment at 980°C	X	X		
Aluminium alloys (7 RPS in Tome 2, 1 RPS in RPP)				
A3.1A : 5754-O	X	X	X	
A3.2A : 6061-T6	X	X	X	
Zirconium alloys (4 RPS in Tome 2)				
A3.1Z : Zircaloy 2	X	X	X	
A3.2Z : Zircaloy 4	X	X	X	

Table 14 Mechanical characteristics available in Appendix A3

The following table 15 shows the correspondence of ASME Section III and RCC MRx code.

RCC-MRx	AFCEN RCC-MRx Code	ASME Nuclear Boiler and Pressure Vessel Code
Section I	General provisions	Section III Subsection NCA
Section II	Additional requirements and special instructions	
Section III	Rules for nuclear installation mechanical components	Section III Division 1
Section III Tome 1	Design and construction rules	Section III Division 1
Subsection A	General provisions for Section III	Subsection NA
Subsection B	Class 1 reactor components, its auxiliary systems and supports	Subsection NB + NF + NG + NH
Subsection C	Class 2 reactor components, its auxiliary systems and supports	Subsection NC + NF + NG + NH
Subsection D	Class 3 reactor components, its auxiliary systems and supports	Subsection ND + NF
Subsection K	Examination, handling or drive mechanisms	
Subsection L	Irradiation devices	
Subsection Z	Technical appendices - A3 / A9 : Properties groups for base metal and welded joints - A16 : guide for prevention of fast rupture, LBB and defect assessment - Other technical appendicies	
Section III Tome 2	Part and product procurement specifications	Section II (Nx 2000)
Section III Tome 3	Destructive tests and non-destructive examination methods	Section V (Nx 5000)
Section III Tome 4	Welding	Section IX (Nx 4000)
Section III Tome 5	Manufacturing operations other than welding	(Nx 4000)

Table 15 Correspondence between ASME III and RCC-MRx (27)

The added values of RCC-MRx are:

- The use of specific materials (aluminium alloys, Zirconium alloys, etc.) required to manage compromises involving mechanical behaviour, low activation under neutron fluxes, low capture of neutron fluxes, thermal conductivity, etc. specific of research reactors;
- modern welding processes (Electron Beam, laser, diffusion);
- taking into account the thermal creep:
  • High temperature (up to 700° C depending on the materials),
  • Other material such as aluminium alloys at intermediate temperature;
- consistent with the PED and ESPN regulations;
- taking into account:
  • Important level of irradiation inducing an evolution of the characteristics of materials (stainless steel, aluminium or zirconium alloys),
  • Up to date European standards such as “harmonised standard.

4.7.1 Promotion of the code in Europe

AFCEN has been working with the EC, in the framework of Sustainable Nuclear Energy Technology Platform (SNE-TP), to promote RCC-MRx as the European code for the design and construction of the next generation of nuclear plants.
The first applications of this code will concern innovative prototypes and demonstration plants supported by the European Sustainable Nuclear Industrial Initiative (ESNII) to be built in Europe during the next decade.

Thus in 2011, it was decided to federate the stakeholders of these projects, and other interested parties (e.g. ITER) in the development of the dedicated code, namely RCC-MRx, through a CEN Workshop (CEN/WS 64) leaded by AFCEN. The workshop was financially supported by the European Commission.
In parallel, the CEN-CENELEC Focus group investigated the situation of nuclear standards in Europe and the

potential need to create a European technical structure dedicated to this area under its umbrella.

Based on the results of the CEN Workshop 64 and the recommendations of the CEN-CENELEC Focus Group, in April 2013 the European Commission, in the framework of ENEF (European Nuclear Energy Forum), decided to support a CEN/WS 64 phase 2 with the target to enlarge the scope to the codes for mechanical equipment and civil engineering of GEN II to GEN IV nuclear installations. Indeed, seeing the perspective of the Long Term Operation for GEN II NPPs and the potential new NPPs of GEN III, the aim would be to offer an opportunity to a wide range of stakeholders to gain a deep understanding of the AFCEN codes and their evolution process, to allow them to define their long term requirements for the codes modification and the adaptation to their needs. It would also lead to the definition of pre-normative research priorities. At the end this would help initiating the harmonization process of codes at EU level (28).

5. Fusion structural materials

Structural materials for in-vessel components (like TBM and first wall) of future energy generating fusion reactors will be exposed to high neutron radiation and thermo-mechanical loads. Degradation of microstructure of structural materials under neutron irradiation as a result of displacement damage accumulation up to damage doses of 150 dpa and production of transmutation products, i.e. helium and hydrogen, will strongly influence materials' performance (29).

ITER materials activities were largely influenced by its time schedule. Advanced low activation materials such as vanadium alloys and SiC_f/SiC composites (SiC fibers in SiC matrix), and even martensitic steels were discarded in favour of the more robust stainless steel type 316L(N).
DEMO materials activities did not have the same time constraint and in addition it was well known that the solution annealed austenitic stainless steels such as 316L(N) are not suitable for high dose applications (in DEMO > 70 dpa) due to their irradiation swelling (29,30,31).
Vanadium alloys and SiC_f/ SiC composites developments marginalised due to several unresolved shortfalls (e.g. low temperature irradiation embrittlement and absence of reliable protective coating for vanadium alloys, and low thermal conductivity and low fracture toughness for SiC_f/SiC (16, 31)).

The reduced activation ferritic/martensitic (RAFM) steels and their oxide dispersion strengthened (ODS) variants are considered as primary candidate structural materials for in-vessel components with operating temperature range between 350 and 650 ◦C (29). This because metals and alloys with "body-centred cubic (Bcc)" crystal lattice structure, including iron and ferritic steels, show better resistance to prolonged irradiation than metals with "face-centred-cubic (Fcc)" lattices (32).
A comparison of the swelling under fission irradiation for ferritic steels against austenitic (fcc) steels is shown in fig. 15.

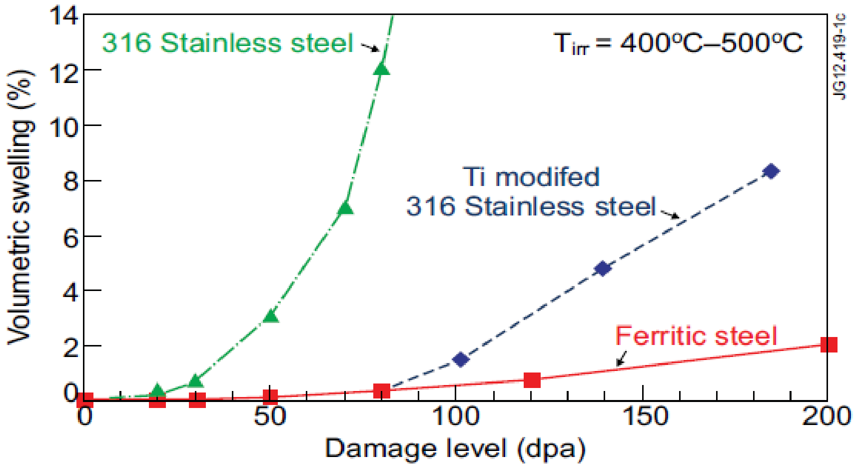


Fig 12 Comparative swelling under fission neutron irradiation for ferritic and austenitic steels (32)

Some properties of ferritic/martensitic steel (F/M), V alloy and SiC_f/SiC (called simply SIC in the table) are compared in table 16 (33).
Fusion structural materials activities evolved achieving code qualified status (13).

	Units	F/M steel (~500 °C)	V4Cr4Ti (~600 °C)	SiC
Melting temperature	K	~1,800	2,170	
Density	kg/m^3	~7,900	6,050	2,500–3,200
Young's modulus	GPa	190 (Tavassoli)	120	200–300
Poisson's ratio	–	0.29–0.31	0.37	0.16–0.18
Thermal expansion coefficient	10^{-6} K^{-1}	11.5	10	3–4
Specific heat	J/kg K	630 (Tavassoli)	550	600
Thermal conductivity	W/(m K)	32.5 at 400 °C 33.0 @ 500 (Tavassoli)	36	5–20
Thermal stress figure of merit (Zinkle and Ghoniem 2000)	kW/m	~5.4 Fe–8–9Cr martensitic at 400 °C	~6.4 at 450–700 °C	2.0 at 800 °C
Maximum allowable combined stress	MPa	≈160	≈180	≈190
Maximum allowable burnup	%			≈3 %
Maximum temperature for compatibility with Li	°C	550–600	650–700	550
Maximum temperature for compatibility with PbLi	°C	450	650	800
Electrical resistivity	Ω m	0.07E−6	0.71E−6	0.002–0.05

Table 16 - Some properties of F/M, SIC and V4Cr4Ti (33)

5.1 Eurofer97

Fission experience has strongly influenced fusion program. Nevertheless, extensive screening tests were performed on different Fe–Cr compositions before narrowing down the chromium range to 7–9% and finally converging towards a composition similar to that of the conventional Modified 9Cr–1Mo steel.
In 1995 IEA defined a reference low activation steel to be produced in Japan and characterised by all members. The term RAFM (Reduced Activation Ferritic/Martensitic) steel was used.
Two industrial heats were fabricated and designated IEA-F82H heats, to distinguish them from an earlier F82H heat, also produced in Japan (31). These heats were characterised by all partners and the results were collected in dedicated relational database. In 1997, EU opted for a higher chromium and lower tungsten composition grade, Eurofer (also called Eurofer97) (31). Other countries are investigating similar compositions and targeting industrial productions for ITER TBMs. Table 17 presents chemical compositions of four RAFM steels produced in different countries along with the composition of the conventional Modified 9Cr–1Mo steel.

	Mod. 9Cr–1Mo	F82H	Eurofer	INRAFM	CLAM
Cr	8.5	7.7	9	9	9
C	1	0.09	0.11	0.11	0.1
Mn	0.4	0.16	0.4	0.5	0.45
P	<0.020	0.002	<0.005	<0.002	
S	<0.01	0.002	<0.005	<0.002	
V	0.21	0.16	0.2	0.22	0.2
B		0.0002	<0.001	<0.001	
N2	0.05	0.006	0.03	0.03	
O2		<0.01	<0.01	<0.01	
W		1.94	1.1	1.35	1.45
Ta		0.02	0.07	0.07	0.15
Ti		100 ppm	<0.01	<0.005	
Nb	0.08	1 ppm	<0.001	<0.001	
Mo	0.92	30 ppm	<0.005	<0.002	
Ni	<0.20	200 ppm	<0.0050	<0.005	
Cu	<0.10	100 ppm	<0.0050	<0.002	
Al	<0.040	30 ppm	<0.0010	0.005	
Si		1100 ppm	<0.050	<0.05	
Co		50 ppm	<0.0050	<0.005	
Sn		(<20 ppm)			
As		(<50 ppm)	As + Sn + Sb + Zr < 100 ppm	AS + Sn + Sb < 0.03	

Table 17 - Nominal chemical compositions of RAFM steels (31).

The basic difference between RAFM steels and the conventional steel is in replacement of Mo and Nb with their equivalent low activation elements (W and V) (31). Other high activation residual elements are kept as low as possible. Tantalum is added for grain size control. Heat treatment specifications and acceptance values of RAFM steels are similar to the conventional 9Cr–1Mo steels (31).

Their main advantages over austenitic stainless steels are related to the excellent dimensional stability (creep and swelling) under neutron irradiation (16). In addition, they present lower coefficients of thermal expansion and higher coefficients of thermal conductivity at high temperatures (31). Development of RAFM steels has been the object of extensive cooperation between the different fusion parties; Eurofer97 is one of the variants developed in the European Union within the framework of the EFDA.

Table 18 reports the minimum ductility requirements of the materials as defined by ESPN/RCC-Mx and that of Eurofe97 in irradiated/un-irradiated conditions.

	ESPN/RCC-MX Limit	Eurofer97 at BOL	Eurofer97 at 3 dpa
ε_t (RT)	>14%	22%	~14% (300 °C)
Toughness (0 °C)	>40 J	220 J	~200 J (300 °C) ~23 J (RT)
S_y/S_u (RT)	<0.85	0.83	0.92 (300 °C)

Table 18 - Minimum ductility requirements of the materials as defined by ESPN/RCC-MX and that of Eurofer in irradiated/un-irradiated conditions (16).

Eurofer achieved code qualification status after several years in 2013 with its entry in RCC-MRx edition 2012 (under Section III, Tome 1, Sub-Section Z and denomination A3.19AS) (31). F82H is expected to follow with its entry in the Japanese Code.

5.1.1 Eurofer mechanical properties (un-irradiated)

5.1.1.1 Tensile strength and ductility

Eurofer97 shows excellent mechanical properties in the unirradiated condition from −80 up to 550 ◦C. Above this temperature range the structural application of the steel is limited mainly by considerable loss of tensile and creep strengths (29).

The yield strength (Sy) and ultimate tensile strength (Su) of Eurofer97 with temperature compared to those of the 316L(N)-IG steel are reported in fig. 13. In the whole range of temperatures relevant for TBM operation (300-550 °C), Eurofer97 exhibits a much lower strain hardening capability compared to 316L(N)-IG. The lack of strain-hardening capability is also evident in Fig. 14 which compares the uniform (ε_u) and total elongation at rupture (ε_t) for Eurofer97 and 316L(N)-IG. The reduced ductility/strain-hardening capability of Eurofer97 respect to 316L(N)-IG is shown in Fig. 15 which reports stress–strain curves for the two steels at 500 °C.
According to the values Eurofer97 could be considered as a semi-brittle or even brittle material even in the absence of irradiation damage (16).

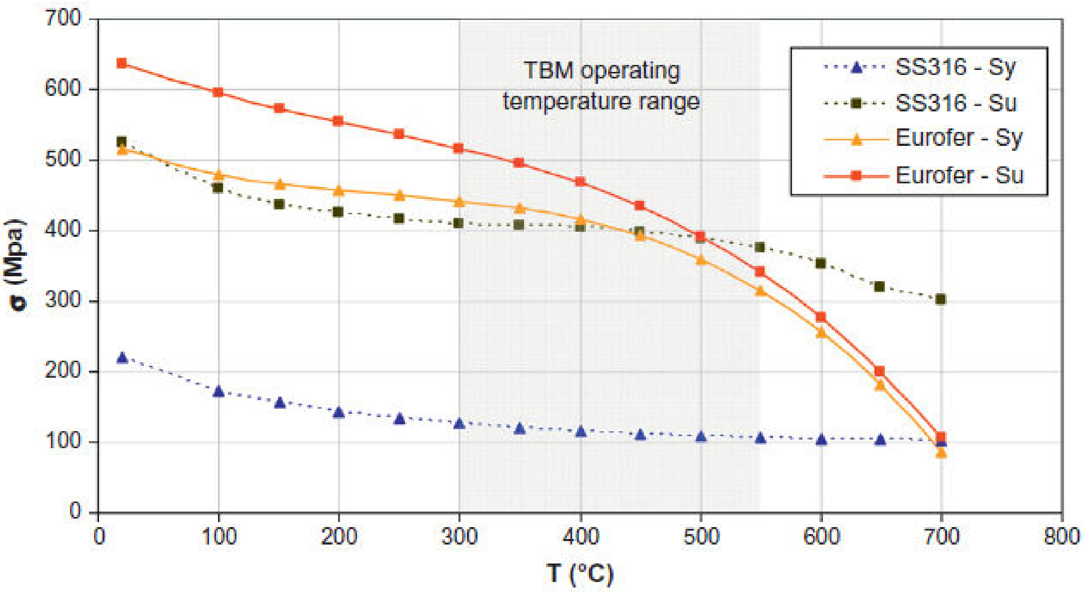


Fig 13 - Eurofer97 and 316L(N)-IG yield strength (Sy) and ultimate tensile strength (Su) (16)

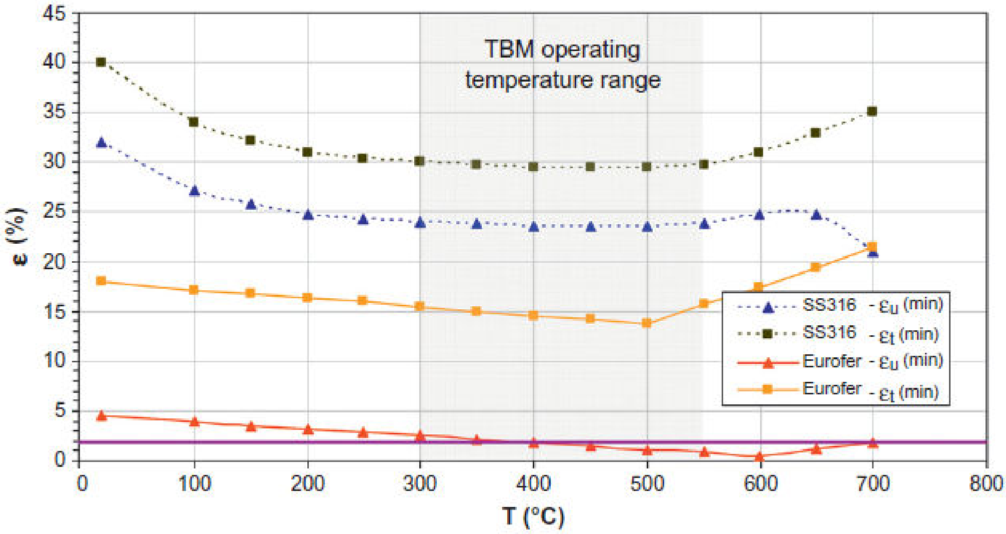


Fig. 14- Uniform elongation (ε_u) and total elongation (ε_t) values for Eurofer97 and 316L(N)-IG (16).

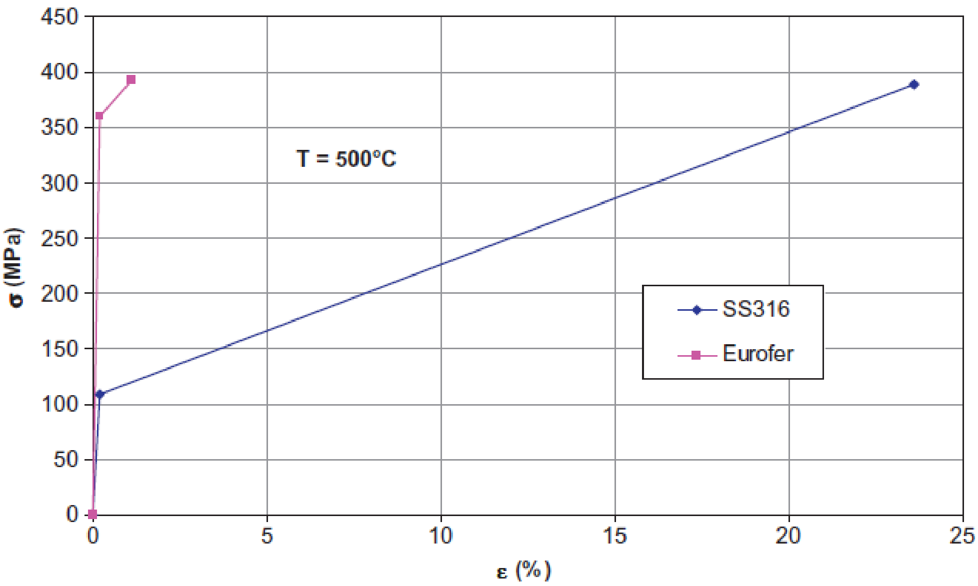


Fig. 15 - Schematized stress-strain curve for Eurofer97 and 316(N)-IG at 500 °C (16)

5.1.1.2 Thermal creep

The creep behaviour of Eurofer97 is well characterised. Eurofer97 shows adequate creep rupture strength, comparable to that of other RAFM steels. Values of allowable stress S_t for Eurofer97 are shown in Fig. 16.

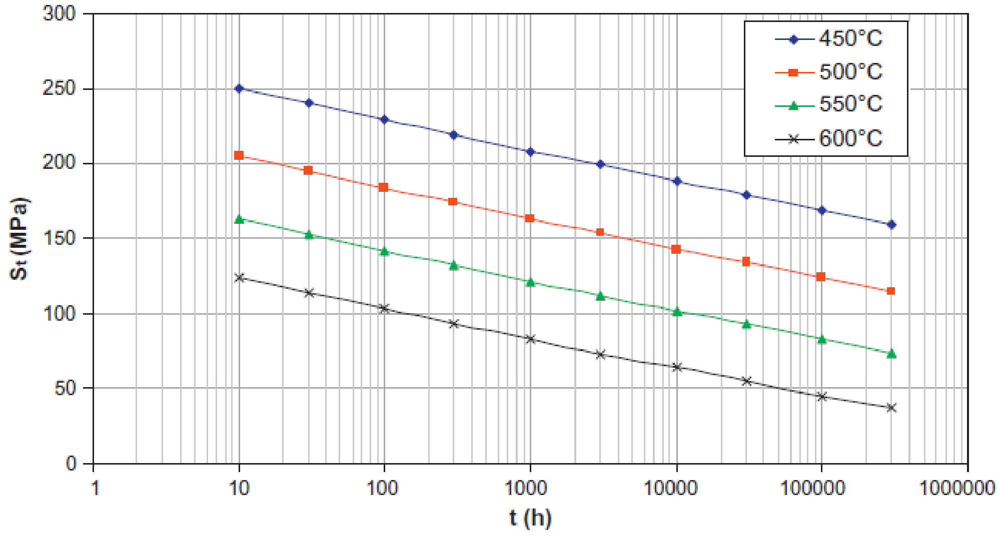


Fig. 16 – Eurofer allowable stress S_t values (16)

5.1.1.3 Fatigue proprieties

Eurofer97 has very pronounced softening under Low cycle Fatigue (LCF) tests: 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. During fatigue tests, when saturation is reached, the range of the stable stress $\Delta\sigma$ can be plotted as a function of the imposed strain range $\Delta\varepsilon$.
The curve thus determined is called the cyclic stress–strain curve. Cyclic stability is not achieved in LCF tests of Eurofer97: in this case, the cyclic stress–strain curves corresponding to half-lives are reported as cyclic stress–strain curves. Fig 17 reports the cyclic stress-strain curves compared with monotonic stress-strain curves (16).

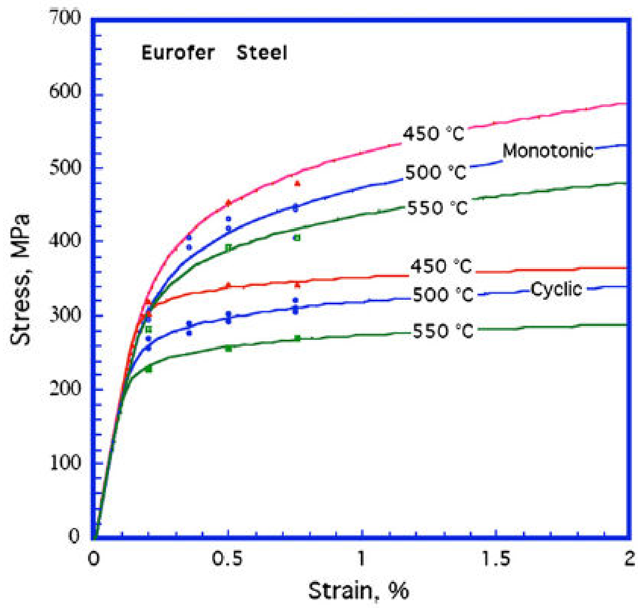


Fig. 17 - Reduced cyclic stress-strain curves for Eurofer97 at different temperatures (16)

5.1.1.4 Creep-fatigue interaction

In vessel components like TBM or Divertor components will operate under cyclic, non-symmetrical thermo-mechanical loadings that can lead to creep damage accumulation. These operating conditions can be simulated by introducing a hold time in LCF tests. The fatigue behaviour of Eurofer97 becomes much more complicated when introducing hold times in LCF (Low Cycle Fatigue) tests, showing very different characteristics with tension, compression or symmetrical hold times (16). Fig. 18 reports the influence of hold times on the fatigue life of Eurofer97.

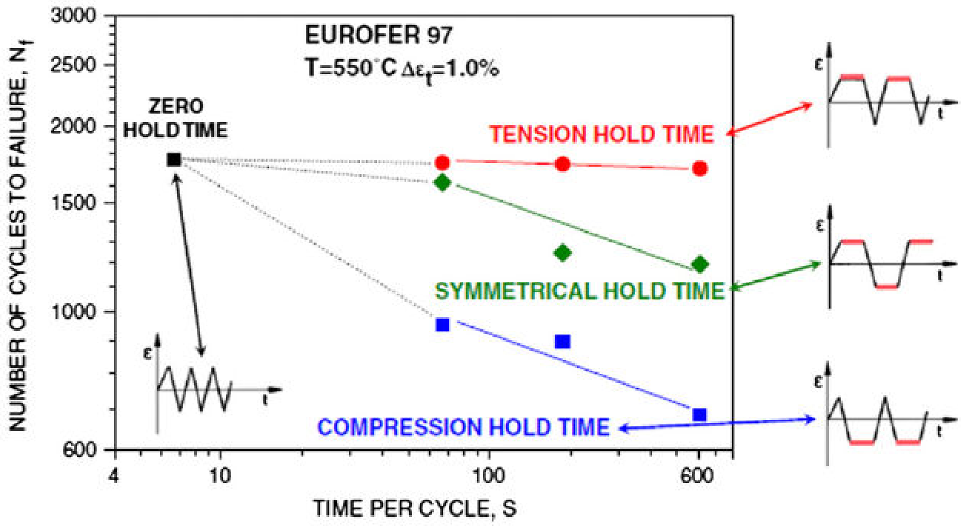


Fig. 18 - Influence of hold time on the fatigue life of Eurofer97 (16)

5.1.2 Neutron irradiation effects on eurofer97 and other RAFM steels

5.1.2.1 Tensile properties

Fig. 19 shows Yield Stress (R_{p02}) vs. test temperature (T_{test}) for EUROFER97 in the unirradiated condition and after neutron irradiation in different medium and high dose European irradiation programmes at target irradiation temperatures (T_{irr}) between 300 and 350 ◦C (29).

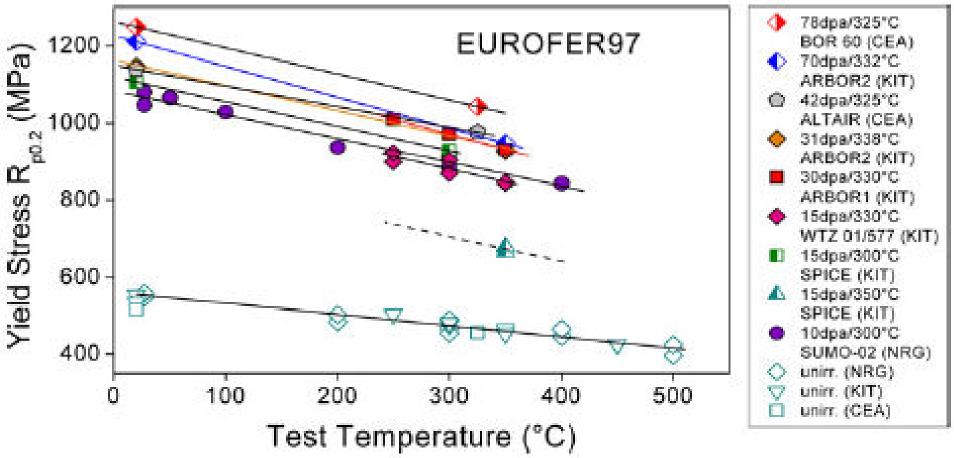


Fig. 19 - Yield stress vs. Temperature for Eurofer97 in the unirradiated condition and after neutrons irradiations in different irradiation programme (29, 34).

Neutron irradiation leads to substantial increases in the Yield Stress which is sensitive to the irradiation parameters, i.e. irradiation dose and temperature. Furthermore, for given irradiation conditions the Yield Stress increase depends on the test temperature and is larger at low test temperatures.
The test temperature and dose dependences of the Ultimate Tensile Strength (Rm) resembles to that of the Yield Stress. Close values of the Ultimate Tensile Strength (Rm) and Yield Stress (Rp0.2) in the irradiated conditions indicate a strong suppression of the strain hardening capability under neutron irradiation.

The evolution of the hardening, as the increase in Yield Stress, with damage dose for different product forms of EUROFER97 and for other selected RAFM steels for Tirr = 300–335 ◦C and Ttest = 300–350 ◦C is summarized in Fig. 20.

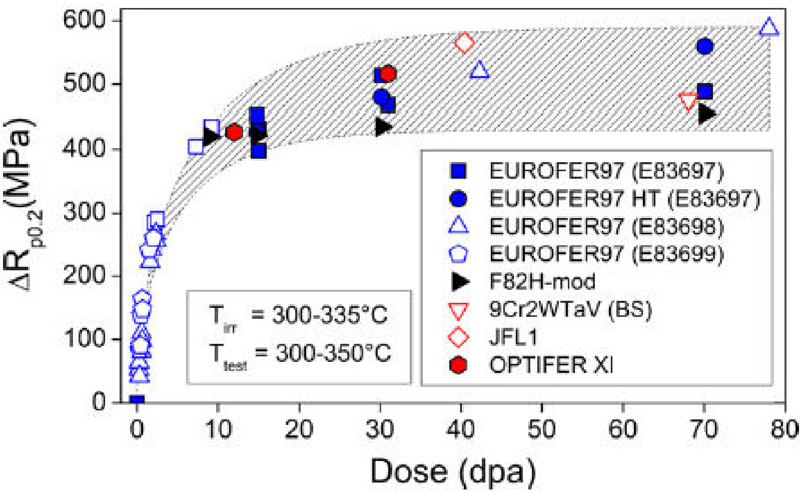


Fig. 20 Irradiation hardening vs. Irradiation dose for eurofer97 and other RAFM steel for Tirr= 300-335 °C and Ttest= 300-350 °C.

Neutron irradiation leads to a substantial increase in the Yield Stress of RAFM steels with the damage dose. The Yield Stress increase is rather steep at doses below 10 dpa. The hardening rate appears to be significantly decreased at the achieved damage doses and a clear tendency towards saturation is identified.

For the analysis of high dose irradiation behaviour of EUROFER97 differentiation has to be done between different product forms as well as different heat treatment conditions. In fact there is a strong sensitivity of materials' mechanical properties and irradiation performance to metallurgical parameters.

The hatched area marks the scattering band of high dose hardening for different RAFM steels.
It is important to note that to the reasons for the data scattering belong not only the differences in the metallurgical variables, but also variations and uncertainties in the irradiation conditions.

Fig. 21 shows Uniform Strain and Fracture Strain of selected RAFM steel specimens irradiated to 70.1 dpa at 331.5 ◦C and tested at 350 and 20 ◦C. Rather low values of the Uniform Strains below 1% are observed for all investigated RAFM steels. The Fracture Strains, in contrast, remain at a high level above 9–10%.

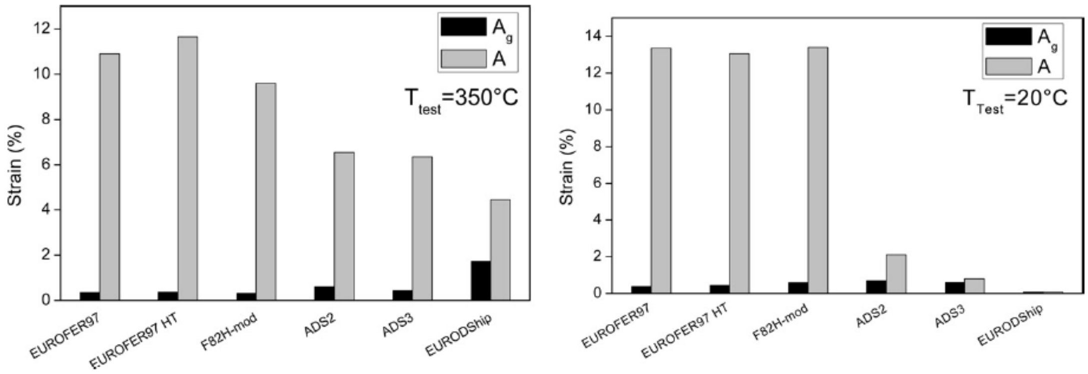


Fig. 21 - Uniform strain (Ag) and fracture strain (A) of 70.1 dpa, 331.5 °C irradiated RAFM materials tested at 350°C (left) and 20 °C (right).

The boron doped Steels ADS2 and ADS3 (to study the effect of helium generation) show larger Uniform and reduced Fracture Strains in comparison to base RAFM steels.

The evolution of Strain with irradiation dose for different product forms of EUROFER97 for irradiation temperatures between 300 and 335 ◦C and test temperatures between 300 and 350 ◦C is shown in Fig. 22. The both tensile properties show a strong reduction from their values in the reference unirradiated state already at a damage dose of 2.1 dpa. The Uniform Strain (Ag) is scattered mostly below 0.5% in the irradiated state. Though strongly reduced, the Fracture Strain (A) remains mostly at or above 10% for irradiated specimens.

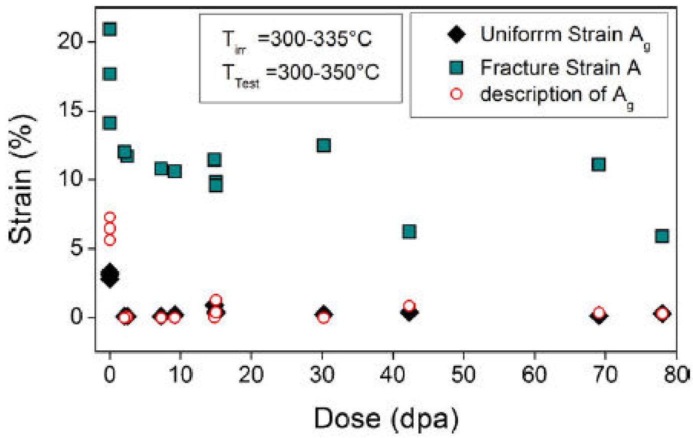


Fig. 22 - Uniform and fracture strain vs. Damage dose for different product forms of Eurofer97 (29, 34)

5.1.2.2 Impact properties

The role of irradiation temperature on the impact properties of EUROFER97 and other RAFM steels was studied in SPICE program.

Fig. 23 shows the DBTT vs. irradiation temperature for EUROFER97 and other RAFM steels from SPICE irradiation programme.

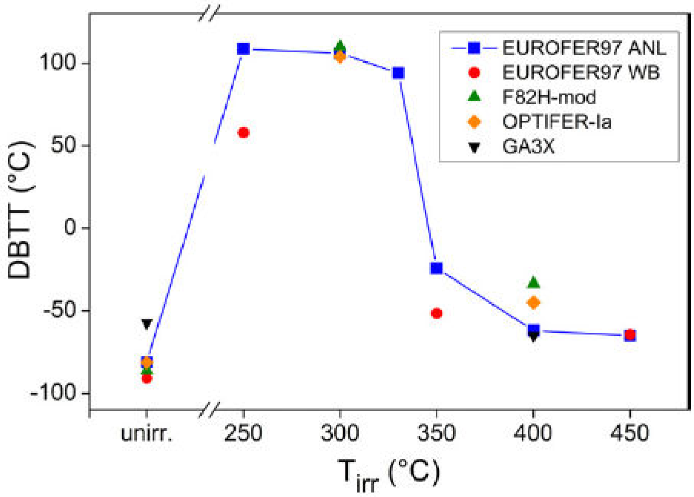


Fig. 23 - Change in DBTT as a function of irradiation temperature for various Eurofer alloys at 16 dpa exposure (29, 34)

The DBTT of all steels is influenced most at low irradiation temperatures (Tirr ⩽ 330 ◦C). The DBTTs of the materials irradiated at or above 350 ◦C remain below −24 ◦C and, hence, are well below the material application temperature. On the basis of this diagram a temperature window between 350 and 550 ◦C is proposed for operation of the FW, Divertor and Blanket components.

Fig. 24 shows the evolution of the neutron irradiation induced embrittlement (measured in impact tests) with dose for EUROFER97 and F82H steels at irradiation temperatures between 300 and 337 ◦C (29, 34).

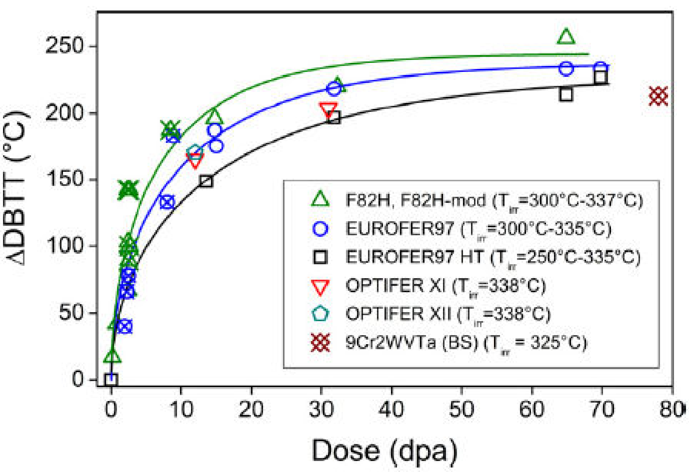


Fig. 24 - Irradiation shifts of the DBTT vs. Dose for eurofer97 and other RAFM steels.

In case of EUROFER97, differentiation is made between specimens machined from as-delivered products and specimens machined from the plates subjected to pre-irradiation heat treatment (HT). The results on F82H and F82H-mod are plotted together for different heat treatments and material compositions. The pre-irradiation heat treatment (HT) of Eurofer97 leads to considerable improvement of the irradiation resistance at doses up to 30 dpa. At the achieved damage doses, however, the embrittlement of Eurofer97 HT becomes comparable to that of Eufofer97. All RAFM steels show steep increase in the ΔDBTT with dose below 15 dpa. With farther increasing the damage dose the embrittlement rate decreases and a clear tendency towards saturation is observed at the achieved damage doses.

5.1.2.3 Helium effects

The helium effects are studied in different simulation experiments including fission reactor irradiation of boron doped steels, irradiation under spallation target environment, multi-ion beam-irradiation, etc.

Helium effects are not sufficiently understood due to absence of an irradiation facility with fusion reactor relevant neutron spectrum and have to be a subject of future investigations. Different simulating techniques used, e.g. fission reactor irradiation of boron doped model steels or spallation neutron and proton irradiation of RAFM steels most probably strongly overestimate helium effects. Moreover, helium effects will strongly depend on the irradiation temperature. Boron doping technique yielded progressive embrittlement with increasing helium concentration (10 – 430 appm) at Tirr ⩽ 350 ◦C; the embrittlement rate was however reduced at achieved helium contents. Helium embrittlement was considerably reduced at Tirr = 450 ◦C. Synergistic helium and dpa effects under spallation target environment at Tirr ⩽ 380 ◦C yielded linear increase of DBTT shift with produced helium up to 1600 appm.

5.1.2.4 Swelling

The swelling of RAFM steels is sensitive to irradiation or implantation temperature, helium and dpa production rates and steel virgin microstructure. The neutron irradiation of boron and nickel doped RAFM steel at 300 °C yielded nearly no swelling while irradiation at 400 °C, however, 1.2% and 1.1% volumetric swelling was observed.
The absence of irradiation facility with fusion relevant neutron spectrum does not allow the assessment of swelling under in-service condition as there still remain uncertainty in the helium content and irradiation temperature for the onset of pronounced swelling for fusion relevant dpa and helium production rates.

5.1.2.5 Irradiation effects of on fatigue

Few data are available on the consequences of neutron irradiation on the fatigue behaviour of Eurofer97 and other RAFM steels. As a general rule, the fatigue lifetime of the irradiated material should be shorter because of loss of ductility and significant increase in cyclic softening (16).

5.1.2.6 Application of Eurofer under PWR conditions

Eurofer97 can also be considered to be used in water cooled breeding blanket under pressurized water reactor (PWR) condition with coolant inlet/outlet temperatures of 275/325 °C (at a pressure of ~15.5 MPa) taking into account that neutron irradiation of this steel in the temperature range between 275 and 325 °C results strong degradation of the mechanical properties (30). In fact the DBTT will be raised above room temperature already after few dpa and at 70 dpa at 330 °C the value of DBTT is at around 152 °C.
Helium effects are expected to lead to further degradation of DBTT
From first experiments the swelling seems not to be a problem under PWR condition.

Application of ERUOFER97 under PWR condition will be problematic because the irradiation induced embrittlement must be considered at least for normal and abnormal reactor shut downs particularly where the temperature goes below the DBTT of the irradiated material. Recovery heat treatments prior to reactor cool down can be utilized to avoid handling of brittle components which however might be a challenge with regard to needed homogeneous heating. Under such circumstances implementation of the recovery heat treatments (550°C/3 h) will be required in order not to allow the DBTT to come close to a coolant inlet temperature of 275 °C.

Furthermore, corrosion behaviour of EUROFER97 under PWR condition is considered as another concern. Stress corrosion cracking (SCC) phenomena need to be investigated in dedicated experiments under unirradiated and irradiated conditions.

5.1.3 Databases

Distinction is to be made between material's properties data and code-qualified materials properties data (31). In the case of code-qualified properties, all data collected must be harmonised and validated before they are entered in the databases. As a result, the quality of materials properties data collected in the fusion program has varied with time. For instance, materials properties data from the literature were initially included in the ITER MPH (Materials Properties Handbook) to allow conceptual design analysis to proceed. For the ITER Interim Structural Design Analyses only the code-qualified part of the data was used (25). SDC-IC harmonised ITER MPH containing mostly code-qualified data.

All Eurofer data entered in the EU databases are now code qualified. Extracts from these databases are used at first to derive design allowable for ITER Test Blanket Modules (TBMs), since TBM design (not requiring irradiation data higher than 3 dpa) is in an advanced stage.
For DEMO, where doses are much higher, design is still evolving and effects of irradiation damage are more complicated. Furthermore, some experimental data generated and analysed are still for understanding the irradiation damage, particularly the effects of higher helium formation under 14 MeV neutrons and its synergistic effects when combined with dpa.

Presently the EU RAFM database has suited data concerning Eurofer steel.

5.2 ODS Eurofer

Advanced Oxide Dispersion Strengthened(ODS) EUROFER steels with 0.3 wt.% Y2O3 exhibit good mechanical properties in the unirradiated state in the temperature range from -50 to 650 °C . Yield and tensile strength are about 35% higher than for EUROFER97 up to 700 °C. The impact properties are somewhat worse than that of EUROFER97 but still in an acceptable range. Low temperature neutron irradiation (Tirr ⩽ 340 ° C) leads to strong embrittlement of ODS EUROFER steels.
Irradiation at 450 and 550 °C of recently developed ODS EUROFER steel yielded nearly no changes of the mechanical properties . Having in mind a strong recovery of the radiation damage in base EUROFER97 steel above 350 °C, the operation temperature window between 350-400 and 650 °C can be proposed for ODS EUROFER (9%Cr) steel. The influence of helium on the mechanical properties of ODS EUROFER is not available yet and a dedicated study is required in the future (29).

5.3 Improved structural materials for fission and fusion energy

Existing structural materials have rather limited operating temperature regimes where they can be utilized in a neutron irradiation environment as summarized in Fig. 25 for moderate damage levels of 10 to 50 dpa.
At low temperatures, the reduced ductility associated with low temperature radiation hardening creates

conditions where modified (larger safety margin) engineering design rules must be used. In cases where fracture toughness* is reduced by low temperature irradiation, the operating temperature is restricted to higher temperatures where embrittlement does not occur. The upper operating temperature limit is typically determined by thermal creep strength or high temperature helium embrittlement considerations.

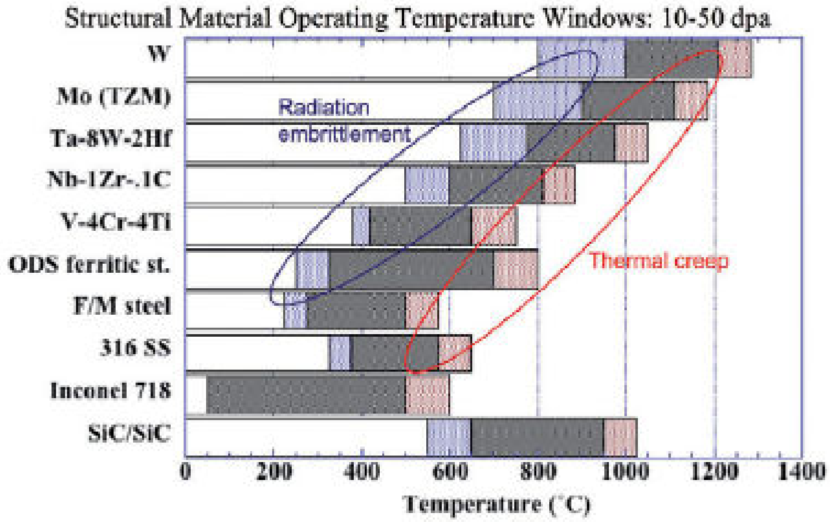


Fig. 25 - Estimated operating temperature windows for structural materials in nuclear energy systems for damage levels of 10 to 50 dpa (35).

The "Fourth Generation Ferritic-Martensitic steels" (that are FM steel with Thermo-Mechanical Treatment (TMT) to improve the microstructure) have been developed with a focus on improving strength compared to the Eurofer generation steel and creep resistance at high temperature (up to ~ 650 °C) (fig 26).

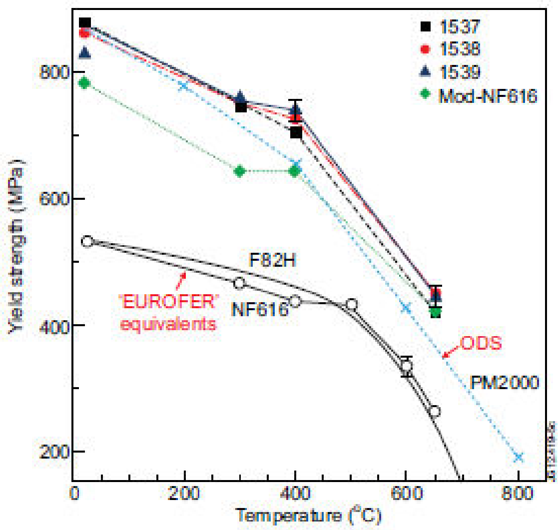


Fig. 26 - Yield stress of US Generation IV HT FM steels (32).

* A quantitative way of expressing a material's resistance to brittle fracture when a crack is present, a material with high fracture toughness is more prone to ductile fracture. Brittle fracture is characteristic of materials with less fracture toughness

The microstructure produced in these latest generation steel are expected to provide improved radiation resistance over a broad range of temperatures due to higher precipitate densities and other radiation defect sinks, which have the potential to improve the low temperature embrittlement issue. These steels are just entered fission irradiation testing in the US. More testing is needed, but it has been clearly demonstrated that substantial improvement in multiple mechanical properties before irradiation, including thermal creep strength, fracture toughness (DBTT and upper shelf toughness), etc., can be achieved compared to 1980-1990 grades steel.
Development of structural materials for large-scale energy application is historically a long and costly process, due to the extended period to develop a new alloy followed by an even longer proof testing period to validate the performance of the material in prototypic environments for appropriate licensing authorities.

5.4 Summary on neutron irradiation effects on Eurofer and other RAFM steel

Following are summarised key issues related to neutron irradiation on TBM and divertor made on Eurofer (29, 34).

1. As reported by several sources (16, 29, 34), 335 °C is the most critical irradiation temperature in terms of hardening and loss of ductility for RAFM steels. In fact low temperature neutron irradiation ($T_{irr} \leq 335$ °C) leads to strong hardening and embrittlement of Eurofer97. Minor hardening and embrittlement observed at Tirr = 350 ◦C indicates considerable healing of radiation damage. Neutron irradiation at Tirr ⩾ 400 ◦C has nearly no impact on the mechanical properties.

2. The increase of the Yield Stress of RAFM steels is rather steep at doses below 10 dpa at Tirr ⩽ 335 ◦C. In spite of large scattering of low temperature, high dose hardening data a clear reduction of the hardening per dose increment is observed at achieved damage doses of 70–80 dpa. Eurofer show saturation of hardening at 70 dpa. Ductility properties (Uniform Strain, Total Strain) seem to saturate already above 2.5 dpa.

3. All RAFM steels show steep increase in the ΔDBTT with dose below 15 dpa at Tirr ⩽ 335 ◦C. At 70 dpa Eurofer97 indicates saturation behaviour of low temperature embrittlement.

4. Post irradiation annealing at 550 ◦C for 3 h leads to a nearly complete recovery of the impact and tensile properties of low temperature (Tirr = 335 ◦C) irradiated Eurofer97 indicating substantial healing of radiation defects. In fusion reactor this heat treatment has to be done by He. This knowledge leads to the following proposed life cycle operation: six month operation followed by few hours annealing at 550 °C with He. During this annealing the impact properties strongly recover. Afterwards four month operation followed by few hours annealing at 550 °C with He. And finally one month of operation, than replacement (30).
Helium effects cannot be healed out by post irradiation annealing.

5. Helium effects are not sufficiently understood due to absence of an irradiation facility with fusion reactor relevant neutron spectrum and have to be a subject of future investigations. Different simulating techniques used, e.g. fission reactor irradiation of boron doped model steels or spallation neutron and proton irradiation of RAFM steels most probably strongly overestimate helium effects. Moreover, helium effects will strongly depend on the irradiation temperature. Boron doping technique yielded progressive embrittlement with increasing helium concentration (10 – 430 appm) at Tirr ⩽ 350 ◦C, the embrittlement rate was however reduced at achieved helium contents. Helium embrittlement was considerably reduced at Tirr = 450 ◦C. Synergistic helium and dpa effects under spallation target environment at Tirr ⩽ 380 ◦C yielded linear increase of DBTT shift with produced helium up to 1600 appm.

6. The RAFM steels are highly suited for the special fusion reactor design with the operating temperature range between 350 and 550 ◦C for the FW and BB. In the irradiation temperature range between 350 and 550 ◦C the application of Eurofer97 will be further limited by helium effects. Helium effects are not sufficiently

understood due to absence of an irradiation facility with fusion spectrum. By considering DEMO relevant helium to dpa ratio of 10 appm/dpa helium effects will be tolerable up to a damage dose of 40 dpa and up to helium contents of 400 appm He in the temperature window between 350 and 550 ◦C. The mechanical performance of RAFM steel at low irradiation temperatures (Tirr < 350 ◦C) will be mainly limited by dpa induced embrittlement. By utilizing post irradiation annealing of FW and BB structures for the recovery of dpa effects at low irradiation temperatures the helium effects will be tolerable up to a damage dose of at least 40 dpa

7. Irradiation behaviour with respect to impact properties is rather poor for early developed ODS Eurofer steels.

6. RCC-MRx design rules and corresponding stress limits

The selected C&S for the design of the divertor cassette is the RCC-MRx (13) because of the following reasons:

- it provides an ensemble of consistent design, manufacturing and materials rules, revised in 2013, in order to implement, in particular, regulation requirements related to PED directive (17) and French ESPN order (18);
- it provides rules for high temperature operation;
- it provides rules for the design of irradiated components;
- eurofer97 is already included in the material properties tables given by the code, although these tables must be completed;
- it is the reference C&S selected for the design and construction of the test blanket for ITER. The code will continue to be developed in the framework of the TBM programme for fusion relevant aspects (16).

The RCC-MRx code has three quality classes in the design and construction rules:

- Class 1 N1Rx;
- Class 2 N1Rx;
- Class 3 N1Rx.

They correspond from 1 to 3 to a decreasing confidence in the security level regarding the different mechanical damage modes to which the equipment could be submitted due to the loading induced by operating conditions.

The use of the RCC-MRx design and construction code in the context of contractual relations between customer and supplier is based on a list of the components to which the code applies and their “entrance keys” which make it possible to determine what rules are applicable. The “entrance keys” are quoted in the Equipment Specifications:

- Key 1. This key is applicable to:

1) Components of nuclear reactor and its auxiliary systems,
2) or examination, handling or drive mechanisms,
3) or components of irradiation devices.

- Key 2. This key gives the required RCC-MRx class:

1) Class N1Rx
2) Class N2Rx
3) Class N3Rx.

For Safety classified components, the relations between “Safety Classes” and “RCC-MRx classes” that must be applied are defined before the application of the Code. The Manufacturer must apply the required class to each type of component and its support. It may however apply a higher class, if needed, to simplify parts manufacturing for example. It must inform the Contractor of working to higher class as an exception. It must apply all the provisions of the higher class.

- Key°3. This key indicates the type of component to which the component is attached:

- Vessels, tanks, containers;
- Pumps;
- Valves;
- Piping;
- Bellows;
- Box structures;
- Heat exchangers.

- Key°4. This key indicates whether "Catalogue Component" or not, for irradiation devices, class 3 components of nuclear reactor and its auxiliary systems.

- Key°5. This key indicates whether component is subjected to pressure equipment regulations applicable in France (ESP Decree/ESPN Order).

- Key°6. Based on the value of keys 2 and 3, this key indicates all applicable rules among the following (Table 19):

- Section III;
- Section II REC 2200 as part of applying standard NF EN 13445;
- Section II REC 2300 as part of applying standard NF EN 13480;
- Section II REC 2400 as part of applying standard NF EN 1993-1-1.

As well as the following, based on the value of key 5:
- Section II REC 3200 if the component is subject to pressure equipment regulations - decree ESP/order ESPN.

The list of all applicable rules is defined in the Equipment Specification.

Key 1 **= Reactor components and associated auxiliaires** **= Examination or handling mechanisms** **= Irradiation devices**	**Applicable sets of rules**		**Key 5 = ESP/ESPN Special instructions**
Key 2 = Class $N1_{rx}$	Section III - **RA 4000**		Section II **REC 3200**
Key 2 = Class $N2_{Rx}$	Section III - **RA 4000**		Section II **REC 3200**
Key 2 = Class $N3_{Rx}$	Section III - **RA 4000**	Section II **REC 2200** (vessels) Section II **REC 2300** (piping) Section II **REC 2400** (metallic frameworks)	Section II **REC 3200**

Table 19 - Sets of applicable rules and special instructions.

The following table 20 shows the rules applicable according to keys 2 to 5 for ESP/ESPN component.

Table 1.3a: sets of applicable rules and specific provisions for an ESP or ESPN

	Specific design and manufacturing provisions	**Key 2 $N1_{Rx}$**	**Key 2 $N2_{Rx}$**	**Key 2 $N3_{Rx}$**
ESP Cat. I to IV	Arts. 3 to 6 and EES Annex 1 of ESP Decree.	Subsection B + REC 3231	Subsection C + REC 3231	Subsection D or REC 2000 + REC 3231
ESP art.7	Art.7 of ESP Decree. Best practices.	Subsection B + REC 3232	Subsection C + REC 3232	Subsection D or REC 2000 + REC 3232
ESPN (1) $N1_{ESPN}$ Cat. I to IV	EES Annex 1 ESP Art. 7, 8 and Annex 1 ESPN Order. Art. 9 and ER Annex 4 of ESPN Order.	Subsection B + REC 3233	Subsection B + REC 3233	Subsection B + REC 3233
ESPN (2) $N2_{ESPN}$ Cat. I to IV	EES Annex 1 ESP Art. 7, 8 and Annex 2 ESPN Order. Art. 9 and ER Annex 4 of ESPN Order.	Subsection B + REC 3234	Subsection C + REC 3234	Subsection C + REC 3234
ESPN $N3_{ESPN}$ Cat. I to IV	EES Annex 1 ESP Art. 7, 8 and Annex 3 ESPN Order. Art. 9 and ER Annex 4 of ESPN Order.	Subsection B + REC 3235	Subsection C + REC 3235	Subsection D or REC 2000 + REC 3235
ESPN $N1_{ESPN}$ Cat. 0	Art.6-II of ESPN Order. Best practices: trade manual submitted to applicable ministry. Art. 9 and ER Annex 4 of ESPN Order.	Subsection B + REC 3236	Subsection B + REC 3236	Subsection B + REC 3236
ESPN $N2_{ESPN}$ Cat. 0	Art.6-II of ESPN Order. Best practices: trade manual submitted to applicable ministry. Art. 9 and ER Annex 4 of ESPN Order.	Subsection B + REC 3237	Subsection C + REC 3237	Subsection D or REC 2000 + REC 3237
ESPN $N3_{ESPN}$ Cat. 0	Art.6-II of ESPN Order. Best practices. Art. 9 and ER Annex 4 of ESPN Order.	Subsection B + REC 3238	Subsection C + REC 3238	Subsection D or REC 2000 + REC 3238

(1) does not include pipes with NPS less than or equal to 100.
(2) includes Category I and II pipes from level $N1_{ESPN}$ with NPS less than or equal to 100.

Table 20 - Sets of applicable rules and specific provisions for an ESP or ESPN component.

As illustrated in the following table 21, Tome 1 Volume B-C-D L 3000 and Appendix A3 contain the design. and analysis rules.

Moreover table 21 shows the outline of RCC-MRx code (RDG 1200) and provides the acronyms used to identify the Sections, Tomes and Subsections.

Sections	**Tomes**	**Titles**	**Acronyms**
• **Section I**		**General provisions**	RDG
• **Section II**		**Additional requirements and special instructions**	REC
• **Section III**		**Rules for nuclear installations mechanical components**	
	• **Tome 1**	• ***Design and Construction rules***	
		- *Subsection A: general provisions for Section III*	RA
		- *Subsection B: class $N1_{Rx}$ reactor components, its auxiliary systems and supports*	RB
		- *Subsection C: class $N2_{Rx}$ reactor components, its auxiliary systems and supports*	RC
		- *Subsection D: class $N3_{Rx}$ reactor components, its auxiliary systems and supports*	RD
		- *Subsection K: examination, handling or drive mechanisms*	RK
		- *Subsection L: irradiation devices*	RL
		- *Subsection Z: technical appendices*	A1, ...
	• **Tome 2**	• ***Part and product procurement specifications***	RM
	• **Tome 3**	• ***Destructive tests and non-destructive examination methods***	RMC
	• **Tome 4**	• ***Welding***	RS
	• **Tome 5**	• ***Manufacturing operations other than welding***	RF
	• **Tome 6**	• ***Probationary Phase Rules***	RPP

Table 21 - Outline of RCC-MRx code and acronyms to identify the Sections, Tomes and Subsections.

6.1 Design by analysis

The general analysis rules are provided in chapter RB 3200 (Class N1Rx), RC 3200 (Class N2Rx), RK 3200 (Examination and handling mechanisms) and RL 3200 (Irradiation devices) of Section III. In addition the specific design rules for box structures (RB3800) are well adapted to the Divertor cassette structural layout.

These general analysis rules are applicable to structures made of material whose properties are listed in one of the Properties Groups in Appendices A3 and A9 for the specified operating conditions. In the case of a new product or new grade purchased under a Particular Procurement Specification, it must be accompanied by a new Properties Group as shown in Appendices A3 and A9.

In addition to the negligible creep test (RB 3216.1) a negligible irradiation test, based on ductility criteria, has been added in RB 3216.2. This test makes it possible to disregard the effects of irradiation if the fluence received by the component in question remains below a value specified in Appendix A3 (Properties Group) at the service temperature in question. If the test is meeting, the fluence is considered to be negligible and the rules "without irradiation" apply unrestrictedly.

The structure of RB 3200 is as follows:

	Negligible creep	Significant creep
Negligible irradiation	RB 3251.1 (Type P damages) RB 3261.1 (type S damages) identical to RCC-MR	RB 3252.1(Type P damages) RB 3262.1 (Type S damages) Identical to RCC-MR
Significant irradiation	RB 3251.2 (Type P damages) RB 3261.2 (Type S damages) New rules	RB 3252.2(Type P damages) RB 3262.2 (Type S damages) New rules

Table 22 - RB structure.

They aim of RB3000 rules is to ensure that the components are sufficiently safe under the various mechanical damages to which they could be exposed under loads in specified operating conditions:

-damages called **"P-type damages"** are those which could result from applying constantly increasing loads to a structure:

- excessive deformation,
- plastic instability,
- fracture.

-damages called **"S-type damages"** are those which could only result by repeatedly applying loads:

- progressive deformation,
- fatigue.

- buckling.

- fast fracture; usually two types of fast fracture are considered:

- from ductile tearing,
- from fragile or semi-fragile tearing.

These rules do not cover all measures to be taken to avoid other types of deterioration that could occur, for example from erosion or corrosion.
These rules also do not cover the measures to be taken to ensure proper operation of component with moving parts such as pumps and valves, because they are essentially aimed at confinement and support functions.

RB 3000 includes general design rules (RB 3100) and general analysis rules (RB 3200) as well as specific design rules for particular components (RB 3300 through 3900).
When these specific rules lead to configuration conditions more restrictive than allowed in the general rules,

the specific rules must be used.
The Equipment Specification (RB 3170) and the set of additional reference notes (operating condition notes, notes defining limit conditions, interface notes, load definition notes) include all the data needed to check the rules according the specified criteria levels.
All calculations made to check the rules contained in the chapters 3000 of Tome 1 shall be written in design calculation. These notes constitute the Design Report (RB, RC, RD, RK, and RL 3180).
The Design Report justifies that the requirements of the chapters 3000 of Tome 1 are satisfied when the component is subject to all the loading conditions specified in the Equipment Specification. This report also covers compliance with any additional requirements likely to appear in the Equipment Specification.

In the following we will explain only RB rules for level N1Rx. In fact design by analysis rules for RC 3200 is identical to those in RB 3200. RD 3200 is not supplied.
For irradiation devices RL 3000 rules, depending on the class, rules in RB 3200 (Class N1Rx) or RC 3200 (Class N2Rx) shall be applied.

6.2 DAMAGES (RB 3120)

6.2.1 Type P damages (RB 3121)

Types of damage referred to by the expression "type P damages" are those which can result from the application to a structure of a steadily and regularly increasing loading or a constant loading.

Type P damages have the same meaning as in ASME code; P means Primary, damages that are to be considered in first, because they could lead to the burst or the collapse of the structure if they are not limited. These damages are caused by primary loads (like constant pressure, forces, etc.) and not by displacement controlled loads (like temperature gradients), for ductile and hardening materials. Of course, it is well known than any displacement corresponds to a force, so that controlled displacement can be neglected in Type P damage analyses if the produced plasticity is not important.
Type P damages for the level A criteria will cover excessive strain, plastic instability and rupture. They are divided in two parts:
- instantaneous, i.e. damages due to plastic strains;
- differed, i.e. damages due to creep strains.

6.2.1.1 Immediate excessive deformation RB 3121.1

For a structure made of elastic, ductile material to which is applied a loading multiplied by a gradually increasing coefficient, the following behaviour can be observed: with lower coefficient values, the structure behaves elastically and deformation is reversible. At higher values, irreversible plastic deformations occur such that if the loading were to be cancelled, the structure would not return to its original dimensions or shape. These plastic deformations are firstly contained by elastic zones which limit them and then, the plastic zones being sufficiently extended, yielding takes place easily. The overall permanent deformation of the structure thus increases faster the higher the loading coefficient. It is when the overall permanent deformation begins to increase rapidly that it is said to be excessive.

6.2.1.2 Immediate plastic instability (RB 3121.2)

When, in the previous case, the loading continues to increase, the behaviour of the structure depends on any variations in its shape and the strain hardening increase of the yield strength of the material. These two effects rapidly become counteracting: any change in shape tends to weaken the structure whereas an increase in the yield strength of the material tends, on the contrary, to reinforce it. As long as the first effect is dominated by the second, the structure is deformed in a stable manner, when the first becomes dominant, deformation is

unstable and fracture is not far behind if the loading is maintained.
Plastic instability considered here is an overall phenomenon. It must be distinguished from ductile tearing which is a form of fast fracture and must be examined separately.

6.2.1.3 Time-dependent excessive deformation (creep) (RB 3121.3)

When a structure is subjected to loadings maintained for a sufficiently long time at high temperatures, deformations evolve with time and can consequently produce excessive deformation. This type of damage is called a time-dependent excessive deformation.

6.2.1.4 Time-dependent plastic instability (RB 3121.4)

Although inducing no immediate damage when applied, a loading can, because of creep, induce plastic instability over a certain period of time. This type of damage is called time-dependent plastic instability.

6.2.1.5 Time-dependent fracture (RB 3121.5)

In certain conditions, changes in shape prior to fracture can be small. The sometimes considerable reduction in the elongation at the time of rupture means that this phenomenon must be taken into account both globally (under the effect of external forces) and locally (fracture before complete release of internal stresses).

6.2.1.6 Elastic or elastoplastic instability (RB 3121.6)

Apart from the instabilities described above, other elastic or elastoplastic instabilities may occur, in which elastic deformation, by the changes in shape it induces, considerably weakens the strength of a structure and its ability to withstand the applied loading. The typical case of this type of damage is buckling as raised in RB 3123.

6.2.2 Type S damages (RB 3122)

Types of damage described by the expression "type S damages" are those which can only result from repeated application of loadings.

6.2.2.1 Progressive deformation (or ratcheting) (RB 3122.1)

When we consider a structure subjected to cyclic loading, at the end of the first cycle, the structure may show signs of permanent deformation. During the following cycles, two cases may arise:
- After a few cycles, the overall permanent deformation is stable;
- The permanent overall deformation continues to increase as every loading cycle induces additional deformation and the structure gradually changes from its original shape. This behaviour is called progressive deformation.

6.2.2.2 Fatigue (progressive cracking) (RB 3122.2)

When the loading applied to a structure evolves with time, in particular in a cyclic fashion, the material is subjected to deformation variations. These variations, if sufficiently numerous and if of large amplitude are capable of causing cracking. The damage here is defined by the appearance of small macroscopic cracks which do not compromise the strength of the structure with regard to the other types of damage to be considered. When the temperature is sufficiently high, creep deformation occurs during each cycle thus accelerating the appearance of cracking.

6.2.3 Buckling damages (RB 3123)

The Buckling correspond to the behaviour of some slim or slender structures which are subject to compression stresses. For a certain level of loads (critic load), the deformed shape of the structure change suddenly.

6.2.4 Fast fracture damages (non-ductile damage modes) (RB 3124)

Fast fracture is any fracture initiated from an existing defect or defects under monotonic loading which occurs without being preceded by an appreciable global plastic deformation. Fast fracture is generally caused by unstable propagation of a crack.
Two types of fast fracture are generally considered, one by ductile tearing, the other by fragile or semi-fragile tearing:

- ductile tearing: the material cracks with local detectable plastic strain;
- brittle tearing: the material cracks without local plastic deformation (plastic deformation within a microscopic volume of material).

6.3 Operating conditions and criteria level

6.3.1 OPERATING CONDITIONS (RB 3130)

During operation a component may be subjected to a number of different operating conditions which are classified under four categories according to considerations which do not necessarily concern only the problems posed by its mechanical strength as a structure.
The operating conditions for each component are classified as follows:

- the 1st category operating conditions (SF1) and 2nd category operating conditions (SF2) are the conditions to which component may be subjected in the course of normal operation, including normal operating incidents, start-up and shutdown.
- the 3rd category operating conditions (SF3), emergency conditions, corresponding to very low probability of occurrence but which must nonetheless be considered, and which imply shut down and appropriate inspection of the component or of the plant.
- the 4st category operating conditions (SF4), which are highly improbable but whose consequences on component are studied among others for safety reasons.

The list and classification of operating conditions must be defined in the Equipment Specification (RB 3170).

6.3.2 LOADING CONSIDERATIONS (RB 3140)

There is a set of environmental effects (pressures, forces, heat flux, irradiation, corrosion) corresponding to each operating condition. Some of these effects, which may produce mechanical work depending on the component deformation, are referred to as loads. Sets of simultaneous loads are referred to as “loadings”.
The loads include the following:
- Internal and external pressures,
- The weight of the component and its contents, and the static and dynamic loads produced by liquids under each condition analysed;
- Forces resulting from weight, thermal expansion, and pressure and dynamic loads which originate outside the zone studied and which are applied at its boundaries;
- Loads resulting from earthquakes and vibrations;

- Reactions of supports;
- Temperature effects, either constant or transient;
- Forces resulting from non-free swelling in irradiation conditions.

6.3.3 CRITERIA LEVELS (RB 3150)

The level (A, C and D) of criteria to be met must be defined in the Equipment Specification (RB 3170) for each loading associated with an operating condition or set of operating conditions.

These criteria levels have been established according to the objective defined in RB 3110 and they aim at the prevention of a certain number of damages for the component in question.

Table 23 below shows the level of Criteria and relative damages.

- Criteria of level A
  - Type P damage:
    - Fracture
    - Excessive deformation
    - Plastic instability / Failure
  - Type S damage:
    - Progressive deformation
    - Fatigue
  - Elastic or elastoplastic instability
  - The respect of the level A criteria ensures the security beside these damages
- Criteria of level C
  - Type P damage:
    - Excessive deformation
    - Plastic instability / Failure
  - Elastic or elastoplastic instability
  - There is not type S damage analysis. The number of cycle must be less than 10.
- Criteria of level D
  - Type P damage:
    - Plastic instability / Failure
  - Elastic or elastoplastic instability

Table 23 - Level Criteria and relative damages (36).

6.3.3.1 Level A criteria (RB 3151)

The aim of level A criteria is to protect the component against the following types of damage:

- Immediate or time-dependent excessive deformation;
- Immediate or time-dependent plastic instability;
- Time-dependent fracture;
- Elastic or elastoplastic instability, immediate or time-dependent;
- Progressive deformation;
- Fatigue.

The respect of level A criteria guarantees the level of safety required with regard to these types of damage throughout the life of the component, for operation as specified.

6.3.3.2 Level C criteria (RB 3153)

The aim of level C criteria is to protect the component against the following type of damages:

- Immediate or time-dependent excessive deformation;
- Immediate or time-dependent plastic instability;
- Time-dependent fracture;
- Elastic or elastoplastic instability, immediate or time-dependent.

6.3.3.3 Level D criteria (RB 3154)

The aim of level D criteria is to protect the component against the following types of damage:
- Immediate or time-dependent plastic instability,
- Time-dependent fracture,
- Elastic or elastoplastic instability, immediate or time-dependent.

But with lower safety margins than those with level C criteria.

6.3.4 MINIMUM CRITERIA LEVELS (RB 3160)

The level A criteria shall be met for the first and second category operating conditions (SF1 and SF2).
The criteria to be met for third category operating conditions (SF3) shall be at least as severe as those of level C.
The criteria to be met for fourth category operating conditions (SF4) shall be at least as severe as those of level D (table 24).

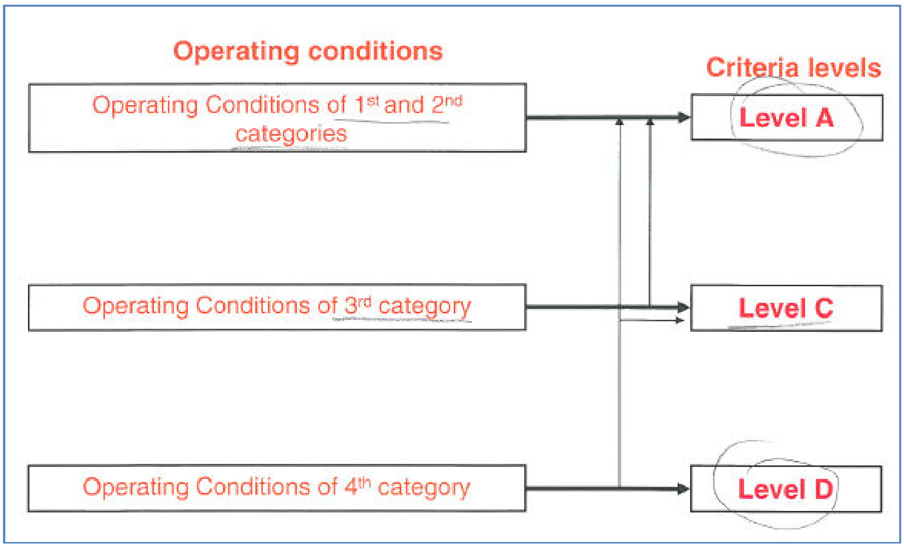


Table 24 - Category operating conditions and criteria levels (36)

In the following of the thesis only Level A criteria is considered since they are both the most conservative and comprehensive of all possible damage modes.

Limits for Level C and Level D criteria are usually derived from those of Level A using lower safety factors.

6.3.5 EQUIPMENT SPECIFICATION (RB 3170)

The Equipment Specification comprises at least:
- Description of the component with plans needed for calculation and providing functions, shapes, dimensions of component parts, limits and interconnections with other components;
- The list and definitions of conditions and corresponding loads;
- Identification of materials as well data for design;
- Levels of criteria A, C or D;
- Type and extent of maintenance and inspection actions;
- Provisions associated with seismic and fast fracture analysis.

6.3.6 DESIGN REPORT (RB 3180)

The Manufacturer shall prepare a Design Report for all components manufactured in accordance with this Code.
This report shall demonstrate:
-That the rules of this RB 3000 are respected for all loadings specified in the Equipment Specification;
-Demonstrate that any additional design requirements contained in the Equipment Specification are met.

The stress report has containing a design computation report.

The typical organization of these Design and Calculation Reports is:
- Introduction;
- Basic data;
- Materials;
- specified criteria;
- Calculations;
- Results and analysis;
- Conclusions.

6.4 General Analysis rules: RB 3200

The RB 3200 rules apply to structures made from materials whose properties are given in Section III (Appendix A3) (table 13) of RCC-MRx for the specified operating conditions.
The purpose of mechanical analyses is to demonstrate that a component does not undergo certain types of damage when subjected to the loadings associated with the conditions specified in the Equipment Specification.
Analyses consist in verifying compliance with criteria selected on the basis of the method of analysis and the level of criteria and the type of damage.

Three methods of analysis can be used in the code:
- Elastic analysis,
- Inelastic analysis,
- Experimental analysis.

The term elastic analysis designates analyses carried out on the assumption that the behaviour of the material is elastic and linear; that the displacements are small (geometrical linearity) and that there is no initial stress (or residual stress).
The term inelastic analysis designates all the other methods except for experimental analysis.
Experimental analysis consists in subjecting models representing the component or some of its elements to loadings in order to determine the deformation and stresses or margins with regard to the damage under study.

Elastic analysis should be the most commonly used method, the other methods of analysis only when some criteria associated to the elastic methods are not verified.

In the following thesis only elastic analysis will be pointed out. This means:
-the behaviour of the material is elastic and linear,
-the material is isotropic,
-the displacements and strains are small,
-the initial stresses are nil.

It may be necessary for technical reason to break a component up into several zones of calculation (RB3214) in order to analyse a single type of damage. In this case, an overall analysis of the component shall be made.

6.4.1 Applicable rules (RB 3216)

The rules to be respected differ according to:

- Level of criteria: A, C or D;
- Method of analysis: elastic, inelastic or experimental;
- Damage envisaged: type P or type S damages.

These rules also depend on three parameters: temperature, time and irradiation:

- Temperature because the properties of the material (allowable stresses, fatigue curves, etc.) often depend upon it;
- Time because its influence on the behaviour of the materials shows up in phenomena which can often be neglected at moderate temperatures but which, on the other hand, must be taken into account at high temperatures. These phenomena due to the effect of time are generally designated by the term creep;
- Irradiation because neutrons could, at moderate temperature, generates movement of atoms and transmutation that could lead to changes in the material's properties. At high temperature, irradiation could influence the behaviour of materials over time and thus contribute to creep ("irradiation" creep combined with "thermal" creep). At high temperature, neutrons could also cause gas swelling, coming from the formation and growth of cavities due to the vacancies created by irradiation.

6.4.2 Negligible creep tests (RB 3216.1)

The first step to apply in any case is the negligible creep rule. It is generally better to keep a structure in the negligible creep domain. Here it is necessary to consider the maximum temperature of the structure. The reason is that the negligible creep rule covers both Type P damages (related to mean stresses and strains in the thickness) and Type S damages (taking care of local stresses and strains).
For the first ones, the mean temperature could be sufficient to describe the effect of creep. But for the second, it is necessary to have the maximum local temperatures. To keep the rules in the RCC-MRx code in a simple form, the maximum temperature has been defined for a conservative approach.

The negligible creep rule is divided in two tests.

Test 1 - is defined with two parts:
- is the maximum temperature in the thickness during the whole life of the shell less than the negligible creep temperature of the material?

- does the structure respect $\sum_{i=1}^{N} \frac{t_i}{T_i} \leq 1$? Which means that for any operating situation i to be analysed with level A, C, D criteria, and corresponding to a precise maximum temperature in the thickness, the sum of the ratio of the operating time t_i to the allowed time T_i is limited to 1. The T_i is read on the negligible creep curve, given in the A3 appendix.

Test 2 - the same approach of test 1 but without considering situations analysed with level D criteria.

6.4.3 Negligible irradiation test (RB 3216.2)

For a component the irradiation is said to be negligible or insignificant for the entire duration of its operational life if the fluence received during entire operational life is lower than the value defined in chapter A3.32 (table 25) at the maximum operating temperature for the material involved.

θ (°C)	20°C to 375°C	375°C to 400°C	400°C to 425°C	425°C to550 °C
D_{lneg} (dpa)	2.75	2.6	2	2
D_{lmax} (dpa)	53	40	30	24

Table 25 - Irradiation damages for X2CrNiMo17-12-2(N) austenitic stainless steel (from A3.1S.32).

6.5 Stresses (RB 3224)

It is possible to build the following table 26 that summarizes all possible combinations of loading and rules to be applied with negligible or significant creep and irradiation.

	NEGLIGIBLE CREEP	SIGNIFICANT CREEP
NEGLIGIBLE IRRADIATION	RB 3251.1 (Type P damage) RB 3261.1 (Type S damage) RB 3271.1 (Buckling)	RB 3252.1 (Type P damage) RB 3262.1 (Type S damage)
SIGNIFICANT IRRADIATION	RB 3251.2 (Type P damage) RB 3261.2 (Type S damage)	RB 3252.2 (Type P damage) RB 3262.2 (Type S damage)

Table 26 - All possible combinations of loading and rules to be applied with negligible/significant creep and negligible/significant irradiation (34).

6.5.1 Classification of stresses (RB 3224.3)

Primary, secondary and peak stresses are defined.

The primary stress is defined as the fraction of the total stress which does not disappear after small permanent deformation (stresses that balance the mechanical efforts: e.g. pressure, etc.).

Secondary stress is the fraction of the total stress which can disappear as a result of small permanent deformation, minus the peak stresses (e.g. thermal stresses, due to imposed displacements, etc.).

The peak stress is the fraction of the total stress which meets the following two conditions:

- It is the additional stress applied by a geometrical discontinuity of the structure or by non-linearity in the distribution of stresses within the thickness;
- This additional stress cannot cause deformation of the whole structure.

6.5.2 Negligible creep and negligible irradiation

In the following only rules for Level A criteria will be outlined.

6.5.2.1 TYPE P DAMAGES (RB 3251.1)

Two modes of failure directly related to the primary stress intensity in the material are the immediate plastic collapse and immediate plastic instability (37). If a structure is loaded above the yield strength of the material, plastic deformation occurs until the structure collapses either because of excessive deformation or necking.
We have to verify the classical limitation of primary stresses:

$$\overline{P_m} \leq S_m$$

$$\overline{P_L} \leq 1.5\ S_m$$

$$\overline{P_L + P_b} \leq 1.5\ S_m$$

Where:
P_m= primary membrane stress
P_b= primary bending stress
P_L= local membrane stress
S_m = Allowable stress given in A3.43 of the code with $R_{p0.2}^{t}$ and R_m

The explanation for the 1.5 coefficient multiplying bending P_b and local primary stress P_L is that these kinds of stresses cannot lead to necking and rupture but they can produce large strains, and they must be limited. But the limitation is not so severe compared to the limitation of the general primary membrane stress P_m. The temperature Θ_m to be considered here is the mean temperature in the thickness, since the load is sustained by the whole thickness of the component.

There is no limitation of secondary stresses.

Moreover we have to verify the measuring defect susceptibility.

$$J(a, C) \leq J_{IC}$$

Where:
C= the mechanical or thermal load;
a= the depth of conventional defect;
J_{IC}= characteristic value of the initiation;
J= driving force.

6.5.2.2 Type S damage (RB 3261.1)

6.5.2.2.1 Ratcheting damage (RB 3261.111)

Ratcheting is the accumulation of plastic deformation in structures subjected to cyclic stressing with a non-zero primary stress. When 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, two cases may arise:

a) After a few cycles, the overall permanent deformation is stable. The subsequent structural response is elastic or elastoplastic and progressive incremental inelastic deformation is absent;
b) The permanent overall deformation continues to increase as every loading cycle induces additional deformation and the structure gradually changes from its original shape until it eventually collapses. This behavior is called progressive deformation or ratcheting.

With ratcheting, there is a risk that a structure could malfunction. Also, the rules to prevent fatigue damage are not valid if ratcheting occurs. Ratcheting rules require the definition of a period, which is considered as the assembly of all the situations to be analyses with level A criteria: here, this period is the sum of all the cycles, and the corresponding time is the total time life of the structure.

The criterion is based on efficiency diagram.

The objective is to calculate the equivalent of primary stress giving the same damages of the combination of a constant primary stress and a secondary stress range. This equivalent stress is compared to S_m.

The approach used in RCC-MRx to prevent ratcheting is based on the concept of the ‘‘effective primary stress’’ (P_{eff}), that is an equivalent stress that would give the same immediate deformation as the actual cycling load combination. Values of P_{eff} are determined experimentally. Limits are then defined in terms of this effective primary stress. Before obtaining the effective primary stress intensity, it necessary to calculate the relative variation of secondary stress in relation to the primary stress considered. This introduces the notion of secondary ratio (SR):

$$SR_1 = \frac{\overline{\Delta Q_{\max}}}{\max(\sigma_m)}$$

$$SR_2 = \frac{\overline{\Delta Q_{\max}}}{\max(\sigma_L + \sigma_b)}$$

Where max (σ_m) and max ($\sigma_l + \sigma_b$) are stresses depending of the type of loadings:
- Presence of the secondary membrane stress Q_m;
- Short term overstress.

Four methods exist for the calculation of $Max(\sigma_m)$ and $Max(\sigma_L + \sigma_b)$.

Once the values of the secondary ratios are known, the corresponding effective primary stresses can be obtained by means of the efficiency diagram (Fig. 31):

$$P_1 = \frac{Max(\sigma_m)}{V_1}$$

$$P_2 = \frac{Max(\sigma_L + \sigma_b)}{V_2}$$

Where V_1 and V_2 are the efficiency indexes obtained from SR1 and SR2 using the efficiency diagram.

The following limits need then to be verified:

$$P_1 \leq 1.3\, S_m$$

$$P_2 \leq 1.3 \text{ x } 1.5 \text{ x } S_m$$

This means that the effective primary membrane stress intensity P_1, calculated on the basis of a period covering all the events with loadings for which compliance with level A criteria is required, should not exceed 1.3 times the value of S_m.
The effective primary stress intensity of the sum of primary stresses P_2, calculated over the same period as previously, should not exceed 1.3 x 1.5 times the value of S_m.

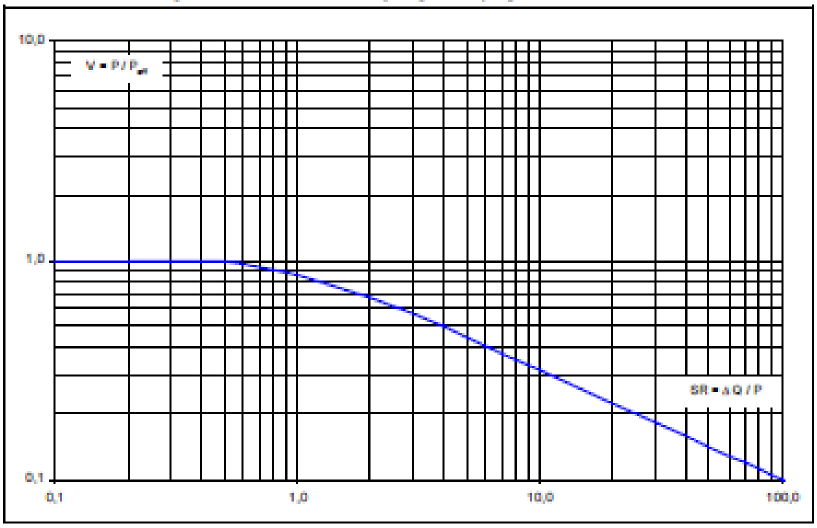


Fig. 27 - Typical efficiency diagram for progressive deformation.

Or as alternative rule (RB3261.1118)

$$\text{Max } \overline{P_L + P_b} + \overline{\Delta Q} \leq 3 \, S_m$$

It must be noted that while 3Sm rule is extremely conservative for Eurofer (and 9%Cr steel in general); the present efficiency diagram in the code is not suited for this type of steel.

6.5.2.2.2 Rule for fatigue damage (RB 3261.112)

Fatigue is a damaging mode that may appear under repeated loading. If the cyclic loads are too high, cracks can appear, propagate, and lead to the rupture of the component. Such cracks will appear most of the time at the surface of the component, where the stresses are higher. The total stress variation has to be considered everywhere in the structure, and more precisely, the total strain variation during a cycle.

The rules of RB 3261.112 only apply if the rules of RB 3261.111 (progressive deformation) are satisfied.

Use of fatigue curves (given in A3.47, fig. 28) and calculation of fatigue usage factor V= specified N_{cycles} /allowable N_{cycles} that must be lower than 1:

V $\leq$ 1

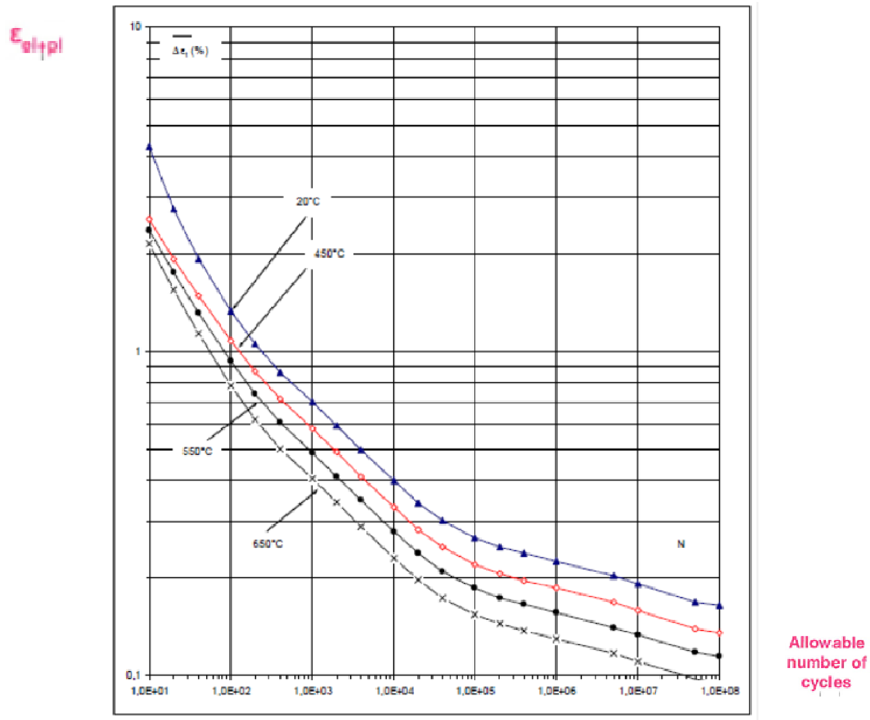


Fig. 28 - Typical fatigue curve.

6.5.3 Negligible creep and significant irradiation

6.5.3.1 Type P damage (3251.2)

Primary stresses being limited by S_m, conventional design codes rely on sufficient ductility of the materials to simplify the analyses, ignoring secondary and peak stresses (apart from their effect on cyclic loadings like ratcheting and fatigue) since they are intrinsically self-limiting and cannot cause, by themselves, the failure of the structure. When the ductility of the material is reduced (intrinsically or because of neutron irradiation), it is however necessary to insure that the combined strain due to primary plus secondary stresses does not exceed the remaining elongation of the material.

Two different modes of potential failure due to the limited ductility of the materials can be defined: immediate plastic flow localization and immediate local fracture due to exhaustion of ductility (which is associated with a low total elongation). Given that irradiated Eurofer97 retains considerable ductility after necking, as is shown by the high values of the total elongation and reduction of area, the latter is not an issue. Immediate plastic flow localization is instead a concern for Eurofer97 (16).

We have to verify the rules for non-irradiated material with limitation of primary, secondary and peak stresses:

$$\overline{P_m + Q_m} \leq S_{em}^A$$

$$\overline{P_L + P_b + Q + F} \leq S_{et}^A$$

And also the defect of susceptibility:

$$J(a, C) \leq J_{IC} \text{ (irradiated)}$$

Where:

$\overline{P_m + Q_m}$ = general membrane primary and secondary stress

$\overline{P_L + P_b + Q + F}$ = total primary and secondary equivalent stress

S_{em}^{A} = the allowable elastic membrane stress (in A3.63)

S_{et}^{A} = the allowable elastic total stress (in A3.63)

6.5.3.2 Type S damage (RB 3261.2)

6.5.3.2.1 Ratcheting

Apply analysis with non-irradiated material (RB 3261.111)

6.5.3.2.1 Fatigue

Fatigue curves under irradiation. If these curves are not given in appendix A3 it is possible to use fatigue curves without irradiation

6.5.4 Significant Creep and Negligible Irradiation

6.5.4.1 Type P damage (3252.111)

First the rules of 3251(negligible creep) must be checked (37).

Then calculate the creep usage fraction (defined in RB 3226.1) associated with primary membrane stresses multiplied by a creep correction factor Ω:

Rules:

$$U_{A,C}(\Omega\overline{P_m}) \leq 1$$

Where Ω is determinate as following:
If all membrane stresses, whatever the cause, are classified under general primary membrane stresses, Ω is then equal to 1;

If not $\Omega = \Omega_1 + \Omega_2$

With $\Omega_1 = 1 + 0.2\,(\frac{\overline{Lm}}{P_m})$ if $\overline{P_L} > \overline{P_m}$ and $\Omega_1 = 1$ if $\overline{P_L} < \overline{P_m}$

Calculate the creep usage faction (RB 3226.1) associated with the sum of the primary membrane stresses and primary bending stresses multiplied by coefficient Φ (depending of the geometry of cross section concerned)

And

$$U_{A,C}(\overline{P_L + \Phi\, P_b}) \leq 1$$

The primary bending stress is multiplied by factor Φ that takes into account the effect of creep on this stress category. Coefficient Φ depends on the geometry of the cross-section concerned; it is equal to 0.8 for plate elements and thin-wall shells with a rectangular cross-section.

$U_{A,C}$ is calculated by a linear damage rule summation:

$$U_{A,C}(\bar{\sigma}) = \Sigma_{i\varepsilon A,C}\frac{t_i}{T_i}$$

Where t_i is the time of duration of loading situation i, T_i the allowed time under a given stress, the latter being calculated on the S_t creep stress curves at the mean temperature in the thickness Θ_m.
S_t has the same meaning as in ASME code and is given inside A3 appendix.

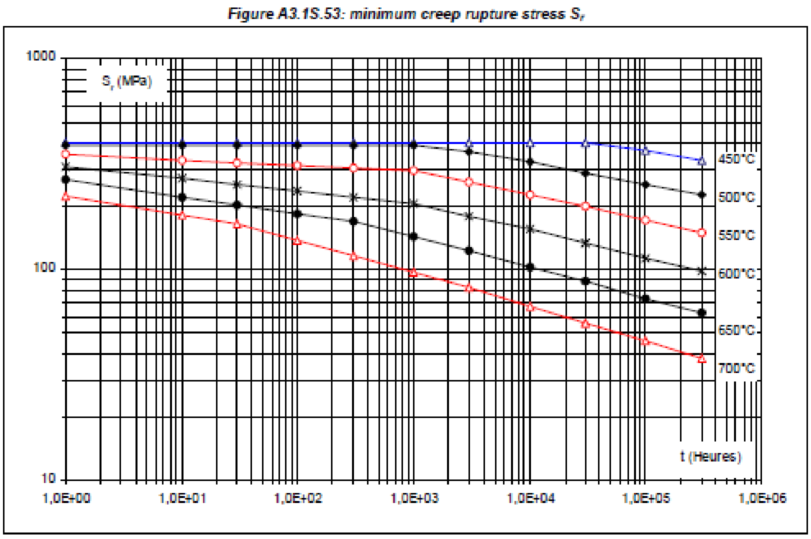


Fig. 29 - S_t curves from A3.53.

6.5.4.2 Type S damage (RB 3262.1)

6.5.4.2.1 Ratcheting

1) Verification of negligible creep rule (RB 3262.11)
2) The following limits must be checked at all points of the structure:

- Plastic strain + associated creep strain at 1.25 times the effective primary membrane stress intensity P_1 (RB 3262.1114) should not excide 1%;
- Plastic strain + associated creep strain at 1.25 times the effective primary stress intensity of the sum of primary stresses corrected by a factor of creep P_3 (RB 3262.1116) should not excide 2%

These rules are for austenitic stainless steels. No rules for the other steel for ratcheting (study in progress).

6.5.4.2.2 Fatigue-creep

Fatigue is a damaging mode that may appear under repeated loading. If the cyclic loads are too high, cracks can appear, propagate, and lead to the rupture of the component. Such cracks will appear most of the time at the surface of the component, where the stresses are higher. The total stress variation has to be considered everywhere in the structure, and more precisely, the total strain variation during a cycle. Moreover, the mechanical properties have to be taken at the local temperature, in the potentially damaged zone.
The design rules of RCC-MRx propose to avoid crack initiation by mean of the fatigue usage factor V_A for all possible cycles among level A:

$$V_A = \Sigma_i \frac{n_i}{N_i(\overline{\Delta\mathcal{E}})}$$

n_i being the number of cycles really applied to the structure, and N_i being the allowed number of cycles for the strain variation $\overline{\Delta\mathcal{E}}$. This strain variation is the addition of the elastic plus plastic strain variation $\overline{\Delta\mathcal{E}_{el+pl}}$ and of the creep strain variation$\overline{\Delta\mathcal{E}_{fl}}$: these latter accounts for the fact that creep or relaxation will increase the fatigue damage.

Limitation of creep-fatigue damage are reported in RB3262.1122:

-Calculate the fatigue usage factor V = specific number of cycles/allowable number of cycles;
-Calculate the creep fracture usage fraction (W = specific holding time/ allowable holding time with relaxation stress).

As criteria we have that all points V, W are located within an allowable area defined in creep-fatigue interaction diagrams in A3.55 (fig. 30).

Figure A3.1S.55: fatigue-creep interaction diagram

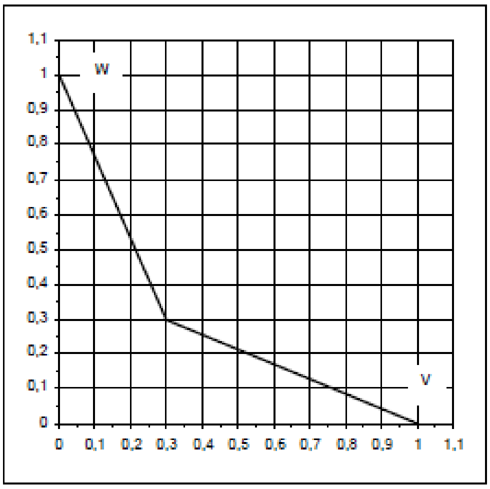


Fig. 30 - Creep-fatigue diagram.

6.5.5 Significant Creep and significant irradiation

For solution annealed or work –hardened austenitic stainless steel type of 316 and 316 L:
-The temperature domain is less than 625 °C;
-The irradiation damage remains less than the maximum allowable dpa number (A3.33)

6.5.5.1 P damage (RB 3252.2)

As a rule we have:

The rules of 3252.1 (significant creep and negligible irradiation) must be checked using the limits on usage fractions U and W equal to 0.1 instead of 1.

6.5.5.2 S damage (RB 3262.2)

The rule of 3262.1 (significant creep and negligible irradiation) must be checked multiplying the creep rupture usage fraction W by 10.

6.5.6 Rules for Buckling RB 3270

Buckling is a phenomenon which can occur in structures with an average centerline or average surface area. It consists in the development of deformations different from those which manifest themselves at low loading levels.
Buckling is not strictly speaking a type of damage but its appearance generally induces damage such as elastoplastic instability or excessive deformation or fatigue.

6.5.6.1 Negligible creep and negligible irradiation RB 3271.1

No elastic nor elastoplastic instability under a loading equal to 2.5 the specific loading.

6.5.6.2 Negligible creep and significant irradiation RB 3271.2
Later

6.5.6.3 Significant creep and negligible irradiation RB 3272.2
No elastic or elastoplastic instability under a loading equal to 1.5 the specific loading.

6.5.6.4 Significant creep and significant irradiation RB 3272.2
Later

6.6 Equipment subject to French regulations ESP/ESPN and RCC-MRx

According to French regulations, Pressure Equipment (ESP) and Nuclear Pressure Equipment (ESPN) (see Annex 5) are classified into categories based on pressure, by type of equipment (vessels, pipes and accessories), by type of fluid (gas or liquid) and by fluid group (group 1: explosive, dangerous – group 2: other fluids). On the other hand, Nuclear Pressure Equipment (ESPN) are classified into three levels (from $N1_{ESPN}$ to $N3_{ESPN}$), based on the amount of radiation released if they fail.

For equipment subject to both RCC-MRx and French ESP/ESPN regulations, all RCC-MRx rules and specific minimum provisions are therefore a function of the specified RCC-MRx classes (N1Rx, N2Rx or N3Rx) and of the ESPN levels ($N1_{ESPN}$, $N2_{ESPN}$, $N3_{ESPN}$) as shown in table 20.

REC 3200 of section II of RCC-MRx lists the specific mandatory provisions for design and manufacturing, matching all the applicable rules of RCC-MRx to those in the ESP decree and the ESPN order.

For example the requirements for the values for acceptance of material for the pressure parts and the requirements for regulatory hydraulic pressure test are:

-the minimum acceptance values for percentage elongation after fracture A at ambient temperature and the average values of the energy KV absorbed by breakage on three Charpy V-notch test specimens used in manufacturing level $N1_{ESPN}$, $N2_{ESPN}$ and $N3_{ESPN}$ pressure equipment are shown in table 27; they can be found under REC 3254.1, REC 3264 and REC 3274, as well as in Tome 2 in RM 012-1.

-the French regulatory hydrostatic pressure test is covered specifically in REC 3257.4 because the test is performed differently in each country. Based on the ESP decree and the harmonized standard NF EN 13445, the hydrostatic pressure test is performed on each equipment. It is deemed satisfactory if the pressure is reached without leakage or visible deformation based on direct visual examination. For class 1 (N1Rx) or 2 (N2Rx) vessels sized under RB 3000 or RC 3000, Pressure Test(PT) is defined by the following formula:

$$PT = Max\left[\left\{1.25 \cdot PD \cdot \frac{S_{mA}(T_{test})}{S_{mA}(T)}\right\};\{1.43 \cdot PS\}\right]$$

The purpose of the pressure test is not to dimension pressure equipment, the test pressure PT shall therefore be limited so as not to cause a higher membrane stress than is allowed for level C:

$$S_{mC}(Ttest) = Min[\ 1.35xS_{mA}(T_{test})\ ;\ R_{p0.2}(T_{test})\ ].$$

Where:
-PS: maximum allowable pressure. PS cannot be less than the pressure in the equipment for 1st category (SF1) and 2nd category (SF2) operating conditions to which the equipment would be subjected during normal operation conditions, including normal operating incidents, start up and shut down;
-PD: Design pressure (RB 3324.2)
-T: temperature associated with PD
-T_{test}: test temperature
-$S_{mA}(T_{test})$: allowable stress at the test temperature T_{test} (A3.43)
$S_{mA}(T)$: allowable stress at the temperature T (A3.43)

The ratio between $S_{mA}(T_{test})/S_{mA}(T)$ depends on the material used for the section in question. The value of the ratio used when calculating PT must not be less than the smallest ratio obtained by assuming different materials for the main pressurized sections (e.g. shells, heads, exchanger tube sheets, tube bundles, main body flanges, but ignoring the bolts associated with the main body flanges).

For Class N3 vessels sized using RD 3000, test pressure is defined by the same formula except for replacing S_{mA} by the allowable stress SA (A3.43), provided that the test pressure PT is limited so as not to cause a higher membrane stress than this allowed for level C: $SC(T_{test}) = 1.5\ SA(Tt_{est})$.

Table 1.3b: acceptance values for steels of Nuclear Pressure Equipment ESPN

Acceptance values for $N1_{ESPN}$ Nuclear Pressure Equipment		
Ferritic steels	A ≥ 20 % at room temperature	KV ≥ 40 J at 0 °C, if R_m < 600 Mpa KV ≥ 60 J at 0 °C, if 600 MPa ≤ R_m ≤ 800 MPa (R_m at A at room temperature)
Austenitic and austenitic-ferritic steels	A ≥ 35 % at room temperature	KV ≥ 100 J at room temperature and R_m ≤ 800 MPa at room temperature
	If A ≥ 45 % at room temperature	No Charpy impact test requirement on ISO V test specimen
Martensitic steels	A ≥ 14 %, at room temperature	KV ≥ 40 J at 0 °C and $R_{p0.2}$ / R_m ≤ 0,85 at room temperature
Bolting steels	A ≥ 12 % at room temperature	KV ≥ 40 J à 0 °C(1) and reduction in area ≥ 0,45 if 12 % ≤ A < 14 % at room temperature
(1) or, for bolting austenitic steels, KV ≥ 50 J at room temperature		
Acceptance values for $N2_{ESPN}$ Nuclear Pressure Equipment		
Ferritic steels	A ≥ 14 % at room temperature	KV ≥ 27 J at 0 °C
Austenitic and austenitic-ferritic steels	A ≥ 25 % at room temperature	KV ≥ 60 J at 20°C
	IF A ≥ 45 % at room temperature	No Charpy impact test requirement on ISO V test specimen
Bolting steels	A ≥ 12 % at room temperature	KV ≥ 40 J at 0 °C (1) and reduction in area ≥ 0,45 if 12 % ≤ A < 14 % at room temperature
(1) or, for bolting austenitic steels, KV ≥ 50 J at room temperature		
Acceptance values for $N3_{ESPN}$ Nuclear Pressure Equipment		
Ferritic steels	A ≥ 14 % at room temperature	KV ≥ 27 J at the lowest operating temperature or at 20 °C

Tab. 27 - Acceptance values for steels of ESPN.

7. Rules for prevention damages for Demo divertor cassette body

On the basis of a preliminary evaluations, the divertor cassette body can be classified as ESPN N2 level (release of 370 GBq after failure), Category IV (depending of pressure hazard basis). As RCC-MRx class let's consider this in-vessel component inside SF1 operating category and N1Rx (RB design class rules) (it could also be inside the N2Rx class corresponding to a decreasing levels of assurance of ability to withstand different types of mechanical damages to which the component might be exposed).

In this thesis we will also illustrates the competent design rules in case we can classify the cassette body outside the ESP/ESPN legislation as it has been done with ITER divertor. In fact for ITER divertor, according to point h), Part II, article 2 of the Decree 99-1046 (38), the equipment has been considered as "...not significant design factor" as defined in ESP, even if it operates at absolute pressure greater than 1.5 bar. For divertor design the main driving loads are electromagnetic, which are significantly larger than water pressure for the current reference design of the divertor, which allows to conclude that divertor is outside of scope of ESP and consequently of ESPN.
In addition to this let's also consider that in case of failure of the divertor cassette the first confinement barrier for a fusion reactor should be the vacuum vessel.

As it has been written in par. 3.6 there are 2 options for the cooling system of divertor system and so we can consider the cassette to be cooled with water at low temperature (150-200 °C and 5 MPa) or with higher temperature water –PWR condition (300 °C 15-16 MPa). This means that we have to verify the component or at "significant creep and significative irradiation" or at "negligible creep and significative irradiation" (see table 28).

Negligible creep and significative irradiation (low temperature)	Significative creep and significative irradiation (high temperature)
Inside the PED/ESPN	Inside the PED/ESPN
Outside PED/ESPN	Outside PED/ESPN

Table 28 - Possible combination of rules for divertor cassette.

7.1 Inside PED/ESPN at negligible creep and significative irradiation – low temp

Applying table 20 we have to verify the component to rules of sub-section B and REC 3234.
Let's consider before the subsection B and separately the REC 3234.
For the first category, according to RB3160, level A criteria shall be met (see table 23)

First we have to perform the negligible creep test RB 3216.1 as reported in par. 6.4.2.

The negligible creep curve is not present in A3 appendix of RCC-MRx; the S_t curve in fig. 16 par. 5.1.1.2 is available (16).

Appendix A3 for Eurofer is in the probationary part of the code (Tome 6 of section III). Probationary phase rule is a document which defines a rule for a technical field with a current status which is insufficiently defined to permit direct inclusion in the Code.

For Eurofer is reported: "Given the information currently available, use of the material is limited to the creep and negligible irradiation range".
In the paragraph A3.19AS.31 of negligible creep curve is reported: "**For temperatures less than 375 °C, creep is negligible whatever the holding time** (Appendix A3 Eurofer)".

When creep is negligible the following table 29 shows the various damages related to subsection B to take into account:

	Criteria level	Negligible irradiation	Significant irradiation
Type P damages	Level A Level C Level D – test 1 satisfied Level D – test 2 satisfied	RB 3251.11 RB 3251.12 RB 3251.13 RB 3252.13 (1)	RB 3251.21 RB 3251.22 RB 3251.23
Type S damages	Level A	RB 3261.1	RB 3261.2
Buckling		RB 3271	later
Bolts		RB 3280	later
(1) with a limit of 0.9 instead of 1 if only test 2 of RB 3216.12 is satisfied.			

Table 29 - Damages related to negligible creep.

7.1.1 Negligible creep and significative irradiation

Type P damage (3251.2)

We have to verify the rules for non-irradiated material with limitation of primary, secondary and peak stresses as in par. 6.5.2.1

Type S damage (RB 3261.2)

Ratcheting

Apply analysis with non-irradiated material (RB 3261.111) – see par. 6.5.2.2.1

Fatigue

Fatigue curves under irradiation. If these curves are not given in appendix A3 it is possible to use fatigue curves without irradiation.
Fatigue curves for non-irradiated Eurofer are given in A3.47 (Tome 6 of section III) of the Code.

Table A3.19AS.47: fatigue curves $\overline{\Delta\varepsilon_t}$ (%)

Number of cycles	20 - 450°C	500-550°C	650°C
20	1.549	1.335	1.41
40	1.306	1.066	1.125
100	1.043	0.801	0.841
200	0.882	0.652	0.679
400	0.747	0.536	0.552
10^3	0.602	0.422	0.426
2.10^3	0.513	0.358	0.354
4.10^3	0.439	0.309	0.298
10^4	0.358	0.259	0.242
2.10^4	0.309	0.232	0.21
4.10^4	0.268	0.210	0.185
10^5	0.224	0.189	0.16
2.10^5	0.196	0.177	0.146
4.10^5	0.174	0.168	0.135
10^6	0.159	0.159	0.124

Tab 30 - Fatigue dates for Eurofer.

Figure A3.19AS.47: fatigue curves $\overline{\Delta\varepsilon_t}$ (%)

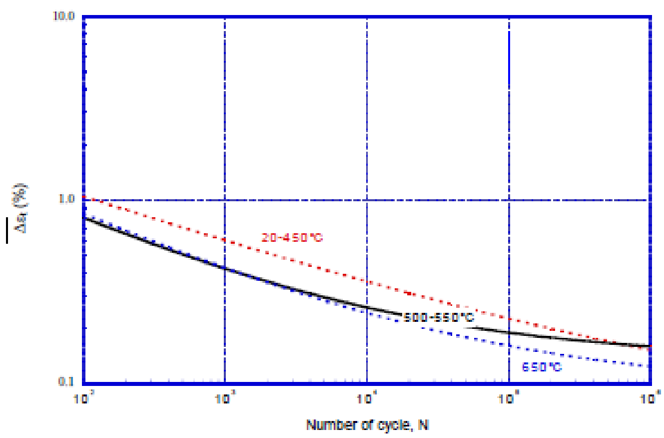


Fig. 31 - Fatigue curves for Eurofer.

7.1.2 REC 3234

For the Level $N1_{ESPN}$ pipes from Categories I and II with nominal pipe sizes less than or equal to 100, Class 2 RCC-MRx (N2Rx) is required at least for the ESPN Categories I to IV that are classified as $N2_{ESPN}$,along with the following specific provisions:

- REC 3250 for compliance with the Essential Safety Requirements of Annex 1 of the ESP Decree,
- REC 3270 for compliance with the Essential Safety Requirements of Annex 2 of the ESP Order,
- REC 3290 for compliance with the radiation protection requirements of Annex 4 of the ESPN Order.

Class 1 RCC-MRx (N1Rx) is required if needed for the entrance key in RCC-MRx (RDG 4000).

7.2 Outside PED/ESPN at negligible creep and significative irradiation – low temp

The same as par. 7.1 but not applying REC 3234.

7.3 Inside PED/ESPN at significative creep and significative irradiation – high temp

The following table 31 shows the damages to take into account when creep is significative

	Criteria level	Negligible irradiation	Significant irradiation
Type P damages	Level A Level C Level D	RB 3252.11 RB 3252.12 RB 3252.13	RB 3252.2
Type S damages	Level A	RB 3262.1	RB 3262.2
Buckling		RB 3272	later
Bolts		RB 3280	later

Tab. 31 - Damages for significative creep.

7.3.1 Significative creep and significative irradiation

P damage (RB 3252.2)

See par. 6.5.5.1

S damage (RB 3252.2)

See par. 6.5.5.2

Fatigue-creep

As criteria we have that all points V, W are located within an allowable area defined in creep-fatigue interaction diagrams in A3.55.

In the code there isn't such diagram but it is available the following one (30).

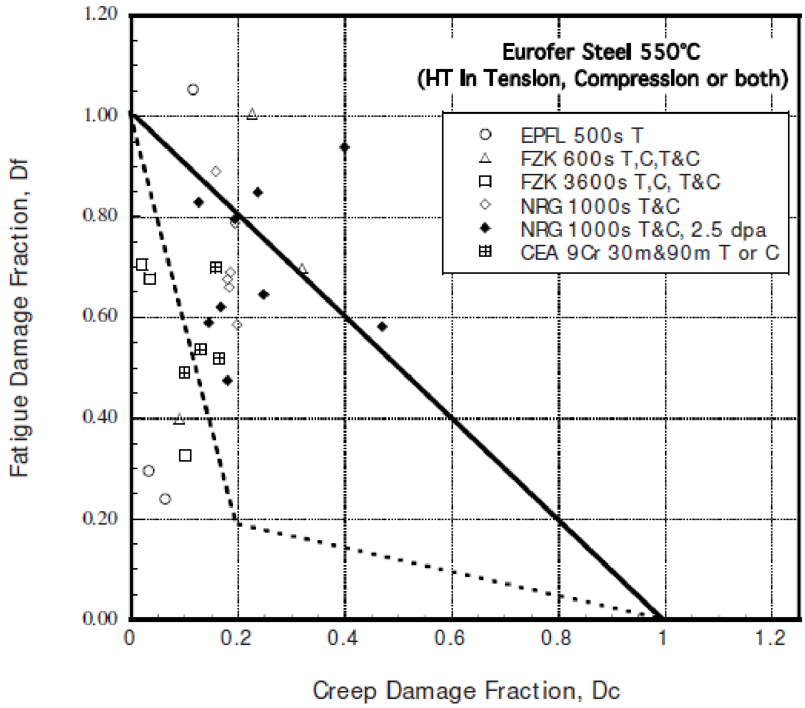


Fig. 32 - Tentative creep-fatigue interaction diagram recommended for Eurofer (30).

7.3.2 REC 3234

See par. 7.1.2.

7.4 Outside the PED/ESPN at significative creep and significative irradiation – high temp

The same as par. 7.3 but not applying REC 3234.

8. Conclusions

The development of a divertor concept for post ITER fusion power plants is deemed to be an urgent task. The divertor is one of the high heat flux components of the fusion reactor. Its developing involves many uncertainties including physical boundaries and candidate materials.

The principal objectives that we posed at the beginning of present thesis have been:

1. describe EU-DEMO and particularly its divertor cassette body identifying all the significant loads (and their combinations) for verifying its structural integrity (chapters 2 and 3, annex 1 and 2);
2. investigate all design codes and standards (C&S) for nuclear components (defined as set of rules to assure structural integrity of components through service and compliance against regulatory requirements) (chapter 4);

3. investigate on possible structural materials for in-vessel components (chapter 5);
4. investigate on mechanical properties of Eurofer97, with or without neutron irradiation, as principal candidate for in-vessel structural material (chapter 5 and annex 3);
5. analyse RCC-MRx design rules for mechanical components of fusion reactor (chapter 6 and annex 4);
6. apply RCC-MRx design rules to the DEMO divertor cassette body (chapter 7 and annex 5).

The main conclusions of the present thesis are:

Demo and its divertor cassette body

Demo has to resolve all physics and technical issues foreseen in ITER and demonstrate reactor relevant technologies.
The study on EU Demo divertor design is on-going. Some preliminary engineering estimation of main loads and combinations on divertor cassette (mainly derived from ITER design) have been done and analysed.

C&S for nuclear components

All relevant C&S available for fusion reactors have been investigated. Attention has been posed on RCC-MRx code developed by AFCEN since this code is emerging as a European code for mechanical components of nuclear installations. In RCC-MRx Eurofer is in the probationary part of the code and various data are not present.
This year in December we will expect new release of RCC-MRx code by AFCEN and also a first version of Division 4 (fusion devices) of ASME III.

Fusion structural materials

A brief overview on fusion structural materials such as SiC_f/SiC composites (SiC fibers in SiC matrix), Vanadium alloy and mainly "Reduced activation ferritic/martensitic steels" (RAFM) (and particularly Eurofer97) has been done. In particular we found that Reduced Activation Ferritic Martensitic (RAFM) steel is the primary choice material for Test Blanket Modules structural application for ITER and in future fusion power plants with operating temperature range between 350 and 550 ◦C .
These steels have been developed in order to simplify special waste storage of highly radioactive structures of fusion reactor after service. Eurofer 97 is the reference RAFM steel developed for the requirements of the European fusion technology program. The reasons for selection of RAFM steels are:

- good overall balance of mechanical properties (strength ductility, fracture toughness, creep resistance, fatigue resistance);
- the properties can be adjusted and to some extent tailored (e.g. by chemical composition but even more from heat treatments);
- there is broad industrial experience with FM steels in nuclear and non-nuclear facilities (as compared to other materials like Vanadium alloys);
- they have superior properties to austenitic steels:
  - in terms of thermo-physical properties and thermal stress factor (factor 1.7 to 2 compared to AISI 316);
  - in terms of neutron irradiation stability of bcc steels (swelling is very low compared to AISI 316);
  - there is either no or much reduced "high-temperature" embrittlement.

The main issues with Eurofer are:

-low temperature radiation embrittlement. Eurofer embrittlement fairly rapidly under fission irradiation. Post irradiation tests,after fission irradiation, show large hardening and a ~ 200 °C increase in the ductile to brittle transition temperature (DBTT) occurs for irradiation temperature of ~ 300-350 °C after ~ 20 dpa. This raises serious concerns about the viability of these steels for fusion structure operating at or below 300-350 °C without mitigation measures, particularly since potential additional embrittlement from the effects of neutrons with fusion relevant energy levels have not been included in the results. Helium content, which will be produced with fusion neutrons, may cause additional matrix hardening and may migrate to grain boundaries and increase the tendency to fracture;

-loss of uniform elongation and swelling under irradiation;

-water corrosion and irradiation assisted cracking;

-degradation of mechanical properties in weld heat-affected-zones (HAZ) under irradiation;

-fatigue and creep-fatigue cause “cyclic softening”.

In addition to this, since most of irradiation data up to 80 dpa for Eurofer have been obtained in fission reactors and do not cover higher He/dpa found under fusion spectra, Eurofer qualification at higher doses, above 20 dpa, is not achieved (39).

A short introduction has been done to the “Fourth Generation Ferritic-Martensitic steels” (that are FM steel with thermo-mechanical treatment (TMT) to improve the microstructure). These steels are just entered fission irradiation testing in the US. More testing is needed, but it has been clearly demonstrated that substantial improvement in multiple mechanical properties before irradiation, including thermal creep strength, fracture toughness (DBTT), etc., can be achieved compared to 1980-1990 grades steel.
Development of structural materials is historically a long and costly process, followed by an even longer proof testing period.

RCC-MRx code applied to Divertor cassette body

All design rules and corresponding stress limits of RCC-MRx code have been summarized. In particular all possible damages of level A criteria and elastic analysis under creep/irradiation (that can be negligible or significant) of subsection B have been taken into account.
This methodology has been applied to divertor cassette body “inside/outside” French nuclear pressure legislation at negligible/significative creep and significative irradiation conditions.

The following criticalities are still open in RCC-MRx:

RCC-MRx must take into account consistency of rules and recommendations for a group of materials (in the case of Eurofer steel, for the martensitic steels). At present there is no recommendation on irradiation behavior of the conventional martensitic steel (9Cr-1Mo) used in the fission reactors (mostly in unirradiated state in the steam generators) (39)

The 2015 edition of RCC-MRx, not only will pass from probatory to definitive but also will contain high temperature rules and fracture toughness (39).
The only part that is still debated is the negligible creep limit. Tavassoli will propose for the negligible creep limit for Eurofer a temperature higher of 400°C (39).

Annex 1 "ITER divertor cassette body"

ITER cassette body is ~3.3 m long, ~ 2.5 m high and 0.4 to 0.8 m wide. The mass of a CB is ~4.7 tons.
The CB is actively cooled by pressurized water (40).

Typical design parameters are:

- Design pressure: 5.0 MPa
- Baking pressure: 4.4 MPa
- Water temperature during baking: ~240 oC
- Hydraulic test pressure: 7.15 MPa

The standard CB consists of four main parts:

1) **Steel structure**; it is the supporting structure of the PFCs and provides the coolant distribution to them.
The main part of the steel structure of the CB is made of 316L(N)-IG and XM-19. The multilink attachments, the upper plate below the Dome, the parts interfacing with the nose and the knuckle and the lifting sockets are made of XM-19;

2) **Nose**; this is bolted onto the inner side of the CB by means of high strength 660 Steel bolts. The Nose is made of aluminium-bronze. It is used for the fixation of the CB onto the Divertor inner toroidal rail mounted on to the vacuum vessel;

3) **Knuckle**; this is used for the fixation of the CB onto the Divertor outer toroidal rail mounted on to the vacuum vessel.

4) **CB pipe stubs**; these are made of 316L and have alignment features for the hydraulic connection of the Divertor to the Tokamak cooling water system.

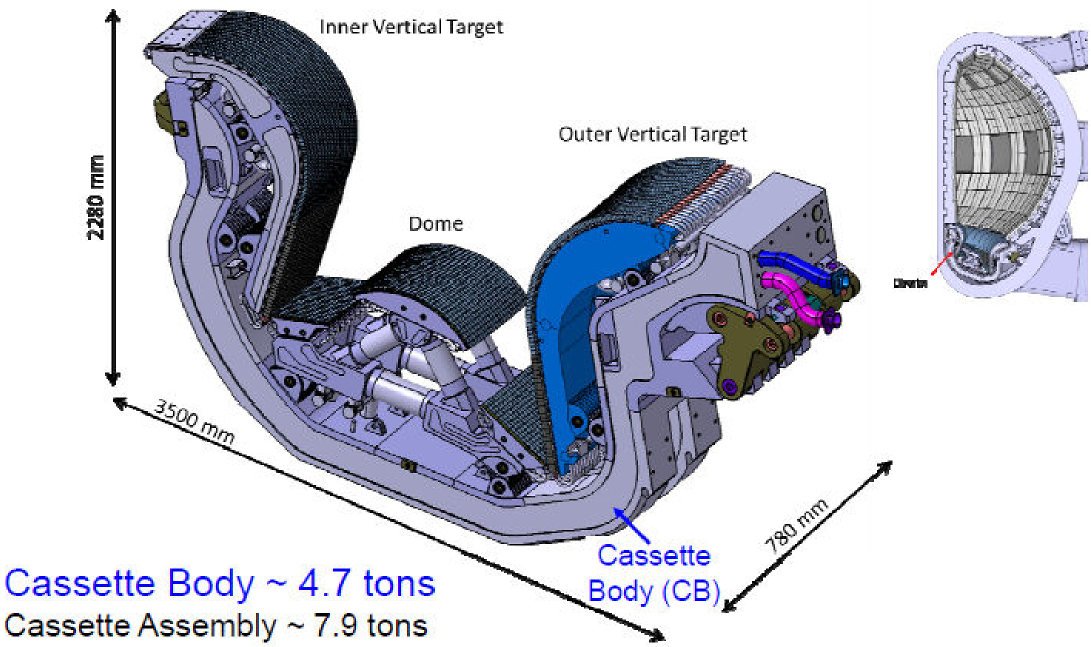


Fig. A1.1 - Iter cassette body

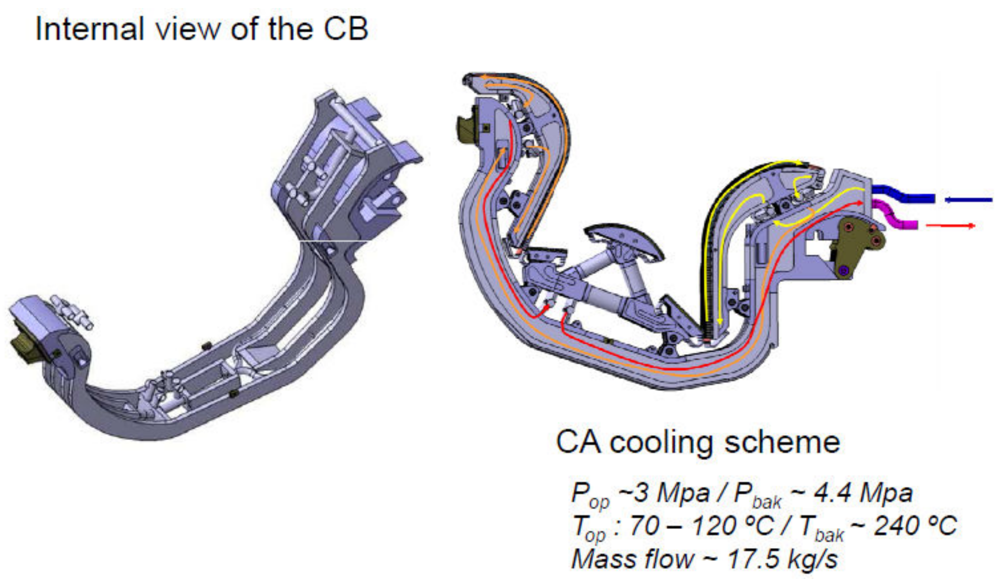


Fig. A1.2 - Internal view

A1.1 Materials

316L(N)-IG AUSTENITIC STAINLESS STEEL PLATES

316L(N)-IG steel is grade 316L steel with narrower alloying element ranges and controlled impurities. The closest analogy is X2CrNiMo17-12-2 controlled nitrogen content austenitic stainless steel described in the RCC-MRCode, Edition 2007

Element	Alloying elements and impurities content, wt. %.	
	min	max
C		0.030
Mn	1.60	2.00
Si		0.50
P		0.025
S		0.010
Cr	17.00	18.00
Ni	12.00	12.50
Mo	2.30	2.70
N	0.060	0.080
B		0.0020
Cu		0.30
Nb		0.10
Ti		0.10
Ta		0.01
Co		0.05

Table 2. Requirements for tensile properties

Tensile properties			
Test temperature, °C	Tensile Strength, min, MPa	Yield Strength (0.2%), min, MPa	Elongation, (5d) min, %
Room	525	220	45
250	415	135	-

Table A1.1 - 316L(N)-IG steel

The thickness of the plates is 40 to 60 mm.

XM-19 AUSTENITIC STAINLESS STEEL FORGINGS

Chromium-manganese-nickel austenitic stainless steel type XM-19 (UNS No S20910, grade in ASTM standards also called FXM-19) forgings.

Table 1. Chemical composition of XM-19 stainless steel

Elements	Content in wt.%	
	Min.	Max.
Fe	balance	balance
C		0.06
Mn	4.0	6.0
Si		1.00
Cr	20.5	23.5
Ni	11.5	13.5
P		0.040
S		0.030
Mo	1.50	3.00
N	0.20	0.40
V	0.10	0.30
Nb	0.10	0.30
Ta		0.01
Co		0.05

Table 2. Requirements for tensile properties

Tensile Properties	Test temperature	
	20°C	250°C
Yield Stress (0.2%), min	380 MPa	270 MPa
Tensile Strength, min	690 MPa	561 MPa
Total Elongation (over 5d), min	35 %	-
Reduction of area , min	55 %	-

Table A1.2 - XM-19 steel

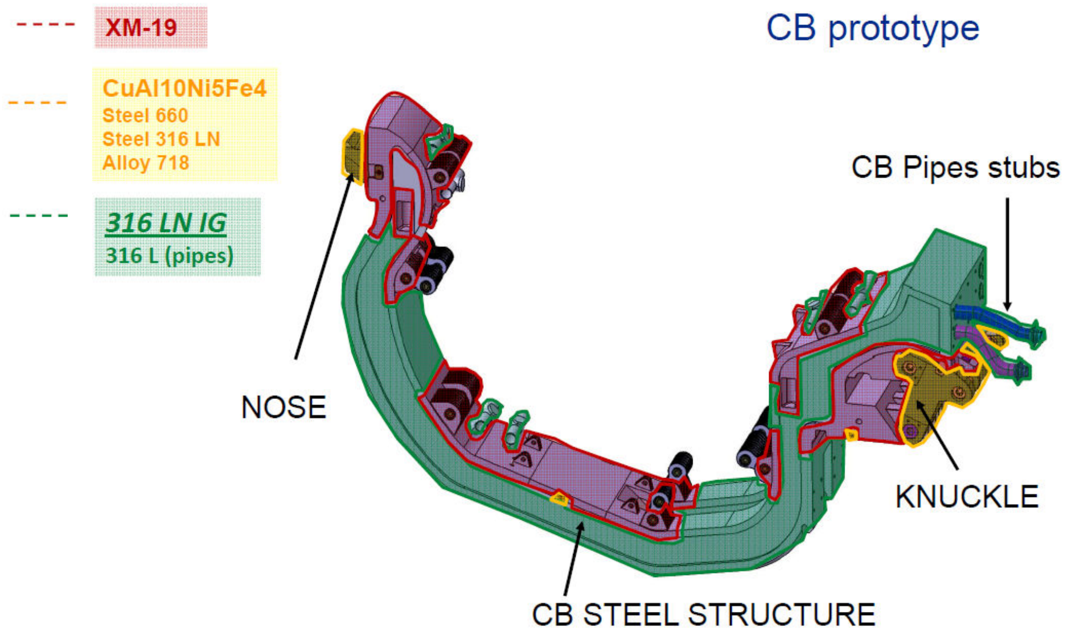


Fig. A1.3 - Cassette body materials

WEIGHT & PRICE

XM19 forging: Raw materials ~7000 Kg Price estimation €/Kg 30,00
AISI 316LN-IG plates: Raw materials ~5000 Kg Price estimation €/Kg 15,00
CB Structure finished: ~4100 Kg (without Nose and Knuckle)
Filler metal for XM 19 Materilals Price estimation €/Kg 60,00
Filler metal for AISI 316LN-IG Price estimation €/Kg 15,00

A1.2 NDT & INSPECTION

All the welds are fully penetrated welds and shall be 100% checked by volumetric examination

- For all VQC 1A (in vessel divertor cassette water cooling), water boundaries and vacuum boundary welds which become inaccessible, 100% volumetric examination of production welds shall be performed, unless a method of pre-production proof sampling is approved.
- Radiographic testing is preferred, but in the case where it is not possible due any reason, also ultrasonic testing (UT) examination can be used. When the CB is finished, only UT inspections are feasible.

Annex 2 "Demo divertor"

A2.1 Divertor main function.

The divertor (Fig. A2.1) main function is to withstand the plasma heat loads and the great amount of neutral particles and impurities produced by the fusion process. If these particles and impurities went into the core plasma, then the plasma temperature would decrease quickly because of the increased radiation heat loss.

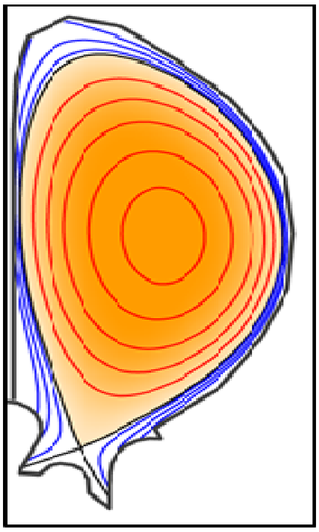


Fig. A2.1 - Principle of tokamak divertor

In poloidal cross section of vacuum vessel the magnetic field lines generated by the external poloidal coils change locally from the typical configuration of poloidal magnetic field because of the superposition of the magnetic field generated by the divertor poloidal coils. They are circular closed in the core plasma region, while in the so called scrape-off layer (SOL) they are opened and intersect the divertor target plates. With this adoption of magnetic field configuration, once the charged particles coming from the central plasma region go across the last closed flux surface (LCFS), most of them will be addressed to the divertor target plates by the magnetic field lines. At the same time new impurities will be generated by the plasma interaction with the divertor surface, and the high energy owned by charged particles will be deposited on the target plates. It can be said that the high energy particles and the absorbed energy are brought in the divertor private region where they can be efficiently pumped away and exhausted, respectively, by dedicated system.
The function of divertor to exhaust the power located in the scrape-off layer interests the divertor target plates with plasma conduction and convection (particles) and by radiation (photons and neutrals) from the divertor plasma volume. This function is realized by keeping an acceptable level of plasma impurities (obtained with the plasma-surface interaction). As a consequence, the divertor must withstand high heat loads and at the same time shield the vacuum vessel and the magnet coils in its neighbourhood from the neutron flux.
In the divertor region the intensity of neutral particles with lower energy is much higher because a big amount of charged particles are neutralized after interaction with the targets and other impurity gases are injected by feeding system. So the charged particles coming from the core plasma often collide with the neutral particles. As a result, the energy of the charged particles decreased quickly and is low enough when collided with the target. During this process, the charged particles are neutralized. The neutral particles including the sputtered and puffed impurities and neutralized particles, which cannot be confined by the magnetic field lines, will be exhausted by the divertor pumping system.

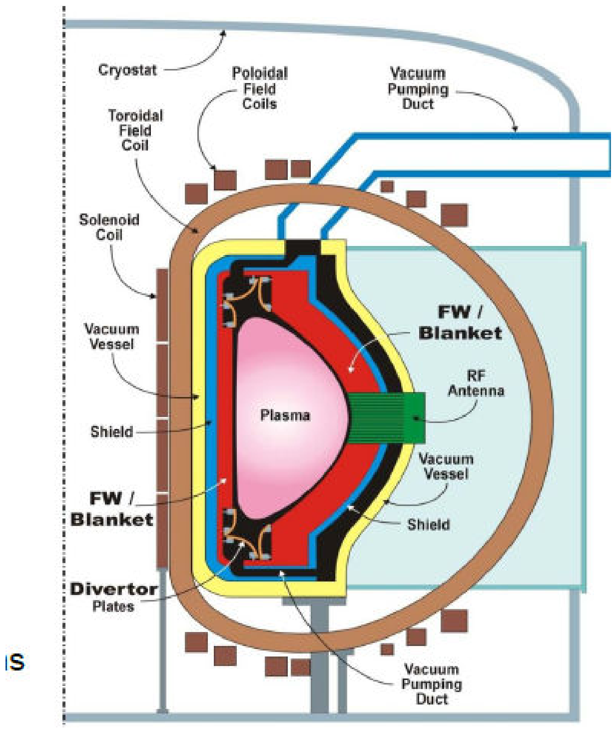


Fig. A2.2 - Tokamak poloidal section

A2.2 Physical reason of working principle

Although the plasma facing surface of the divertor will not be damaged too much if the particles in the divertor region are well controlled, it will still be at high temperature due to the high heat loads. The heat loads can be up to some tens of MW/m^2 on divertor target surface, especially near the strike points. Then high quality and efficient cooling system must be developed.

Besides these, the divertor structure must withstand huge electromagnetic forces caused by the halo current and induced current interact with high magnetic field. The PFMs directly exposed to particle impact and heat radiation generated by the plasma and the impurities will be released from the PFM surfaces as the result of plasma material interaction (PMI). So the PFM must be the best compromise between the following properties: high thermal conductivity, high melting point, high mechanical strength, high thermal shock resistance, low coefficient of thermal expansion (CTE), low outgassing, low sputtering yield and low neutron activation. After several theoretical studies, high Z material tungsten (W), low Z material beryllium (Be), and carbon-based material (CBM) are the most favourable PFMs for fusion device.

As proved in past plasma experiment, the shape of the divertor plays an important role in the particle and impurity control. The dome plate and the inner and outer target plates form a “W” shape. The angle of the “V” region formed by the dome plate and (inner and outer) target plate is a sensitive parameter. The deeper and smaller angle of the “V” region, the higher particle pressure could be obtained in this region and easier flux control for particles. In contrast, the divertor pumping system should have high enough pumping speed for exhausting these particles and balance the pressure of neutral particles in case of back streaming to the core plasma of too many particles.

More the divertor shape has made in such a way and orientation to reduce the high heat flux. Indeed the magnetic field lines follow the toroidal direction twisting themselves as an helix forming a toroidal angle and a poloidal angle that intersect the solid surface of the PFCs. The divertor is always continuous in toroidal

direction and it intersects the magnetic field lines with a constant angle. In a poloidal cross section, the magnetic field lines have acute angles with the inner and outer target plates. So the heat flux density coming from the SOL is decreased by the factor sinα where α is the intersection angle. The shape of the divertor target should be designed in such a way that this angle is as low as possible. Another way is to increase the distance from the X point to the target plate, consequently the high energy particles spend more time in the process of approaching the target and then they have many possibilities to collide with the low energy particles in the divertor "V" region. So high energy will be radiated to the target and less energy is transferred by convection. The heat load distribution is more uniform and the pick heat flux is lower around the strike points, which can extend the life time of the divertor target plates.
Also important are the Electromagnetic Loads during transient plasma instability as disruption and VDE. At the beginning of the disruptions, the induced eddy current will occur on the divertor structure which is in the region of high magnetic field, resulting in big electromagnetic force. More, when the plasma VDE happens, the core plasma will move up or down in the VV and touches the divertor structure, causing the halo currents flow (30–50 % Ip) in the divertor itself. High electromagnetic loads will be generated. It is commonly used to reduce (keeping the structural strength) the section surface of the flowing path of the eddy and halo current in order to increase the electrical resistivity and finally to decrease the forces.
The divertor structure should also withstand the thermal stresses resulted from the temperature gradient in the same material or the difference of CTE of unlike materials during the plasma operation phase or baking phase.

Annex 3 "Eurofer pecularieties"

A3.1 Why Eurofer?

Reduced Activation Ferritic Martensitic (RAFM) steel is the primary choice material for Test Blanket Modules structural application for ITER and in future fusion power plants (32). These steels have been developed in order to simplify special waste storage of highly radioactive structures of fusion reactor after service (32). With this objective some alloying elements such as Mo, Nb and Ni, present in the commercial martensitic steels, have been replaced by other elements which exhibit faster decay of induced radioactivity such as Ta, W and V.

The reasons for selection of RAFM steels are:

- good overall balance of mechanical properties (strength ductility, fracture toughness, creep resistance, fatigue resistance);
- the properties can be adjusted and to some extent tailored (eg by chemical composition but even more from heat treatments);
- there is broad industrial experience with FM steels in nuclear and non-nuclear facilities (as compared to other materials like Vanadium alloys);
- -they have superior properties to austenitic steels:
    - in terms of thermo-phisical properties and thermal stress factor (factor 1.7 to 2 compared to SS316);
    - in terms of n-irradiation stability of bcc steels (swelling is very low compared to Stainless Steel);
    - there is either no or much reduced "high-temperature" embrittlement.

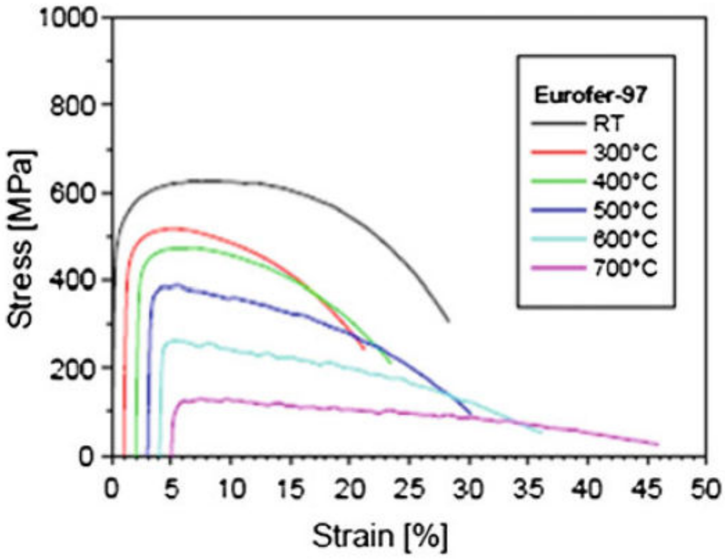


Fig. A3.1 – Stress versus strain of Eurofer at various temperature (33)

The main issues with Eurofer are:

-low temperature radiation embrittlement. Eurofer embrittles fairly rapidly under fission irradiation. This occurs at temperature about 325-350 and He content which will be produced with fusion neutrons is expected to make this worse, since Helium may cause additional matrix hardening and may migrate to grain boundaries and increase the tendency to fracture;

-loss of uniform elongation and swelling under irradiation;

-water corrosion and irradiation assisted cracking;

-degradation of mechanical properties in weld heat-affected-zones (HAZ) under irradiation;

-fatigue and creep-fatigue cause “cyclic softening”.

In addition to these technical issues there is a lack of materials database for 14 MeV neutron irradiation effects and this impedes meaningful design code development. Another issue is the lack of design rules for high-temperature creep-fatigue.

The irradiation damage which causes the embrittlement and hardening is known to be ‘annealed’ if irradiation occurs at higher temperatures (above 300-350°C). It is due to the thermally-induced dissolution of small defect aggregates as the temperature (and hence energy) is increased. Post irradiation tests, on steel after fission irradiation, show large hardening and an ~ 200 °C increase in the ductile to brittle transition temperature (DBTT) occurs for irradiation temperature of ~ 300-350 °C after ~ 20 dpa. This raises serious concerns about the viability of these steels for fusion structure operating at or below 300-350 °C without mitigation measures, particularly since potential additional embrittlement from the effects of neutrons with fusion relevant energy levels have not been included in the results.

The AISI 316 steel has a limited power exaust capability and limited resistance to radiation effects.
The ductility and fracture toughness of 316 steel are severely degraded during irradiation around 300 °C and doses of ~ 10 dpa. At intermediate temperature of 400-600 °C they are susceptible to significant volumetric void swelling for doses >20 dpa, depending on the irradiation spectrum and other experimental variables.

AISi 316 steels have been shown to suffer from severe He-embrittlement at high temperature (>550 °C).

A3.2 Eurofer Code qualification

Eurofer has entered in an approbatory status in Tome 6 (Section III) of RCC-MRx code. This status allows industry and interested parties to comment the files before they are made permanent (13).
The first files written for Eurofer code qualification were materials procurement files. These files had to be:
(a) sufficiently detailed to allow any manufacturer to bid for fabrication;
(b) adequately precise to ensure that the end products would comply with the specified properties. Materials procurement files submitted for Eurofer steel covered plates, tubes, bars and forgings. Welding specifications are yet to be finalised. All the base metal files were backed up with support documents that include results of inspections, examinations, macro and microstructural examinations, hardness, etc. for each product. However, only the plate products were approved for RCC-MRx 2012 edition.

The second set of files written for RCC-MRx was materials properties data files. These are organised, as in the RCC-MRx, in 5 sections:
– Physical properties.
– Borderlines.
– Analysis data 1.
– Analysis data 2.
– Analysis data 3.

Physical properties have been evaluated for F82H and Eurofer steels (28). They include: coefficients of thermal expansion (average and instantaneous), modulus of elasticity, Poisson's ratio, density, specific heat, thermal conductivity, and thermal diffusivity, magnetic and electrical properties.

Physical properties of RAFM steels are similar to those of the conventional 9Cr–1Mo steels.

Borderlines define conditions where certain analyses are not needed. For instance, thermal creep analysis is not required if the creep deformation does not exceed 0.05% (negligible creep) under a stress equal to 1.5 S_m at the service temperature. RAFM steel data are still insufficient for its precise evaluation. One can reasonably assume that thermal creep can be ignored at temperatures below 400 °C. At temperatures higher than 500 °C, time to reach 0.05% creep is very short and thermal creep cannot be ignored even in the ITER like pulsed operation mode.

Like thermal creep negligible irradiation creep is defined as a dose below which irradiation creep deformation does not exceed 0.05% under a stress equal to 1.5 S_m. For ITER TBMs, where the maximum dose is less than 3 dpa, irradiation creep can be effectively ignored except for bolts. Irradiation creep in general does not provoke physical damage.
At higher doses irradiation creep cannot be ignored (31) and its effects should be added to those of the thermal creep and irradiation swelling. Irradiation swelling provokes physical damage and at levels higher than 3% in volume (1% in dimension) needs special design procedures. At above 6% it is beyond design and not tolerated. Fortunately, martensitic steels show excellent resistance to swelling and even at DEMO dose levels their swelling is expected to remain low.
Analysis data 1 deal with time independent properties such as tensile, impact, fatigue and fracture. Tensile strength data were used to derive design stress intensity values S_m and S. Tensile ductility and toughness values were used to ensure that Eurofer has adequate resistance to brittle fracture.
Tensile properties of RAFM steels (16) are close to those of the conventional Mod. 9Cr–1Mo steel and are higher than those of the SS 316L(N) steel at temperatures less than 500 °C, Fig. 16.
At higher temperatures, both RAFM and conventional martensitic steels, lose their strength and are seldom recommended for service at above 550 °C. Other grades of these steels, such as oxide dispersion strengthened steels (ODS), are developed to increase their temperature window (31).
These steels show a Ductile-to-Brittle Transition Temperature (DBTT) at ≤ 50 °C. The fracture toughness of RAFM steels in the ductile region is very high, e.g. J_Q of Eurofer at 60 °C is in the range of 300–620 kJ/mm2.

Fatigue is the only time independent property of major concern in time independent properties because of continuous cyclic softening and breakdown of martensitic microstructure (31).
In fact, due to cyclic softening tensile resistance of martensitic steels can become lower than the 316L(N) steels even at low temperatures, rendering their time dependent damage calculation more difficult.

Analysis data 2 deals with time dependent properties such as creep and creep-fatigue (31). Creep results of Eurofer steel were used to determine stress to rupture, stress for time to reach 1% deformation and stress to reach the onset of tertiary creep criteria. Combination of these criteria was used to determine the stress intensity criterion St. In comparison with stainless steel, conventional and RAFM martensitic steels have a lower creep resistance at 550 °C and higher.

Present RAFM creep test durations may not be sufficiently long for reliable extrapolation to nuclear reactor design time of > 300,000 h.

Analysis data 3 deals with irradiation effects on time independent and time dependent properties. Modified 9Cr–1Mo steel is used essentially in conventional power plants or steam generators of nuclear power plants, i.e. in unirradiated zones. The results obtained for RAFM steels show that despite progress in improvement of low temperature irradiation embrittlement, these steels cannot be recommended at present for service at doses higher than 3 dpa at T < 350 °C. The same results show that at irradiation temperatures higher than 350 °C the toughness of RAFM steels remains adequate even at much higher doses, unless future irradiation experiments under fusion relevant neutron spectra prove otherwise.

Cyclic softening observed in unirradiated state also occurs in the irradiated state (31). Under these conditions, calculation of creep-fatigue interaction diagram, already difficult in the unirradiated state, becomes even more difficult. Furthermore, since most of the results are from fatigue tests with relatively short hold times (<1 h relevant to TBM and early DEMO); they do not cover the higher creep damage region relevant to DEMO operation.

Annex 4 "RCC-MRx peculiarities"

A4.1 RCC-MRx radiation damage

When subject to neutron irradiation, the material behavior is modified. Involved phenomena are very complex; thus let's identify here the main effects which have been considered in the establishment of the design rules.
The effect of irradiation on metallic material can be summed up as follows:

- Atoms displacement, induced by neutrons with a sufficient energy level. It is a dominating effect for steels and is measured with the number of displacements per atom (dpa). These displacements lead to disturbance and reconstruction of the regular atomic ordering in the crystal lattice. These processes are accompanied by the formation of characteristic lattice defects which affect both the microstructure and the macroscopic structurally sensitive properties of the irradiated material. The consequences may be a decrease of ductility and a hardening of the material, a swelling or irradiation creep;
- Changes in the chemical composition by stopping of the bombarding particles (called ion implantation) or capture of particles in the atomic nucleus with consequent transmutation;
- Excitation of electrons and ionization of atoms (which does not produce permanent damage in metals).

The design rules of the codes are established to prevent from mechanical damages: excessive deformation, plastic instability, fatigue, progressive deformation, creep, and fast fracture. Generally, the rules are based on a comparison between the estimation of the consequences of a given load in the structure (stresses, strains,

cracking force etc..) and the capability of the structure material to resist to this effect (in link with the mechanical properties of the material, yield stress, ultimate stress, toughness etc..).
To establish rules preventing from these damages for the RCC-MRx code, only the consequences with an impact on the mechanical behavior of the components are considered.
The effect of irradiation is integrated in these design rules through a material effect and also through the estimation of the load effects (41).
From a macroscopic point of view, the irradiation leads to an increase of the yield strength and of the tensile strength (see fig. A4.1). In the same time, uniform and rupture elongations are decreased.

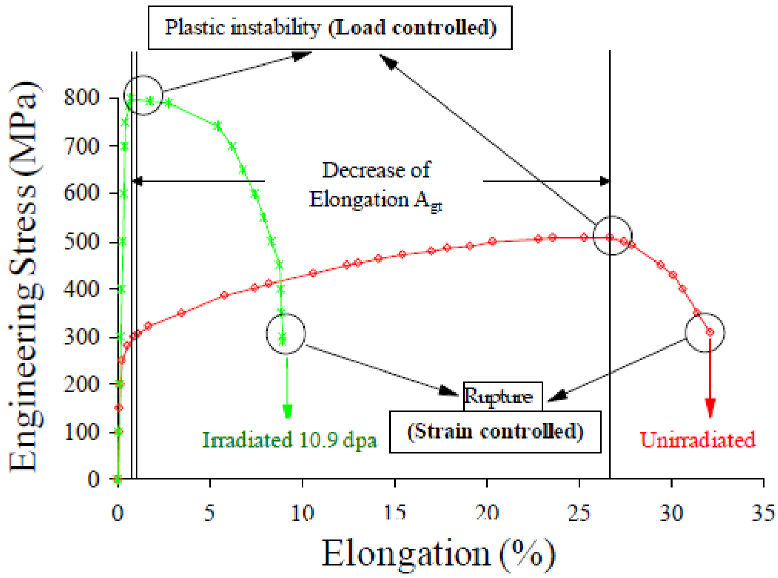


Fig A4.1 - Evolution of a conventional tensile stress-strain curve with irradiation (AISI 316L(N) steel).

These changes in the material behavior challenge the usual rules preventing mechanical damages because the prevention uses the principle of stress relaxation.

Hence, it results in 2 issues to be addressed:

- Knowledge of the mechanical behavior of the material when irradiated;
- The rules used to prevent mechanical damage of the structures.

These multiple considerations led to an approach in three steps for the definition of the rules to prevent irradiation damage:

- First, mechanical characteristics depending on relevant irradiation parameters have to be collected;
- Then the rules themselves have to be defined, considering the consequences of the loss of ductility on the existing classical rules;
- This implies the definition of the border lines for the application domain of the irradiated material prevention rules.

The drivers for the use of the rules are the following:

- Prevention of the mechanical damages of the structure;
- Use as far as possible of proven approaches and methodologies for the establishment of the rules;
- Easiness of use of the rules by designers.

In some specific cases, depending on the functions of the mechanical components some additional considerations regarding the effects of irradiation are to be done on a case by case basis especially for some components ensuring specific safety or mechanical functions requiring set-up or removal of assemblies, mechanical motions,… for which geometrical deformations are to be considered.

The details of the selected approach and rules are detailed hereafter.

A4.1.1 Rules overview

A4.1.1.1 Material properties

Data shall be given in function of parameters considered as driving the material mechanical behavior. Three materials families are concerned in the RCC-MRx code: stainless steels, aluminum alloy and zirconium alloy.

For stainless steels it is considered that material mechanical behavior is driven by dpa.
For aluminium alloys, it is considered that material mechanical behavior is driven by Si production in the alloy.
For zirconium alloys the material mechanical behavior is considered driven by irradiation flux in fast neutrons.

To feed the code, a huge effort in the collection and analysis of available experimental irradiation data has been done by the CEA. An experimental program to complete this data base is still on-going. The mechanical characteristics are available in appendix A3 of the code. There are two types of characteristics: the border lines and the properties needed to apply the design rules.

A4.1.1.2 Border lines

Two border lines are defined in the code in relation with irradiation:
- Negligible irradiation curve: below this curve, the design of the component is made without including irradiation effects;
- Maximum allowable irradiation: above this curve, the rules given by the code are not validated anymore.

To define these curves, the effects of irradiation on the mechanical properties have been analyses. The borders have been selected on the basis of the following considerations:
- Negligible irradiation is based on a variation of ductility criterion;
- Maximal irradiation is based depending on the nature of the material either on the residual ductility or maximal swelling criteria.

The selected borders lines can be sum up in the following table:

Material designation	Negligible irradiation	Maximum allowable irradiation
A3.1S: « 316L(N) »	↘ Ductility	↗ Swelling
A3.3S: « 316L»	↘ Ductility	↗ Swelling
A3.7S: « 316L work hardening»	↘ Ductility	↗ Swelling
A3.4S: « 304L »	↘ Ductility	Not supplied
A3.1A: 5754-O (solution annealed)	↘ Ductility	↘Ductility
A3.2A: 6061-T6 (structural Hardening)	↘ Ductility	↘Ductility
A3.1Z: Zircaloy 2	↘Ductility	↘Ductility
A3.2Z: Zircaloy 4	↘Ductility	↘Ductility

Tab. A4.1 - Effects of irradiation for structural materials.

The selected borders are related to the applicability of the design rules and may have to be completed on a case by case basis according to a specific analysis covering:
- Deformation of the structure;
- Others considerations issued from the safety analysis and operating conditions of the component.

Fig. A4.2 shows the negligible, significant and maximum allowable irradiation borders curves for Cr-Ni-Mo stainless steels material family.
The criterion for the negligible irradiation curve is the loss of ductility (percentage of the minimum value of A% (unirradiated) – A is the percentage total elongation (elastic plus plastic) at fracture).

The criterion for the maximal irradiation curve is the variation of ΔV/V or ΔL/L due to swelling (where ΔV/V, ΔL/L are the variation of volume or linear dimension due to swelling).

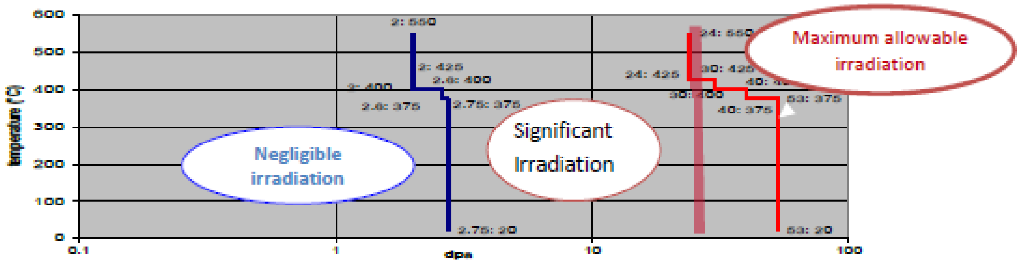


Fig A4.2 - Border line for CR-NI-Mo stainless steels.

A4.1.1.3 Design rules

Mechanical codes prevent from the usual damages: Excessive deformation, Plastic instability, Elastic and elastic-plastic instability, Progressive deformation, Fatigue, Creep, Fast fracture.

Design codes propose different possible methods:
- Direct verification: experimental analysis, elastic-plastic analysis;
- Elastic analysis: detailed elastic analysis, simplified elastic analysis (use of flexibility and stress indices for standard components).

The selected approach for the irradiated component rules establishment is based on the following approach:

- Additional rules for irradiated structures;
- When the ductility decrease, the idea is to complete the non-irradiated set of rules preventing the excessive deformation and plastic instability damages by a total stress and strain limitation criterion based on the irradiated material residual ductility;
- Thanks to the well-known elastic follow-up method (41) used by several nuclear design and construction codes for mechanical components, the use of an elastic analysis allowing to properly represent the actual stresses obtained through an elastoplastic behavior was selected.

This approach allows the use of the elastic analysis in order to assess the elasto-plastic stresses and strain.

Only the detailed elastic analysis will be treated here. As said before, the way to implement irradiation effect in the RCC-MRx code consists in a definition for the already existing rules of an "irradiation effect" through pertinent parameters.
In detail, it leads to the following sets of rules in the case of the detailed elastic analysis:

Rules to be met without irradiation effect	**Additional rules to integrate the irradiation effect**
Excessive deformation, plastic instability $\overline{P_L + P_b} \leq 1.5 \cdot S_m$ $\overline{P_L} \leq 1.5 \cdot S_m$ $\overline{P_m} \leq S_m$	Excessive deformation, plastic instability $\overline{P_m + Q_m} \leq S_{em}^A$ $\overline{P_L + P_b + Q + F} \leq S_{et}^A$
Fast fracture *$J(a,C) \leq J_{IC}$*	Fast fracture *$J(a,C) \leq J_{IC}$ (irradiated)*
Progressive deformation *P and ΔQ to compare to kSm*	Progressive deformation *Analyse non irradiated material*
Fatigue *Usage factor V = specified Ncycles / allowable Ncycles < 1*	Fatigue *Fatigue curve without irradiation: $\Delta\varepsilon$ increases*
Creep *Creep factor U or W ≈ application time / allowable time < 1*	Creep *Creep factor U or W ≈ application time / allowable time < 0,1*

Tab. A4.2 – Summary of design rule in RCC-MRx.

Annex 5 "ESP/ESPN French Legislation"

A.5.1 Pressure Equipment 97/23/EC (PED) and French Decree 99-1046 (ESP)

The Pressure Equipment Directive (acronym – PED) was adopted by the European Parliament and the European Council in May 1997 (17). It entered in force in all European Union from 29 May 2002. The directive provides an adequate legislative framework at the European level for equipment subject to a pressure hazard.

The French Decree No 99-1046 dated 13th December 1999 (amended by further Decrees) and Order dated 21st December 1999 concerning classification and conformity assessment (acronym ESP will be used) put the Pressure Equipment Directive in force in France (38).

The Directive and French Decree apply to the design, manufacture and conformity assessment of pressure equipment and assemblies with a maximum allowable pressure greater than 0.5 bar over atmospheric pressure (1.5 bars absolute).

These regulations introduce a categorization (Category I–IV, Category IV being the highest) of the pressure equipment, depending on the hazard due to pressure, volume of the vessel or diameter of the pipe, type of fluid and temperature. There is an additional category which is Sound Engineering Practice (SEP). For each category the so-called modules for conformity assessment in accordance with the Essential Safety Requirements are established. For equipment in Category II–IV the conformity assessment has to be performed by a Notified Body.
Some equipment operating under a pressure greater than 0.5 bar can be excluded from the scope of directive.

The ESP and PED formulate the Essential Safety Requirements (ESR) (see Annex 1 of ESP), which includes technical and legal conditions which have to be satisfied. These requirements are related to the design, manufacture, materials and other specific conditions. The manufacturer of equipment is under an obligation to analyze the hazards and must then design and construct it taking account of this analysis. The selection of C&S is the responsibility of the manufacturer; he shall demonstrate that the selected C&S provides conformity with ESR. The use of European harmonized standards in the design and manufacture of a product will give the presumption of conformity with those ESRs listed in Annex ZA of the particular harmonized standard.

After completion of the conformity assessment the manufacturer shall declare conformity and issue a CE mark. Pressure equipment's are subject to the provisions applicable to operation and re-qualification, as required. The specific rules for the implementation of such requirements are described in the special order concerning the operation of pressure equipment (38).

A.5.2 French order concerning nuclear pressure equipment (acronym-ESPN)

This French law (18) defines nuclear pressure equipment as equipment that meets the following conditions:
– Is pressure equipment as defined by ESP;
– Is used in a Basic Nuclear Installation;
– Directly ensures containment of radioactive substances, and
– In case of failure leads to release of activity above 370 MBq.

ESPN has practically extended the application of the methodology foreseen by ESP and PED (ESR, conformity modules, etc.) to nuclear pressure equipment in France. ESPN has double classification of the equipment:
– Pressure hazard based on ESP rules, Category I–IV, and Category 0 (equivalent to SEP);
– Nuclear level – N1, N2 and N3.

ESPN includes some additional requirement on ESR depending on the nuclear level of the equipment.

As far as C&S are concerned, the ESPN does not define specific requirements for the selection of the codes, but requires that the conformity with ESR is demonstrated.
In accordance with ESPN requirement, the operator of a nuclear facility shall provide the description of operational conditions, and the manufacturer of the equipment (who is responsible for the design, fabrication and conformity with regulation) shall select an applicable code which is used as a tool for demonstrating conformity with Essential Safety Requirements.

The ESPN also defines rules for maintenance and monitoring, periodic inspections, installation and operation and periodic requalification of nuclear pressure equipment. A manufacturer of nuclear equipment shall contract an Agreed (by the French regulator) Notified Body (ANB) and after completion of the conformity assessment shall declare conformity.

Abbreviations

BB	Breeding blanket
BoP	Balance of Plant
CEA	French Atomic Energy Commission
CB	Cassette Body
CP	Coolant pressure
CS	Center solenoid
CTE	Coefficient of thermal expansion
DAF	Dynamic Amplification Factor
DBTT	Ductile-brittle transition temperature
DO	Dome
DW	Dead Weight
EM	Electromagnetic
FDVDE	Fast Downward Vertyical Displacement Event
FM	Ferritic/martensitic steel
FPY	Full power year
FTTT	Fracture toughness transition temperature
FW	First wall
HAZ	Heat-affected-zones
IC	Heat Transfer Coefficient
IVC	In vessel component
IVT	Inner Vertical Target
LCF	Low-cycle fatigue
LCFS	Last closed flux surface
MD	Major discharge
MFD	Magnet fast discharge
NOC	Normal Operating Conditions
ODS	Oxide dispersion strengthened
OVT	Outer Vertical Target
PED directive	Pressure equipment directive
PF coils	Poloidal field coils
PFC	Plasma Facing Component
PMI	Plasma material interaction
PFM	Plasma facing material
PFU	Plasma Facing Unit
RAFM	Reduced Activation Ferritic Martensitic
RT	Room Temperature
SCC	Stress corrosion cracking
SDVDE	Slow Down Vertical Displacement Event
SIC	Safety importance class
SL-1, SL-2	Seismic Level 1 and 2
SOL	Scrape-off layer
SS	Stainless steel
Tbd	To be defined
TBM	Test blanket module
TH	Thermal Loads
TMT	Thermo-mechanical-treatment
VDE	Vertical Displacement Event
VT	Vertical Target
VV	Vacuum vessel

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