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Materials science problems of blankets in Russian concept of fusion reactor

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Abstract

Structural materials, beryllium and tritium breeding materials proposed for blanket of Russian reactor DEMO and Test Modules for ITER are discussed. Main requirements for the materials are concerned with basis current designs of blankets and modules and possibility meet of ones for presence and developed alloys and materials discussed considered. Main properties and results of test of ferrite-martensite and vanadium alloys for DEMO and Test Modules are cited. Beryllium compositions used as component of First Wall and neutron multiplier are discussed. Liquid lithium and ceramic (lithium orthosilicate) are treated as tritium breeding materials. Russian development of reactor experimental unit for tritium breeding zone using beryllium, lithium ceramic and ferrite-martensite alloys for structural materials is presented.

1. Introduction

At present the Russian conception of a fusion reactor based on TOKAMAK contemplates our participation in the ITER Project and development of the domestic design of DEMO.

DEMO is assumed to generate electric power on a commercial scale [1] with simultaneous demonstration the operational reliability and safety (Fig. 1). The major technical simulations worked out for this reactor are assumed to be applied to the commercial fusion reactor that will follow DEMO.

The DEMO Project developments were started in Russia in 1992 and the significant place in the plans of the development of this reactor blanket is allocated to Test Modules (TM) for ITER.

The concepts under consideration are based on the use of two options of tritium breeding:

- a helium cooled ceramic blanket and
- a self-cooled liquid Li hot metal blanket.

In the ceramic blanket option neutrons are assumed to be bred using beryllium.

The choice and design of materials (including structural, tritium and neutron breeding ones) is a sophisticated goal and depends on their operation conditions in the systems under consideration.

This report discusses:

- steels and vanadium alloys as structural materials for the first wall (FW), tritium breeding and beryllium zone of the blanket;
- lithium orthosilicate, metasilicate and aluminate as materials for tritium breeding;
- beryllium as a FM coat and neutron breeder.

2. Main parameters of DEMO reactor and initial data for ITER test modules

The main parameters of the DEMO Project that primarily affect the choice of the blanket design and its materials are listed in Table 1. The second and third columns of the table tabulate the parameters of two DEMO options: DEMO-I for operation under induction-pulse conditions for ~1 h and DEMO-S optimised for the maximal (up to 100%) current stability based on the boot strap effect. The fourth column tabulates respective parameters of ITER [2].

They are initial data for the development Test Module (TM) projects and a choice of materials for TM (Fig. 2).

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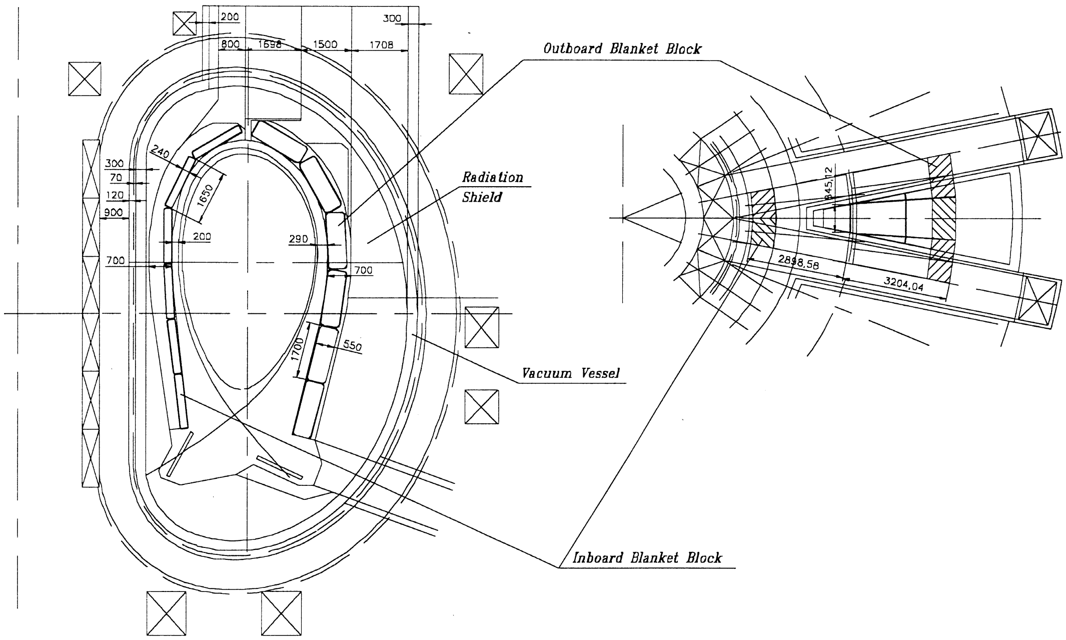

Fig. 1. The general view of DEMO ceramic block blanket.

Table 1
Main parameters of DEMO and ITER reactors

Parameters	DEMO-I	DEMO-S	ITER [2]
1	2	3	4
Geometry:			
Aspect ratio	3.31	4.5	–
Large radius, m	6.950	9.0	8.14
Small radius, m	2.100	2.0	2.80
Plasma volume, m^3	1080	~1000	2000
First wall surface, m^2	922.0	~1000	1200
Plasma parameters:			
Thermonuclear power, MW	3345	3760	1500
Neutron load on first wall, MW/m^2	2.91	3.0	1.0
Axial field, T	6.86	9.0	5.68
Pulse duration, s			1000
Energy retention time, s	1.96	1.56	
Economics:			
Thermal power, GW	4.55	~5.0	2.2 [a]
Inherent needs, MW	298	298	100
Electric power (net), GW	1.52	~1.5	
Energy conversion ratio	0.33	0.33	

[a] Heat power of blanket.

The design studies of the DEMO blanket and TM demonstrated the rigid conditions of the selected material operation and corresponding the rigid technical requirements for those materials.

The main damaging factors are:

- cycling variable and time synchronised effect of 14 MeV neutrons and powerful heat fluxes from plasma and internal release;
- thermal and mechanical loads with the peak values close to the yield strength;
- an interaction between coolant and structural materials;
- nuclear transformation of blanket materials resulting in the formation of hydrogen, helium and other elements including changes in the initial material properties.

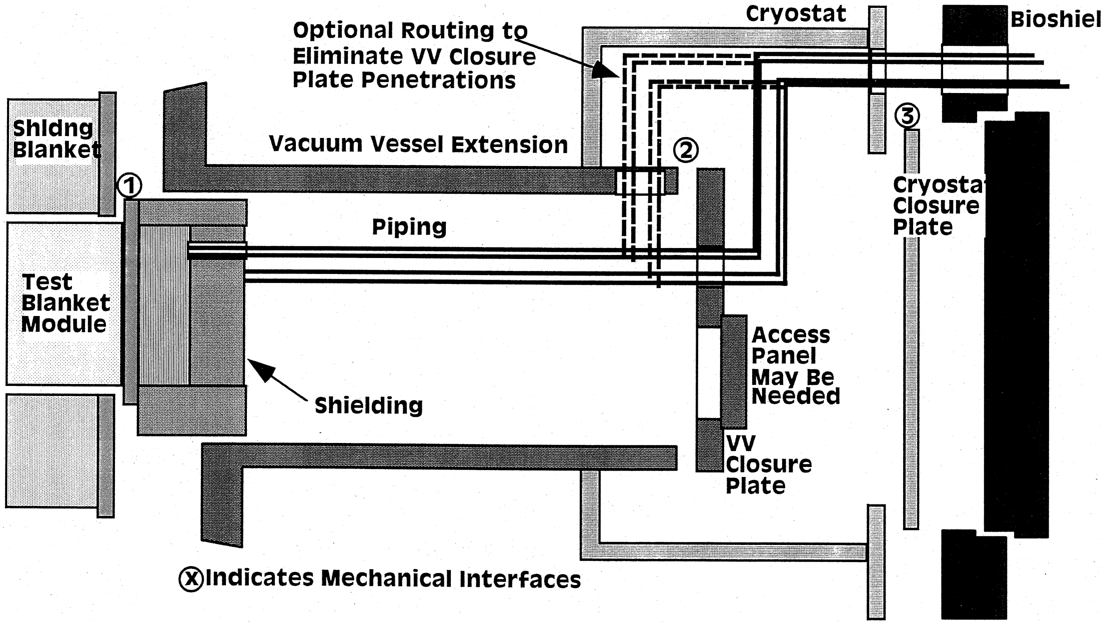

Fig. 2. Schematic of test blanket module systems in horizontal port.

The major parameters defined the operational conditions of materials in DEMO and TM are shown in Table 2.

The choice of structural materials is primarily affected by the following two factors: repairability and minimum possible activation. These two conditions arise

Table 2
Major parameters needed for development of DEMO blanket

Parameters	DEMO-I	DEMO-S
1	2	3
Neutron load ($E_n = 14$ MeW) on first wall, MW/m^2	2.9	~3
Full service life of blanket, MW year/m^2	15–20	
Full service life of reactor, year	30 (6 blanket replacement)	
Time of plasma burning, s	3400	Steady burning
Tritium breeding ratio	1.05–1.10	–
Number of pulses	~5×10^4	~500
Thermal flux to first wall, W/s:		
Mean	~40	
Maximum	80	
Neutron flux on first wall, neutron/m^2 s	~6×10^{14}	
DPA	~200 dpa	
Coolant temperature (Helium):		
for ceramic:		
T_{in}, °C	300	
T_{out}, °C	550	
for liquid lithium metal:		
T_{in}, °C	300 (steel)	
T_{out}, °C	550 (steel), 600–750 (V)	
Temperature of material in ceramic blanket, °C		
Steel	600	
Beryllium	550	
Temperature of V alloy in lithium blanket, °C	800	
Burn-up of Li, %	20	

from the detailed analysis of the all feasible situations that may take place in the process of fusion reactor operation including maintenance and decommissioning. At that it is necessary to take into consideration great dimensions and weight of blanket modules and difficulties encountered in their replacement as well as requirements of radioactive safety at normal modes and accident and a possibility of repeated use of the exhausted materials.

The possibility of accomplishing those goals tans account of the pre-holding of materials removed from a reactor for several days or months. Further on, for the normal conditions the main objective is a maximally attainable reduction of high and intermediate activity waste in accordance with the classification that prevails in the Russian Federation [3,4]. As regards the refabricability, it is proposed to study some aspects of this problem within the scope of the DEMO Project.

The above described parameters and requirements are valid for all materials used the DEMO and Blanket TM. However one to take into account that for the discussed structural and tritium breeding materials there are specific operational conditions and extra requirements are placed on them.

3. Structural materials for the first wall and blankets of fusion reactors

3.1. 9–12% Cr steels

Chromium steels may be considered as one of possible structural materials for the first wall and blanket. The specific feature of those materials in their high yield strength at adequately high ductility in combination with a favourable complex of physical properties and irradiation resistance at temperatures up to 600–650°C.

The analysis of the DEMO components reveals as the temperature (600–650°C) fast neutron damaging dose (up to 200 dpa), cyclic mode of loading (10^5 cycles), compatibility with He, Li, Pb (up to 650°C) etc. the candidate ferritic and ferritic-martensitic steels are to be thermocycle strong, irradiation and corrosion resistant and have as low as possible activation (in other words not to contain at all or contain limited amounts of Ni, Nb, Mo, Cu, and other). This combination of the requirements drastically narrows the range of suitable materials. Therefore, the major emphasis in the current investigations was put to steels that have already demonstrated their high serviceability as structural materials of fuel rod claddings, FA wrappers and other components of experimental and commercial reactor of the BN (BN-600, BN-350. BOR-60 and other) types. These are primarily heat resistant 12% chromium steels EP450 and EP823 (Table 3) [5–7] at the same time as a candidate material consideration is groin to 10Cr9MoVNb steel [5] that is less prone to irradiation induced damage but is much inferior to the above mentioned steels in terms of its high-temperature stringent (Table 4) and has a limited corrosion resistance in air, water etc.; therefore a special protection of its surface is needed in storage, transportation and so on which entails some difficulties.

The physical characteristics of the above steels are approximately at same level. As compared to akstenitic (chromium–nickel) steels the thermal conductivity of chromium steels is a factor of ∼1.5 higher. This circumstance is of high significance since structures fabricated from chromium steels will demonstrate a much lower level of thermal stresses upon cyclic loads than that in similar components made from austenitic steels.

The essential advantage of chromium steels is their low neutron irradiation induced swelling at the damaging doses up to 1000 dpa in a wide temperature range. The irradiation effected swelling of 12Cr13Mo2VNbB

Table 3
Chemical composition of ferritic-martensitic steels

	Steel type		
	EP450 (12 Cr13Mo2VNbB)	EP823 (16 Cr12MoWSiVNbB)	10Cr9MoVNb
	Contents of elements, % mas.		
C	0.10–0.5	0.14–0.18	0.08–0.12
Si	<0.5	1.1–1.3	0.1–0.3
Mn	<0.8	0.5–0.8	0.3–0.6
Ni	0.5–0.30	0.5–0.8	<0.5
Cr	11–13.5	10–12	8.6–10
V	0.1–0.3	0.5–0.8	0.1–0.2
Mo	1.2–1.8	0.6–0.9	0.6–0.8
W	–	0.5–0.8	–
Nb	0.3–0.6	0.2–0.	0.2
B	0.004 (calc.)	0.006 (calc.)	–

Table 4
Physical, tensile and refractory properties of ferritic-martensitic steels at 300–650°C

Property	Steel type	
	10Cr9MoVNb	EP-450 and EP-823
Linear expansion coefficient, $\alpha \times 10^6$ (20–400°C)	10–12	11–12
Thermal conductivity coefficient λ, W/m K (means value) (400°C)	27	28
σ_u, MPa (300–600°C)	470–300	900–400
$\sigma_{0,2}$, MPa (300–600°C)	340–240	520–370
δ_0, % (300–600°C)	15–26	17–27
$\sigma^{600°C}$, MPa	110	180
$\sigma^{650°C}$, MPa	60	80

type steel, e.g., is 0.1% at the dose at 90 dpa, $T_{irr} = 400°C$ [5].

However, as is demonstrated by the investigations at the damaging dose of 110–120 dpa the dose dependence of EP450 steels swelling is observed to go from the incubation stage to the accelerated one at the rate of (0.05–0.07)%/dpa. The swelling of the steel attained 2% at the damaging dose of 142 dpa which is an order of magnitude lower than the swelling of austenitic steels, but the investigations need be continued comprising all candidate steels to reach design damaging doses.

Another advantage of ferritic-martensitic steels (compared to austenitic ones) is absence of high temperature brittleness.

The common limitation of the steels discussed is their propensity to low temperature irradiation induced embrittleness that shows up both as a shift of the cold brittleness threshold under irradiation to the positive temperature side and as a reduction in impact strength and percent elongation at some doses and temperatures.

The strengthening effect is particularly noticeable at low irradiation doses (10–30 dpa) in a narrow temperature range (300–365°C). As the doses is increased partial recovery of the initial level of the strength takes place (Fig. 3) [5,6].

A similar dependence is also observed for impact strength. After it is substantially reduced at 20–30 dpa it starts growing with an increase of the dose. The recovery of percent elongation is observed to a lesser extent. In the studies range of damaging doses (up to 32 dpa) 10Cr9MoVNb steel is less prone to low temperature embrittlement. The investigations demonstrate that the

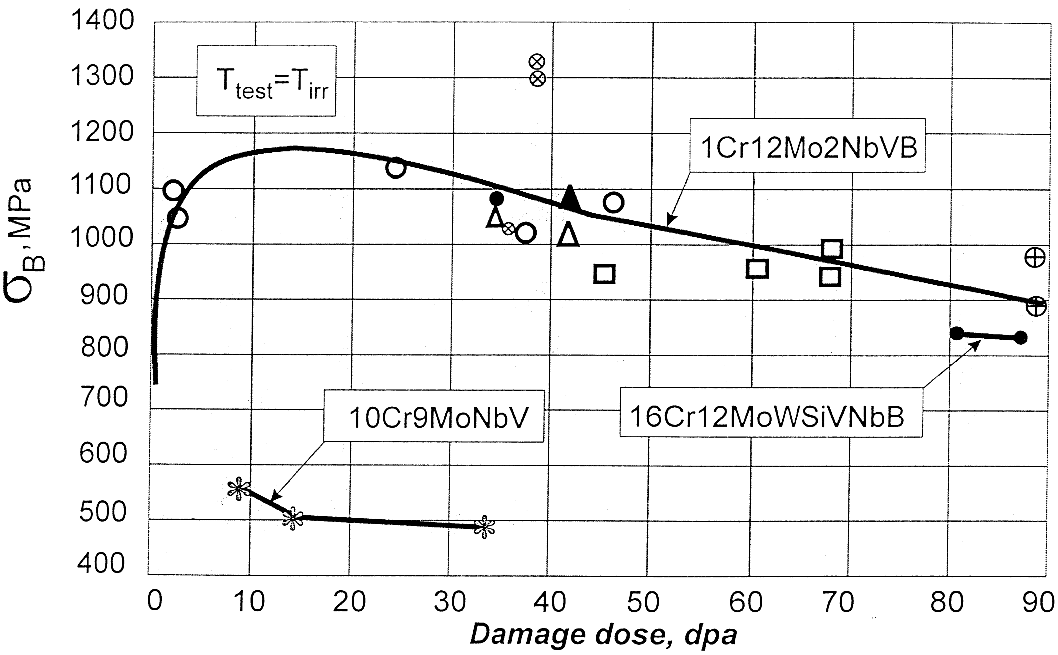

Fig. 3. Ultimate tensile strength of steel 1Cr12Mo2NbVB, 16Cr12MoWSiVNbB and 10Cr9MoNbV as irradiated at 350–365°C as a function of fluence.

embrittlement propensity of chromium steels is substantially affected by the δ-ferrite contained by their structures as well as the steel purity (impurities of low melting metals S, P and other).

While assessing the serviceability of a specific material one is to focus attention on the absolute impact strength but not only on the amount of a shift T_{br} so, when BN-600 irradiated at 340–390°C to 40–60 dpa the impact strength of 16Cr13Mo2VNbB steel within −50°C–250°C remains at an adequately high level (Fig. 4). The high value is also characteristic of irradiated EP460 steel (Fig. 4) but in the temperature range of 20–250°C.

With a rise of the irradiation temperature to ∼500°C and the irradiation dose to 100–108 dpa 12% chromium steel is observed to increase its strength insignificantly, its ductility is monotonously raised, no embrittlement shows up. An example of this behaviour is the results obtained after the BN-600 irradiation of 16Cr12MoWSiVNbB and 12Cr13Mo2VNbB steels at $T_{irr} = 385$–500°C and the damaging neutron dose of 40–108 dpa (Fig. 5). 16Cr12MoWSiVNbB steel is insignificantly strengthened while its yield strength 0.02 is somewhat lowered. The ductility level of the material does not much change to 600°C as compared to its initial state but at $T > 650$°C it is drastically increased. No indications of embrittlement are observed this is also corroborated by both micro structure and microfractographic examinations. The behaviour of 12Cr13Mo2VNbB steel is similar (Fig. 5); the ductility characteristics of this steel being lower.

The favourable properties of 12% Cr steels comprise the recovery of ductility and impact strength upon rather low temperature anneals (450–550°C) which may be considered to be a mechanism of increasing the life time of 12% Cr steel components.

Some complication in the application of Cr steel in the blanket may proceed from the need in the additional healing due to welding thick-walled components 10–20 mm thick as well as the high-temperature tempering of weldments (700–750°C) for several hours. However, there are ways of resolving those problems.

The studies of 9–12% chromium steel behaviour in various liquid metal environments (700–750°C) for

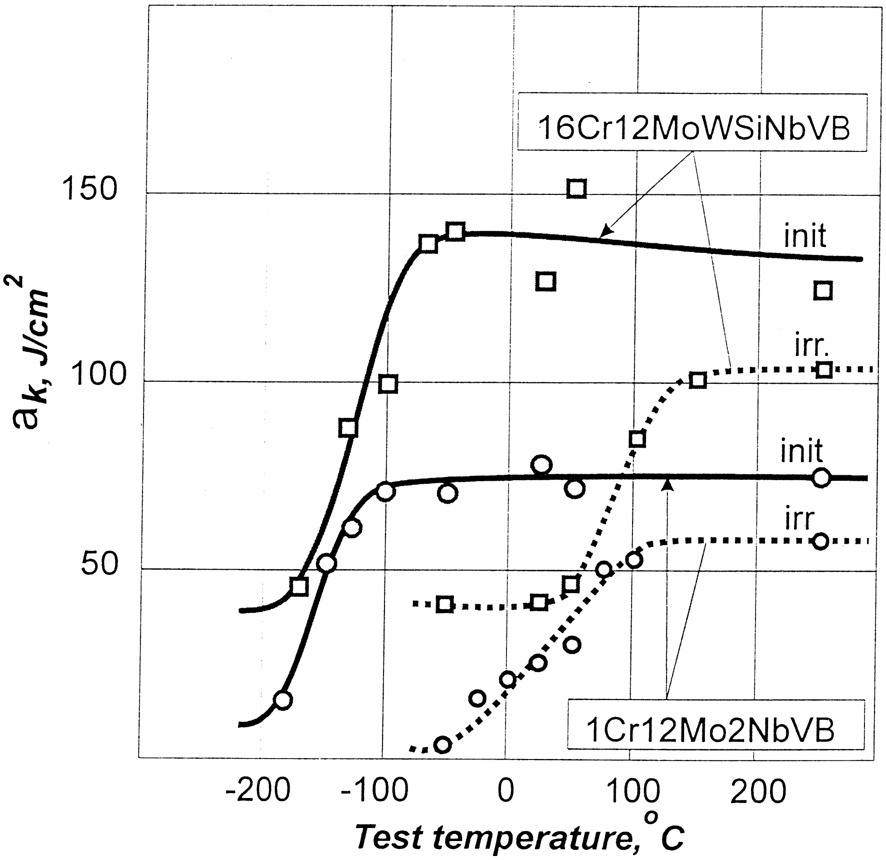

Fig. 4. Temperature dependencies of impact strength of steels 16Cr12MoWSiVNbB before and after BN-600 irradiation at 370–390°C (1Cr12Mo2NbVB – 49 dpa, 16Cr12MoWSiVNbB – 60 dpa).

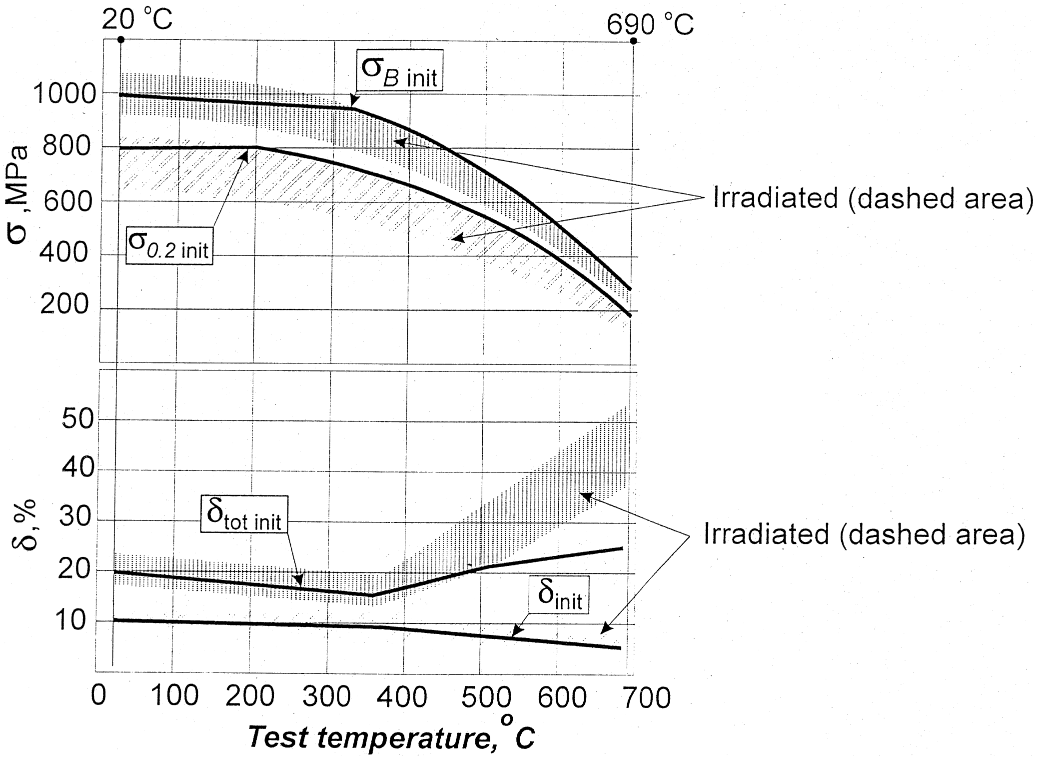

Fig. 5. Tensile properties of 16Cr12MoWSiVNbB steels as BN-600 irradiated at 385–500°C to 40–108 dpa.

several hours. However, there are ways of resolving those problems.

The studies of 9–12% chromium steel behaviour in various liquid metal environments (Li, Pb, Na. Na–Li etc.) show that these steels are compatible with the above coolants up to 600–650°C depending upon the specific alloying of the steel [8] and protective coats applied [9]. Under conditions of thermal tests, neutron irradiation and residence in a liquid metal coolant the weldments of 12% Cr steels have somewhat inferior properties compared to the base material; but the component serviceability does not give rise to concern.

The service characteristics of 9–12% Cr steels are dealt with in more detail in communications [10].

A future investigation of chromium steel properties supposes following ways of study:

- obtaining dose-time dependence of physical and mechanical properties 9–12% chromium steels for dose 200 dpa;
- investigation of effects of replacement Mo, Nb, Ni, etc. to W, Ta, etc. on physical, tensile and technology properties;
- safety problems:
  - development of coating against tritium penetration, abnormal situations;
  - waste management;
- reactor tests of the materials in MODEL (RF Reactor Lithium–Beryllium Program);
- development of industrial program for new materials;
- validification of developed materials.

3.2. Vanadium alloys

Promising as low activation structural materials (SM) for DEMO and future power fusion reactors are vanadium base alloys of the V–Ti–Cr (V–4Ti–4Cr and V–10Ti–5Cr) system. This is accounted for by the favourable combination of their physical and mechanical properties, prominent irradiation resistance at temperatures above 400°C, adequate compatibility with Li up to 700–800°C and favourable thermal resistance parameter *M*. V–Ti–Cr alloys (with the total content of alloying elements not higher than 20–30%) are workable and readily weldable (see Table 5). However, the rigid adherence to the processes is needed that eliminate the interaction of alloys with elements such as oxygen, and nitrogen at high temperatures and also hydrogen at low ones (up to 250°C).

Vanadium alloys as alloyed with Ti and Cr are characterised by the yield strength from 400 to 500 MPA that varies insignificantly with an increase of the alloying element concentration by more than 5% (Table 6).

Table 5
Short-term tensile properties

T (°C)	σ_u (MPa)	$\sigma_{0.2}$ (MPa)	δ (%)
20	580	450	26
	(500–800)	(320–700)	
	[600–1000]	[500–850]	[10–25]
600	560	400	18
	(400–700)	(250–600)	
	[450–850]	[400–700]	[10–17]
800	500	350	16
	(300–550)	(200–450)	
	[180–350]	[140–250]	[28–35]

Notes: The number above denotes properties of specific pilot alloy 10–5 (produced by VNIINM);
The numbers in the round brackets denote the range of the properties of all V-alloys of this system;
The numbers in the square brackets denote the range of the properties of 9–12% chromium steels.

Vanadium as alloyed with up to 10% Ti has a noticeably lower temperature of the ductile-brittle transition (DBTT) (<−200°C). A further increase of the Ti content raises DBTT up to −40°C. The alloying with chromium also rises DBTT.

The vanadium alloying with Ti and Cr to 15–20% in total also increases the rupture strength and creep resistance up to 850–900°C. In terms of the heat resistance vanadium alloys are much superior to met only chromium but also heat resistant chromium–nickel alloys (Table 7).

The reactor irradiation of V-alloys improves their strength that becomes lower with temperature rise. The data are listed in Table 7.

The available (although limited) data on their thermal fatigue evidence their superiority over austenitic and chromium steels.

A dangerous corrosion effect of Li on alloys of V (and also Nb, Ta) is its penetration at the expense of the interaction with O, N and H impurities. However, the threshold oxygen concentration for vanadium that is needed to initiate the process is maximal compared to that for Nb and Ta and is more than 0.2% at 600°C. This gives grounds not to practically take into account this kind of the corrosive interaction of V alloys. Hydrogen is also accumulated in liquid Li. Thanks to this fact vanadium the mechanical properties of which are highly sensitive to a hydrogen impurity will not be affected by tritium generated in Li.

The Li solubility of pure vanadium is much lower than the solubility of the main components of steel and is equal to less than 0.03% at 1000°C.

The alloying of vanadium with Ti and Cr also results in a drastically decreased weight gain (from 12–14 g/m^2 for pure vanadium to 1–2 g/m^2 for its alloys) when testing in a steam-water mixture at 300°C. The tests of this type simulate to some extent the behaviour of V alloys in helium containing oxygen and water impurities. The alloys of the VTC-15 type do not alter their strength and ductility properties after being tested under the above conditions. This is likely to be explained by the available thin protective films (TiO_2, Cr_2O_3) that inhibit the saturation of the alloys with hydrogen.

The study of the solubility of hydrogen, deuterium and tritium in V–4Ti–4Cr and V–10Ti–5Cr [11] in the range of temperatures within 973–1073 K and pressure within 1.3–13 kPa has demonstrated that the isotopic effect of H,D,T on these alloys is not present, while the hydrogen isotope solubility in V–10Ti–5Cr alloy is a factor of 2 higher than in pure V or V–4Ti–4Cr alloy. It is also shown that processes at sample surfaces play a determining role in the rate of hydrogen penetration from a gas environment and its release by the alloy.

The most substantial effect is the influence of a nitrogen impurity on the V alloy – Li compatibility. This impurity while being distributed rather quickly over the metal section lowers down the alloy ductility. The titanium alloying of V and V–Cr alloys results in a nitrogen impurity that is absorbed from Li being localised in thin surface layers of the metal where titanium is available as titanium nitride. Thin titanium nitride film is likely to be protective which results in a lower mass transfer rate in Li and a retention of the ductility by the main mass of the metal.

The designed V–Ti–Cr alloy are readily weldable. The argon-arc welding (the rate of 8 m/h, the current of 70 A) does not induce hot cracking in the weldment; the

Table 6
Long-term tensile properties

Material and temperature	Ultimate tensile strength on 10 000 h basis	Creep rate (%/h)		
	σ_{10000} kg/mm^2	$\sigma = 5$ kg/mm^2	$\sigma = 10$ kg/mm^2	$\sigma = 15$ kg/mm^2
V alloys				
650°C	25–40			0.0003–0.003
850°C	4–22	0.0001–0.5	0.001–1.0	0.02–3
950°C	1.5–15	1–0	7.0	15–50
Chromium steels,				
650°C	6–12	0.01–0.04		

Table 7
Comparative results of tensile tests before and after neutron irradiation

Property	Conditions		Material	
	T_{irr}/test (°C)	F (dpa)	V alloys	Cr steels
σ_u, MPa	400/20	0	500–800	600–1000
		50	1000–1300	800–1300
	400/400	0	400–700	450–850
		50	700–1150	700–1000
	600/600	0	400–700	250–550
		50	500–700	260–570
	400/20	0	320–700	500–850
		50	900–1150	800–1200
$\sigma_{0.2}$, MPa	400/400	0	250–600	400–700
		50	600–850	600–950
	600/600	0	200–450	230–470
		50	400–600	250–500
σ_0, %	400–20	0	10–30	10–25
		50	0–10	2–10
	400/400	0	10–30	10–17
		50	2–5	0–2
	600/600	0	10–30	15–35
		50	1–5	2–30
T_{br}, °C	365–400	0	<−200–25	−75–25
		7–15	−100–350	−60–300
	550	0	<−200–25	−75–25
		13	−100–250	−75–60

tensile properties (σ_u, $\sigma_{0.2}$) of the welded joints are 0.8–0.9 of the σ_u and $\sigma_{0.2}$ values of the base metal. Currently the processes of argon-arc and electron beam welding sheets 1 mm thick have been optimised to give good quality welded joints. The argon-arc method was used to weld sheets 5 mm thick.

Today there are several commercially applicable flow sheets to produce high purity V metal. The most promising and commercially mastered one deals with the alumothermal reduction of vanadium pentoxide followed by twofold electron beam melting to produce ingots 80–150 mm in diameter and up to 1.5 m long. Presently vanadium is produced as ingots up to 1 t in mass.

The industry has mastered the production of vanadium alloy items as sheets ($80 \times 1000 \times 1200$ mm), ($5 \times 1000 \times 12000$ mm) and tubing 5–100 mm in diameter. The vacuum furnaces available at the Ti production plants allow ingots 10 t and more in mass to be produced. Generally speaking, the technology of producing ingots and items from V-base alloys does not differ from that used for alloys on Ti, Zr, Nb base which makes it possible to produce in future billets up to 10 t and more in mass.

We suggest to fabricate pilot units and components of the first wall structure at "Atmosphere 17" facility that is in operation in RF.

The induced radioactivity of FR materials will substantially affect the condition of reactor servicing during its operation and after shut-down for maintenance and refabrication. Table 8 lists the properties of steels 316L, Manet2 (Fe–10Cr–Mn) and V–5Ti–5Cr alloy containing

Table 8
Characteristics of some steels and vanadium alloys as irradiated to fluence of 35 MW year/m^2

Material	Contact dose rate (mkSv/h) at different moments of time after reacter shut-down, years						
	1	5	10	20	40	100	300
316L	2×10^{10}	6×10^9	3×10^9	5×10^8	4×10^7	1.2×10^5	1×10^5
Manet 2	1×10^{10}	1×10^9	3×10^8	7×10^7	7×10^6	5×10^5	4×10^5
V	2×10^6	2×10^2	5×10^1	4×10^1	3×10^1	7	1
V–5Cr	3×10^6	2×10^4	3.5×10^2	4×10^1	3×10^1	7	1
V–5Cr–5Ti	1.1×10^8	3×10^4	4×10^3	3×10^3	2×10^3	5×10^2	7
V–5Cr + impurities [a]	1.3×10^7	8×10^6	5×10^6	1.2×10^6	1×10^5	3.5×10^3	3.5×10^3
V–5Cr–5Ti + impurities [a]	1.2×10^8	8×10^6	5×10^6	1.2×10^6	1×10^5	4×10^3	3.5×10^3

[a] Impurities (wppm): Co – 3, Mo – 200, Ni – 20, Fe – 120, Nb – 6, Al – 200, Pt – 2, Ag – 0.3, Eu – 0.1, Sm – 0.1, Tb – 0.1, Tu – 10.

impurities after irradiation to the fluence of 35 MW year/m^2 in the position of the DEMO reactor (12 year continuous irradiation at the average neutron load of 2.91 MW/m^2) [12,13].

It follows from Table 4 that the V–Ti–Cr alloy activation is determined by impurity elements the removal of which is a problem faces by metallurgists.

The analyses of the results studying V-alloys of the approximately similar compositions differ substantially. This results from both the appreciable impurity dependence of the properties and the alloy inhomogenity in terms of alloying elements (Fig. 6). The method of the metallurgical production by vacuum-arc melting when the electrode is assembled of plates or particles of metals having different melting temperatures provokes alloying element segregation, aside from this the properties of the alloys are much influenced by heat treatment schedule and conditions (Fig. 7), therefore these factors are also to be taken into account when comparing V-alloys between themselves.

The physical and mechanical properties of V–Ti–Cr system alloys depend on the alloying and impurity element content. To isolate the influence produced by each individual element on same property (we have chosen the shift in the cold shortness threshold and irradiation induced swelling) the data base on the impact toughness, irradiation effected swelling and yield strength was analysed statistically.

The derived mathematical relations for the effect produced by the composition on the properties under study demonstrate that there exist a favourable range of alloying and the optimal composition for each property investigated.

Based on the results of testing for impact strength the V–6.5Ti–3.4Cr is assumed to have the minimal shift in the DBTT at Ti/Cr = 1.7.

In terms of the irradiation induced swelling the V–13Ti–7.5Cr alloy is assumed to have a favourable alloying range that ensures the maximum shift of the swelling peak to the side of higher irradiation doses.

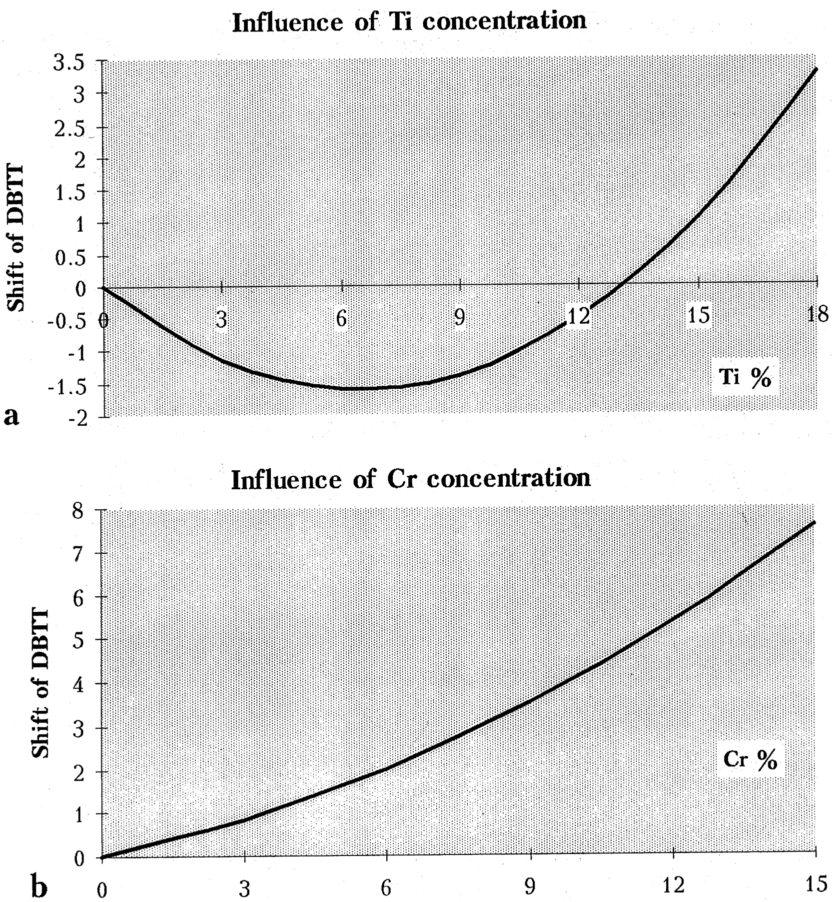

Fig. 6. Influence of Ti (a) and Cr (b) concentrations on temperature shift of DBTT.

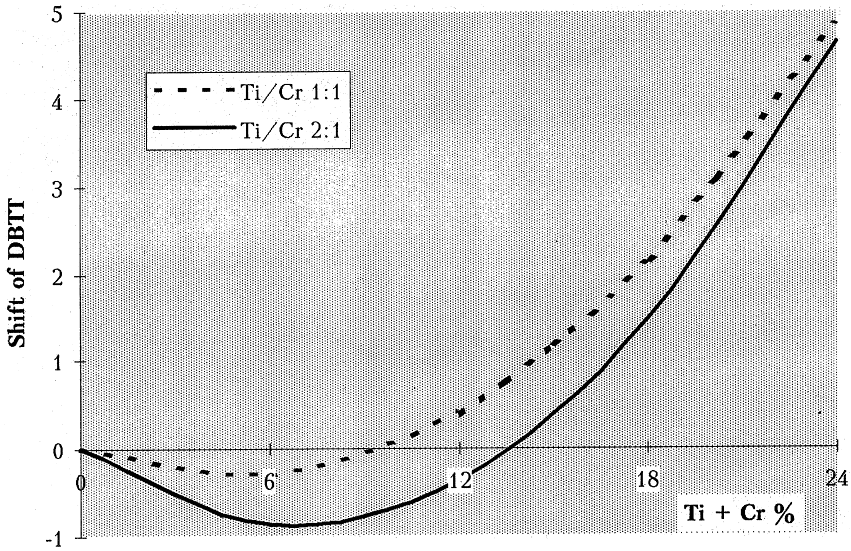

Fig. 7. Influence of Ti and Cr concentration variations on temperature shift of DBTT at Ti/Cr ration 1:1 (...) and Ti/Cr ratio 2:1 (——).

4. Lithium-contained materials in tritium breeding blankets of fusion reactors

Two variants of tritium breeding zone are considered in the Russian Program:

- breeding zone for DEMO with ceramic lithium-contained materials cooled by helium or water circuits using stainless steel as structural material;
- Test Module for ITER with hot lithium using vanadium alloys and stainless steel as structural materials.

4.1. Ceramic materials for breeding zone

Ceramic lithium-contained materials are promising for FR blankets. In comparison to hot metal systems they have several important advantages:

- compatibility with structural materials;
- no problems related to the initiation and end of the operations as for hot metal circuits;
- much less serious problems from the radiation safety standpoint at the systems development.

Tritium recovery system from lithium ceramic blanket is based on a combination of two technological processes: a hot metal tritium gas purification (HMTGP) and a hydrogen isotope separation (IS) [14,15].

At present an adequately large quantity of oxygen-containing lithium ceramics are known that may considered to be candidate materials for tritium breeding blankets. Lithium orthosilicate, metasilicate and aluminate are most suitable from the standpoint of physical–chemical properties and possibility fabricating various products (pellets, microspheres, balls) from this ceramics [16]. One of the constructive decisions assumes their application as pellets or microspheres in breeding zone.

The experiments were carried out to irradiate lithium ceramics in a graphite-water nuclear reactor; their results were used to asses changes to their strength properties at the 2–3% burn-up of lithium with tritium and helium extraction under irradiation.

A pellets fabrication technology by pressing with followed backing were developed, special units for irradiation were designed and manufactured (Fig. 8).

Pellets some 10 mm in diameter and 10–14 mm high were loaded into special ampoules that are aluminium tubes made of three successive sections ~300 mm and having the total height of ~1 m. Three ampoules were placed into irradiation unit that was loaded in a channel of nuclear reactor. This assembly was connected to special tritium installation for a control of gas composition and irradiated ceramic temperature under irradiation.

Two series of irradiation experiments in a graphite-water reactor channel were carried out. The first series involved irradiation of three assemblies to the fluence of 7×10^{19}, 4×10^{20} and 1.4×10^{21} neutron/cm^2, respectively; in the second series the irradiation was implemented in the range of 7×10^{19}–2.2×10^{20} neutron/cm^2.

Below discussion is given to the main results generated in the study of the two assemblies of the first series, that used ceramics with the Li-6 concentration within 3.5–5% (Table 9).

After the end of the irradiation experiment the pellets were removed from the ampoules. The ones that fully retained their geometrical shape (without cracks or chips etc.) were used to determine the ultimate compressive

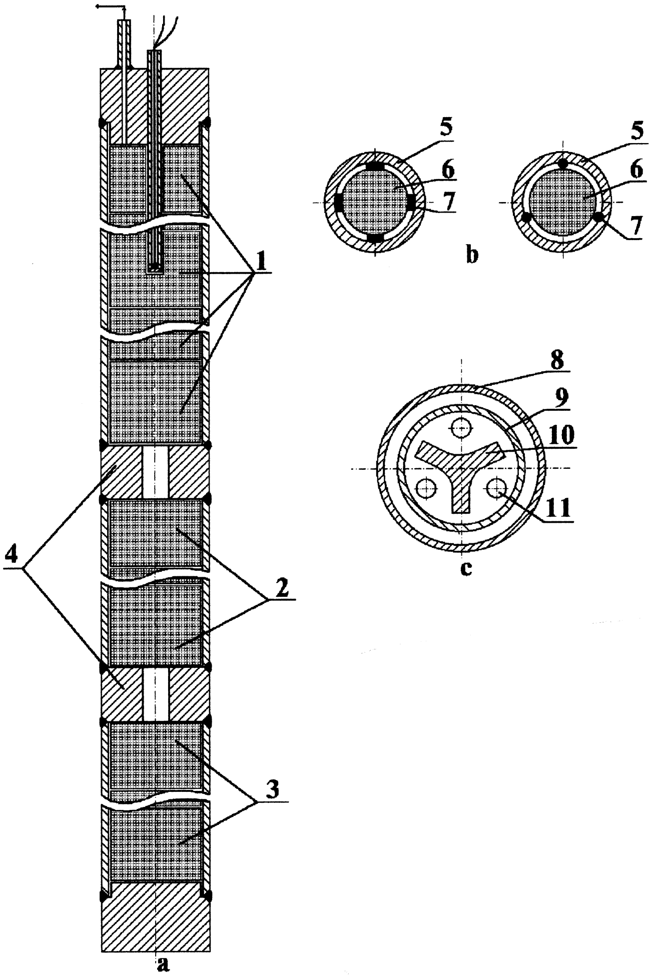

Fig. 8. Unit for reactor irradiation of ceramic pellets (a) ampoule, (b) cross-section of ampoule, (c) cross-section of assembly. 1,2,3 – ceramic pellets; 4 – dividing disk; 5 – shell of the ampoule; 6 – pellet; 7 – spacer strip; 8,9 – inside and outside tubes of the assembly; 10 – load carrying support; 11 – ampoule.

Table 9
Variation in compressive strength of ceramic pellets under irradiation

Material	Fluence, n/cm^2	Lithium burn-up (%)	Irradiation temperature (°C)	Compressive strength (MPa)	
				Before irradiation	After irradiation
Aluminate	4.8×10^{20}	1.72	204–280	79	79
Orthosilicate	4.4×10^{20}	1.56	290–300	16	16
Metasilicate	4.6×10^{20}	1.44	330–386	140	87
Aluminate	1.4×10^{21}	2.61	58–185	56	56
Orthosilicate	1.3×10^{21}	2.49	94–249	71	31
Metasilicate	1.35×10^{21}	2.50	148–215	102	60

strength (s_{comp}) that did not change in lithium aluminate, but was much reduced in ortho- and metasilicate. However, the retained strength margin was responsible for the pellet integrity and sharp retention.

The results are tabulated in Table 10 from which it follows that in the process of the experiment the generated tritium in the quantity of 29–84% was released at relatively low temperatures (see Table 9).

Table 10
Tritium extraction from irradiated lithium ceramic pellets under nuclear reactor irradiation and in post-irradiation vacuum heat treatments

No.	Material	Fluence (neutron/cm^2)	Calculated accumulation of tritium (cm^3/g)	Tritium extracted				Tritium decay	
				Under irradiation		In vacuum heat treatment			
				cm^3/g	(%)	(cm^3/g)	(%)	(cm^3/g)	(%)
1	2	3	4	5	6	7	8	9	10
1.	Aluminate	1.3×10^{20}	1.37	0.67	49	0.65	47	0.05	4
2.	Aluminate		1.37	0.89	65	0.43	31	0.05	4
3.	Aluminate		1.37	0.97	70	0.35	26	0.05	4
4.	Orthosilicate	$(4.4–4.8) \times 10^{21}$	7.85	6.60	84	1.1	14	0.15	2
5.	Orthosilicate		7.85	5.34	68	2.36	30	0.15	2
6.	Metasilicate		5.01	2.85	57	2.01	40	0.15	3
7.	Aluminate		3.87	2.12	55	1.64	42	0.11	3
8.	Metasilicate	$(1.3–1.4) \times 10^{21}$	7.24	3073	52	3.32	45	0.19	3
9.	Orthosilicate		10.85	7.48	69	3.00	27	0.37	3
10.	Aluminate		4.85	1.43	29	3.25	67	0.17	3
11.	Aluminate		4.85	1.99	41	2.69	55	0.17	3
12.	Orthosilicate		10.85	6.59	60	3.89	37	0.37	3
13.	Metasilicate		7.24	2.92	40	4.14	57	0.18	3

4.2. Liquid lithium for ITER test module

The system of tritium extraction from liquid lithium metal at the ultimately low concentration (~1 appm) as designed at VNIINM is based on the technological process of a jet return-flow non-equilibrium molecular distillation the theoretical-design validation of which was implement in the ITER project framework [17].

The process comprises two stages:
- concentrate tritium in lithium to 10^4 appm;
- extract tritium from the concentrate.

The process is implemented in a single apparatus that is a vacuum vertical column inside which a stretched lithium jet (~100 m) is flowing in spiral and is condensed on a condenser. Helium is extracted and pumped from the top of column; in its bottom part where the enriched concentrate is collected tritium is extracted at the extraction coefficient of ~0.5–0.7 at the evacuation rate of ~50 1 Pa/s.

Structural material for the column is Ni–Cr–SS. Mo and W can be used for coating.

The given technology allows to achieve concentration of tritium in lithium as small as and by this means the significant from the viewpoint of radioactive safety problem of tritium inventory in blanket reduction is solved.

5. Beryllium in fusion reactor blanket

5.1. First wall

The SSC VNIINM developed physical model of polycrystalline Be allowed the formulation of requirements for the metal structure as applied to different operation conditions [18]. It follows from the model and that is experimentally corroborated that it is not feasible to create Be that would simultaneously have the high thermal strength and high irradiation resistance since the high thermal strength Be is to be relatively coarse grained and contain insignificant oxide inclusions along the grain boundaries, while the irradiation resistant one is to have fine grains at a higher content of inter granular oxide inclusions. Therefore, the special grades of Be were fabricated, one of which (TR-30) was irradiation resistant, the other one (DShG-200) was thermally strong and the third one (TGP-56 or TShG-56) had the above properties at intermediate level. The DShG-200 and TShG-56 grades were produced by hot pressing followed by upsetting be powders with particle size less than 200 and 56 μ, respectively, while the TR-30 grade was produced by hot pressing the powder with the particle size less than 30 μ. The materials were SM-3 reactor irradiated at 650–700°C to the fast neutron fluence $(5.6-6.2) \times 10^{21}$ n/cm^2 (2.7–3.0 dpa, the helium concentration up to 1150 appm) $E > 0.1$ MeV. As it was

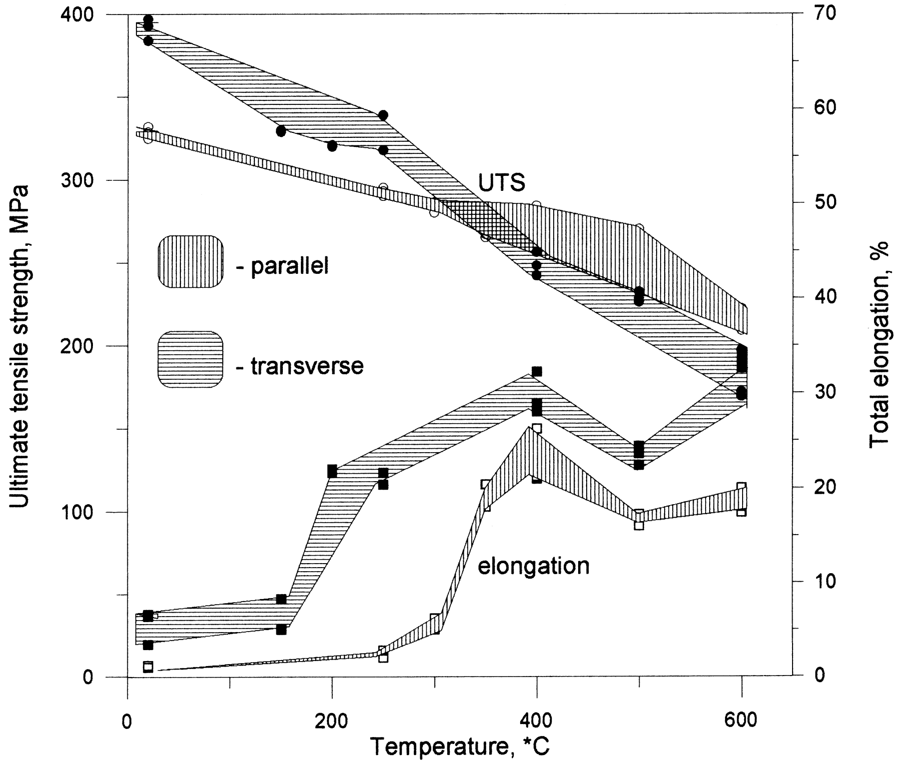

Fig. 9. Temperature dependence of mechanical properties for TGP-56 beryllium grade.

expected under those conditions minimal swelling not more than 0.3% was demonstrated by the TR-30 grade. For further investigations The TGP-56 grade (Fig. 9) was chosen as a basic one.

Depending on the test results one is capable of purposefully improving one of the properties (thermal strength or irradiation resistance) at the expense of the other. All the changes are carried out in the framework of the same technological process by varying its parameters (chemical composition and particle size distribution of initial powders, temperature and time characteristics of the compacting process etc.)

It is evident that the requirements for ITER and DEMO Be may differ due to different operation parameters and, correspondingly, different Be modifications can be needed. It is to be emphasised that all the available and assumed hypothetical grades of Be may be produced using the available equipment by the well-known technological processes.

5.2. Joints

Many adequately well studied routes of joining Be and Cu-alloy or stainless steel bases are known (soldering with various solders, different kinds of diffusion welding) [19], however, today it is not possible to single out anyone since the results are not yet available on the influence of neutron irradiation on the quality of the joint. The more so, in thermal cycling tests of the unirradiated joint specimens the fracture as a rule takes place in the joint, not in Be [20]. Even the limited results available give grounds to suppose that the problem of the joint reliability may prove to be critical in terms of the service life of a fusion reactor as a hole.

It is to be pointed out that when developing "Be–Cu alloy" or "Be-stainless steel" pairs one is to take into account not only the different thermal expansion coefficients but also the propensity for swelling shown by Be and the base material. It is known [21], e.g., that the extent of swelling of Cu-base alloys can vary in wide ranges a dependent on the alloy composition and irradiation temperature. This circumstance may influence the final choice of Be grade aimed at reducing extra stresses in the joint under neutron irradiation.

5.3. Breeding blanket

Be and Pb may be used as neutron multipliers in that blanket. It is more preferable to use Be due to its higher multiplication factor, higher melting temperature and higher protective properties in relation to neutron irradiation. Various concepts of the ceramic breeder blanket contemplate the use of Be in the compacted or porous states as well as in the form of pebbles [22]. In all instances it is to meet the following two requirements: first, it is to resist gas swelling since to compensate for it gaps are to be contemplated which will inevitably deteriorate the heat transfer, and, second, it is to provide for the free release of tritium. In our view, the application of porous Be having the open porosity is of a good promise. Although the irradiation tests of porous Be have not yet been implemented there is every ground to suggest that it will maximally meet the above requirements since the open porosity is to provide a slight removal of tritium and helium, thus, reducing the danger of gas swelling. On the other hand, compared to a compact Be, porous Be is much more dangerous under emergency conditions [23], however, this danger is not to be exaggerated against the background of other unpleasant effects of an accident (tritium contamination and so on).

At SSC VNIINM a promising method have been worked out and patented or producing porous Be having an inherent open porosity. The method is based on the fact that beryllium hydride BeH_2 when heated to ≈200°C is decomposed to form hydrogen gas and Be metal. The mixed Be and BeH_2 powders are heated with the simultaneous application of pressure under conditions that provide the removal of released hydrogen that forms through tunnels of open porosity. The amount of porosity is determined by the Be/BeH_2 ration the mixture and the process parameters (pressure, temperature etc.). The metal produced in this way may have a very low oxygen content which reduces the chemical retention of tritium. This method was used to fabricate 15–20% porosity mock up specimens clad in stainless steel with the good thermal contact between Be and cladding; it is planned to fabricate a model element of a breeding zone of the DEMO ceramic blanket.

6. Suggested reactor tests of ceramic breeding zones and neutron breeders in blanket

As it was pointed out above in present in Russian Federation the preparation is in progress for in-pile testing models of tritium breeding zones on the base of lithium silicate and beryllium neutron multipliers.

The goal of these investigations is to systematically study the tritium generation in ceramics and beryllium and tritium and helium extraction under irradiation at the temperatures similar to those in the blanket system of DEMO and TM.

The cross-section of the breeding zone model is shown in Fig. 10.

In the centre of the model there is a lithium element with a system for collecting by pellets under irradiation and also with a thermocouple to measure the centreline temperature of the ceramics. A similar system is also available in the beryllium zone.

Two model options are suggested to be tested:

- an option using ceramic pellets of ~20% porosity and beryllium too of ~20% porosity;
- an option using ceramic and beryllium spheres of ~1.5 mm diameter.

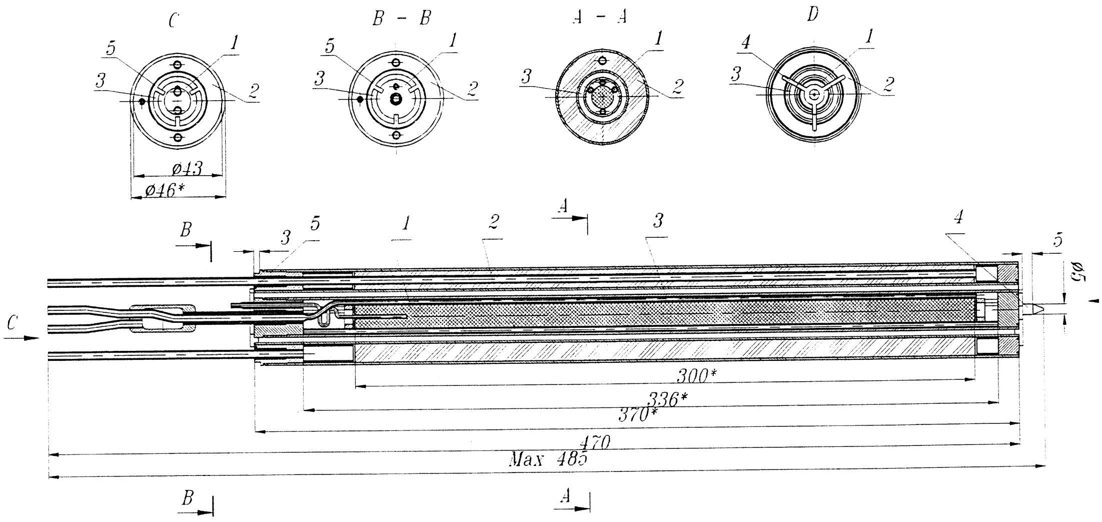

Fig. 10. DEMO blanket breeding zone element model. 1 – breeder elements; 2 – multiplier element; 3 – codant channels; 4,5 – distance elements.

7. Conclusion

1. The materials discussed in the report may be considered to be candidates for DEMO blanket and their testing in Test Modules is advisable.
2. The studying properties of 9–12% Cr ferrite-martensite steels demonstrate that the suggested materials meet the requirements placed on the materials of the FW and Test Modules of the ITER, however to choose the materials for the DEMO items under discussion is needed on the serviceability of the discussed steels under conditions of thermocycling and high dose irradiated (above 140 dpa).
3. Vanadium alloys are promising structural materials:
   - capable of ensuring the blanket operational temperature of 700–800°C at the expense of high short-term and long-term strength at acceptable ductility and creep;
   - have good thermal conductivity and relatively high heat resistance which allows high wall thickness;
   - have a high corrosion resistance in liquid lithium and pure helium at the temperatures up to 800°C;
   - have a high irradiation resistance (do not subject to swelling or embrittlement) at 400°C and higher;
   - as compared to chromium steels their initial level of induced activity is an order of magnitude lower and becomes seven orders lower after 30 years which makes the refabrication possibility.
4. It is to be emphasised that currently the large-scale production of vanadium alloys is not available and the data base is not yet adequate to validate their wide-scale introduction.
5. Vanadium alloys are highly hydrogen permeable and prone to hydrogen effected embrittlement at temperatures below 250–300°C.
6. The implemented irradiation of lithium orthosilicate, metasilicate and aluminate pellets demonstrated that they may be used to generate tritium under irradiation by thermal neutrons at fluence of $1.5 \times 10^{21} cm^{-2}$ and the lithium burn-up above 2.5%. Results are shown that the substantial amount of tritium is extracted at relatively low temperatures, that allows one to consider the particular ceramics to be candidates for DEMO and TM.
7. The suggested process of tritium extraction from liquid lithium ensures its tritium concentration below 1 appm which is important in view of the resolution of the problem related to the minimisation of the tritium inventory in DEMO.
8. The currently achieved ultimate values of the thermal strength and irradiation resistance of beryllium are unlikely to be much improved by the technological processes presently available; materials are to be designed having needed priority properties at the comprise values of minor ones.
9. Due to no demand the commercial production of high porosity beryllium for a breeding blanket has not been set up, however, all the technical decisions needed for its organisation are available.
10. The experimental results are badly needed on the porous beryllium behaviour under neutron irradiation as well as on gas extraction from beryllium.
11. The suggested program of nuclear reactor tests of Li–Be models will make it possible to determine the sphere of the application of Li and Be in more detail.

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