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Strategy of fusion reactor materials R&D in China

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Abstract

There are two phases of fusion reactor materials R&D in China. The first phase will focus on fusion materials R&D for fusion breeders (or breeders for short) and in the second phase fusion materials for commercial pure fusion reactors will be developed. China is now in the first phase. Now development of reduced activation oxide dispersion strengthened ferritic steels, low- and high-Z plasma facing materials, including carbon base materials, Be, advanced W-alloys and functionally graded material, will be emphasized. Limited research activities for the second phase, e.g. V–Cr–Ti alloys and SiC/SiC composite development, are also supported. Fusion reactor materials R&D status in China is introduced and some research results are also given in present paper.

1. Introduction

Chinese scientists have devoted themselves to the early application of fusion power for decades. According to the requirement of China national economy development the electrical generating capacity of nuclear power should be ~450 GW in the 2050s. Only fusion breeders (or breeders for short), i.e. so-called 'fusion–fission hybrid reactors' as we named them before, development could make the requirement becoming feasible [1]. So breeders are now an inevitable choice for China [2–4], while advanced pure fusion power application is an ultimate objective, rather than the present one, for China. For this reason, there are two phases of fusion reactor materials R&D in China. The first phase will focus on fusion materials for breeders and a primary scientific and technological basis for pure fusion reactor development would be built up simultaneously. In the second phase, fusion materials of commercial pure fusion reactors will be developed. The first phase might last 30 years in the first half of this century.

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2. Fusion reactor materials R&D strategy

2.1. Tritium breeding materials for breeders

2.1.1. Liquid metal tritium breeding materials

Both liquid metal and solid tritium breeding materials have been developed in parallel in China for many years [5]. Many recent breeder designs prefer to adopt liquid metal (LM) blanket with pebble bed. LM tritium breeding materials used in the designs were lithium and lithium lead eutectic alloy ($_{83}Pb\ _{17}Li$, or PbLi for short) respectively. Their compatibility etc. have been explored [6], while many research activities focus now on magnetohydrodynamic (MHD) effect, which is one of the feasibility issues for LM self-cooled blanket. In order to alleviate MHD pressure drop in LM self-cooled blanket, insulator coatings, especially self-healing ones, have been developed.

2.1.2. Solid tritium breeding materials

A solid tritium breeding material blanket is also the alternative choice for breeders in China. Li_2O was the primary choice for tritium breeding materials because of highest lithium atomic concentration. Considering better thermal and irradiation stability of tritium breeding materials, γ-AlO_2 [7], Li_2ZrO_3 [8] and Li_4SiO_4 were also developed. A in-pile tritium production, release and transportation test has been performed [9]. In this test,

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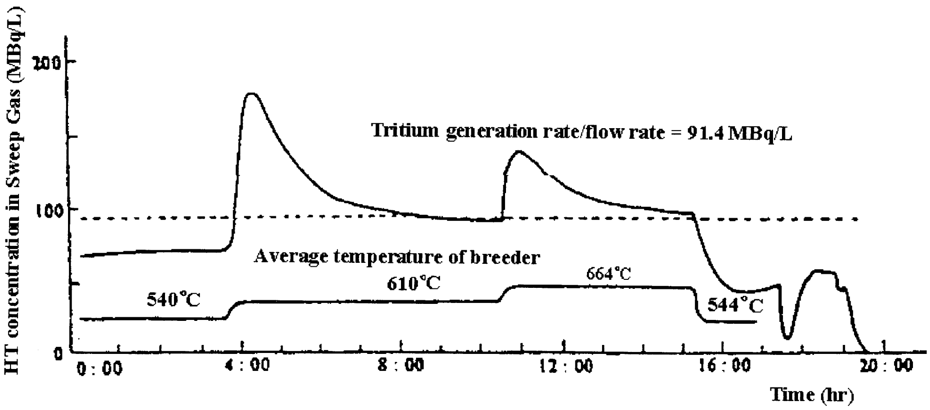

Fig. 1. Tritium production and release curves of γ-AlO_2 sintered in hollow cylinders vs. fission reactor operation time. Thermal neutron flux at the channel the capsule located: 8.3×10^{16} n/s · m^2, sweep gas through the capsule: He + 0.11%H_2 with 100 ml/min.

ten pieces of γ-AlO_2 sintered in hollow cylinders were put in a capsule, located in a fission reactor, and sweep gas (helium mixed 0.11% hydrogen) flowed through the capsule to carry the tritium produced (~97.5% of it in form of HT) out of the fission reactor. One of the test curves is shown in Fig. 1, where HT concentration in sweep gas increased sharply at every increment of γ-AlO_2 temperature and then tended to the common equilibrium tritium generation rate calculated.

2.1.3. Tritium permeation barriers

Tritium permeation through the wall is an essential problem in the blanket and the first wall for economy and safety reasons. Single and multi-layer ceramic coatings developed in China offered good tritium permeation barriers (TPB) on the surface of the metal wall [10]. TPB of (TiN + TiC + TiN) and (TiN + TiC + SiO_2) coatings on 316L substrate formed above 300 °C and were stable below 500 °C. Their permeability were 4–5 and 4–6 orders of magnitude lower, respectively, compared to 316L with Pd coating in the temperature range of 200–500 °C. The influence of ion irradiation on TPB has also been studied [11]. Tritium permeation in PbLi blanket is also a serious issue since the tritium solubility in PbLi is quite low. A permeation test of tritium permeating through the pipe wall with aluminized coating has been performed [12]. Self-healing coatings for TPB formation is also useful in this case.

2.2. Low activation materials R&D for structural application

2.2.1. Activation reduced oxide dispersion strengthened ferritic steels

In the first phase of fusion reactor materials R&D in China, activation reduced materials will take priority over the inherent low activation materials.

Ferritic steels are promising candidates for LM blankets because of their good compatibility and neutron irradiation resistance. Many ferritic steels, especially oxide dispersion strengthened (ODS) ferritic steels, have been developed in China. There are two kinds of ODS ferritic steels, Fe–Cr–Mo–Ti–Y_2O_3 and Fe–Cr–W–Ti–Y_2O_3 alloys, have been developed [13]. But the later, activation reduced ODS Fe–Cr–W–Ti–Y_2O_3 alloys, where ultrafine complex yttrium oxides (Y_2O_3 or $Y_2Ti_2O_7$) were ~2–3 nm in diameter, is more preferable. Some ODS ferritic steels have been manufactured into cladding tubes for fast neutron reactor applications [14]. Electron and dual-beam irradiation tests for activation reduced ODS ferritic steels have been performed [15] and the results indicated that the alloy showed good swelling resistance even at $e + He^+ + D^+$ irradiation condition (see Fig. 2). Effect of neutron irradiation on ODS ferritic steels has been studied by means of fission reactor and irradiation hardening at a very small dose (1.73×10^{19} n/cm^2 at 290 °C) [16], strengthening and plasticity degradation at 20 dpa (410 °C) were explored [17].

2.2.2. Activation reduced austenitic steels

Activation reduced austenitic steels might be candidates for solid tritium breeding material blankets in near term even though their operation temperature would be no more than 600 °C. A kind of Fe–Cr–Mn (W, V) alloy series has been developed in the recent decade. Addition of oversize solutes of W, V and long term aging of deformed austenites play an active role to their phase stability [18]. The irradiation tests have been performed using electrons and $e + He^+$ dual-beam technique, respectively [19]. Main results up to 9 dpa were much better than that of Fe–Cr–Ni series alloy under the same conditions, while phase stability was good enough and segregation of Cr and Mn at grain boundaries was not obvious. In the Ar ion (of 92 MeV) irradiation test, the

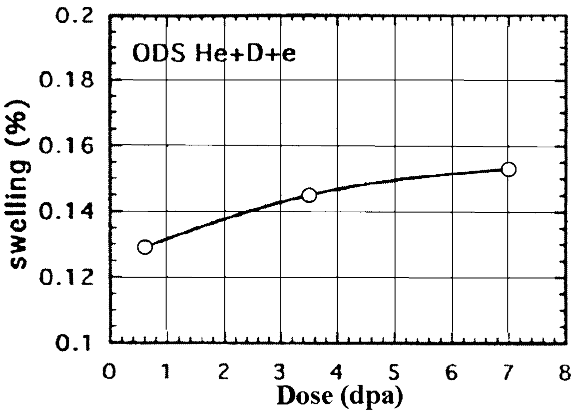

Fig. 2. Swelling of ODS ferritic steel, Fe–12.5Cr–2.5W–0.25Ti–Y^2O_3, vs. ($e + He^+ + D^+$) irradiation dose at 723 K. 170 keV He^+ and 140 keV D^+ implanted simultaneously into the specimen up to $6 \times 10^{16}/cm^2$ while the electron irradiation damage rate was 3×10^{-3} dpa/s.

α-phase formation in the alloy was only observed at a peak dose of 90 dpa [20].

2.2.3. Inherent low activation materials

Inherent low activation materials, e.g. V–Cr–Ti alloys and SiC/SiC composites, will be developed in the second phase. Considering the long term R&D needs, some research activities in this field have started right now. A small heat of V–4Cr–4Ti alloy has been produced for an oxidation behavior test, where the alloy was put into a flowing Ar gas with a small fraction of oxygen impurity (6.7–12 ppm in volume) at elevated temperature of 450–600 °C for 8 h [21]. The weight increase and V_2O_4 formation on the surface were observed. Three dimension weave reinforced SiC/SiC composites have been developed using Hi-Nicalon™ silicon fibers and chemical vapor infiltration technique [22–24]. Their main mechanical properties are given in Table 1.

2.3. Plasma facing component materials with high performance

2.3.1. Low-Z plasma facing materials

Plasma facing materials (PFMs) are an essential part of plasma facing components (PFCs) and also used as armor materials on the first wall. PFM R&D closely relate to the plasma material interactions (PMIs), which have been explored since the beginning of fusion research in China [25]. Both low- and high-Z (atomic number) PFMs, e.g. graphite, carbon/carbon composites, tungsten, molybdenum etc., have been used in current tokamak experiments. Many kinds of graphites have been adopted and their thermal shock resistance has been tested in specially designed simulation experiments [26], where the impingement energy deposited on the specimen surface for high heat flux and plasma disruption simulation was supplied by intense laser pulses. Obvious ablation and hot graphite particle emission were observed in the thermal shock experiments.

Table 1
Main mechanical properties of 3-D SiC/SiC composite

Density ($g\,cm^{-3}$)	2.49
Flexural strength (MPa)	860
Dynamic toughness ($kJ\,m^{-2}$)	36.0
Work of fracture ($kJ\,m^{-2}$)	28.1
Open porosity (%)	15
Fracture toughness ($MPa\,m^{1/2}$)	41.5
Fracture displacement (mm)	1.2

Coated graphite is also widely used in tokamak machines, where coatings face the plasma instead of graphite. Thermal shock resistance tests on various coatings have been performed in both laser simulation experiments [27,28] and tokamaks, and the behavior of some coatings seemed better than that of pure graphite. Unfortunately, some coatings were exfoliated from substrates because of strong thermal stress at the interface and poor bond strength there. TiC- and SiC-coated graphite was put into the edge plasma of HL-1 tokamak [29], and physical sputtering and chemical erosion seemed serious because of high oxygen content, which decreased the bond force between coating and substrate and accelerated chemical erosion of the coatings.

Many efforts have been made for increasing the bond strength at the interface. Most successful one among them is functionally graded material (FGM) development for this purpose, where the coating was replaced by FGM, e.g. SiC/C FGM replaced the SiC coating on graphite and B_4C/Cu FGM replaced the B_4C coating on Cu [30]. The thermal stress in PFC during thermal shock obviously decreased when FGM is adapted.

It is well known that graphite shows strong chemical sputtering (CS) and radiation enhanced sublimation (RES) at elevated temperature. A multi-element doped graphite has been developed and the CS of B-, Si- and Ti-doped graphite was successfully decreased by a factor of 5 (see Fig. 3) [31] and their RES was also improved. The thermal conductivity of Si- and Ti-doped graphite recently developed was 278 W/m K at room temperature [32]. Now SiC coated multi-element doped graphite has successfully been applied as PFM in HT-7 tokamak and it has been selected as PFM of superconducting tokamak HT-7U in China. The tungsten-coated doped graphite has also been developed in China and starts to be used in tokamak experiments. Carbon/carbon composite has also been used in tokamak and they showed better thermal shock resistance [26]. Many C/C composites with high performance have been developed in China.

Beryllium is a good PFM but has not been used as PFM in tokamaks in China because of its drastic

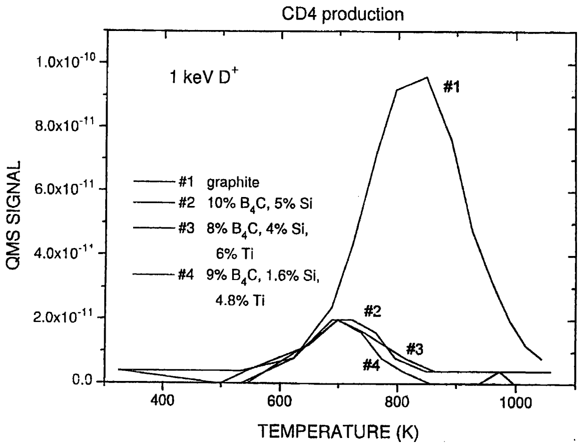

Fig. 3. Chemical sputtering of pure graphite and three kinds of B-, Si- and Ti-doped graphite under 1 keV D^+ bombardment.

toxicity. It has been used as neutron multiplier in breeder designs. Some analyses on Be, including its neutron irradiation effects, has been conducted in international cooperation [33–35].

2.3.2. High-Z plasma facing materials

High-*Z* PFMs, e.g. W, Mo etc., have been used as limiter materials in tokamaks in China. They have higher sputtering threshold energy and hence lower erosion rate and longer lifetime in comparison with that of graphite under the same condition. Considering low activation property, W would be preferable in future breeders. A new ODS W-alloy proposal has been put forward [36] and it is expected that the low temperature toughness, high temperature strength and neutron irradiation resistance of the new alloy would be further improved.

2.3.3. Functionally graded materials

As a candidate material for heat sink in PFCs, many copper alloys and ODS Cu-alloys have been developed [37]. The joining between PFM and heat sink material in PFC is a key issue. Because of the big difference in thermal expansion between, for example, W and Cu, the obvious residual stress would be retained in the interface region when conventional joining technique was adopted. Cracks often appeared and grew in this region during high heat flux impingement and plasma disruptions. FGM supplied a good solution for joining between PFM and heat sink materials. W/Cu FGM developed recently offered transition layer between W and Cu, where the metal composition gradually changed from (100%W + 0%Cu) at W side to (0%W + 100%Cu) at Cu side [33].

A new liquid divertor and liquid first wall concept has been put forward and Chinese scientists are studying their possibility theoretically for commercial pure fusion reactors.

2.4. Further R&D issues of fusion materials for breeders

2.4.1. Tritium breeding materials

Relating to LM tritium breeding materials, the coatings for MHD (in self-cooled blanket) and TPB are necessary. The compatibility of LM tritium breeding materials with structural materials and others should be explored. Tritium extraction from LM breeding materials, MHD pressure drop and heat transfer are essential issues of blanket technology which should be investigated.

Manufacture technology of solid tritium breeding materials has been well developed. Comparison and selection study among various solid tritium breeding materials have to be processed. The investigation on neutron irradiation effects, compatibility and temperature window enhancement should be explored for the selection of solid tritium breeding material. In-pile test for tritium production and release is also necessary.

According to breeder designs, one kind of tritium breeding materials, e.g. LM tritium breeding materials, might be emphasized. This would be determined by blanket design in more detail in near future.

2.4.2. Structural materials

ODS ferritic steels should not be used below 300 °C because of significant irradiation hardening and embrittlement at lower temperatures while the highest operation temperatures are limited by their corrosion behavior in LM. The temperature limits of 450 and 550 °C for PbLi and liquid lithium, respectively, in LM self-cooled blankets are expected. If the tritium breeding materials and coolants are separate, corrosion would be alleviated and the temperature limits would be higher, e.g. 650–700 °C (for He coolant). ODS ferritic steels can also be used in solid tritium breeding material blankets. Insulator coatings for MHD effect and TPB formation, especially self-healing ones, are needed. Further investigation on coatings, compatibility, ferromagnetic effects and the property degradation due to neutron irradiation are required. Much research should be conducted for setting up a complete database.

Fe–Cr–Mn (W, V) alloy is a potential candidate of austenitic steels for structural application in solid tritium breeding material blankets even though its operation temperature limit is only 600 °C (He coolant). A complete database on conventional mechanical properties, their degradation due to neutron irradiation, TPB coating is needed.

V–Cr–Ti alloys and SiC/SiC composites are good structural materials for both of LM and solid tritium breeding material blankets. They are mainly addressed in the second phase and so only limited research activities would be supported now. The investigation on alloy metallurgy, conventional mechanical properties, impurity pick up from coolant and embrittlement, coatings for MHD effect and TPB formation, especially self-healing ones, are needed. SiC/SiC composites have been developed for other industries and it provides a good R&D basis for fusion. Now limited tests for fusion are supported.

2.4.3. PFC materials

PFCs in plasma driver of breeders have to suffer long pules or even steady-state operation and so PFM with high performance are required. Both low- and high-Z PFMs would be further developed for breeders in the first phase of fusion materials R&D in China. Big concerns about low-Z PFMs are their significant erosion rate, re-deposition and tritium retention. The formers not only limit the lifetime of PFC but also lead to impurity transporting into plasma, while the later not only affects fuel recycling but also the fuel availability and safety. Further R&D on these critical issues are required and common efforts of scientists on materials and plasma physics are needed. Beryllium as a PFM application should be considered for breeders.

PFMs for breeders would be mixed carbon base materials, beryllium, tungsten and FGM. Advanced tungsten alloys would be the alternative candidates for carbon base materials but their melt might be removed by Lorentz force during plasma disruptions. So disruption mitigation is also required. High-Z impurity production and its transportation into the plasma center are also a big concern. Decrease of high-Z impurity production at the surface of PFCs is not the only way, and effective impurity transportation control, especially towards the plasma center, might be a good solution. FGM can be used as both coating interface and joining between PFM and heat sink materials in PFCs but there is lack of experience and database. FGM should further be developed for breeders.

2.5. International cooperation

Fusion power R&D is a worldwide effort. Extensive international cooperation is needed for China and other countries. In fact, many counties are also interested in breeders, e.g. transmutation breeder, to manage their high level radioactive waste (HLW, e.g. ^{135}Cs, ^{129}I, ^{99}Tc etc.) and minor actinides (MA, e.g. ^{237}Np, ^{241}Am, ^{243}Am, ^{244}Cm) produced in their fission reactors. The fusion materials R&D effort in China is similar to that in other countries in the world. This provides a good basis for the cooperation. China has had widespread international cooperation with many countries and China is willing to extend the cooperation, including on ITER-FEAT. China would make its contribution to our common mission of fusion power development.

3. Conclusions

The strategy of fusion reactor material R&D in China is divided into two phases. The first phase will focus on fusion materials for breeders and meanwhile a primary scientific and technological basis for pure fusion reactor development would be built up simultaneously. In the second phase, fusion materials for commercial pure fusion reactors will be developed. Advanced pure fusion power development and application is only the ultimate objective, rather than the present one, for China.

According to detailed breeder designs, which would be performed in the near future, one kind of tritium breeding materials, e.g. LM one, might be emphasized. Activation reduced ODS ferritic steels would be the prime candidates for structural application while activation reduced austenitic steels would also be a potential choice for breeders. The limited research activities on V–Cr–Ti alloys and SiC/SiC composites would be supported. Low- and high-Z PFMs, including carbon base materials, Be, advanced W-alloys and FGM, would be developed. Advanced W-alloys might replace the carbon base materials for PFCs of breeders in near future.

Not only China but also many countries are interested in breeders, e.g. transmutation ones, to manage their HLW and MA produced in their fission power plants. The fusion materials R&D effort in China is similar to that in other countries in the world. This provides a good basis for cooperation. China is willing to develop more extensive international cooperation and make its contribution.

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