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# Neutron irradiation effects on plasma facing materials

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**Abstract**

This paper reviews the effects of neutron irradiation on thermal and mechanical properties and bulk tritium retention of armour materials (beryllium, tungsten and carbon). For each material, the main properties affected by neutron irradiation are described and the specific tests of neutron irradiated armour materials under thermal shock and disruption conditions are summarized. Based on current knowledge, the expected thermal and structural performance of neutron irradiated armour materials in the ITER plasma facing components are analysed. 

## 1. Introduction

Beryllium, tungsten and carbon-based materials are typically considered as plasma facing armour materials for the Next Step fusion device. Their main functions are to provide adequate conditions for plasma burn (minimize the plasma core impurity contamination), to protect the wall structures from high heat fluxes and contact with the hot plasma, and to satisfy the required erosion lifetime. Operational conditions for armour materials are very complex and include particle bombardment, thermal fatigue, neutron irradiation, different types of thermal shocks, etc. The critical issues related to plasma–wall interaction for fusion devices and for ITER can be found in [1,2]. For the Next Step fusion machine, neutron irradiation is an important factor that has to be taken into account during the selection of the armour materials and during analysis of the performance of the plasma facing components.

Armour materials typically have no structural function and the requirements for their performance are lower in comparison with structural materials. However, the degadation of the properties by neutron irradiation could lead to a decrease in the erosion lifetime, to a loss in the mechanical integrity of the material itself and in the joining with heat sink etc.

This paper reviews recent data on neutron irradiation influence on the properties of plasma facing materials and analyses possible changes in the performance at typical fusion operational conditions. For the conditions expected in ITER the performance of armour materials is analyzed.

## 2. Neutronic features of fusion spectrum

It is well known that the displacement damage and transmutation products during neutron irradiation depend on neutron spectrum. For armour materials (Be, W, C) the values of damage (in dpa), the amount of He, H, T and some other elements for the 1998 ITER design at neutron a fluence of 0.3 MW*a/m$^2$ are presented in Table 1, [3]. The presence of high energy neutrons leads to high He generation in Be (~1000 appm He/dpa) and in carbon (~380 appm He/dpa). For W the He production is negligible, however, some solid transmutants, Re, Ta, Hf, Os and others, will be produced.

Special analysis is required to predict the property changes based on data from fission irradiation. After irradiation in fission reactors, the typical values of ration (appm He/dpa) for Be is ~100–250 and only 2–3 for

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Table 1
Operational conditions for armour materials for the 1998 ITER design, fluence of 0.3 MW*a/m$^2$ ($\sim 2.0 \times 20^{21}$ n/cm$^2$), [3]

| Material | Damage (dpa) | He (appm) | Other impurities (appm) | Irradiation temperature (°C) | Transient temperature/time (°C/time) |
|---|---|---|---|---|---|
| Be | 1 | 1000 | Tritium: ∼16 | 240–280$^{FW,L}$ 290–480$^{B}$ | 390 – 700$^{L}$/∼30 h up to MP/ms |
| W | 0.3–0.5 | 0.2 | Re: 4800, Ta: 170, Os: 75 | 200–1000 | up to MP/ms |
| CFC | 0.1–0.2* | 60–120 | – | 200–1000 | up to 3000 /ms |

FW – First Wall, L – Limiter, B – Baffle, * – with planned replacement of divertor, MP – melting point

carbon. For Be, it seems that for high temperature (>500°C) properties, such as helium embrittlement and swelling, the value of He generation is important and has to be used as a basis for comparison, whereas to predict the mechanical properties at low temperatures (less than ∼300°) the value of the displacement damage has to be compared [5]. For carbon there is no possibility (at least not without special designed experiments) to take into account the influence of the high He content. For W the presence of solid transmutation products has to be taken into account. Greenwood [4], has performed the calculation of Re and Os transmutation products in W in reactors with different neutron spectra, and significant production of Re and Os, especially for mixed neutron spectra (up to ∼(1–2 at.%)/dpa) already at low fluence, has been reported.

## 3. Effect on beryllium

The latest reviews on the effect of neutron irradiation on the properties of beryllium have been presented in [5,6]. The data presented in these reports have been generated in the last 10–20 years. Due to developments in the Be production technology, the new and old grades differ greatly in the morphology of Be powder, impurity content, etc., and the old data can only be used only for qualitative prediction of the Be performance.

In this review the recently generated data for the modern Be grades are summarised.

### *3.1. Change of beryllium properties*

*Thermal conductivity*. Some data exist on the effect of neutron irradiation on the thermal conductivity of beryllium. For S-65C Be irradiated up to $10^{21}$ n/cm$^2$ (∼0.74 dpa) at ∼300°C the thermal conductivity was similar, within experimental error, to that of the unirradiated value [7]. For Be S-200F irradiated at 200°C up to a fluence of $4.5 \times 10^{20}$ n/cm$^2$ (E > 1 MeV), (∼88 appm He, ∼0.6 dpa), the thermal conductivity of irradiated Be decreased to about 90–95% of the initial value [8]. After high temperature annealing, in specimens with 29% and 63% swelling, thermal conductivity at ambient temperatures decreased to about 70% and 40% of that of the unirradiated specimen, respectively. The similar behaviour of the thermal conductivity of porous Be has been described in [9].

*Swelling*. A comprehensive review of the data for old Be grades and the proposed empirical correlations have been reviewed in [5]. Recently the ANFIBE code has been developed, which allows calculation of the swelling in the Be over wide temperature and fluence ranges [10].

Nevertheless, new data for the recent Be grades are needed for code verification. In recent years some additional data have been published [7,11–15]. Moons et al. [11] have studied the swelling of the S-65C VHP, S-65 HIP, S-200F VHP and S-200F HIP Be grades irradiated to a fluence of $1.5–1.6 \times 10^{21}$ n/cm$^2$ (∼520–570 appm He, 1.8–2.4 dpa) in the temperature range 235–600°C. It was shown that at high temperature (>500°C) the swelling of the S-65C grades is slightly higher than for other grades. Low temperature irradiation (300°C, fluence $\sim 1 \times 10^{21}$ n/cm$^2$) revealed very low swelling (∼0.027%) of S-65C Be [7].

The results of a study of the modern Russian Be grades have been published recently [13–15]. Different types of Be with different grain size ∼8–26 μm, BeO content (0.9–3.9 wt%), dispersity of BeO (0.04–0.5 μm) and with different consolidation methods have been studied. Neutron irradiation has been performed at 550–780°C and a fluence of $1–3.7 \times 10^{21}$ n/cm$^2$. The results demonstrated that Be grades with the smallest grain size and high BeO content (grades TRR, DRR, TIP, DIP), have the lowest swelling behaviour at high temperatures. The main reason is that the presence of a large quantity of small beryllium oxides prevents the migration of the He and formation of the large He bubbles which are responsible for the swelling. The selected data for S-65C Be and for Russian grades are presented in Fig. 1. Further modification of existing codes is needed with a goal to include the influence of the microstructural parameters such as grain size and BeO morphology.

*Mechanical properties*. The general qualitative trends in the strength and ductility behaviour of neutron irradiated Be are the following [5]:

- at low and moderate (20–500°C) irradiation temperatures the strength is typically increasing, while ductility is decreasing, in some cases to zero value;

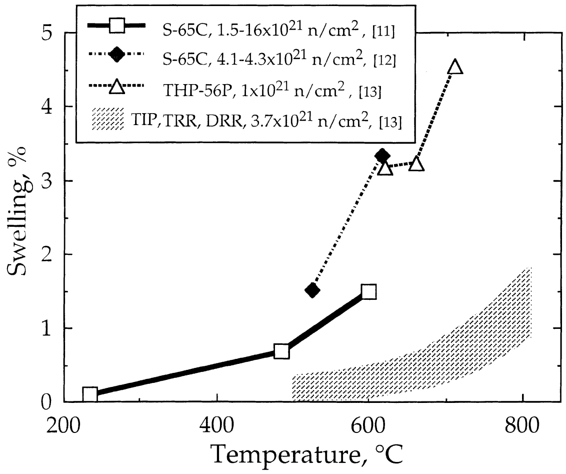


Fig. 1. Swelling of S-65C VHP Be and some Russia Be grades as a function of temperature at different neutron fluence [11,13].

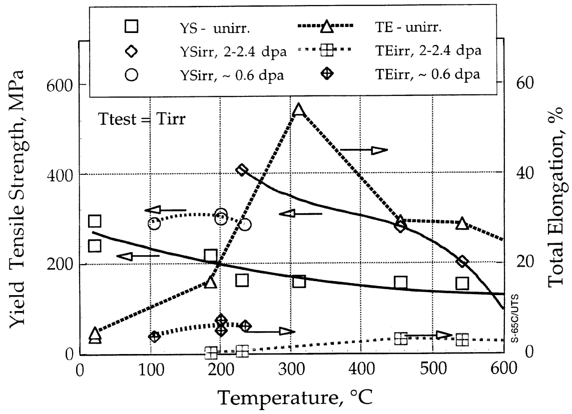


Fig. 2. Temperature dependence of the yield strength (YS) and total elongation (TE) of Be S-65C after neutron irradiation [7,17].

- at high temperature (more than ~600°C) the ductility decreases without increase in the strength;
- increase of fluence leads to saturation in the behaviour of the strength and ductility.

Recently published papers give some quantitative data on the mechanical property changes for modern Be grades. For S-65C VHP Be, tensile data after neutron irradiation have been reported in [7,16,17]. Snead [7] has measured the mechanical properties after irradiation at temperature 100–275°C and fluence 0.05–1 $\times$ $10^{21}$ n/cm$^2$ (0.04–0.74 dpa). The elongation of S-65C Be in the temperature range 100–240°C after irradiation up to damage dose of 0.6 dpa remains at the ~3–5% level. Mechanical properties of S-65C VHP, S-65C HIP, S-200F VHP and S-200F HIP beryllium grades after irradiation at 185–610°C and fluence 0.85–1.6$\times$$10^{21}$ n/cm$^2$ has been reported in [16,17]. At the lowest irradiation temperature (185°C, 1 dpa) significant strengthening and severe embrittlement have been observed, and the remaining ductility was ~0.1%. At higher irradiation temperatures the ductility increases up to maximum ~3–5% at 350–500°C at a fluence of ~2.5 dpa. In Fig. 2 the data from [7,17] for Be S-65C VHP are summarised.

Similar behaviour for the Russian grades has been observed by Fabritsiev [18]. The interesting features related to strong anisotropic materials (e.g. DShG-200, TShG-200) have been observed: the elongation of DShG-200 in the direction transverse to the moulding pressure at $T_{irr}$ ~350–400°C and a fluence of ~4–5 dpa, was still ~30%, whereas in the longitudinal direction the elongation drops from 25% down to 0%. The saturation of hardening and embrittlement of DShG-200 occurs already at a fluence of ~<0.7 dpa ($T_{irr}$ = ~ 550°C) and remains almost without change up to a fluence of ~5 dpa [18].

Irradiation at high temperatures (more than ~600°C) leads to a decrease in the ductility. The main reason is the He embrittlement. This is accompanied by a loss of strength and the fracture mode is intercrystalline [18–20]. This behaviour of irradiated Be was observed for S-65C irradiated up to a total fast neutron fluence of 1.3–4.3$\times$$10^{21}$ n/cm$^2$ ($E > 1$ MeV) at 327–616°C [21].

Kuprijanov [13,14,22], has studied the mechanical properties of the modern Russian Be grades with different content and morphology of the BeO, grain size etc. after high temperature irradiation (700–750°C). The typical loss of ductility has been observed. For grades with high swelling resistance (high BeO content and low grain size) some ductility at the level of a few per cent still remains, while for other grades the failure was brittle. As for swelling resistance, this could be explained by the reduced He migration in these materials.

A few reports deal with the effect of neutron irradiation on fracture toughness of Be in the temperature range 20–600°C. Beeston [23], reported that fracture toughness of Be decreases from 12 MPam$^{1/2}$ by 60% after irradiation at 66°C and fluence 3.5–5 $\times$ $10^{21}$ n/cm$^2$. Chaouadi [17], studied the effect of neutron irradiation (fluence 0.65–2.45 $\times$ $10^{21}$ n/cm$^2$, temperature 200–600°C) on the behaviour of the four Be grades S-65C VHP, S–65C HIP, S-200 VHP and S-200 HIP. Initial fracture toughness of all these grades is very similar. Irradiation decreases (by factor 4–5) the fracture toughness, especially in the temperature range 400–450°C.

### 3.2. Behaviour of irradiated Be at high heat fluxes

Armour materials are also subjected to thermal fatigue, thermal shocks, disruptions, etc., during operation. The behaviour of the neutron irradiated materials at these specific conditions cannot be predicted based on knowledge of the materials properties and has to be studied. This work has started recently for neutron irradiated Be [24]. Several Be grades (S-65C, plasma sprayed Be, TShG-56, etc.) have been irradiated at 350°C and 700°C at a fluence of ~0.35 dpa. After irradiation, samples have been subjected to thermal shocks (15 MJ/m$^2$, 5 ms duration, number of shocks – 5). For all studied materials the thermal erosion was 1.5–2.5 times higher than for unirradiated materials. The main reason is that thermal erosion is not simple evaporation, but also loss of some particles due to the brittle destruction of the surface. Metallographic examination has shown more crack formation in the surface of neutron irradiated beryllium.

### 3.3. Tritium retention in neutron irradiated Be

An investigation of neutron effects on bulk tritium retention in Be has been reported in [25,26]. Tritium loading in these experiments has been performed from the gaseous phase. Kwast [25] has reported the data on tritium retention in S-200E Be (irradiated at 40–50°C and a fluence of $4 \times 10^{22}$ n/cm$^2$ or 40 dpa) and B-26 Be grade (irradiated at 270°C, fluence of $6 \times 10^{20}$ n/cm$^2$). It was concluded that the tritium retention increases with neutron fluence by a factor 10 at 40 dpa. Wu [26] has reported data for Be S-200 HIP after irradiation at 235–600°C and damage dose ~1.5–1.7 dpa. The results show that at this condition the bulk retention increases by a factor 3.

However, tritium retention in beryllium is expected to be less serious than previously anticipated [1]. Recent implantation experiments at low energies, high fluxes and high particle fluences in beryllium showed that after reaching fluences of about $10^{22}$ atoms/m$^2$, further implantation results in a negligible increase in tritium inventory. The possible reason is the creation of surface-connected porosity that provides a rapid return path back to the plasma. This is also expected to affect the contribution to trapping at damage sites created by neutrons. Further experiments are needed to determine the depth dependence of tritium concentration in the bulk of irradiated material.

## 4. Effect on tungsten

### 4.1. Change of tungsten properties

*Physical properties*. Some data are available on the influence of the neutron irradiation on electrical conductivity [27–30]. An increase of electrical resistance of 24% was measured in pure W after a maximum dose of $2 \times 10^{22}$ n/cm$^2$ (~4 dpa) in a fast reactor. After irradiation in a reactor with a mixed spectrum an increase of ~15% has been observed after irradiation at a fluence ~$10^{21}$ n/cm$^2$ [26]. Since in accordance with the Wiedemann–Franz law the electrical resistance of metals is inversely proportional to thermal conductivity, the latter should change accordingly.

*Swelling*. There is no systematic study on the swelling behaviour of W and W alloys. Only in two papers [31,32] include the swelling temperature dependencies and in other papers only isolated data points have been reported, [33–38]. There is no possibility to describe the fluence dependence: swelling values for pure W reported in [31] was very similar to data from [32], but the difference in neutron fluences was ~10 times. The swelling maximum is reached at ~800°C and there is no additional swelling peak at least up to a 1500°C. The maximum reported swelling was ~1.7% at 9.7 dpa and 800°C [31]. Addition of rhenium suppresses swelling [32,39].

*Mechanical properties*. W, as is typical for a bcc metals, is embrittled after neutron irradiation due to radiation hardening and loss of strength at grain boundaries due to contamination by interstitial impurities [28,40–42]. These lead to an increase in the ductile-to-brittle transition temperature (DBTT). It should be noted that the value of DBTT depends strongly on the testing method and for the correct comparison of different results this has to be taken into account.

In [40], sintered W was irradiated at 100°C up to a dose of $4.2 \times 10^{21}$ n/cm$^2$ ($E > 0.1$ MeV). The DBTT of unirradiated W in this study was reported as ~400°C and after irradiation the DBTT increased up to 600°C. In [41], W samples were irradiated at 371–380°C up to a fluence of $0.5–0.9 \times 10^{22}$ n/cm$^2$ (~1–2 dpa). The DBTT measured by tensile testing in the unirradiated condition was ~60°C and after irradiation the DBTT was ~230°C. In [28] loss of strength after irradiation was observed. At $T_{irr} = T_{test} = 300$°C, the W specimen had a tendency to brittle fracture at stress levels 5–10 times lower than in the unirradiated condition. Conversely, irradiation and testing in the range 500–800°C result in appreciable hardening of W. These data differ from those reported in [40,41], but this could be explained by the different chemical composition and method of testing. As also reported in [28] annealing at high temperature (1200°C, 1 h) can partially restore the properties of W (especially ductility).

A comparison of the influence of neutron irradiation on DBTT in W, W-10%Re alloy and W3.4Ni1.6Fe (Densimet18) has been reported in [42]. Neutron irradiation has been performed at 252–302°C up to a fluence of $5.4 \times 10^{22}$ n/cm$^2$. Three-point bend tests at different temperatures have demonstrated a strong increase of DBTT with increasing of neutron fluence, especially for Densimet and W-10%Re alloy. The results of this work are surprising: addition of Re, which in the unirradiated condition improves DBTT, leads after irradiation to more rapid embrittlement. The possible reason is the formation of Re-rich phases, which are also observed in

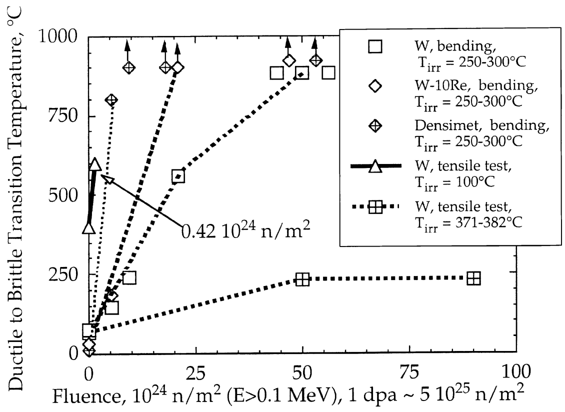


Fig. 3. Influence of neutron irradiation on the DBTT of W, Densimet (W3.4Ni1.6Fe) and W-10%Re alloy [40,42].

Mo–Re alloys [43]. The influence of neutron irradiation on the change of DBTT of W and W alloys is summarised in Fig. 3.

Based on this limited data, we can conclude that no matter what W grade is chosen, tungsten will become brittle after low fluence irradiation and especially at low irradiation temperatures.

## 5. Effect on carbon based materials

Among various available carbon based materials carbon fibre composites (CFC) are considered as armour for fusion applications, because they have high thermal conductivity and high thermal shock resistance [44]. For this reason, only neutron effects on the properties of CFCs will be reviewed.

### *5.1. Change of CFCs properties*

*Dimensional stability*. The recent review has been published by Burchell [45]. Wu and Bonal have published a series of papers at wide irradiation temperature and fluence ranges (~300–1000°C, 0.3–2 dpa), [26,46–49]. Additional data can be found in [50–53]. Different CFCs have been included in the irradiation programs (1D: UFC, MKC; 2D: CX 2002U, A05, DMS 678; 3D: FMI 222, FMI 223, N112, etc.). The following general trends have been revealed:

- dimensional changes in CFCs dominated by the behaviour of fibres: neutron irradiation leads to shrinkage in the direction parallel to the fibres and to swelling in the perpendicular direction;
- dimensional changes in 2- and 3D materials are more isotropic in comparison with 1D CFCs, turnaround to growth for 1- and 2D CFCs appeared earlier than in 3D CFCs;
- dimensional changes of 2- and 3D CFCs such as A05, CX 2002U, N112, N312B in the temperature range ~350–840°C and damage dose of ~1 dpa are in the range of 0.5%;
- increase of the irradiation temperature leads to an increase of the dimensional changes;
- CFCs with pitch precursor fibres have more resistance to growth;
- materials with a low initial density have better dimensional stability;
- silicon doped CFC (NS31) has higher dimensional stability in comparison with N112, N312B [26].

*Elastic modulus*. Limited data are available on the influence of neutron irradiation on the Young's modulus of CFCs. Similar to fine grain graphites, irradiation leads to an increase in the elastic modulus. Sato [54] has reported that after irradiation (~1.1 dpa and temperature 750–810°C) the Young's modulus has increased from 26.3 to 34.2 GPa for CFC with PAN fibres and from 13.5 to 19.2 GPa for CFC with pitch fibres. Eto [55] has reported data on neutron irradiation at 1000°C and a fluence of $2 \times 10^{21}$ n/cm$^2$, (~2 dpa) on the Young's modulus of MFC-1 and CX 2002U. The elastic modulus of CX 2002U increased by ~25% (from 14.9 to 18.5 GPa), and for MFC-1 in the fibre direction the Young's modulus increased by 32%, but in the perpendicular direction no change was observed.

*Coefficient of thermal expansion*. Data on the effect of neutron irradiation on the coefficient of thermal expansion (CTE) of CFCs is also very limited. Burtseva [56] and Platonov [53] have reported no change of the CTE of Russian PAN type 3–4D CFCs after irradiation at 200–600°C and up to a fluence of $2.6 \times 10^{20}$ n/cm$^2$ (~0.2 dpa). Bonal and Wu [57] have reported that for felt type CFCs (A05 and CX 2002U), the ratio ($CTE_{irr}$/$CTE_{unirr}$) was almost the same independent of the variation in the irradiation conditions (600–1000°C, 0.8–1.8 dpa) and that ratio was in the range 0.74–1.33, depending on materials and direction of measurement. The ratio $CTE_{irr}/CTE_{unirr}$ for 3D CFC N112 was ~2.7 (changes from 0.69 to $1.85 \times 10^{-6}$ K$^{-1}$) at a damage dose of 1.8 dpa and 620°C irradiation temperature.

*Mechanical properties*. Platonov [53] has reported the compression strength of CFC with PAN fibres UAM-5 and UAM-90 after irradiation at 230–600°C and up to a fluence of $2.6 \times 10^{20}$ n/cm$^2$ (~0.2 dpa). For UAM-5 the strength increased by a factor of 1.5–2 (up to ~150–200 MPa), while for UAM-90 no changes were observed.

Sato [54] has reported data on neutron irradiation effects (damage dose ~1.1 dpa and irradiation temperature 750–810°C) on strength and fracture toughness. In all cases the strength increased after irradiation by ~20% and fracture toughness also increased by 20–30%. Eto [55] has published data on the influence of neutron irradiation (1000°C, fluence of $2 \times 10^{21}$ n/cm$^2$, ~2 dpa) on the strength of MFC-1 and CX 2002U. For CX 2002U, irradiation leads to an increase of the strength by 25–27% and to a decrease of the fracture strain by 10–30%.

Burchell [50] has studied the influence of irradiation at 600°C up to a damage dose of 2.5 dpa on strength of FMI 222 and FMI 223. An increase in the strength was also observed, however, for FMI 222 (PAN fibre) some decrease in the strength after fluence ~0.6 dpa was found. Increase of strength and failure strain have been reported by Tanabe [58] after irradiation at 240°C, 640°C and 0.64, 0.87 dpa, respectively.

*Thermal conductivity*. It is well-known that neutron irradiation decreases the thermal conductivity of the carbon based materials. There is a relatively large body of available data for CFCs [26,45–53,56,59,60,62,63]. Some available data of normalised thermal conductivity ($K_{irr}/K_0$) at a measurement temperature equal to the irradiation temperature is presented in Fig. 4. as a function of damage dose.

The following general tendencies can be observed:

- irradiation decreases the thermal conductivity of CFCs at damage doses as low as $\sim 10^{-3}$ dpa [62];
- the level of degradation of thermal conductivity significantly depends on irradiation temperature: $K_{irr}/K_0$ of CFC FMI 222 and MFC-1 irradiated at 150°C and a damage of ~0.24 dpa was ~0.1 [61]. An increase of the irradiation temperature leads to an increase of $K_{irr}/K_0$;
- an increase of the neutron fluence leads to a decrease in the thermal conductivity: in the temperature range 400–1200°C, $K_{irr}/K_0$ is a logarithmic function of damage dose. Saturation in the thermal conductivity changes has been observed: at an irradiation temperature of 600°C the saturation began at 1–4.5 dpa [63], whereas at 150–200°C the saturation is observed at 0.1–0.24 dpa [61].

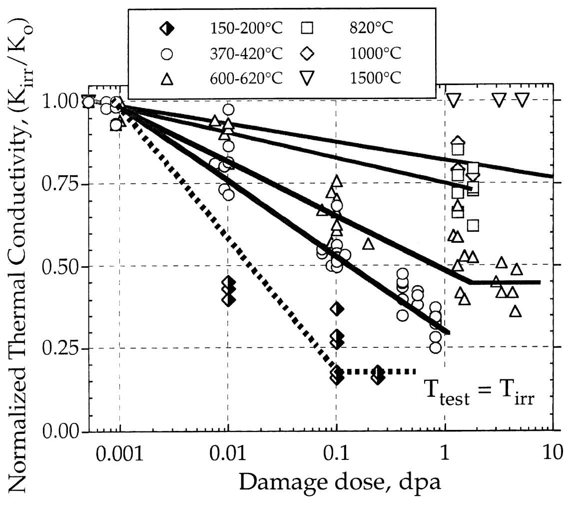


Fig. 4. Normalised thermal conductivity of different CFCs as a function of damage dose [45,53,59–63].

Thermal conductivity of irradiated CFCs may be partially restored by high temperature annealing [51,52,59,61,63,64]. The level of the thermal conductivity recovery depends on the annealing temperature and neutron fluence. Maruyama [52] has shown the full recovery of thermal conductivity of CFC irradiated up to 0.01 dpa, while for CFC irradiated up to 0.84 dpa the recovery was not complete.

The behaviour of the thermal conductivity at temperatures higher than the temperature of irradiation has been studied recently by Bonal [65]. Thermal conductivity of CFC NS11 irradiated at 335°C and a damage dose 0.31 dpa was measured during fast heating up to 800°C. The main result is that thermal conductivity at this condition was almost constant in the temperature range RT – 800°C.

### 5.2. *Thermal erosion of neutron irradiated CFCs*

The behaviour of neutron irradiated CFCs at specific conditions simulating thermal shocks and disruptions has been studied recently [66,67]. CX 2002U was irradiated at 290°C and fluence $\sim 5.6 \times 10^{20}$ n/cm$^2$ (~0.3 dpa) and after irradiation the samples were subjected to thermal shocks (500–800 MW/m$^2$, 25–40 ms) [66]. The measured thermal erosion was – twice as high as the thermal erosion of the unirradiated material. Roedig [67] has performed similar tests: CFCs such as NS11, NB31, Dunlop and CX 2002U were irradiated at 350°C and 750°C to a damage level ~0.3 dpa and thermal shock tests have been carried out at 8.4 MJ/m$^2$. Generally, the thermal erosion of materials irradiated at 350°C was higher than the erosion of unirradiated CFCs and CFCs irradiated at 750°C. The main reason is the reduction of the thermal conductivity by neutron irradiation.

### 5.3. *Tritium retention in neutron irradiated CFCs*

Tritium retention in neutron irradiated carbon based materials has been reported in [25,26,68–71]. The results for fine grain graphites [69,70] show that tritium retention saturates at a damage level ~0.1 dpa and is equal to ~7000 appm at damage doses above this level. Later, Causey [68] was found that the bulk retention in the CFCs was significantly less (by a factor 10 at a least at fluence up to 1 dpa) than that reported for graphite. The possible reason is the difference in the microstructure, mainly due to a lower fraction of edge sites in the CFCs where tritium is trapped. Atsumi [71] has reported hydrogen trapping in fine grain graphites and CX 2002U after neutron irradiation to $2 \times 10^{20}$ n/cm$^2$ and also has found significantly less tritium retention in CFC in comparison with graphites. Wu [26] reported the increasing of tritium retention with increasing neutron fluence. However, a saturation level of 1000 appm at a neutron damage of 0.1 dpa has been observed.

## 6. Discussion and assessment of the neutron effect on armour for ITER

As was described, neutron irradiation affects the physical and mechanical properties of the Be, W and CFC. Some of the changes are important, others are not significant. Using the operational conditions of ITER as an example (Table 1), the consequences of the neutron irradiation on the performance of the armours in the ITER will be discussed in this section.

*Beryllium*. Change of thermal conductivity and swelling are not important due to the low irradiation temperature and low fluence. The bulk tritium retention in neutron irradiated Be is expected to be significantly less than tritium retention in the carbon codeposited layers [1].

The most critical consequence of neutron irradiation under ITER conditions is embrittlement. It seems that for a damage level of ~1 dpa there is some remaining ductility (less than 1%). This is typical of all beryllium grades. The structural integrity of neutron irradiated brittle Be is a key issue. There are positive experimental results which indicate that Be could meet the requirements:

- high heat flux tests of neutron irradiated mock-ups did not reveal any damage in Be and in Be/Cu joints [72,73], although the irradiation conditions were not fully ITER relevant (damage dose of ~0.3 dpa instead 1 dpa for the end of life);
- an increase in crack formation has been observed in the surface of neutron irradiated Be [24]. However, the high heat flux tests (at more severe conditions than needed for the first wall) of the cracked unirradiated Be did not reveal any detrimental behaviour and loss of material due to cracking [74].

From an engineering point of view to avoid possible crack formation and delamination of the brittle Be it is recommended to use Be tiles without any stress concentrations. Still more tests of neutron irradiated Be at relevant conditions (1000 appm He, 1 dpa) are needed, especially to simulate the VDE conditions and further thermal cycling behaviour of Be.

*Tungsten*. There is no problem with swelling. The generation of solid transmutation products is still small due to the low expected fluence and the expected change of thermal properties is negligible. Embrittlement is most crucial for W. It is not possible to select a grade, from among the existing ones, which will remain ductile after neutron irradiation at the fluence and temperature ranges typical of ITER. Almost no data exist to validate the performance of neutron irradiated W at high heat fluxes. More experiments are needed to tests the structural integrity of the irradiated brittle W at high heat flux conditions. As all W grades are brittle, it is recommended to avoid using tungsten in a geometry with crack initiators and to use a W armour design which minimises stresses in the armour (W armour in form of brush or rods) and with the orientation of the texture perpendicular to the surface of the W/Cu joints.

*Carbon fibre composites*. Under ITER condition (damage dose ~0.1–0.2 dpa) there are only minor changes (not more than 5–10%) in elastic modulus, CTE and strength, and the dimensional changes are also very small. Bulk tritium retention due to the presence of the radiation defects in CFCs is expected to be significantly lower than tritium retention in codeposited layers [1].

Neutron irradiation at this low fluence affects mainly the thermal conductivity of the CFCs, especially in areas near the cooled heat sink. Few correlations are available for calculation of the thermal performance of the CFC armoured components [75,76]. The change of the thermal conductivity of CFCs due to neutron irradiation has been included in the analysis of the erosion lifetime of the divertor components, [77]. The main conclusion was that also with reduced thermal conductivity the erosion lifetime of CFC armoured components was acceptable. However, the more detailed analysis is needed.

## 7. Conclusion

Beryllium, tungsten and carbon fibre composites are leading candidate armour materials for plasma facing components of the Next Step fusion device. In recent years, the data base on the influence of neutron irradiation on the properties of Be, W and CFC has been significantly improved.

- Mainly in the frame of ITER R&D program quantitative data on the properties for the modern Be grades and CFCs have been generated.
- The study of the combined effects of the neutron irradiation and high heat fluxes on the behaviour of the armour materials has started; for future activities it is important to define the expected operational conditions.

For ITER design Be, W and CFCs have been selected as armour materials. For each material the key problem is mechanical integrity and maintaining their function of protecting the wall structure. The main issue for Be and W is the integrity of neutron irradiated brittle material. The first results on the tests of the neutron irradiated armour materials at high heat fluxes have demonstrated that the performance of armour could be acceptable. Still more activity is needed to validate the performance of neutron irradiated armour materials.

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## References

[1] G. Federici et al., these Proceedings, p. 110.
[2] V. Barabash, G. Federici, R. Matera, A.R. Raffray, Physica Scripta T 81 (1999) 74.
[3] V. Khripunov, private communication, September 1999.
[4] L.R. Greenwood, F.A. Garner, J. Nucl. Mater. 212–215 (1994) 635.
[5] D. Gelles et al., J. Nucl. Mater. 212–215 (1994) 29.
[6] M.C. Billone et al., Fus. Eng. Des. 27 (1995) 179.
[7] L. Snead, private communication, 1998.
[8] E. Ishitsuka, H. Kawamura, T. Terai, S. Tanaka, in: C. Varandas, F. Serra (Eds.), Fus. Technol. 1996, vol. 2, Elsevier, Amsterdam, 1997, p. 1506.
[9] M.C. Billone, W. Dienst, T. Flament et al., ITER solid breeder blanket materials database, ANL report ANL/FPP/TM-263, November 1993.
[10] M. Dalle Done, F. Scaffidi-Argentina, C. Ferrero, C. Rocci, Modeling He inuced swelling in beryllium during fast neutron irradiation, in: Proceedings of the First Workshop on Beryllium Technology for Fusion, KFZ report 52711, 1993, p. 105.
[11] F. Moons et al., J. Nucl. Mater. 233–237 (1996) 823.
[12] E. Ishitsuka, H. Kawamura, T. Terai, S. Tanaka, Microstructure and mechanical properties of neutron irradiated beryllium, in: Proceedings of the Third IAE Workshop on Beryllium Technology for Fusion, Mito, Japan, October 1997, p. 261.
[13] I.B. Kuprijanov, V.A. Gorokhov, G.N. Nikolaev, V.N. Burmistrov, Development of radiaition resistant grades of beryllium for nuclear and fusion facilities, in: Proceedings of the Second IEA International Workshop on Beryllium Technology for Fusion, Jackson Lake Lodge, WY, September 1995, p. 249.
[14] I.B. Kuprijanov, V.A. Gorokhov, G.N. Nikolaev, V.N. Burmistrov, J. Nucl. Mater. 233–237 (1996) 886.
[15] G.A. Sernyaev, A.S. Pokrovsky, R.M. Bagautdinov, J. Nucl. Mater. 233–237 (1996) 891.
[16] F. Moons, L. Sannen, A. Rahl, J. Nucl. Mater. 233–237 (1996) 823.
[17] R. Chaouadi, F. Moons, J.L. Puzzolante, Tensile fracture toughness test results of neutron irradiated beryllium, in: Proceedings of the Third IAE Workshop on Beryllium Technology for Fusion, Mito, Japan, October 1997, p. 251.
[18] S.A. Fabritsiev, A.S. Pokrovsky, The effect of the He, radiation damage and irradiation temperature on the mechanical properties of beryllium, in: Proceedings of the Third IAE Workshop on Beryllium Technology for Fusion, Mito, Japan, October 1997, p. 284.
[19] S.A. Fabritsiev, A.S. Pokrovsky, R.M. Bagautdinov, The neutron irradiation effects on the mechanical properties and structure of beryllium, effect radiation on materials, in: R. Nanstad (Ed.), Proceedings of the 18th International Symposium, ASTP STP 1325, ASTM, 1997.
[20] A.S. Pokrovsky, S.A. Fabritsiev, R.M. Bagautdinov, Yu. Goncharenko, J. Nucl. Mater. 233–237 (1996) 841.
[21] E. Ishitsuka, H. Kawamura, T. Terai, S. Tanaka, Microstructure and mechanical properties of neutron irradiated beryllium, in: Proceedings of the Third IAE Workshop on Beryllium Technology for Fusion, Mito, Japan, October 1997, p. 267.
[22] I.B. Kuprijanov, V.A. Gorokhov, R.R. Melder et al., Neutron irradiation behaviour of ITER candidate beryllium grades, in: Proceedings of the Third IAE Workshop on Beryllium Technology for Fusion, Mito, Japan, October 1997, p. 277.
[23] J.M. Beeston, Effects of Radiation on Structural Materials, ASTM-STP 683, ASTM, Philadelphia, PA, 1979, p. 309.
[24] A. Lodato, M. Roedig, R. Duwe et al., High heat flux performance of beryllium before and after neutron irradiation, in: Proceedings of the Fourth IAE Workshop on Beryllium Technology for Fusion, Karlsruhe, Germany, September 1999, p. 206.
[25] K. Kwast, H. Werle, C.H. Wu, Physica Scripta T 64 (1996) 41.
[26] C.H. Wu, J.P. Bonal, H. Kwast et al., Fus. Eng. Des. 39&40 (1998) 263.
[27] L.K. Keys, J. Moteff, J. Nucl. Mater. 34 (1970) 260.
[28] I.V. Gorynin, V.V. Rybin, V.R. Barabash et al., J. Nucl. Mater. 191–194 (1992) 421.
[29] M.S. Anand, B.M. Pande, R.P. Agarwala, Radiat. Eff. 39 (1978) 149.
[30] L.K. Keys, J.P. Smith, J. Moteff, Phys. Rev. 176 (3) (1968) 851.
[31] J. Matolich, H. Nahm, J. Moteff, Scr. Metall. 8 (1974) 837.
[32] F.W. Wiffen, Effect of irradiation on properties of refractory alloys with emphasis on space power reactor application, in: Proceedings of Symposium on Refractory Alloys Technology for Space Nuclear Power Application, Oak Ridge, CONF-8308130 (DE84001745), 10–11 August 1983.
[33] V.I. Shcherbak, M.I. Zakharova, V.N. Bykov, Fiz. Met. Metalloved. 40 (5) (1975) 1095.
[34] V.N. Bykov, G.A. Birzhevoy, M.I. Zakharova, Atomn. Energ. 32 (4) (1972) 323.
[35] V.N. Bykov, M.I. Zakharova, N.A. Artemov, A.G. Bahtin, Radiation Effects and Properties of Refractory Metals, IPPE-1001, Obninsk, 1980.
[36] R.C. Rau, R.L. Ladd, J. Moteff, J. Nucl. Mater. 33 (1969) 324.
[37] V.K. Sikka, J. Moteff, J. Appl. Phys. 43 (12) (1972) 4942.
[38] S.A. Fabritsiev, A.S. Pokrovsky, these Proceedings, p. 1215.
[39] R.K. Willjams, F.W. Wiffen, J. Bentley, J.O. Stiegler, Meter. Trans. A 14A (1983) 655.
[40] I. Alexandrov, I.V. Gorynin, Metallovedenie 22 (1979) 35.
[41] J.M. Steichen, J. Nucl. Mater. 60 (1976) 13.
[42] P. Krautwasser, H. Derz, E. Kny, Influence of fast neutron fluence on the DBTT of W, W10Re and W3.4Ni1.6Fe, RM 77, 673.

[43] S.A. Fabritsiev, A.S. Pokrovsky, J. Nucl. Mater. 252 (1998) 216.
[44] V. Barabash et al., 149 (1998) 258–263.
[45] T. Burchell, T. Oku, J. Nucl. Fus. 5 (Suppl.) (1994) 79.
[46] J.P. Bonal, C.H. Wu, Phys. Scr. T 64 (1996) 26.
[47] C.H. Wu, J.P. Bonal, B. Thiele et al., J. Nucl. Mater. 212–215 (1994) 416.
[48] J.P. Bonal, B. Thiele, G. Tsotridis, C.H. Wu, J. Nucl. Mater. 212–215 (1994) 1218.
[49] J.P. Bonal, C.H. Wu, J. Nucl. Mater. 228 (1996) 151.
[50] T.D. Burchell, W.P. Eatherly, J.P. Strizak, Effect of irradiation on materials, in: Proceedings of the 16th International Symposium on ASTM-STP 1175, 1993, p. 1266.
[51] S. Ishiyama, T.D. Burchel, J.P. Strizak, M. Eto, J. Nucl. Mater. 230 (1996) 1.
[52] T. Maruyama, M. Harayama, J. Nucl. Mater. 195 (1992) 44.
[53] P.A. Platonov et al., Plasma Dev. Operat. 3 (1994) 79.
[54] S. Sato, A. Kuramada, K. Kuwamata, R. Ishida, Fus. Eng. Des. 13 (1990) 159.
[55] M. Eto, S. Ishiyama, H. Ugachi et al., J. Nucl. Mater. 212–215 (1994) 1223.
[56] T.A. Burtseva, O.K. Chugunov, E.F. Dovguchits et al., J. Nucl. Mater. 191–194 (1992) 309.
[57] J.P. Bonal, C.H. Wu, J. Nucl. Mater. 230 (1996) 271.
[58] Y. Tanabe et al., Carbon 29 (7) (1991) 905.
[59] C.H. Wu, J.P. Bonal, B. Thiele, J. Nucl. Mater. 212–215 (1994) 1168.
[60] J.P. Bonal, C.H. Wu, J. Nucl. Mater. 228 (1996) 155.
[61] L.L. Snead, T.D. Burchell, J. Nucl. Mater. 224 (1995) 222.
[62] B.A. Thiele et al., Effect of Irradiation on Materials, ASTP-STP 1175, 1992, p. 1304.
[63] T.D. Burchell, Irradiation induced structure and property changes in tokamak plasma-facing, carbon–carbon composites, in: Proceedings of the 39th International SAMPE Symposium, vol. 39, p. 2423.
[64] J.P. Bonal, Thermal annealing of irradiated carbon fibre composites used for fusion application, CEA report, Rapport DTM N97/150, June 1997.
[65] J.P. Bonal, Thermal conductivity of irradiated Si doped CFC after annealing at 800°C, CEA report, Rapport DTM SEMI/ML2E/RT/99/009/A, June 1999.
[66] M. Uda et al., SOFT-20 Marseille, September 1999, p. 161.
[67] M. Rödig et al., these Proceedings, p. 1161.
[68] R. Causey, W. Harbin, D. Taylor, L. Snead, Phys. Scr. T 64 (1996) 32.
[69] W.R. Wampler, B.L. Doyle, R.A. Causey, K. Wilson, J. Nucl. Mater. 176&177 (1990) 983.
[70] R.A. Causey, K. Wilson, W.R. Wampler, B.L. Doyle, Fusion Technol. 19 (1991) 1585.
[71] H. Atsumi, I. Iseki, T. Shikama, J. Nucl. Mater. 191–194 (1992) 368.
[72] M. Roedig, private communication, 1998.
[73] I. Mazul, R. Giniatulin, R. Melder, Post irradiation examination of Be–Cu divertor mock-up after in-pile thermocycling experiment with active cooling, in: Proceedings of the Third IEA Workshop on Beryllium Technology for Fusion, Mito, Japan, October 1997, p. 281.
[74] A. Gervash, Fusion Eng. Design (1999) to be published.
[75] L.L. Snead, T. Burchell, in: Proceedings of the Second Biennal Conference on Carbon, San Diego, 16–21 July 1995, p. 774.
[76] V. Barabash, C. Wu, L. Snead, to be published.
[77] H.D. Pacher et al., J. Nucl. Mater. 241–243 (1997) 255.