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Assessment and selection of materials for ITER in-vessel components

G. Kalinin *, V. Barabash, A. Cardella, J. Dietz, K. Ioki, R. Matera [1], R.T. Santoro [2], R. Tivey, The ITER Home Teams

ITER Garching JWS, Max-Planck-Institute fur Plasmaphysik, Boltzmannstrasse 2, 85748 Garching bei München, Germany

Abstract

During the international thermonuclear experimental reactor (ITER) engineering design activities (EDA) significant progress has been made in the selection of materials for the in-vessel components of the reactor. This progress is a result of the worldwide collaboration of material scientists and industries which focused their effort on the optimisation of material and component manufacturing and on the investigation of the most critical material properties. Austenitic stainless steels 316L(N)–IG and 316L, nickel-based alloys Inconel 718 and Inconel 625, Ti–6Al–4V alloy and two copper alloys, CuCrZr–IG and CuAl25–IG, have been proposed as reference structural materials, and ferritic steel 430, and austenitic steel 304B7 with the addition of boron have been selected for some specific parts of the ITER in-vessel components. Beryllium, tungsten and carbon fibre composites are considered as plasma facing armour materials. The data base on the properties of all these materials is critically assessed and briefly reviewed in this paper together with the justification of the material selection (e.g., effect of neutron irradiation on the mechanical properties of materials, effect of manufacturing cycle, etc.). © 2000 Elsevier Science B.V. All rights reserved.

1. Introduction

Material selection must encompass a total engineering approach, by considering not only physical and mechanical properties and processing, but also the maintainability, reliability, replaceability, and recyclability of each material. The chemical composition must also be optimised to reduce waste disposal problems.

The choice of materials for the main international thermonuclear experimental reactor (ITER) components during the earlier engineering design activities (EDA) is presented in several publications [1–4]. The detailed information on the material selection, their properties, the effect of component manufacturing cycle, and the environmental effect on the material behaviour, are given in the reference ITER documents. These are the following:

- Material sections of the Design Description Document for the blanket and divertor (DDD) [5]. The DDD is the main design reference document, where the material working conditions are defined and a brief account of the designation of the selected materials or materials-grade and the related manufacturing technology are described.
- Material assessment report (MAR) [6]. This gives the rationale for the selection of materials and joining technologies for in-vessel components, on the basis of the available information from open literature, the research and development (R&D) results, the existing experience, and any other available information. The emphasis is on the reference solution, plus in some cases one back-up option, and the related data base. A brief justification of the reasons for discarding the other options are also given. Material and component manufacturing processes (welding,

* Corresponding author. Tel.: +49-89 3299 4123; fax: +49-89 3299 4163.

E-mail address: kalinig@itereu.de (G. Kalinin).

[1] Member of ITER JCT during 1993–1998. Present address: Joint Research Centre Ispra, TP 750-1-21020 Ispra (Va), Italy.

[2] Member of ITER JCT during 1993–1997. Present address: Oak Ridge National Laboratory, P.O. Box 2008, MS 6363, Oak Ridge, TN, 37831-6363, USA.

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HIPing, casting, etc.) and their impact on the material properties are included. The effects of in-service conditions on the structural integrity of the component are taken into account.
- Material properties handbook (MPH) [7]. The MPH is a collection of design relevant data on the physical and mechanical properties of a large variety of materials of general interest for fusion technology and particularly for the ITER design. It includes data available in the open literature as well as data from the R&D programme.
- Interim structural design criteria (ISDC) [8]. The ISDC for in-vessel components contain interim rules for the structural design of the in-vessel components of the ITER. The scope of these criteria is limited to design. Appendix A of the ISDC defines the allowable structural materials for the various failure mechanisms considered in the construction code.

Design parameters define the materials-choice. A list of materials proposed for the ITER vacuum vessel (VV) and in-vessel components is presented in Table 1 [5,6]. All materials proposed for the ITER components can be conventionally divided into two groups:

1. Industrially available standard materials with well-established manufacturing techniques. No additional R&D, or only minimum R&D, is required for such materials. These are steels AISI 304L, 316L, 316LN, 304B7, 430, titanium alloy Ti–6Al–4V, nickel alloys Inconel 625, Inconel 718, tungsten alloys PM–W, W–1%La_2O_3, and ceramic materials Al_2O_3, $MgAl_2O_4$.
2. Readily available standard materials with minor modifications in order to better satisfy the component design. These are structural materials, stainless steel 316L(N)–IG and copper alloys CuCrZr–IG, CuAl25–IG (Glidcop©Al25). Beryllium S–65C or DShG-200 grades and the carbon-based composite carbon fibre composite (CFC) are used for plasma facing components. Modifications of composition, structure and additional R&D have been performed for the qualification of these materials. The modified stainless steel 316L(N)–IG and the copper alloys CuCrZr–IG and CuAl25–IG have the suffix IG (ITER Grade) to indicate modification.

The paper deals with the second group, and the modification of the materials and their properties that are needed to meet the design conditions. Selection of standard materials is not included in this paper.

2. Austenitic stainless steels

On the basis of the in-service experience of fission reactors and the R&D results obtained in the fast breeder reactor and fusion programmes, a solution annealed 316L type steel is thought to be the most suitable material to resist a high dose of radiation, relatively high loads and direct contact with water. Within the family of austenitic 316L stainless steels (SS), different grades with slight variations of the material specification are available in different countries. There are AISI 316L, AISI 316LN, AISI 316NG, SUS 316L, SUS 316LN,

Table 1
Materials proposed for the ITER component design[a]

Description and application	ITER or commercial designation
Austenitic SS for cryogenic applications	SS Cryo
Austenitic SS for ex-vessel components	316 LN
Austenitic SS for cryoline application	316L
Austenitic SS for cryostat and ex-vessel water cooling pipe lines	304L
Nickel-based-alloy for keys and bolts	Inconel 718
Nickel-based-alloy for structural use	Inconel 625
Low thermal expansion Fe–Ni alloy	Invar
Austenitic SS for components fabricated by HIPing	316L(N)–IG1
Austenitic SS for components to be re-welded after neutron irraditation	316L(N)–IG2
Cast austenitic SS for the divertor cassette body and shielding part of modules	316L(N)–IG3
Thin wall tubes of austenitic SS for the first wall	316L(N)–IG4
Thick wall tubes of austenitic SS for piping	316L(N)–IG5
Boronised SS for high neutron shielding efficiency	304B7
Ferritic steel for ferromagnetic insert	430
DS Copper for the heat sink of plasma facing components	CuAl25–IG
PH copper alloy for the heat sink of plasma facing components	CuCrZr–IG
Beryllium for protective armour of plasma facing components	S65C
Tungsten for protective armour of plasma facing components	Tungsten
Carbon-based composite for protective armour of PFCs	CFC
Electrical insulators of modules	Ceramics Al_2O_3 or $MgAl_2O_4$

[a] *Note*: Materials used for standard components are not included in the table.

Table 2
Source elements for key isotopes in steel

Product isotope	Why important	Main source element
Cr^{51}	Maintenance	Cr
Mn^{54}, Mn^{56}	Accident safety, maintenance	Fe
Fe^{55}	Accident safety, maintenance, waste management	Fe
Co^{57}, Co^{58}, Ni^{57}	Accident safety, maintenance, waste management	Ni
Co^{60}	Accident safety, maintenance, waste management	Co via $Co^{59}(n,\gamma)Co^{60}$
Ni^{59}, Ni^{63}	Waste management	Ni
Nb^{91}	Waste management	Nb
Nb^{94}, Mo^{93}	Waste management	Nb
Mo^{99}	Accident safety	Mo
Tc^{99}	Waste management	Mo

X2CrNiMo17-12-2, 0.3X17H14M2, 316L(N)–SPH, 316L(N)–FBR, etc. Unfortunately properties and design allowable are different for the same type of SS in different national standards. To provide a uniform designation and reference to the stainless steel used for in-vessel components it was suggested by the Home Team experts to designate it as 316L(N)–IGX. This designation is specific to the SS selected for ITER and implies the following:

- 316 – type of steel;
- L – low carbon;
- (N) – controlled nitrogen content;
- IG – ITER Grade reflects the adaptation of the steel to specified requirements, e.g., to the allowable range of alloying elements and impurity content, to the quality of mill products and delivery conditions;
- *X* is a progressive number indicating the application of steel in specific ITER components reflecting in different procurement specifications which define the requirements in terms of product form, impurity content, quality assurance procedure, and delivery conditions needed for the different components. The steel is to be in a solution-annealed condition;
- solution annealed AISI 316L is the back-up option.

2.1. Requirements for the steel composition

Most of the requirements for steel are based on the 316L(N)–SPH steel developed for the European fast breeder reactor programme. There is a comprehensive data base for this SS grade, including heat-to-heat variations and product size.

For application in ITER, only minor modifications are required, in order to cope with radiological safety limits and with the re-welding requirement. Additional modifications have also been implemented to take the different manufacturing requirements for each of the components into account. The chemical composition of 316L(N)–IG steel is given in [1,6].

The radiological hazard of SS alloying elements and impurities has been discussed in several ITER papers [9–11]. Activation is dominated by the isotopes Mn^{54}, Mn^{56}, Fe^{55}, Co^{57}, Co^{58}, Co^{60}, Ni^{57}, Cr^{51} produced by transmutation of the elements initially present in steel, namely Fe, Ni, Mo, Cr, Co, Nb [9,10] (see Table 2). All these elements, with the exception of Co and Nb, are constituents, whose contents cannot be changed without affecting the steel properties. Co and Nb are present as impurities in the Fe and Ni ores.

The safety implications of the Co and Nb contents show the following:

- For a fusion reactor, activated cobalt plays an important role in determining the occupational dose level during maintenance and in the case of severe accidents (like a LOCA).
- Reducing the Co content from 0.25% to 0.05% decreases the total decay heat in the VV by $\sim$20% and helps to reduce the activation of components.
- A further decrease of Co to 0.01% does not reduce the decay heat in comparison with 0.05% because the $Ni^{60}(n,p) \rightarrow Co^{60}$ reaction becomes dominant.
- Cobalt is one of the main components of activated corrosion products in the water cooling system. Activated corrosion products are responsible for about 90% of the occupational dose in fission reactors [9].
- Nb produces long-lived radioisotopes that could become important for the decommissioning and waste disposal of in-vessel components. In 316L(N)–IG, niobium is present as a trace element picked up during the melting process from the ferroalloy addition. Existing data indicate that it is possible to reduce Nb to $\leqslant$ 0.01% in 316 SS. A minimum level of Nb that seems technically feasible in industrial heats is $\sim$0.005%. The subsequent reduction of the Nb content will not result in a decrease of the steel specific activity because the main steel alloying elements will dominate.

The VV, back plate (BP), manifold and branch pipe connections must remain weldable throughout the machine's lifetime in spite of the He generated by neutron irradiation. Welding an irradiated steel is possible, provided certain precautions are taken, such as low weld heat input, butting of surfaces, avoiding high tensile stresses, etc. The helium content generated by gaseous transmutation should be kept below a threshold value, which depends on the welding method. The current assumption, based on existing data, is that re-welding of 316L(N)–IG can be successfully carried out when the He content is less than 1 appm. He generation can also be minimised by reducing the boron content of steel.

Neutronic calculations [10] show that decreasing the boron down to 10 wppm results in a decrease of 31% in the He generated in the VV (less then 1 appm of He for 1 MWa/m^2), while decreasing the boron content to 5 wppm results in a decrease of 55% in the He generation. The effect of boron on He generation is most significant for the steel close to the water cooling channels due to thermalisation of neutrons by water. In these regions, the He/dpa ratio increases by a factor of 2–2.5 in a steel containing 20 wppm of B in comparison with a 5 wppm boron steel.

In consequence, the following impurity limitations have been implemented in the steel specifications for different ITER components:

- Steel used for the modules of the primary wall should have a cobalt content of <0.05%.
- Steel used for the back plate and the vacuum vessel should have a cobalt and niobium content as low as possible. Cobalt and Niobium <0.05% and <0.01%, respectively, have been assumed as industrially feasible limits. Taking into account the requirement of re-welding the irradiated material, the boron content should be limited to 10 wppm.
- Cast steel is recommended as one of the options for the divertor cassette body and for the shielding part of the first wall module. Cobalt is limited to 0.05%.
- The following requirements are given for steel used in in-vessel cooling pipes: Co <0.05% and B <0.0010%.

For the parts of the in-vessel components having only a neutron shielding function or are located far away from the plasma (VV port structures), a lower grade material (304L, 316L, SF3M or SF8M) could be used, with benefits in terms of cost saving.

2.2. *Influence of component fabrication process on material properties*

Established technologies and mill products are proposed for manufacturing the back plate and the manifold. Rolled stock, bars, plates, forged parts and other mill products will be used. Hot and cold die forming, rolling in combination with machining and welding or any other commonly used technologies can be applied.

The solid hot isostatic press (HIP) is the reference option, and powder HIP the back-up, for the manufacture of the first wall primary modules and the shielding blanket. Powder HIP is also one of the fabrication methods envisaged for the divertor cassette body.

The HIPing temperature needed to bond SS to SS is in the range of 1050–1100°C with an exposure of 1–4 h at 100–150 MPa. Cooling time after HIPing may vary from 0.5 to 1 h depending on the HIP facility and on the mass of material. As a result of multiple heat treatments, some of the HIPed SS properties may differ from those of the wrought material [12]. Austenitic grain size becomes non-uniform, with an increasing number of large grains. After one HIP cycle (1100°C, 2 h), the grain size distribution is heterogeneous; the ASTM grain size is ~3–5. Some additional ferrite bands may appear in the structure.

Surface preparation and strict quality control are the key factors for obtaining good quality HIPed SS/SS joints with tensile properties within the scatter band of the wrought material.

Microscopic examination of powder-HIPed SS shows a fine grain size and an isotropic and uniform microstructure. In some cases small pores with a diameter <10 μm are observed [13]. This type of defect is not expected to affect the steel properties. The amount of oxygen is not specified for the reference 316L(N)–IG composition and is defined by the manufacturing process; typically the oxygen concentration is approximately 20 wppm. After powder HIPing, SS may contain ~170 wppm of oxygen [13]. Such an oxygen content does not affect the tensile properties, but could have an impact on the fatigue, fracture toughness and weldability. It is planned to study the effect of the high oxygen content on the latter properties.

The main advantage of the powder HIPing process is the possibility to consolidate in one single step components with intricate shapes; it could be cost competitive with solid HIPing for the manufacture of shielding blanket modules and for the cassette body. Selection of fabrication procedures, quality control of the powder and modelling of shrinkage are key factors in producing near final shape components with narrow tolerances. Large demonstration blocks up to $1250\times650\times250$ mm^3, weighing 1500 kg, with internal cooling channels have been produced and are being characterised [14,15].

The available data on tensile properties of solid-HIPed and powder-HIPed SS are shown in Fig. 1 [6,14–18]. All data of powder-HIPed steel are within the scatter band of wrought SS, the ultimate tensile strength being close to the average value of wrought SS and the yield strength slightly higher. Very little variation of the room temperature yield and the ultimate tensile strength (±5 MPa) is observed in powder HIP blocks fabricated with HIP temperatures varying from 1050°C to 1085°C and pressures of 100 or 150 MPa [14]. Solid-HIPed material has slightly better ductility and lower strength than wrought SS. Most of the strength data is close to the minimum confidence limit (−95%) (see Fig. 1).

Casting was proposed as a manufacturing method for the divertor cassette body, and the shielding part of the

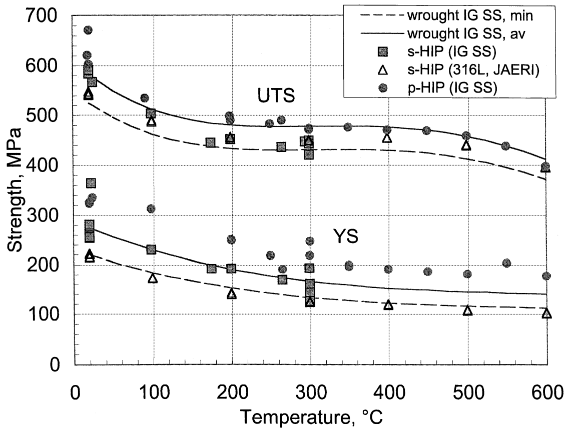

Fig. 1. Tensile strength of powder- and solid-HIPed stainless steel compared with average and minimum specified values of the wrought material [6,14–18]. (*Note*: JAERI data [17] (open triangles) refer to the mechanical properties of SS 316L and not SS 316L(N)–IG after solid HIPing. This explains the low yield strength values.)

first wall modules, because it is cheaper than conventional welding or HIPing. Casting with grades similar to 316L and 316LN is included in ASME (and ASTM) with grade designations CF–3M and CF–3MN, respectively. Requirements additional to the chemical composition of cast SS should be implemented in the material specifications, in particular Co <0.05% and Nb <0.005%. Taking into account the requirement of re-welding the irradiated material, the boron content should be <10 wppm. These requirements are not included in the ASME or ASTM specifications. Nevertheless, it was demonstrated that boron and cobalt can be reduced to ~0.0006% and 0.077%, respectively.

Cast materials are characterised by large grains (>2 mm) and by non-uniform distribution of alloying elements, due to segregation during solidification. The structure of cast austenitic SS is characterised by a columnar structure [19]. In samples examined, small areas of ferrite (~0.5–1%) were detected using ASTM practice A800 and Severn Gauge measurements, consistent with results from magnetic permeability measurements. Nonmetallic inclusions from mould products have not been observed in the castings for the prototype of the divertor cassette body and are not present when employing good foundry practice. Cast SS exhibits relatively good fracture toughness, and very low crack growth rate indicative of high resistance to cracking. However, cracks that tend to grow parallel to the load direction might be a result of columnar grain structure.

The Straus method, the EPR method and crack growth rate measurements did not indicate any susceptibility of cast SS 316L(N)–IG to stress corrosion cracking (SCC) [19,20].

In general, both cast and wrought SSs can be successfully welded using the same or similar processes and techniques [6,19].

2.3. *Effect of irradiation*

After the first wall fluence ~0.3 MWa/m^2, the peak irradiation dose in the SS of the PFCs (plasma facing components – primary wall, limiter, baffle, divertor, blanket) will not exceed 2 dpa with a peak He generation of about 55 appm. Temperatures will be in the range 100–300°C. Hardening of the steel will occur, but predicted uniform and total elongation of the wrought steel will remain higher than 10% and 20%, respectively, at the end of the operation phase. For the solid-HIP steel, the estimated uniform and total elongation will be above 18% and 31%, respectively. Therefore, materials will be relatively ductile and retain work hardening capability.

Initial neutron data on 316L(N)–IG, prepared by powder HIP and irradiated at ~75°C to 2 dpa, indicate that the tensile properties are similar to those of wrought material. However, for powder-HIPed material the fracture toughness decreases with increasing dose from ~1000 kJ/m^2 of the unirradiated steel to ~250 kJ/m^2 after 2 dpa at 350 K [21].

The highest irradiation dose of steel will be in the PW manufactured by solid-HIP. R&D shows that solid-HIPed steel has better initial ductility than wrought SS [22]. After irradiation to a dose of ~10 dpa, the ductility remains relatively high, and the uniform elongation (UEL) remains >2% for both the base metal and the SS/SS joints after the HIP manufacturing cycle.

For the manifold, back plate, and vacuum vessel the irradiation dose will be low, less than 0.06 dpa for a fluence of 0.3 MWa/m^2, and temperatures will be <220°C. This dose will not result in significant property changes, and design allowables for the unirradiated steel have been used for these component designs. Helium generation in the VV will be below the re-weldability limit, <1 appm. For the manifold, the helium generation exceeds the re-weldability limit. Large fluctuations of helium levels occur in the steel near the water channels due to moderation of the neutrons in water and due to streaming effects in the gap between FW modules. The reduction of boron from 20 ppm (as in the reference steel grade) to 10 ppm lowers the helium generation by ~44% while reducing the boron even further results in more than a factor of two suppression in the helium production. Therefore, the boron concentration in steel used for the manifold should be as low as feasible industrially.

3. Heat sink copper alloys

In the design of ITER plasma facing components, a high thermal conductivity material, the heat sink, is interposed between the armour and the cooling channels. The main function of the heat sink is to transport elevated heat fluxes to the cooling water, thus reducing thermal stresses in the structural material.

Two materials have been retained as candidate heat sink materials for the high heat flux components: CuCrZr and GlidCop®Al25. Starting from the original specification, a tighter specification for composition and a specific heat treatment were proposed for the CuCrZr alloy, with the definition of an ITER grade specification, CuCrZr–IG. The fabrication process of GlidCop®Al25 was optimised, with an improvement of ductility and reduction of anisotropy. The optimised grade is indicated as CuAl25–IG for ITER application.

3.1. Composition of copper alloys

The proposed composition of CuCrZr alloy differs from the standard one mainly by its narrower range of the Cr (0.6–0.9%) and Zr (0.07–0.15%) contents. In different national standards the chromium content varies from 0.4% to 1.5%; the zirconium content varies from 0.03% to 0.25%. The reason for limiting the Cr content in the narrower range of the alloy is that it may result in the formation of coarse Cr precipitates which affect the radiation resistance. Zr promotes the hardening of the alloy by providing a good homogeneity of precipitates; moreover, it influences the aging time and the recrystallisation temperature. Limitation of oxygen (<0.002%) and of the total amount of impurities (<0.03%) is required for the same reason and for a better resistance against embrittlement.

Based on industrial experience the reference ITER heat treatment for CuCrZr–IG is as follows: solution anneal at 980–1000°C for 1 h, water quench then age at 450–480°C for 2–4 h. Aging could be performed either before, during or after the component manufacturing.

The chemical composition and quality requirements to the CuAl25–IG products are given in MAR [6]. The material is provided by the manufacturer OMG Americas under the trademark Glidcop©Al25–LOX–CR (low oxygen, cross-rolled). A high temperature annealing (at 950°C) is performed after cross-rolling.

3.2. Influence of component fabrication on the tensile properties of Cu alloys

The tensile properties of reference copper alloys, CuAl25–IG and CuCrZr–IG are shown in Fig. 2 [6,23]. For both alloys, the allowable stress intensity limit, S_m is dominated by the ultimate strength, and not by the yield strength as in the austenitic SS. CuCrZr exhibits a small sensitivity to the strain rate during tensile testing [24]. The yield strength increases 7% when the strain rate increases from 0.00039 to 0.056 s^{-1} at 20°C. Similar

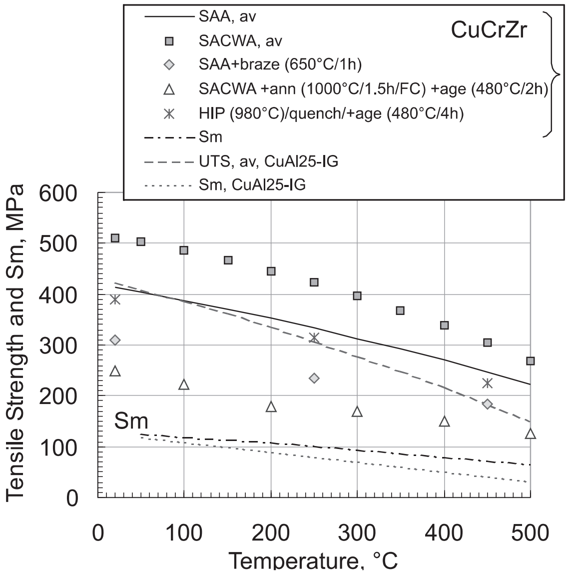

Fig. 2. Temperature dependence of tensile strength (UTS) of CuCrZr alloy after different thermo-mechanical treatment and CuAl25–IG [6,23–27]. Abbreviations: SAA – solution anneal and age; SACWA – solution anneal, cold work and age; FC – furnace cool.

behaviour has been observed at higher temperatures up to 300°C.

During component manufacturing the base material is generally subjected to additional thermal cycles e.g., welding, brazing, HIPing, etc. These heat treatments may affect the physical and mechanical properties of CuCrZr by dissolution and/or by coarsening of precipitates. Over-aging can dramatically affect the mechanical properties by modifying the size and volume fraction of precipitates. It is impossible to give precise data on the residual material properties in general terms, because of the many different manufacturing options for PFCs and because of the poor experimental database. In extreme cases such as HIP bonding with a slow cooling rate, precipitation hardening could be completely lost, the residual strength of CuCrZr being practically that of pure copper [24].

The effect of two typical manufacturing options, brazing and HIPing, on strength is shown in Fig. 2 [23–27]. The SAA heat treatment results in a relatively good combination of strength and ductility. The SACWA treatment gives better strength but the ductility is almost two times lower.

Brazing, even at relatively low temperatures, results in a significant decrease of strength in the case of long-term exposure and/or slow cooling rates (Fig. 2). Brazing or HIPing at high temperature can be combined with solution annealing. In this case the brazing/HIPing temperature should be about 950–980°C, followed by fast cooling and aging at 475–480°C. The most critical step is the cooling rate from the brazing/HIPing temperature. A slow cooling rate results in a low strength of the material. Aging after slow cooling does not restore the full strength because the alloying elements do not remain in a supersaturated solid solution (Fig. 2). If a fast cooling rate can be realised after high temperature brazing/HIPing, the loss of strength can be minimised. Fig. 2 shows the result of HIPing at 980°C using a special HIP-quench facility [27]. The HIPed specimen was cooled at an average cooling rate >2°C/s in the critical temperature range 980–500°C. Helium circulation was used to increase the heat transfer coefficient for the quench up to 1200 W/m^2 K.

CuAl25–IG shows a much less sensitivity to heat treatment compared to CuCrZr. Heating up to 900–1000°C has little effect on the tensile properties [25,28]. The thermal stability of CuAl25–IG has also been confirmed by investigating the properties of specimens taken from mock-ups after a real manufacturing cycle (HIPing at 950–1050°C) [36,37]. Thermal stability is one of the main advantages of CuAl25. This allows a wider flexibility in the use of different joining technologies for high heat flux components. HIPing and brazing can be used without detrimental effects. However, a certain variation of the mechanical properties is observed depending on different manufacturing procedures, test conditions and the measuring laboratory.

Both CuAl25–IG and CuCrZr–IG, with their improved manufacturing technology, have better ductility than the commercial products (extruded MAGT-0.2, HIPed Glidcop® Al25).

3.3. Neutron irradiation effects on mechanical properties of copper alloys

Neutron irradiation effects on the mechanical properties of copper alloys depend strongly on the temperature. At temperatures extending from room temperature to 220–300°C, radiation hardening is the dominant effect. Associated with radiation hardening, a loss of work hardening capability and flow localisation occur [25,29–32].

At temperatures exceeding 220–300°C, radiation hardening disappears and radiation softening starts. The transition temperature between the two phenomena is not defined precisely. In [32] it is shown that radiation hardening of copper decreases significantly above the temperature of about 220°C. Another experiment based on the ductility and tensile strength changes due to irradiation gives a value of 250–300°C depending on the materials [6,25,30,31]. In the temperature range ~220–300°C both precipitation-hardened (PH) and dispersion-strengthened (DS) alloys show ductile fracture and maintain a good level of strength (slightly increasing due to irradiation) in the dose range of interest for ITER. The work hardening capability is also improved.

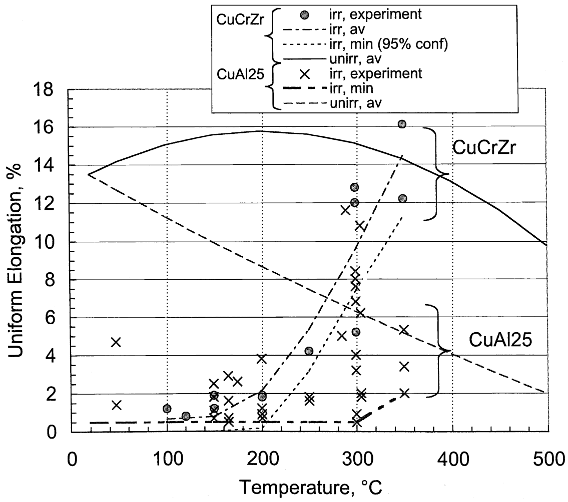

Fig. 3. Uniform elongation of CuCrZr–IG and CuAl25–IG alloys before and after irradiation [29–34].

At irradiation temperatures >300°C, ductility increases and strength decreases [6,31].

The change of uniform elongation after neutron irradiation of the reference alloys CuAl25–IG and CuCrZr–IG is shown in Fig. 3. The ductility of CuCrZr–IG increases with increasing irradiation temperature. At irradiation temperatures <200–250°C the uniform elongation is below 2%, therefore criteria for immediate plastic strain localisation and fracture due to loss of ductility should be assessed. At temperatures exceeding 250°C the material will be ductile.

It is difficult to perform a statistical analysis for the uniform elongation of CuAl25–IG due to significant scattering of experimental data (Fig. 3). There are many experimental points below 2% for the irradiated CuAl25 in the temperature range 100–300°C. Therefore, in all temperature ranges plastic strain localisation and local fracture criteria should be used for structural analysis. A lower values of uniform elongation, 0.5%, was taken for the design analysis as the most conservative approximation for the temperature range 100–300°C.

4. Plasma facing materials

The choice of plasma facing materials for different components is determined mainly by plasma–wall interactions. Beryllium has been chosen as the armour material for ~80% of the total surface exposed to the plasma (primary wall, upper baffle and port limiter). The main reasons for the selection of Be as armour are low risk of plasma contamination, low radiative power losses, good oxygen gettering ability, absence of chemical sputtering (in comparison with carbon), low bulk tritium inventory, and the possibility of in situ (or in hot cell) repair of damaged surfaces. W has been selected as the material for baffle and divertor areas with a high concentration of neutral particles, because in this area the key issue is erosion lifetime, and W has a lower erosion rate, due to its low sputtering yield and its higher sputtering threshold energy, as compared to those of beryllium and carbon. Another advantage of W is its low tritium retention. The plasma compatibility of W is an issue, because a small amount of W in the confined plasma region could lead to a very large power loss from the plasma. In PFC areas exposed to high thermal fluxes during normal operation, and large energy excursions during plasma instabilities (lower vertical target, dump plate), CFC is selected. CFC can resist very high heat fluxes and does not melt. However, its use has to be restricted to these regions, because of chemical erosion and tritium retention, especially in co-deposited layers. These features of the plasma wall interaction in ITER and selection of armour materials have been discussed in [35,36]. The selection of specific grades of Be, W and CFC is driven by resistance to thermal fatigue and thermal shock, and by physical and mechanical properties in the unirradiated and irradiated states.

4.1. Beryllium

Commercially available Be grades from Brush Wellman, USA and from the Russian Federation have been evaluated as candidate materials. The selection of the optimum grade is driven by those properties which are very sensitive to the impurity levels, grain size, method of production, thermomechanical treatment, and which usually differ for the different Be grades. Thermal fatigue resistance of Be is the most important factor because cracking could not only lead to enhanced armour erosion, but also to crack propagation to the heat sink structure. Neutron irradiation resistance is another factor to be taken into account.

First screening experiments, based on thermal fatigue resistance and thermal shock experiments, indicated that the most resistant grades are S–65C VHP (Vacuum Hot Pressed) and, to a lower degree, DShG-200 [37,38]. Based on these considerations, and also on the basis of its availability and better database, S–65C VHP was selected as the reference grade. DShG-200 was selected as the back-up.

Another material proposed to be used in ITER is plasma sprayed beryllium. Be plasma spray has the potential to be used for the manufacturing of the first wall modules and for the in situ or hot cell repair of the damaged armour, which would save maintenance time and reduce radioactive waste. By comparison with

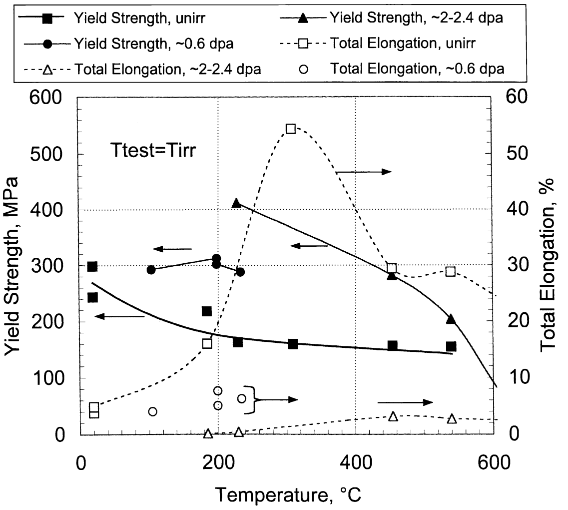

Fig. 4. Influence of neutron irradiation (damage ∼2–2.45 dpa, [40], ∼0.6 dpa [41]) on the mechanical properties of Be S–65C VHP.

S–65C, plasma sprayed Be has lower thermal shock resistance, lower mechanical properties, and higher porosity.

Neutron irradiation typically leads to degradation of the Be properties [39]. For a first wall fluence of 0.3 MWa/m^2 (temperature range 240–480°C, damage level ∼1 dpa, He concentration −1000 appm) there is no effect on the physical properties, and swelling is expected to be less than 1%. The main concern is the embrittlement of Be at low irradiation temperatures (less than 300°C). For S–65C VHP irradiated up to 2–2.45 dpa and tested at 185°C and 230°C, brittle fracture of Be has been observed [40], while for the same material irradiated at 110–275°C up to 0.65 dpa [41], the tensile ductility of irradiated samples was still at a reasonable level of a few percent, (Fig. 4). Similar behaviour has also been observed for other grades. The embrittlement of Be at low temperature could lead to brittle destruction of the tiles and affect the thermal erosion of Be during transient events [42]. The use of Be tiles, without critical defects that may result in crack initiation, could partly solve this problem. Additional R&D is needed to study the fracture mechanics and to clarify the Be behaviour under disruption and vertical displacement events (VDE).

4.2. Tungsten

The preliminary selection of W grades has been made taking into account the near term availability, cost, technological features, and behaviour under thermal fatigue and disruption conditions. Several W grades were selected for investigation: pure sintered W in the cold worked or recrystallised conditions, W–Mo–Y cast alloy, W–Re alloys, dispersion strengthened W–1%La_2O_3 and W–0.3%Y_2O_3 alloys, plasma sprayed W, W produced by chemical vapour deposition (CVD), and single crystal W. The behaviour of W grades under different thermal loads and some mechanical properties are summarised in [43]. Two grades are presently recommended for more detailed investigations: sintered W in the cold worked and stress relieved conditions, and W–1%La_2O_3.

Typically for bcc metals, irradiation leads to an increase in the ductile-to-brittle transition temperature (DBTT). A summary of the available data on the influence of neutron irradiation on the change of DBTT is presented in [44]. Based on these limited data it could be concluded that all W grades have the problem of becoming brittle at an expected fluence of 0.3–0.5 dpa in the region near the heat sink at an irradiation temperature less than ~500°C. It is recommended to avoid the use of W in geometries with crack initiators. One possible way to solve this problem is to orientate W armoured design toward those concepts which reduce thermal stresses in the armour and in the joints (brush, rod or lamella structures).

4.3. Carbon fibre composites

CFC have been selected as reference material in certain parts of the divertor due to their high thermal shock and thermal fatigue resistance (low crack propagation) and their high thermal conductivity in comparison with conventional graphites. The selection of the reference CFC grade is described in detail in [44]. The preferred material is 3-d CFC because of its more isotropic properties and the higher thermal shock resistance. The proposed grades are Sepcarb® NB31 and NIC-01. The available thermomechanical properties of candidate CFCs are collected in [6].

Reviews of the influence of neutron irradiation on the properties of CFCs have been published recently [39,44]. For a fluence ~0.1–0.3 dpa and for the application of CFC only in limited areas relatively far from the plasma core, the changes in properties are not expected to be crucial except for thermal conductivity. CFCs with high initial thermal conductivity retain high conductivity after irradiation, but at low irradiation temperatures (less than 300°C) the thermal conductivity could be ~3–5 times lower than the unirradiated CFC. The decrease in the thermal conductivity due to neutron irradiation leads to an increase in thermal erosion under disruption conditions [45]. After neutron irradiation at a fluence $\sim 5.6\times10^{20}$ n/cm^2 (~0.3 dpa) the thermal erosion was about twice as higher as that of the unirradiated material. However, in spite of an increase of erosion due to irradiation, the total erosion (including ion and chemical sputtering, as well as thermal erosion) is within that allowable for the component lifetime.

5. Conclusions

Materials for the ITER in-vessel components have been selected. Standard industrially available materials (such as 304L, 316L, 316LN, 304B7, 430; titanium alloy Ti–6Al–4V, nickel alloys Inconel 625, Inconel 718; tungsten alloys PM–W, W–1%La_2O_3,) are preferred options for the design. Limited R&D for these materials is in progress to investigate the specific working conditions of ITER components.

Some modifications have been implemented for the other group of materials (steel 316L(N)–IG, copper alloys CuCrZr–IG and CuCrZr–IG (Glidcop© Al25), beryllium S–65C or DShG-200 grades and the CFC) with more extended research and development required, including the study of the effect of component manufacturing cycle and irradiation.

As a consequence of the component manufacturing cycles, the structure and properties of SS have been changed. R&D shows that tensile properties of HIPed steel are within the design allowables. Solid-HIP results in an increase of ductility and a decrease of strength. In some cases the yield strength of solid-HIPed base metal is close to the minimum specified values. Powder-HIP gives tensile properties above the average of the wrought material. The fracture toughness of powder-HIPed steel degrades faster under irradiation than for the wrought material. Cast material shows tensile strength properties below the average of the wrought material. The stress corrosion cracking resistance of cast steel is no worse than that of the wrought steel.

Two commercially available copper alloys have been selected as reference heat sink materials for the PFCs, one age hardenable CuCrZr–IG alloy and one DS alloy (CuAl25–IG). CuAl25–IG is suitable for high temperature manufacturing cycles of ITER components. The properties of CuCrZr can significantly degrade after component manufacturing, so the manufacturing process should be thermally suitable.

For both alloys radiation hardening and loss of work hardening capability occur at low temperatures. For irradiation above 300°C, a decrease in strength and increase in ductility are observed in both alloys. In a comparison of the irradiated properties of CuAl25–IG and CuCrZr–IG, the latter alloy is better.

Reference grades of armour materials have been proposed: Be S–65C VHP (with DShG-200 as a back up) along with plasma sprayed Be; pure W in the cold worked and stress relieved conditions and W–1%La_2O_3; advanced 3-D CFCs Sepcarb® NB31 and NIC-01.

The performance of armour materials, analysed as a part of the design, is adequate in the unirradiated condition. More effort is needed to clarify the combined effects of neutron irradiation and thermal loads.

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