Materials and fabrication technology of modules intended for irradiation tests of blanket tritium-breeding zones in Russian fusion reactor projects

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Abstract

This paper surveys the modules and materials of blanket tritium-breeding zones developed in the Russian Federation for fusion reactors. Synthesis of lithium orthosilicate, metasilicate and aluminate, fabrication of ceramic pellets and pebbles and experimental reactor units are described. Results of tritium extraction kinetics under irradiation in a water–graphite reactor at a thermal neutron flux of 5×10^{13} neutron/$(s\,cm^2)$ are considered. At the present time, development and fabrication of lithium orthosilicate–beryllium modules of the tritium-breeding zone (TBZ), have been carried out within the framework of the ITER and DEMO projects. Two modules containing orthosilicate pellets, porous beryllium and beryllium pebbles are suggested for irradiation tests in the temperature range of 350–700°C. Technical problems associated with manufacturing of the modules are discussed. © 2000 Elsevier Science B.V. All rights reserved.

1. Introduction

The Russian fusion reactor program assumes development of a DEMO design and ITER test modules [1]. The Russian concept for a solid blanket tritium-breeding zone (TBZ) includes a helium-cooled lithium ceramic and beryllium as a neutron breeder. Reactor tests of a lithium ceramic TBZ were carried out at an early stage of fusion research in the USSR [2]. A main focus of these efforts was the study of service life of the lithium ceramics because the cost of electricity produced by a fusion reactor depends strongly on the frequency of blanket replacement. Fabrication technology of a large number of pellets using lithium orthosilicate, metasilicate and aluminate was developed, and the pellets were

irradiated up to a fluence of 1.5×10^{21} neutron/cm^2 and a lithium burn-up of 2–3%. Initial estimates of TBZ service life were carried out on the basis of strength changes of the irradiated pellets. In addition, the tritium retention and kinetics of release from pellets under irradiation was studied. The second stage of this research and development (R&D) was begun in 1997.

Two modules of TBZ have been developed for reactor tests with both breeder–multiplier elements and a cooling system. The first module contains lithium orthosilicate and beryllium particles, and the second contains lithium orthosilicate pellets and porous (~20%) beryllium. In 1998, the two modules were manufactured and tested in Bochvar Institute before reactor tests. The next stage of the activity including manufacture of reactor assembles and reactor test was begun in 1999.

Results of development, fabrication and testing of lithium ceramic pellets are presented here, and the new modules and their fabrication technology are discussed.

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2. Background and manufacture of the TBZ modules

2.1. Results of irradiation of reactor assembles with lithium ceramics

The present development of breeding zones is a continuation of our investigations of mechanical properties and gas extraction from lithium ceramics irradiated in a water–graphite nuclear reactor. Three ceramics – lithium orthosilicate (Li_4SiO_4), metasilicate (Li_2SiO_3) and aluminate ($LiAlO_2$) – were selected for this activity. Assemblies and systems were designed and fabricated to provide experiments with continuous temperature measurements of the pellets and periodic checks of gas emission and composition under irradiation. The main part of the irradiation assembly was a capsule constructed of an aluminium pipe with three sections containing lithium pellets, each with a height of about 300 mm [5]. The three capsules were joined in the assembly by a special loading rod. Each capsule was joined to the tritium system assembled near the reactor by a separate tube for measurement of temperature and gas composition. A method was developed for fabricating pellets with a diameter of about 10 mm and heights of 5, 10 and 14 mm [3]. Almost 5000 pellets and nine reactor assemblies were fabricated for the program.

Two series of irradiation experiments were carried out. In the first series, three assemblies were irradiated to fluences of $(7.1–8.8) \times 10^{19}$, $(4.6–4.8) \times 10^{20}$ and $(1.3–1.4) \times 10^{21}$ neutron/cm^2. In the second series, the neutron fluence was in the range of 7×10^{19}–2.2×10^{20} neutron/cm^2 [2,3]. Results of the tritium release from the pellets under irradiation are given in Table 1. A determination of the compression limit for irradiated pellets (P_c) was made after the reactor tests. The measurements performed showed no change in P_c for the aluminate, and a significant decrease in P_c for the ortho- and meta-silicate [2,3].

The program did not include a plan for complete investigation of gas release from irradiated ceramics [2]. Gas tubes were not heated, and moisture had not been

Table 1
Parameters of gas emission from lithium contained ceramic under irradiation

Experiment index	Irradiation time (s)	Fluence of thermal neutrons (neutron/cm^2)	Lithium contained ceramics	Maximum temperature in a pellet centre (°C)	Calculated burn-out (%)		Calculated gas amount emitted under irradiation (cm^3/kg)
					Li	Li-6	
K	12.0×10^6	4.8×10^{20}	$LiAlO_2$	260	1.72	31.3	7740
			Li_4SiO_4	328	1.56	30.6	15 700
			Li_2SiO_3	350	1.44	30.1	10 000
KK	32.5×10^6	1.4×10^{21}	$LiAlO_2$	119	2.61	69.3	9700
			Li_4SiO_4	193	2.49	67.3	21 700
			Li_2SiO_3	158	2.50	69.5	14 500
T1	2.2×10^6	7.1×10^{19}	Li_4SiO_4[a]	343	0.52	8.7	3900
				462			
				568			
T2	2.7×10^6	8.8×10^{19}	Li_4SiO_4[a]	331	0.70	11.6	5220
				447			
				552			
T3	2.7×10^6	8.8×10^{19}	Li_2SiO_3[a]	256	0.33	8.3	1660
				318			
				420			
T4	8.0×10^6	2.6×10^{20}	Li_2SiO_3[a]	233	1.4	34.9	6970
				270			
				380			
B	2.3×10^6	7.7×10^{19}	$LiAlO_2$	300	0.39	10.7	1320
			Li_4SiO_4	318	0.56	9.3	4200
			Li_2SiO_3	212	0.38	9.4	1870
C	2.2×10^6	7.3×10^{19}	$LiAlO_2$	313	0.43	12.0	1470
			Li_4SiO_4	312	0.62	10.4	4670
			Li_2SiO_3	203	0.42	10.5	2090

[a] Three ampoules with same material but with different clearances and consequently different temperature in pellet centres were simultaneously irradiated in each of experiments T1–T4.

extracted from the pellets before they were loaded into the reactor; so a significant amount of tritium was extracted as HTO and some of this was adsorbed on the inner cold surfaces of the tritium lines. Therefore, the amount of helium produced was used for determining a tritium balance, and the results are shown in Fig. 1.

Some of the pellets were used for quantitative estimation of tritium extracted under irradiation by measurement of tritium and helium release during vacuum–thermal anneals in the temperature range of 50–900°C in a laboratory installation. It was demonstrated that 30–84% of the tritium bred was extracted under irradiation at a relatively low temperature (<400°C) [2]. However, the unheated system lines and low temperature of the pellets under irradiation did not permit the study of the kinetic processes in detail. This problem is supposed to be addressed in the present stage of our research with two modules of TBZ.

2.2. Design of TBZ DEMO module for reactor tests

Reactor tests of breeding zone modules will be carried out in the IVV-2M reactor with only lithium orthosilicate. The main design of the DEMO blanket breeding zone elements modules (BZEM) for in-pile tests correspond to zones developed for ITER and DEMO blankets, but the height and width of the modules are selected with respect to the size of the channel in the reactor in which irradiation will take place. Therefore, the module height is 300 mm and the maximum diameter is less than 46 mm.

The module (Fig. 2) is a coaxial assembly of cylindrical elements: breeding element (1) and multiplier element (2) with a central hole. The annulus (3) is 2.75 mm wide, which is maintained by two separators (4, 5), and it is used for coolant circulation. The breeding element is a cylindrical channel determined by a tube 19 mm in
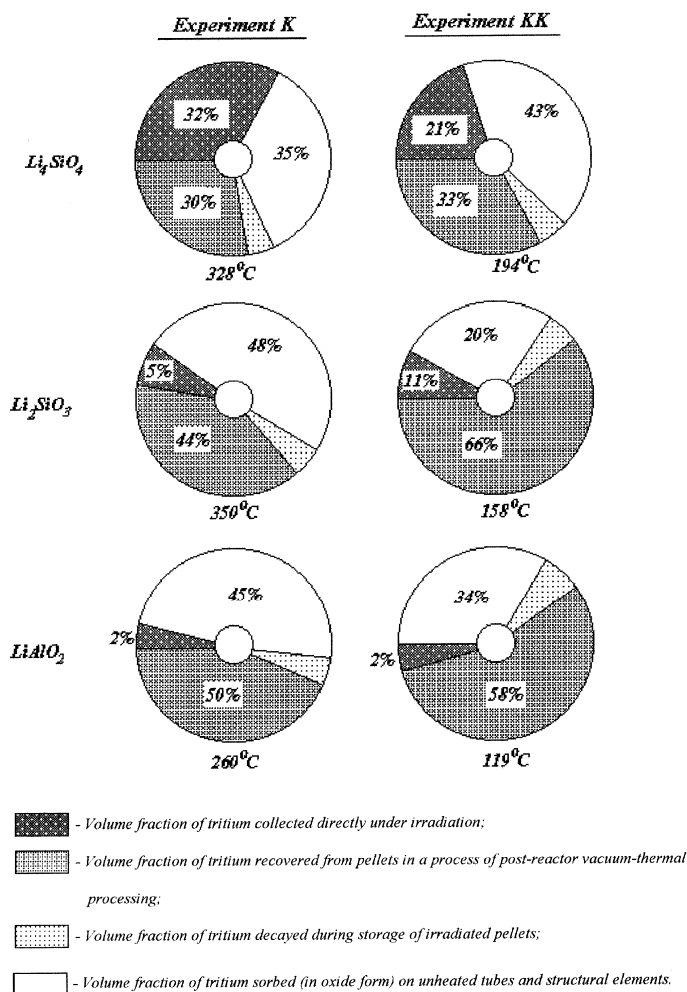

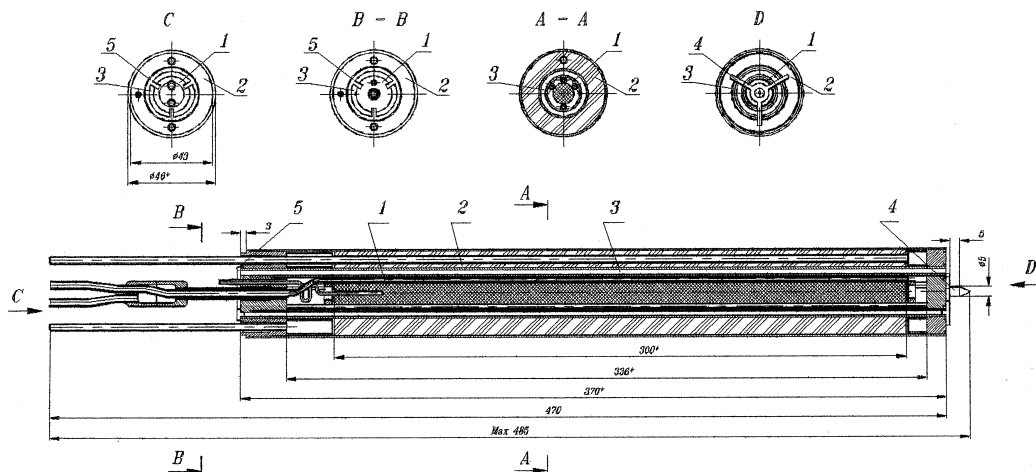Fig. 1. Balance diagrams of tritium recovery from irradiated ceramic pellets (for experiments K and KK).

Fig. 2. TBZ DEMO module for reactor tests.

diameter by 1 mm in thickness. The internal space of the cylindrical channel is the location for the ceramic breeder. The ceramic breeder consists of cylindrical pellets (11 mm diameter and 10 mm height) or spheres (1–1.5 mm diameter) in a pebble-bed. The multiplier element is a system of two coaxial cylindrical channels with a 46 mm diameter and a 26 mm diameter, both with 1 mm wall thickness. The annulus between the cylindrical channels is filled with multiplier made from either porous beryllium (20% porosity) or beryllium spheres in a pebble-bed.

3. Materials for the tritium-breeding zone

3.1. Synthesis of lithium orthosilicate and zirconate

Lithium orthosilicate with close to the natural abundance of lithium-6 and 7 isotopes has been chosen for the present module zone studies. Lithium zirconate is currently considered as an alternative material. Some physical and chemical properties are given in [3]. Some aspects of this materials synthesis are outlined as follows.

The carbonate method consisting of an interaction between lithium carbonate (Li_2CO_3) and silicon dioxide (SiO_2) was used for lithium orthosilicate synthesis. This process has several stages to produce the orthosilicate and manufacture the ceramic [3,4]. Mixing and milling initial components were carried out in a porcelain mill with corundum balls. The degree of synthesis was determined by weighing the charge in several process stages and using X-ray analysis. The material produced was used for pressing and sintering pellets. Two pellet groups with porosity of 19–20% were fabricated and tested for strength [2,5]. Impurities in the pellets were analysed, and a small contamination was found by iron (3×10^{-3}%) from the press and by aluminium (4×10^{-3}%) from the Al_2O_3 bed.

The second variant of the breeding element utilises spherical ceramic particles. This is a simpler form from a manufacturing standpoint, and it has a large ceramic surface area for tritium extraction. We are developing two variants of manufacturing ceramic particles with diameters 1–2 mm. These are: (1) breaking up and spheroidisation of ceramic pellets and (2) stamping threw a draw plate. The first method was used in this program.

$Li_2O–ZrO_2$ and $LiOH \cdot H_2O–ZrO_2$ mixtures are currently used for lithium zirconate synthesis. Concentrations of $Li_2Zr_2O_2$, Li_4ZrO_4, Li_8ZrO_6 and $Li_6Zr_2O_7$ have been found in a first system by X-ray analysis [7,8]. Orthozirconate Li_4ZrO_4 is produced in a range of 900–1000°C. A mixture of Li_4ZrO_4 and Li_2ZrO_3 is observed at low processing temperatures and a mixture with Li_8ZrO_6 is found at higher temperatures. Li_2ZrO_3 is produced by heating an oxide mixture at 750°C. Existence of two forms of metazirconates (Li_2ZrO_3-I and Li_2ZrO_3-II) has been found [9–12].

3.2. Fabrication multiplier zones from porous beryllium and pebbles

Current concepts of beryllium multipliers in ceramic breeding blankets propose utilising Be in compacted or porous states as well as in a pebble form [13]. The main advantage of a porous beryllium multiplier at the operating condition of a fusion reactor is an inherently open porosity, which leads to easier removal of helium and tritium. One promising approach is to use a porous (20%) beryllium. In this study, a unique manufacturing technique of porous beryllium with inherent open porosity [5] was used. This method was used to fabricate 15–20% porosity mock up specimens clad in stainless steel with good thermal contact between the Be and the cladding; fabrication of a module element of a breeding

zone of the DEMO ceramic blanket with this material is planned.

3.3. Structural materials for blanket TBZ

In the Russian Fusion Reactor Program, ferritic–martensitic steels are considered as possible structural materials for blankets. Steels EP-450 and EP-823 with 12% chromium are of this sort. A specific feature of the materials is high strength and adequate ductility in combination with a favourable combination of physical properties and irradiation resistance at temperatures up to 600–650°C.

Some main properties of steels EP-450 and EP-823 have been presented in [14–16]. The steels have already demonstrated their performance as structural materials in fuel rod cladding, fuel assembly wrappers and other components of the experimental and commercial reactors BN-600, BW-350, and BOR-60. The performance of 9–12% chromium steels in these reactors is summarised in [17].

Consequently, EP-450 has been selected as a structural material for the present modules. The drawback of ferritic–martensitic steels is higher hydrogen permeation relative to austenitic steels (almost factor 10). To decrease the permeation on the inside and outside surfaces, the breeding module walls were aluminised. An efficient method of aluminising austenitic steels is to coat them with $Al + Al_2O_3$ powder [6]. A modification of this method was developed for EP-450.

4. Weld and solder joints in the modules

Modules for reactor tests have many weld and solder joints placed opposite to each other. Module designs have to meet not only the technical and functional requirements but also weld and solder feasibility issues. For instance, avoiding overheating of adjacent materials during welding and placing lead-ins of thermocouples with reliably tight fit were major challenges. To surmount this, adapters were welded together on the inlet and outlet gas pipes of the modules to provide reliability of the joints and simplicity of assembling and tightness check on thermocouple feedthroughs. A high-temperature braze and special test methods of the joints were developed for this application.

The technology of argon-arc-weld for welding the outside pipes and caps together was the same as that used for fuel assemblies of nuclear reactors [18,19]. However, a new technology was developed for the welds of the 25 mm diameter by 1 mm thick pipe. The welds were examined for continuity by X-ray methods and the tightness and structure and hardness of the joints were determined on selected samples. The developed technology provides high-quality joints.

5. Conclusion

The service life of lithium orthosilicate, metasilicate and aluminate pellets for breeding zones has been investigated at fluences up to 1.5×10^{21} neutron/(s cm^2), and the orthosilicate form has been selected for the next phase of design. Designs of two-blanket ceramic breeding zones with beryllium multipliers have been carried out and the assembly of one has been developed. The technology of manufacturing lithium orthosilicate pellets and spherical particles, porous and pebbles beryllium has been developed for the blanket breeding zones modules. The two modules of blanket breeding zone for testing in IVV-2M reactor have been fabricated.

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