Fusion Engineering and Design 88 (2013) 1836–1843

Contents lists available at ScienceDirect

Fusion Engineering and Design

journal homepage: www.elsevier.com/locate/fusengdes

Potential and limits of water cooled divertor concepts based on monoblock design as possible candidates for a DEMO reactor

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ARTICLE INFO

Article history:
Received 14 September 2012
Received in revised form 15 May 2013
Accepted 30 May 2013
Available online 9 August 2013

Keywords:
DEMO
Water cooled divertor
Monoblock
CuCrZr
Eurofer

ABSTRACT

In this paper water-cooled divertor concepts based on tungsten monoblock design identified in previous studies as candidate for fusion power plant have been reviewed to assess their potential and limits as possible candidates for a DEMO concept deliverable in a short to medium term (“conservative baseline design”). The rationale and technology development assumptions that have led to their selection are revisited taking into account present factual information on reactor parameters, materials properties and manufacturing technologies.

For that purpose, main parameters impacting the divertor design are identified and their relevance discussed. The state of the art knowledge on materials and relevant manufacturing techniques is reviewed. Particular attention is paid to material properties change after irradiation; phenomenon thresholds (if any) and possible operating ranges are identified (in terms of temperature and damage dose). The suitability of various proposed heat sink/structural and sacrificial layer materials, as proposed in the past, are re-assessed (e.g. with regard to the possibility of reducing peak heat flux and/or neutron radiation damages). As a result, potential and limits of various proposed concepts are highlighted, ranges in which they could operate (if any) defined and possible improvements are proposed.

Identified missing point in materials database and/or manufacturing techniques knowledge that should be uppermost investigated in future R&D activities are reported.

This work has been carried out in the frame of EFDA PPPT Work Programme activities.

1. Introduction

This paper describes a feasibility study of a water-cooled divertor for DEMO (DEMOnstration reactor) based on the optimisation of the ITER W-monoblock design and technology. This is a starting point of an assessment of the water cooling divertor technological maturity in order to check if such a technology is compliant to a DEMO concept deliverable in a short to medium term (“conservative baseline design”).

Several design studies and many experimental campaigns have been performed in the past aiming at evaluating the performances of ITER W monoblock design in ITER conditions. Some studies were also carried out aiming to evaluate the suitability of the ITER W-monoblock divertor design to DEMO and fusion power plant (FPP) reactors. Previously assessed conceptual designs are reanalysed in this paper with regard to newly available material properties and/or design limits (Section 3) and some improvements are proposed (Section 4).

2. DEMO operating parameters impacting the divertor design

Main DEMO design parameters and requirements which impact the divertor design are listed and described thereafter [1,2].

The peak incident heat flux is one of the “most significant” parameters, because it will determine the thermal gradient (and relevant thermal stresses) in the plasma facing wall. It will also prominently contribute to determine the wall heat flux and therefore the margin to critical heat flux (CHF).

The fatigue lifetime was not considered as a divertor dimensioning parameters in previous DEMO and Fusion Power Plants (FPP) studies (e.g. Power Plant Concept Studies, PPCS) because it was assumed that the reactor would have been operated in a steady state mode and that the component must withstand only a low

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0920-3796/$ – see front matter
http://dx.doi.org/10.1016/j.fusengdes.2013.05.114

number of cycles. If a pulsed mode were assumed for DEMO, the fatigue lifetime would probably have a substantial impact. This is also confirmed by experiments carried out on ITER divertor mock ups [3].

The irradiation damage has an impact on the materials physical properties and therefore on their suitability to be used in a DEMO divertor especially for structural materials. Actually this parameter seems to constitute one of the main differences between nowadays High Heat Flux (HHF) components and DEMO divertor requirements. For a given fluence (i.e. neutron wall load integrated over the operating time), the irradiation damage, expressed in displacement per atoms (dpa) will depend on the material [4].

The requirement on high heat grade recovery will have an impact on the coolant temperature and practically will define the structural material operating window and therefore the suitable structural material.

High temperature coolant ($T \sim 300\,^{\circ}$C), e.g. would preclude the use of Cu alloys as structural materials which can only be used for $T_{\mathrm{cool.}} \sim 200\,^{\circ}$C in order to guarantee that it will operate in the recommended temperature window (see Section 3.2). When irradiation is taken into account, the impact of the temperature on various materials properties and therefore on the mechanical capability of the component can be antagonist. In the CuCrZr-IG treatment C, e.g., the yield and ultimate strength, both in irradiated and unirradiated case decrease with temperature. On the other hand the effect of neutron irradiation on the uniform elongation is more important when the material is irradiated at low temperature and the material practically become brittle when irradiated at $T < 200\,^{\circ}$C. These effects strongly depend on the material grade and on the heat treatment, which does not simplify the scenario. Similarly, for Eurofer, the ductile to brittle transition temperature (DBTT) shifts when irradiated at low temperature (<300–350 °C) to values higher than room temperature (see Section 3.2).

Coolant temperature will also have an impact on thermal hydraulic parameters and mechanical capability. To increase the water temperature would mean to increase the water pressure, in order to keep the same CHF margin and therefore primary stresses on the structure.

A non-exhaustive list of other parameters have also to be taken into account, as tritium retention, erosion, electromagnetic loads or criteria regarding waste management, configuration and component classification (pressure, nuclear?) which drives relevant rules for the design of components (including manufacturing techniques, associated qualification) [5–7].

3. Materials

The rationales for the selection of (Plasma-facing materials) PFMs and heat sink/structural materials in ITER and beyond reactors (DEMO, power plant) have been widely described in literature (e.g. [8]). A recent paper [9] has been issued summarising advantages and drawbacks of various materials for various divertor concepts (namely He and water cooled) taking into account latest data available in literature.

Hereafter main characteristics of various envisaged materials are recalled. Three categories are considered, depending on their function, namely plasma facing materials, heat sink/structural materials and compliance layer. Refractory metals (W and Mo), could be envisaged both as structural and plasma facing materials.

It must be highlighted that for all mentioned materials, when effects of irradiation are discussed, neutron-induced helium embrittlement and/or transmutations in a fusion environment are not considered, due to lack of experimental data under a fusion neutron spectrum (most tests have been carried out in fission reactors and accelerators).

3.1. Armour material

With regard to PFM, tungsten is currently considered as the most favoured armour material for PFC of fusion reactors. This is due to its unique characteristics, particularly such as refractory nature, low sputtering erosion, high strength, reasonable thermal conductivity and acceptable activation [9,10]. Main drawback, to use as structural material is its inherent brittleness. Various alloys have been developed in order to improve the base material ductility and W-Ta, W-Y_2O_3, W-Ti alloys seem to be particularly promising [11]. Material data for W exposed to neutron irradiation is rare. The existing fission reactor database on refractory metals indicates low swelling (<2%) for doses up to 10 dpa and higher [12]. Unfortunately a severe embrittlement under neutron irradiation at “low” temperatures (most probably 800–900 °C) can be expected. W can be used as armour materials without structural function. An optimisation of the armour thickness will be necessary taking into account large scale fusion experiences with W (presently only available on inertial component) and synergistic effects as for example bulk (neutron) and surface (charged particle) accumulated damage.

3.2. Heat sink/structural material

The heat sink material would have structural function as well as functions to redistribute the heat flux towards the coolant channels and to provide hermetic coolant confinement. In the frame of 2011 EFDA activities on water cooled divertors, a review of structural materials envisaged in the past has been carried out. Materials potential and limits have been reassessed in view of update knowledge (namely under irradiation) and reactor requirements (e.g. with regard to the possibility of reducing peak heat flux and/or neutron radiation damages). Two materials have been identified as the most promising for a DEMO water cooled divertor, Eurofer and CuCrZr [13,14]. Parameters used for their comparison and main arguments corroborant this choice are reported hereafter.

In order to compare the material capability to transfer high heat flux without mechanical damage, a criterion derived by a simple rule (thermal stress < allowable stress) on a 1D geometry (1D wall) is proposed hereafter. In literature (e.g. [12]) the ultimate strength S_u is usually assumed as allowable limit. According to standard design codes, for ductile (not irradiated) materials, thermal stresses need to be limited to guarantee the component against progressive deformation (under cyclic loads). According to design codes as SDC IC [7], RCC MR [15], RCC-MX [16], and RCC-MRx [17]), e.g., the 3 S_m criterion can be used as allowable stress (membrane plus bending primary plus secondary stress).

The criterion on the heat removal capability can therefore be written as

$$sq'' < \frac{(k3\,S_m)}{\alpha E}[\mathrm{kWm}^{-1}]$$

where s is the wall thickness, q'' is the thermal flux, k is the thermal conductivity, alpha is the coefficient of thermal expansion, E is the Young modulus and S_m is the allowable tensile stress limit which depends on ultimate and yield. The heat removal capability so defined is shown as a function of the temperature for various materials envisaged as possible heat sink in Fig. 1. Curves for both unirradiated and irradiated (when available and more conservative) materials are reported. Cu alloys properties are valid for doses in the range 0.3–5 dpa. For Eurofer and stainless steel, only values calculated for irradiated materials (valid for doses up to 70 and 20 dpa respectively), are reported because more conservative. Irradiated values refer to mechanical properties, the variation of thermal conductivity due to irradiation has not been taken into account, assumption which is verified in the relevant irradiation doses. For CuCrZr, treatment C is the one recommended for FW and