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Journal of Nuclear Materials 307–311 (2002) 823–826

journal of nuclear materials

www.elsevier.com/locate/jnucmat

Influence of neutron irradiation on the strength characteristics of lithium ceramic pellets for fusion reactor blankets

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Abstract

The possibility of using lithium containing ceramic pellets in fusion reactor breeding blankets is estimated. A pilot technology of pellet manufacture of lithium orthosilicate, metasilicate and aluminate was worked out for reactor tests of lithium ceramics. About 5000 pebbles were manufactured by this technology, which were irradiated in a water graphite nuclear reactor up to the fluence of 1.5×10^{21} cm^{-2}. The irradiation assemblies containing 270 pellets were designed and manufactured. A simultaneous irradiation under identical conditions and a comparison of their strength limit σ_{compr} before and after irradiation allow to estimate the neutron irradiation effect on the strength characteristics of the pellets. An estimation of the pellet life has been carried out. The pellets can be applied in a blanket up to a lithium burn-out of 2.5%.

1. Introduction

Controlled fusion is discussed as the most attractive source of energy among many perspectives. A wide development of the program for electric power production demands the creation of blanket systems for tritium breeding. The Russian conception of a fusion reactor involves the development of a DEMO design and the participation in the ITER Project [1]. The conception of tritium breeding zone (TBZ) the blanket are based on:

- solid systems using lithium ceramics,
- liquid-metal lithium blanket.

The solid variant is known to have some advantages:

- safety,
- no corrosion problems,
- simple hydrodynamic in magnetic fields.

But some deficiencies exit and a main item is a restricted service life of the TBZ because lithium ceramics operate under neutron irradiation in just limited time by reason of lithium burn-up and deterioration of physical and mechanical properties of the ceramic. This gives reasons for removable blanket from the reactor and an increase of a produced electricity costs.

Really, a service life is dictated by the design of TBZ. Nevertheless we have to select some parameters of ceramic materials to estimate one. For the proposal a compression limit has been taken for the main parameter considering a service life of the ceramic TBZ.

A pellet as initial model for the research was adopted because of further operation after crush of a pellet and a blanket, and the event may be discussed as a low limit of the service life.

2. Reactor assembly, facilities and systems for gas analysis and investigation of strength properties of ceramic pellets

A complex of reactor assembly, facilities, units and tritium systems were designed and manufactured to

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PII: S0022-3115(02)01299-0

analyse the gas extracted from the lithium ceramics in the in-pile experiment after irradiation and then to investigate the mechanical properties of the pellets in a laboratory. Mechanical properties of the ceramic pellets are known to be characterized by many parameters and a compression limit (σ_{compr}) has been adopted in the present investigation.

A block diagram of the complex is shown in Fig. 1. It contains a reactor assembly, a tritium gas facility touched to the assembly and a tritium laboratory installation for gas extraction after removing the lithium elements from the reactor and a device for the investigation of σ_{compr} after reactor irradiation. The reactor assembly contains three lithium ceramic elements joined by a loading rod (Fig. 2). Each element is an aluminium alloy pipe of 1 m height with three sections containing the lithium ceramic pellets. The lower part of the pipe was the pressurized and the upper part of each pipe was connected to tritium gas facility for gas and temperature measurements [2]. A gap between pipe and ceramic pellets was implemented by means of aluminium gaskets.

Three ceramics, lithium orthosilicate (Li_4SiO_4), metasilicate (Li_2SiO_3) and aluminate ($LiAlO_2$) with an isotope concentration of lithium-6 in the 3.5–5% range were used in the program. The pellet fabrication was carried out by pressing and burning in several process stages. The control of the process was provided by X-ray analysis and some impurities in the pellets were analyzed. In the following a limit of compression (σ_{compr}) of the pellet was determined.

The fabrication process has been developed to provide a repetition work of the pellet properties size, porosity and limit of compression. Root-mean-square errors for height and porosity are $\simeq$1.46 and $\simeq$4.9% (for Li_4SiO_4), $\simeq$0.06 and $\simeq$4.8% (for Li_2SiO_3) and $\simeq$0.06 and $\simeq$3.8% (for $LiAlO_2$). All pellets were tested up to a half of the compression limit and were just checked with the limit. About 5000 pebbles have been manufactured by this technology.

3. Tritium and helium extraction during in-pile reactor irradiation

In the frame of the research program three assemblies were irradiated in a water-graphite reactor [1]. The reactor was assembled with the three described lithium ceramics. The first assembly was tested at the adjustment stage of the pellet fabrication technology and measurement methods. The second and third assemblies were irradiated to a fluence of $\simeq 4.7 \times 10^{20}$ and $\simeq 1.35 \times 10^{21}$ m/cm^2. There are some distinctions for the pellet properties (σ_{compr}) and the size in the second and third assemblies: the pellet height was 1.5 and 1.0 cm consequently. Results of the tritium balance have been discussed in [2,3].

The kinetic parameters of tritium and helium extraction from irradiated pellets are shown in Fig. 3. The helium release rate under reactor irradiation is much higher for Li_4SiO_4 and Li_2SiO_3 and for $LiAlO_2$ in the

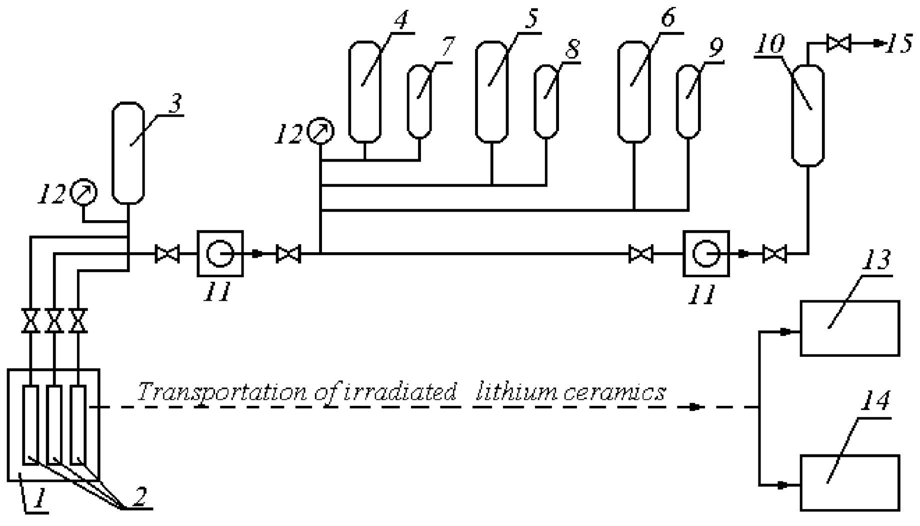

Fig. 1. Flowsheet of gas extraction from irradiated lithium ceramics: 1 – nuclear reactor, 2 – lithium ceramics elements, 3 – neon cylinder, 4,5,6 – tanks for tritium gas collection, 7,8,9 – samples, 10 – pressure gauge, 11 – vacuum pump, 12 – tank for tritium contained gas, 13 – laboratory system for investigation of pebble strength properties, 14 – laboratory system for gas extraction from irradiated pellets, 15 – outlet for gas utilisation.

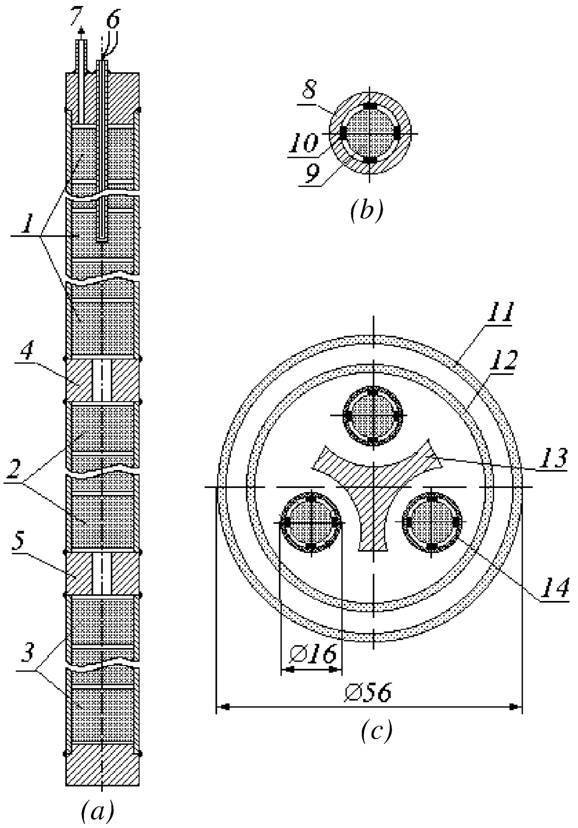

Fig. 2. Unit for reactor irradiation of ceramic pellets: (a) – ampoule, (b) – cross-section of the ampoule, (c) – cross-section of assembly, 1,2,3 – ceramic pellets, 4,5 – dividing disks, 6 – thermocouple, 7 – gas pipe, 8 – ampoule shell, 9 – ceramic pellet, 10 – spacer strip, 11,12 – inside and outside pipes of the assembly, 13 – load carrying support, 14 – ampoule in the assembly.

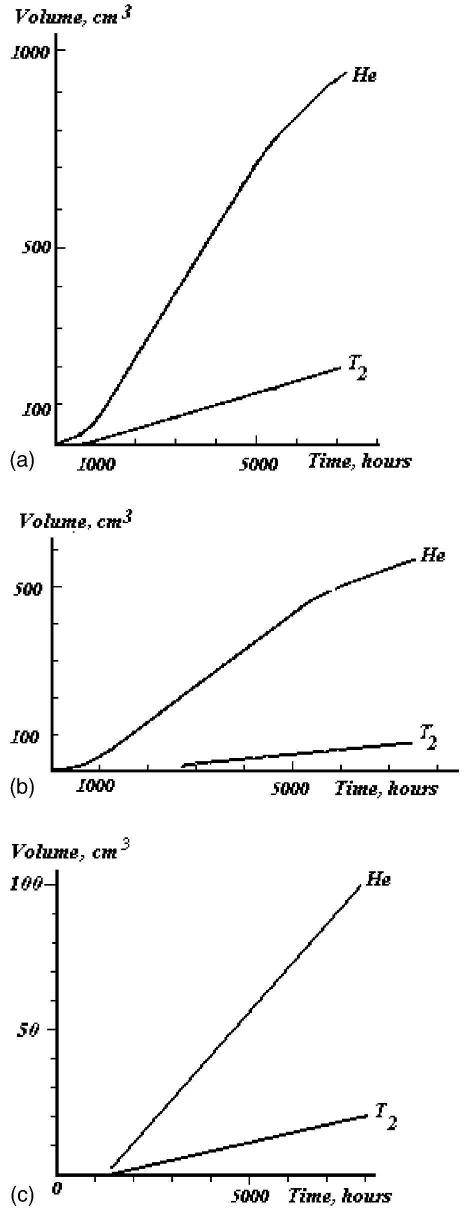

Fig. 3. The kinetic of tritium and helium-4 extraction under irradiation of lithium orthosilicate (a), metasilicate (b) and aluminate (c).

manner indicated in the figures. The values are correlated with the rates of tritium release.

4. Change of strength properties of the ceramic pellets after irradiation

The results of the experiments are referred to the two assemblies irradiated during 0.5 and 1.0 years, and the fluences have been mentioned above.

The pellets were removed after reactor irradiation from the pipes and some of them preserved the geometric shape without cracks and spallings, and were tested to investigate the change of the compression strength. The results are shown in Table 1. The compression strength has been not changed for $LiAlO_2$ and decreased for Li_2SiO_3 (both fluences) and Li_4SiO_4 (just for fluence of 1.3×10^{21} n/cm^2).

The results are correlated to the helium release behavior, an increase of the helium releases rate correlates with a decrease of σ_{compr} for the lithium silicate. An initial presumption can be done for an explanation of the results. By taking into account that the melting temperature of a $LiAlO_2$ is 1.3 times as high as those of the silicates and the helium retention in the test is higher for $LiAlO_2$, then the silicate pellets reasonably guess to

Table 1
Compression strength of lithium acuminate, orthosilicate and metasilicate pellets depending on neutron fluence

Materials	The second assembly			The third assembly		
	$LiAlO_2$	Li_4SiO_4	Li_2SiO_3	$LiAlO_2$	Li_4SiO_4	Li_2SiO_3
Neutron fluence (cm^{-2})	4.8×10^{20}	4.4×10^{20}	4.6×10^{20}	1.4×10^{21}	1.3×10^{21}	1.35×10^{21}
Initial temperature (°C)	280	300	390	185	250	220
Final temperature (°C)	235	250	330	125	195	150
Li-6 burn-up (%)	31.3	30.6	30.1	69.3	69.3	69.5
Li burn-up (%)	1.75	1.56	1.44	2.61	2.49	2.50
Initial compression strength (MPa)	79	16	140	56	71	102
Compression strength after irradiation (MPa)	79	16	87	56	31	60

be more radiation resistant and consequently σ_{compr} of the aluminate pellets may be lower.

The statement for the service life is that some lithium metasilicate, orthosilicate and aluminate pellets can be used as breeder material in a reactor blanket TBZ for 2.5% lithium burn-up and this is just a lower limit.

5. Conclusion

1. A technology of the pellet fabrication of lithium metasilicate, orthosilicate and aluminate with reproducible properties permits the use of the compression limit (σ_{compr}) as a parameter characterizing an appropriation of the properties for a reactor test and describing an irradiation influence on the service life of the pellets.
2. The reactor complex and the laboratory installation enable the investigation of the kinematics of the helium and tritium extraction and make a tritium balance.
3. The reactor experiments have demonstrated a higher radiation resistance of $LiAlO_2$ and the possibility of $LiAlO_2$ and Li_4SiO_4 application in some blankets of controlled reactors up to 2.5% lithium burn-up.

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