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Overview of recent Russian materials and technologies R&D activities related to ITER and DEMO constructions

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Abstract

An overview is given of the activities and major achievements within recent R&D performed in Russia on materials and technologies for ITER and DEMO. In Russia, the basic materials manufacturing and technologies have been selected for ITER, for two reference DEMO breeding blanket concepts and for the related long term R&D. The review on the recent results of investigations on low activation materials (V–Ti–Cr alloys, Fe–12Cr–2W–V–Ta steel EK-181), beryllium and superconducting materials is presented. The fabrication of tubes, sheet and other forms from low activation materials is mentioned. The activity in beryllium materials both in the domestic studies and international cooperation is outlined. The progress in enhancement of the properties of superconducting materials for the ITER magnet system is presented, and the prospect of further developments in superconducting materials for DEMO magnet system is analyzed.

1. Introduction

The strategy of development of atomic energy for Russia includes the creation of industrial fusion reactors. That is why the ITER and DEMO projects are necessary and important steps toward the incorporation of fusion energy in the energy system of Russia. The development and use of high technological low activation, creep resistant and irradiation resistant structural materials in fusion reactors is of primary importance for the attainment of safe and ecologically acceptable fusion energy plants. The development of superconducting materials for fusion reactor magnet systems of also plays a principal and decisive role for providing fusion energy plant competitiveness. The complex of research on the development of low activation materials, beryllium materials and superconducting materials must be treated as a high technology area of material science. Main objectives of the RF research program are the development of low activation and beryllium materials and the appropriate technologies. These will create a sound base for the participation of Russia in the realization of international projects on the development of experimental breeding blanket modules of DEMO reactor and to provide the technical basis for the realization of the national conceptual project DEMO-RF.

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2. Low activation metallic structural materials

To a great extend the fundamental and technological criteria for the development of low activation metallic structural materials (decay time to the residual radioactivity of remote level (10^{-2} Sv/h) is less than 100 years) has been formulated in the countries devoted to the exploration of fusion energy application [1]. Using laboratory scale samples, it was demonstrated that their functional properties could be equal to the properties of the conventional analog alloys. The main activities in the development of low activation materials are focused on the implementation of industrial scale manufacturing processes with minimum impurities contamination, on the optimization of composition of the alloys and on achieving uniformity of microstructure. As a result, the base technologies of low activation V–4Ti–4Cr alloys (USA, Japan, Russia and China) and ferritic–martensitic steels Fe–9Cr–W–V–Ta (EUROFER-97, JLF-1, F82H, in EC, Japan and USA) and Fe–12Cr–W–V–Ta (EE-181, in Russia) were established. In Russia, the works on low activation materials beneficially uses the wide experience of the development and application of ferritic–martensitic 12% Cr steels, such as EP-450, EP-900 (in reactors BOR-60, BN-350, BN-600) and vanadium based alloys (in reactors for space application).

2.1. Low activation V–(4–10)Ti–(4–5)Cr alloys

The V–(4–10)Ti–(4–5)Cr alloys are the most promising materials for application in fusion reactors with liquid metal coolants (Li, Na), which combine low activation properties with attractive functional parameters relative to conventional structural materials and designed for operating temperatures as high as 700–750 °C. The basic composition of V–4Ti–4Cr was proposed in USA [2]. In Russia (in VNIINM), the alloys V–(4–10)Ti–(4–5)Cr are under exploration [3,4]. The 50 kg ingots of the alloy with base composition V–4Ti–4Cr has been fabricated and tested. On the base of developed 'Technical Standard' (VNIINM, OAO 'Uralredmet') the experimental industrial production of high purity vanadium has been established (2000–2002, OAO 'Uralredmet') and a reliable technology for melting V–Ti–Cr alloys has been developed [4]. The industrial technologies for fabrication of different products (sheet, tube and rod) from V–Ti–Cr alloys billets have been developed. The quality of these products has been analyzed [3,5] and confirmed to be high. The increase of the mass of ingots up to 100–300 kg while maintaining the quality is possible using the developed melting process [4]. The work on these enhanced weight ingots and fabrication of different products (sheet, tube and rod) is foreseen in 2006–2007. The established experimental industrial fabrication of the V–4Ti–4Cr products is technologically ready for the fabrication of experimental module of DEMO-RF to be installed in ITER. The recommended and experimentally measured chemical compositions of V–4Ti–4Cr alloys are given in [5]. One of the most important factors defining the quality of V–Ti–Cr alloys is the level of concentration of such impurities as oxygen and nitrogen. These concentrations are presented in Fig. 1. It can be seen that the developed manufacturing processes guarantee the acceptably low levels of O and N concentrations in ingots of V and V–Ti–Cr alloy. The calculated decay rates of residual radioactivity in the so-called 'pure' V–4Ti–4Cr (without any impurities), recommended specification (VV1) and real experimental (VVC2) ingots after their calculated 'irradiation' in the BN-600 and DEMO reactors are shown in Fig. 2. As is shown in Fig. 2 the time to attainment of 'remote level' of radioactivity in 'pure' composition is equal to only 3.5 years for BN-600 reactor and 6.0 years for DEMO-RF. The time to attainment of 'remote level' state for real composition VV2 is 25 years (DEMO-RF) and 20 years for BN-600.

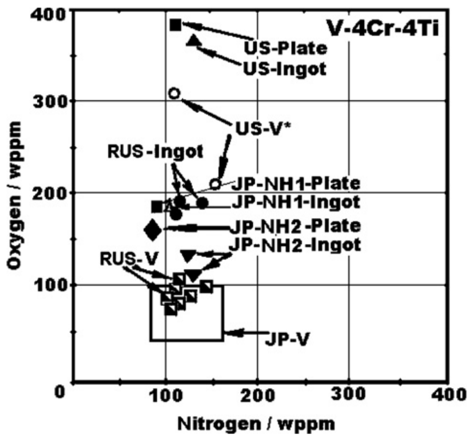

Fig. 1. Contamination by O and N in the ingots of V and V–4Ti–4Cr products (ingots, plates), produced in USA, Japan and RF.

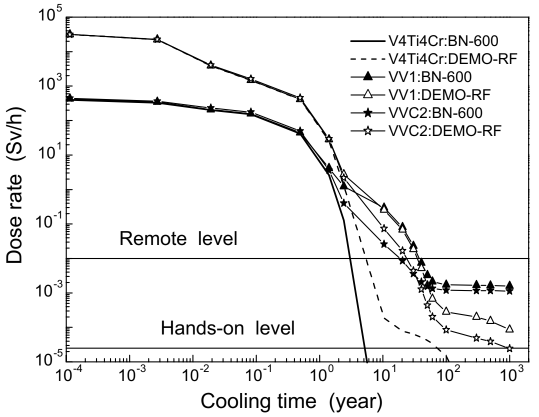

Fig. 2. The calculated decay rates of radioactivity in the so-called 'pure' (V–4Ti–4Cr without any impurities), recommended specification (VV1, VNIINM) and real ingot (VVC2, VNIINM) after their hypothetical 'irradiation' in the BN-600 (flux 6.5×10^{15} neutron/cm^2/s) and DEMO (flux 9.0×10^{14} neutron/cm^2/s) reactors. Effective duration of irradiation 560 days.

2.2. Low activation ferritic–martensitic Fe–12% Cr steel EK-181

Low activation ferritic–martensitic Fe–(9–12)Cr–W–V–Ta steels are designed for application with liquid metal (Na) and gaseous (He) coolants. Usually the operation temperatures for Fe–9Cr steels are limited to 550–600 °C and 650–700 °C for Fe–12Cr steels [6]. The vast experience accumulated in Russia during research on the creation of materials for fast neutron reactors (BN-350, BN-600) has shown that it is necessary to strictly control the processes of formation of specific irradiation resistant microstructures, including the precipitation of different phases (oxide, nitride, carbide, etc.), to control the stability of this microstructure to recrystallization which in turn influences the stability of functional parameters of the material. Nowadays the research is concentrated on the optimization of the developed low activation EK-181 steel with basic composition Fe–12Cr–2W–V–Ta–B [6]. One of the additional objectives resolved during the development of this steel was to attain higher strength (at temperatures 650–700 °C) than in conventional 12% Cr steels (grades EP-450 and EP-900).

This new steel grade (EK-181) is now tested by the Russian industry through the fabrication of industrial scale ingots with weight 500–1000 kg and manufacturing of different types of products (plate, sheet, tube and rod) [6]. The pilot industrial production in Russia has proved to be technologically able to provide the fabrication of the experimental DEMO-RF breeding module for the ITER reactor. The further optimization R&D on the improvement of functional properties of the EK-181 steel is now in progress targeting to achieve better homogeneity of microstructure and composition throughout the large commercial ingots. The irradiation tests are continuing in BOR-60 reactor (up to 5–10 dpa at 325–345 °C) and irradiation tests in BN-600 reactor (up to 80–135 dpa at 370–700 °C) are planning. The calculated decay rates of residual radioactivity in the as-specified 'optimum' steel and real experimental (Fe1) EK-181 steel after calculated 'irradiation' in the BN-600 and DEMO reactors [7] are shown in Fig. 3. Fig. 3 data shows that the time to attain 'remote level' radioactivity in real experimental steel (Fe-1) is 25 years greater for BN-600 and 10 years greater for DEMO-RF than for the 'optimum' steel.

Material science investigation of EK-181 steel has revealed the main microstructural features which are responsible for the attainment of high level of high temperature creep resistance and led to understanding of the mechanisms controlling the creep at temperatures higher than 650 °C. EK-181 steel has the highest high temperature creep resistance among the Russian 12% Cr steels of EP-450 and EP-900 types [7] as demonstrated in Fig. 4.

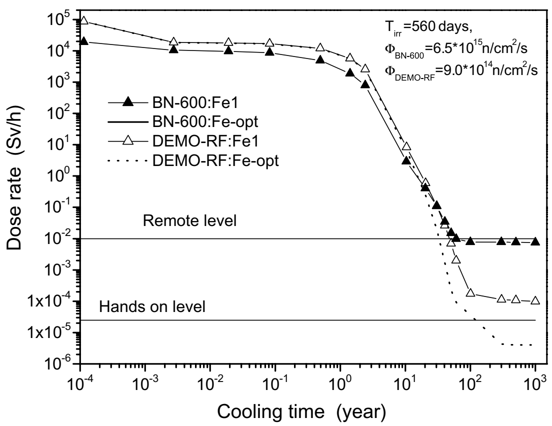

Fig. 3. The calculated rates of radioactivity decay in the as-specified 'optimum' steel (Fe-opt) and real experimental steel (Fe-1) after their hypothetical 'irradiation' in the BN-600 (flux 6.5×10^{15} neutron/cm^2/s) and DEMO (flux 9.0×10^{14} neutron/cm^2/s) reactors. Effective duration of irradiation 560 days.

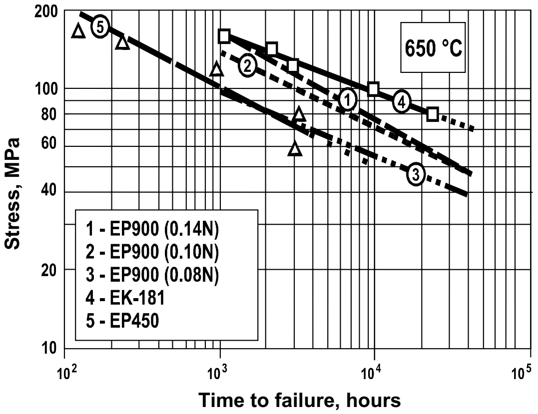

Fig. 4. High temperature creep resistance of 12% Cr steels EP-450, EP-900 (with different nitrogen concentrations) and EK-181 at temperature of 650 °C.

3. Development of Be for ITER first wall components and for DEMO breeding modules

Due to good experience of Russia Institutes in the field of high heat flux components and advanced materials for various applications, the RF participation in the development of ITER Plasma Facing Components, PFC, was rather wide and comprehensive. As a result of activity in this area the RF industry is planning to contribute in ITER construction, in particular, with the following items: first wall panels, divertor dome and liner assembly and high heat flux, HHF, tests [8]. That is why the Russian R&D activity has concentrated on the development of PFC with beryllium and tungsten armour. A very important and challenging task was to select the best armour materials grades and predict their behavior under the most severe loads during plasma disruptions. Development of manufacturing technologies for multi-layered components was also performed. On small mock-ups, the RF team demonstrated the critical incident heat fluxes (without structure damage) of 16 MW/m^2 for Be–CuCrZr composition and 30 MW/m^2 for W–Cu–CuCrZr composites, which are significantly higher than the operational heat fluxes expected for first wall and divertor vertical target, respectively. The semi-industrial abilities were confirmed during the manufacturing of large-scale mock-ups of primary first wall and divertor vertical targets. The mock-ups showed reliable operation for thermal cycling testing under heat loads exceeding the design operational values [9].

Beryllium is planned to be used in both RF experimental tritium-breeding modules (OAM) of DEMO. The basic types of beryllium in these two modules are porous and solid beryllium. For meeting the required parameters of swelling and tritium release, a unique technology for fabrication of Beryllium with predetermined open porosity was developed. For collecting the data base on properties of various modifications of beryllium after irradiation similar to the operation conditions of ITER [8,10] the international project HIDOBE (2003–2009) was initiated. Data from the project will enable estimation of the influence of irradiation conditions (temperature 400–750 °C and two dose levels – $C_{He} = 3000$ appm and $C_{He} = 6000$ appm) and Be characteristics (including: pebble size, grain size, chemical composition, size and type of porosity, rate of cold deformation, and surface oxidation) on swelling, tritium retention and release, thermal and mechanical properties, structural changes, etc. About 950 specimens were fabricated in Russia for this program. The specimens were made of four grades of dense beryllium with different grain sizes (5–60 μm), chemical composition (including $C_{AeI} = 0.3$–3.9 wt%), and of two grades of high porosity beryllium ($P = 20\%$ and 30%). The irradiation of two capsules was started in HFR reactor (Petten, NL) in June 2005. The program on characterization of preirradiation properties of the specimens is in progress.

Within the framework of a cooperation between RF (VNIINM) and EU (ENEA) VNIINM has fabricated beryllium components (some of which are shown in Fig. 5) for the mock-up of the ITER OAM-HCDA [11]. The main purpose of the

Fig. 5. Beryllium components for the mock-up of TBM-HCPB [11].

experiment was to validate the numerical codes and nuclear data used for predicting the tritium production rate in the TBM-HCPB. This was accomplished by measuring the tritium production in the mock-up irradiated with 14-MeV neutrons. The experiment has been performed in ENEA using the 14 MeV neutron generators. The results are expected to be presented by the end of 2005.

4. Development of superconducting materials for ITER magnet system

After the Ministerial Meeting for ITER in Moscow of 28 June 2005, in which a common understanding on the site selection was reached, the Delegations from China, European Union, Japan, the Republic of Korea, Russia and the USA are looking forward to the implementation of the ITER project, to be sited at Cadarache. Among the key activities in the framework of the RF preparation for ITER construction, the R&D on the superconducting materials should be mentioned. The RF is participating in the production of TF Nb_3Sn based conductors, and of PF1/6NbTi conductors. The reason for it is based on the successful experience in development and industrial production of NbTi conductors for the world's first Tokamak, T-7 [12], with a superconducting magnet system, and Nb_3Sn conductors for the world's first Tokamak T-15 [13] with a Nb_3Sn superconducting magnet system. During the EDA stage of ITER, Russia actively participated in the work on the development of superconductors and the testing in model coil-inserts.

The cross-section of bronze processed strand, which met the requirements of the HPII ITER specification, developed in Russia and produced in the amount of 1000 kg, is shown in Fig. 6(a) [14]. This Nb_3Sn strand had been used for the fabrication in RF the model TFCI coil that was successfully tested in JAERI and achieved the design parameters – 46 kA at 13 T magnetic fields [15]. In the ITER EDA framework the NbTi strand for model PFCI coil was developed and produced in the amount of 0.5 tons (Fig. 6(b)). The strand produced was cabled in RF and sent to EU for PFCI coil fabrication. Pilot NbTi composite rods of different layouts were also fabricated from 250 mm billets using industrial conditions for production of 0.73 mm strand with a filament diameter of 6.5 μm, intended for PF1/6 conductors.

The results of model coils tests have confirmed that all coils reached design parameters. It was also revealed that proper interpretation of the performance of the conductor with more than 1000 strands still needs more data to extend the strand database. However some new recommendations were made for strand. The changes in requirements for ITER Nb_3Sn strand in comparison with the requirements for Nb_3Sn strand for T-15 are presented in Table 1. For internal-tin Nb_3Sn strand the critical current density (J_c) of 800–1000 A/mm^2 (12 T, 4.2 K) is easily attainable but for bronze processed wires, due to limitation on the tin content in the bronze matrix this requirement is very tough. The designed for new ITER requirement, internal-tin Nb_3Sn strand is shown in Fig. 7(b). Optimization of bronze processed Nb_3Sn strand also has enabled attaining the target current density in laboratory samples [16]. As a result, J_c higher than 700 A/mm^2 has been reached at the hysteresis loss less than 400 mJ/cm^3 (Fig. 7(a)).

Development and investigation of Nb_3Sn strand for ITER magnet system have had a large impact on the systematization of both fundamental and technological data leading to better understanding of the basic principles of the design and fabrication of large-scale magnet systems. This also opened the

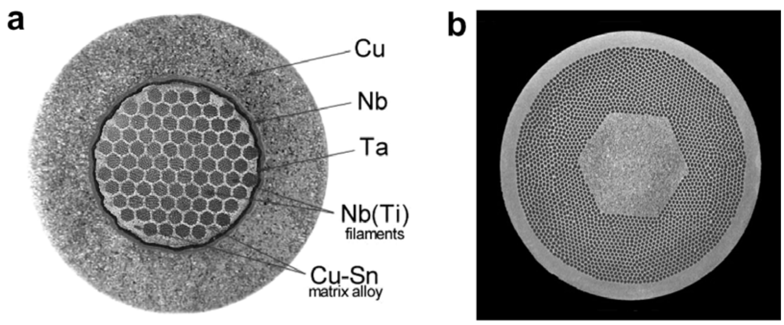

Fig. 6. ITER bronze processed Nb_3Sn strand for TF insert coil (a) and NbTi strand for PF insert coil (b).

Table 1
Requirements for Nb_3Sn strands to be used in TF coils of T-15 and ITER

Specification	Tokamak T-15	ITER			
		Model coils (EDA, 1993)		ITER FEAT (2001)	Enhanced design requirements (2003)
		HP-I	HP-II		
Critical current density (non Cu, 12 T, 4.2 K) (A/mm^2)	>400 (8 T)	>550	>700	>650	800–1000
Hysteresis losses (non Cu, ±3 T, 4.2 K) (mJ/cm^3)	–	<200	<600	<400	<900–1000
Strand diameter (mm)	1.5	0.81		0.71	0.71; 0.81
Cu/non Cu	–	1.5		1.35	1.0
Twist pitch (mm)	25	10		15	15
Unit length (m)	340	1150		1500	3000
'n' (12 T, 4.2 K)	–	>20		>20	>20
RRR	–	>100		>100	>100
Thickness of Cr plating (μm)	–	2		2	2
Last stage of reaction heat treatment	750 °C (48 h)	650 °C (175 h)		650 °C (175 h)	650 °C (100 h)

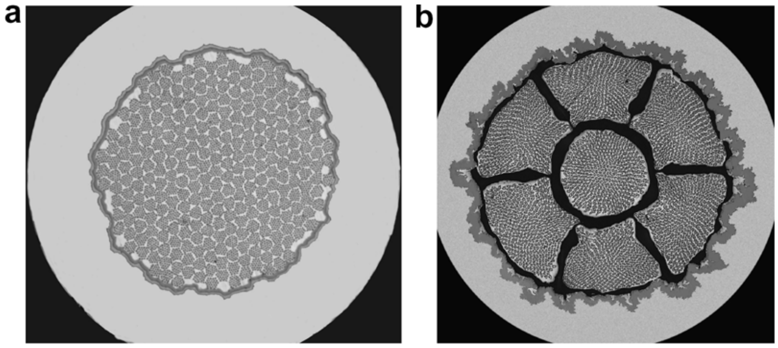

Fig. 7. Advanced bronze processed (a) and internal-tin (b) ITER Nb_3Sn strands for TF conductor (produced by Bochvar Institute) Main properties of the strands: (a) J_c (12 T, 4.2 K) = 774 A/mm^2, hysteresis losses (±3 T) = 337 mJ/cm^3 (b) J_c (12 T, 4.2 K) = 850 A/mm^2, hysteresis losses (±3 T) = 900 mJ/cm^3.

way for the development of Nb_3Sn superconductors for the next generation of fusion magnetic systems, which assume significant increase in operating magnetic field and thus increases in engineering current in the windings. This can be achieved only if current density of strand will be enhanced and the mechanical strength will also be sufficiently high to withstand larger electromagnetic loads.

This can be done if additional high strength material is incorporated in design of the strand. Such investigations have been performed on bronze processed [17] and internal-tin [18] Nb_3Sn superconductors. Reinforcing elements of microcomposite Cu–18 wt% Nb alloy were added to those composite strands. The level of ultimate tensile strength of the strand was increased 1.3–1.4 times [18]. For internal-tin Nb_3Sn wires, J_c higher than 2000 A/mm^2 and the value of the hysteresis losses (±3 T) of 1000 mJ/cm^3 have been reached. For the bronze route strand, J_c was 775 A/mm^2 and the level of the hysteresis loses – 155 mJ/cm^3 [18].

Nowadays the achievements in manufacturing high temperature superconductors (HTS) on an industrial scale are well known. In particular, special HTS wires with low thermal conductance have been developed for economically efficient HTS current leads. As a rule an HTS current lead is a complex device based on HTS single units formed from HTS tapes joined together. For example, in the RF a 1000A-HTS unit has been developed based on BSCCO/AgAu multifilamentary tapes [19]. Summarizing, it should be stated that the superconducting magnet technology has already proved its engineering reliability. The NbTi and Nb_3Sn strands with enhanced properties will still be the workhorse of future fusion reactors magnet systems but HTS superconductors will be playing a more and more important role.

5. Conclusion

In Russia the development of low activation high performance structural materials, beryllium and superconducting materials for application in ecologically acceptable fusion reactors is of primary importance. In the RF the most promising compositions of vanadium based alloys have been chosen, the industrial scaled technologies for fabrication of V–4Ti–4Cr ingots and the different products such as plate, sheet, rod and tube have been proposed and their testing have been successfully initiated. R&D on low activation ferritic–martensitic steel Fe–12Cr–W–V–Ta (EK-181) has reached the stage that makes possible the development and fabrication of the experimental breeding module DEMO-RF. The R&D activity on the development of Plasma Facing Components with beryllium and tungsten armour enables selection of the best armour materials grades and prediction of their behavior in case of plasma disruptions. Development of the bases of manufacturing technologies for multi-layered components was also positive. It was demonstrated that the critical incident heat fluxes of 16 MW/m^2 for Be–CuCrZr composite and 30 MW/m^2 for W–Cu–CuCrZr composite, which are significantly higher than the operational heat fluxes for first wall and divertor vertical target respectively did not led to failure of structures. The NbTi strand for poloidal coils PF1/6 and Nb_3Sn strand for toroidal coils of ITER magnet system, which meet the enhanced ITER requirements, have been designed, experimentally tested and preparation for their industrial production is in progress.

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