Since China joined ITER program, fusion reactor materials research received more supports from China government, universities and institutes. The materials development aimed at their use for ITER components in near term and for Chinese DEMO in future. Recent activities focused on FW/blanket materials, such as the VHP-Be, CuCrZr alloy and 316LN for ITER FW, and CLAM/CLF-1 RAFM steels and breeding materials for ITER and DEMO TBM. Method to bond different materials has been developed in order to fabricate the ITER FW. Small plates of VHP-Be have made, which showed comparable physical and mechanical properties to US S65C. A 300kg CLF-1 ingot has been made, of which various mechanical properties have been measured. Vanadium base alloys, as another choice for advanced blanket structural material, was developed and studied. For plasma facing material, major efforts are placed on W and its coating on heat sink materials. Many researches are carried out by international cooperation, such as the CN-Japan CUP program. This paper will highlight the main fusion materials research activities in China and those for the future.

China will manufacturing 10% of ITER FW panels. Accordingly, ITER FW materials were studied and the technology of Be/CuCrZr and CuCrZr/SS (stainless steel) bonding was investigated. High purity (>99%Be in mass) CN-G01 Be blocks were made by VHP (vacuum hot pressing) with impact grinding powder in size of 7-14μm. Its mechanical and physical properties were measured from room temperature (RT) to 700°C. Estimated from the total O content, the average BeO content is ~1.0% and the minimal RT total elongation is 2.5%, both are not as good as the ITER reference material of S65C VHP-Be made in US. A multi-shot thermal shock test in JUDITH-1 indicated its cracking performance can’t meet the ITER requirement. Subsequently, a modified VHP-Be (CN-G01m) was developed by optimizing the manufacturing process. BeO content was reduced to about 0.7%, while the RT total elongation increased to more than 3%. Single thermal shock test at 1.2-5MJ/m² in a pulse length of 5ms showed the modified Be has equivalent performance as S65C. Formal qualification test of the CN-G01m is ongoing by using Be/Cu mock-ups to simulate the ITER VDE condition (60MJ/m²) and the following heat flux cycling at 2MW/m² in JUDITH-1.

CuCrZr alloy will be used as the heat sink material of ITER FW. To meet the requirement of high strength and small grain size, investigation on heat treatment and forging effects was conducted with the FW panel fabrication process effect into consideration. Results showed that it’s hard to satisfy the requirements by heat treatment, while forging displayed its potential. CuCrZr alloy blocks were forged at 650—800°C with a final thickness reduction of 60%, followed by HIP (hot isostatic pressing) cycles as that required for ITER FW panel fabrication. Tensile test at RT showed the ultimate tensile strength is 290MPa in minimal, while the mean grain size is ~130μm, both meets the requirement of ITER. Further studied will be carried out to reveal the mechanisms. Another material for ITER FW is the 316LN used as supporting back plate. The EB welding of the steel was studied. 5-80mm deep EB welding of the steel blocks or plates was studied. As compared to the base material, the weldment showed good properties of tensile strength and impact toughness, but a small loss of ductility.

The HIP joining technology of Be/Cu/SS for ITER FW panel fabrication is developed. The key for the joining is the interlayer metal between Be and CuCrZr alloy and the HIP parameters. FW qualification
mock-ups in dimension of 80x240x49mm were fabricated and delivered to US and EU for high heat flux test (HHFT). By using Ti/Cu coating interlayer and a HIPing at 580°C/150MPa, good Be/CuCrZr boning was achieved. The bonding strength reached to more than 214MPa when Be/CuCrZr samples were shear test at RT. The mock-up survived the HHFT in US without any damages, for which 12000 cycles at 0.862MW/m² and 1000 cycles at 1.4MW/m² were performed. Another mock-up in EU also withstood the 12000 cycles at 0.62MW/m² without damage. These bonding technologies will be further studied for fabrication of ITER FW semi-prototype.

China has two programs for the development of fusion reactor materials. One is from the nuclear energy development program; another is from the domestic ITER program. Structural materials development is closely bonding to the TBM (tritium breeding blanket modules) design and manufacturing. Several hundred kg CLAM and CLF-1 ferritic/martensitic steels were developed. Various physical and mechanical properties were measured. For the liquid metal TBM concept, corrosion behavior of CLAM steel in LiPb was studied in various liquid metal loops. For solid tritium breeder TBM, the behavior of CLF-1 steel in He will be studied. Both steels have similar chemical composition and properties, and similar mechanical properties to Eurofer 97 and F82H. In the coming years, large ingot of >1 ton will be produced. Major researches will be placed on the corrosion of the steel in blanket environment, neutron irradiation test, its effect on magnetic confinement of a Tokamak machine, and the property data base establishment. Besides steels, vanadium alloy has been studied for many years in China. The studies aimed at an advanced alloy for advanced blanket system. With the collaboration with Japan through CUP and other program, the mechanical properties of various alloys were extensively studied. Results showed that Ti and Al addition could benefit the alloy from hydrogen embrittlement, and cold rolling and aging could strengthen the alloy significantly with some improvement of thermal creep performances. Future studies will focus on development of the alloy in larger scale and new alloy of higher strength with dispersion particles. W coating on carbon based material and W/Cu bonding technology were studied, investigating an armor material and fabrication technology for high heat flux component. The thermal conductivity of the VPS W coating reached up to 90W/m.K. Under 5MW/m² pulse flux condition, it survived 1000 cycles. In the near future from next year, W alloy, thick W coating and mock-up fabrication technologies will be studied for the purpose to support the design and manufacturing of W armor divertor components for HL-2M machine.

For the development of solid tritium TBM, breeding materials were developed, including the 0.8-1.5mm diameter Li₄SiO₄, Li₂TiO₃, Li₂ZrO₃, γ-LiAlO₂ and Be pebbles. Method to produce Li₄SiO₄ powder was investigated. Thermal conductivity was measured and the thermal desorption behavior of the tritium breeder was studied by 3keV D⁺ to a fluence of 10²²/m². Two release peaks were observed. Further studies will be focused on the development of 0.5-1mm diameter Li₄SiO₄.

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