CURRENT EXPERIMENTAL ACTIVITIES FOR SOLID BREEDER DEVELOPMENT

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I. INTRODUCTION

Tritium containing ceramics are among the principal materials being considered for tritium production in future fusion reactors. To ensure a data base adequate for optimization of solid breeder materials, ongoing experimental studies are focused on resolving critical issues related to thermochemical and mechanical behavior, to tritium transport and release, and to material response to a neutron environment.

Thermochemical data provide information on tritium solubility and adsorption-desorption characteristics from the ceramic surface. Low solubility is a prerequisite to ensuring low tritium inventory. Mechanical properties measurements will focus on Young's modulus and elastic stress strain relationships. Neutron irradiation of candidate materials in closed capsules is an appropriate life-test methodology. Both mixed spectrum and hard spectrum irradiation facilities can be used advantageously. Purge flow in-situ tritium recovery experiments can be used to test tritium recovery performance of candidate materials. Along with diffusivity data, this information is critical to developing a firm understanding of the ease of tritium recovery. In this paper we will discuss current laboratory and irradiation experiments for evaluation of candidate breeder materials.

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II. THERMODYNAMIC AND DEFECT PROPERTIES

Tetenbaum and Johnson (1) have reported the solubility of water in Li_2O at temperatures of 973-1273 K. A step-wise change of sample temperature caused absorption or release of water from the sample. Analytically the moisture level in the exiting gas stream was monitored by an electrolytic hygrometer. When plotted the experimental data showed a vapor pressure dependence of ~ 0.5 implying a dissociative solution mechanism. Norman and Hightower (2) also performed similar measurements using a thermobalance technique. The weight change of a sample was measured when exposed to water vapor of a given partial pressure at a fixed temperature. The solubility under a given partial pressure increased with increasing temperature. The half-power dependence on H_2O partial pressure also held. These data were used to calculate the tritium inventory for a ceramic blanket (see Table 1).

Katsuta et al. (3) have measured solubilities of H_2 and D_2 in Li_2O by means of the hot extraction method. The amount of gas released from the sample was measured by a capacitance manometer. The solubility of hydrogen in Li_2O obeyed Sievert's law. The enthalpies of solution of H_2 and D_2 were 32 kJ/mol and 38 kJ/mol respectively. Kudo et al. (4) determined the H_2 (T_2) solubility in Li_2O by equilibrating Li_2O with a HT-H_2 gas mixture and then quenching. The sample was dissolved into water and the amount of tritium measured by a liquid scintillator. Between 1 and 67 kPa dissolved tritium increased as the half power of the total hydrogen pressure. The enthalpy of solution of H_2 was derived to be 48.6 ± 1.8 kJ/mol.

Yoshida et al. (5) have reported adsorption isotherms of water on Li_2O ($0.57 \text{ m}^2/\text{g}$) in the temperature range of 673 to 873 K. Their results favor dissociative adsorption. The isosteric heat of adsorption was found to be constant at a value of 21 kcal/mol. The adsorption contribution to inventory

was found to be very small except at lower temperatures ($\sim 400^{\circ}\text{C}$) where T_2O adsorption could add significantly (10 1/2 of kg to the inventory).

Comparing each contribution it can be seen that the inventory due to T_2O adsorption on Li_2O is most dominant. It has become clear that the contribution of T_2O adsorption is not negligible and can be one of the primary factors that determine a minimum operating temperature for each candidate material.

Defect F^+ centers (an oxygen ion vacancy trapping an electron have been observed by electron spin resonance (ESR) and optical absorption measurements (6-10). Figure 1 shows the schematic illustration of the F^+ center. The concentration of F^+ centers increased with fluence and the concentration of about 10^{24} and 10^{25} centers/ m^2 were attained for a thermal fluence of 10^{23} neutrons/ m^2 and at an oxygen ion fluence of 10^{20} ions/ m^2 , respectively (9). In addition to the F^+ centers, defects associated with F^+ aggregate centers and colloidal lithium metal centers were observable at higher fluences (6-7). The colloidal lithium metal centers were assumed to be preferentially precipitated at grain boundaries in case of Li_2O sintered pellets.

Figure 2 shows typical behavior of F^+ centers observed with ESR or optical absorption methods during isochronal annealing for Li_2O related to various fluences of thermal neutrons or oxygen ions (11). The F^+ centers began to decrease at about 520 K and almost disappeared at about 670 K for the specimens irradiated to 7×10^{18} and 3×10^{19} ions/ m^2 by 100 MeV oxygen ions. In the specimens irradiated to 4×10^{20} and 2.2×10^{21} thermal neutrons/ m^2 , the F^+ centers began to decrease at about 420 K and 470 K and then vanished at about 520 K and 620 K, respectively. The concentration of the F^+ centers in the specimens irradiated to 7×10^{18} and 3×10^{19} ions/ m^2 by oxygen ions can be

estimated to be higher than that in the specimens irradiated to 4×10^{20} and 2.2×10^{21} thermal neutrons/m² (9). Thus, it is seen that the recovery of the F^+ centers in high concentration is delayed in comparison with that of lower concentration. The activation energy for recovery of the F^+ centers was determined to be 135 kJ/mol in the isothermal annealing experiments (8,10).

The colloidal lithium metal centers are more stable than the F^+ centers. In the isochronal annealing experiments, the colloidal lithium metal centers increased in the temperature range from 400 K to 650 K, then began to decrease at about 700 K and to disappear above 870 K (7).

In Li_2O , it is the migration of lithium ions, rather than oxygen ions that contributes to the ion conductivity (12). The diffusivity of lithium ions can be calculated from the ion conductivity by Nernst-Einstein equation. Comparing the diffusivity of lithium ions calculated from the ion conductivity of non-irradiated Li_2O with the diffusivity of tritium evaluated from tritium release measurements it was considered that there was a close relationship between lithium ion diffusion and tritium ion diffusion (13). This hypothesis was supported in a nuclear magnetic resonance (NMR) study of behavior of lithium nuclei in Li_2O (14). It was assumed from the result of ion conductivity measurements for Li_2O irradiated with oxygen ions, that the diffusivity of lithium ions (and tritium ions) in Li_2O decreases in the range from about 489 K to the recovery temperatures of the F^+ centers, and increases at about 443 K by unspecified defects. This assumption is in good agreement with the results of tritium diffusivity measurements for Li_2O single crystals irradiated with thermal neutrons (15).

III. TRANSPORT AND MECHANICAL PROPERTIES

Thermal Conductivity

Thermal conductivity has been determined from thermal diffusivity and specific heat measurements and has been found to increase with material

density, and decrease with temperature. From a limited data base it appears that neutron irradiation does not significantly change the conductivity. Takahashi (16) investigated Li_2O and Li_2ZrO_3 as a function of density and temperature. For LiAlO_2 , Li_2SiO_3 , Li_4SiO_4 , sets of discrete values have been reported by Hollenberg (17) Schulz (18) Rasneur (19) Tam (20) and Kennedy (21). Dienst (22) has presented a comparison of thermal conductivity of all candidate materials for 80% T.D. (see Fig. 3). F. Botter (23), reported that LiAlO_2 samples showed no measurable change of thermal conductivity after irradiation in OSIRIS to 1% Li burn-up at 600°C, and 2% Li burn-up at 850°C.

Thermal Expansion

The current data base, originating from studies by Hollenberg (17), Kurasawa (24), Kennedy (21), Skokan (25) is reasonably complete. A compilation of data of all breeder candidates for 80% T.D. is made by Dienst (22). Figure 4, (26) compares the thermal expansion for several breeder materials and two structural materials. A difference in thermal expansion of breeder and structure generates breeder/structure mechanical interaction, the severity of which will depend on breeder plasticity and structure deformation.

Elastic Moduli

Elastic moduli characterize the elastic behavior of materials under external loads. Only discrete values for the various breeders are available from the works of several authors. Billone (27) calculated Young's modulus, shear modulus and Poisson's ratio for 92% T.D. Li_2O at room temperature from sound velocity measurements and gave formulas for extrapolation to higher temperatures and different porosities. Room temperature measurements of Young's modulus and Poisson's ratio were reported by Kennedy (21,28) for 90% and 80% T.D. Li_2O and 80% T.D. metazirconate and 80% T.D. lithium aluminate

and metasilicate. Room temperature values are reported by Dienst (22) for lithium metasilicate, by Rasneur (19) and Alvani (29) for lithium aluminate. The variation of Young's modulus with temperature was investigated by Rasneur (19) who found a 6 to 8% decrease between room temperature and 700°C. No significant change of Young's modulus of LiAlO_2 specimens was observed after irradiation to 1% burn-up at 400°C; a slight decrease was measured at 600°C (27). An increase in Young's modulus, correlated to specimen shrinkage was observed in irradiations up to 850°C and 2% burn-up (28).

Compressive Strength

Room temperature compressive strength data were reported by Rasneur (19) for LiAlO_2 of several densities and grain diameters, by Kennedy (24) for LiAlO_2 , Li_2SiO_3 , Li_2O , Li_2ZrO_3 , by Dienst (22) for Li_2SiO_3 and Li_4SiO_4 , and by Barra Caracciolo (30) (see Table 2) for LiAlO_2 .

B. Rasneur (19) investigated the variation of compressive strength of 75% T.D., 0.4 μ grain size LiAlO_2 with temperature and found 275 MPa, 153 MPa, 115 MPa, 100 MPa, 78 MPa at room temperature, 500°, 700°, 900°, and 1100°C respectively. Changes due to irradiation were investigated after the ALICE 1 and 2 experiments (23, 31); within the conditions explored, no significant changes were measured as compared to unirradiated values for LiAlO_2 .

Bend Strength

Bend strength of LiAlO_2 , Li_2SiO_3 , Li_2O , Li_2ZrO_3 , 80% and 90% T.D. have been measured at 20°, 400°, and 600°C (21). Except for Li_2O which exhibits a strength reduction at 400°C, no significant change was observed for the others up to 600°C. Bend strength measurements of LiAlO_2 of several densities produced by different fabrication routes are reported by Barra Caracciolo (30). Results, elaborated from Weibull statistics to account for statistical variation and stress volume dependence, are listed in Table 2 along with

compressive strength (Weibull modulus increases with decreasing variability in strength).

Creep Behavior

Plastic deformation of Li_2O single crystals was studied by constant strain rate and constant load compressive tests by Noda (32). The constant strain rate test showed an increasing plasticity with temperature above 623 K but almost no plastic deformation below 600 K. Hollenberg (33) conducted dilational creep experiments on porous Li_2O specimens in the range 500°C to 700°C. At 700°C and at a high strain rate i.e., $2 \cdot 10^{-3}$ in/min, stress levels remain below 34 MPa at 85% T.D. Such low stresses entertain the hope that Li_2O densification may accommodate swelling. Creep data from scoping tests at 700°C, 800°C, and stresses in the range 15-45 MPa also indicate that Li_2O is relatively soft. Those results enable one to anticipate that creep and hot pressing would relax Li_2O mechanical interaction stresses to tolerable design values.

Thermal and mechanical data were used (22) to calculate a reduced thermal shock parameter ($K/E \alpha$ instead of $\sigma_f k(1-\nu)/E\alpha$) for 80% T.D. materials (with k = thermal conductivity at 600°C, α linear thermal expansion at 600°C, and E = Young's modulus at 25°C). The following values were obtained, $2.7 \cdot 10^{-6} \text{ m}^2/\text{s}$ for LiAlO_2 , $1.8 \cdot 10^{-6} \text{ m}^2/\text{s}$ for Li_2ZrO_3 , $1.7 \cdot 10^{-6} \text{ m}^2/\text{s}$ for Li_2SiO_3 , $1.5 \cdot 10^{-6} \text{ m}^2/\text{s}$ for Li_2O , $1.1 \cdot 10^{-6} \text{ m}^2/\text{s}$ for Li_4SiO_4 . Thus, LiAlO_2 has the best crack resistance and Li_4SiO_4 the poorest.

IV. IRRADIATION BEHAVIOR

Transformation of materials during neutron irradiation must be considered because the neutron capture process eliminates one chemical component (Li) and produces two others (T, He). These compositional changes can lead to the formation of second phases either as internal precipitates or as solid

solutions. The impact of a change in composition for a ternary oxide is likely to be reflected in changes in its properties, e.g., melting point, tritium diffusivity, thermomechanical performance. Further, in any detailed evaluation of candidate blanket materials, strong consideration must be given to their tritium breeding potential. Currently, blanket designs use beryllium for neutron multiplication to ensure an adequate tritium breeding ratio (tritium atoms produced per fusion neutron).

Recent FUBR 1A irradiation tests indicate that tritium retention and pellet integrity are reasonably good (34-36). Analysis of helium and tritium retention showed that Li_2O exhibits the greatest helium retention, especially at the intermediate temperature (700°C), while tritium retention was lowest for Li_2ZrO_3 . All four materials show an approximately linear proportionality for tritium and helium retention with burn-up. In the ALICE experiments (37) tritium retention was strongly dependent on the nature of the structural material surrounding the sample during irradiation. Retention was much lower when surrounded by stainless steel rather than an aluminum-magnesium alloy.

The materials from FUBR 1A showed increase in fragmentation in the order Li_2ZrO_3 , LiAlO_2 , Li_2O , and Li_4SiO_4 . LiAlO_2 at 95% T.D., was less fragmented than this same material at 85% T.D. Fragmentation was not burn-up dependent and was thought to result from thermal stresses arising from thermal gradients and thermal expansion. In the ALICE experiments, no significant fragmentation of LiAlO_2 was observed up to 600°C and 1 at.% burnup; severe fragmentation was observed at 750°C with thermal gradients of about 80 K/mm (38).

In the FUBR 1A experiment, swelling at 700° and 900°C was observed to be very low for 85% T.D. LiAlO_2 and Li_2ZrO_3 , slightly higher for Li_4SiO_4 , and rather high for Li_2O . Swelling was thought to arise from the retained helium in the material. A small shrinkage was observed ($\leq 0.03\%$) for 75% T.D. LiAlO_2

in the ALICE1 experiment (38). Significant changes were noticed in grain growth for Li_2O , a slight growth for Li_4SiO_4 , and no change for LiAlO_2 and Li_2ZrO_3 (36).

The thermal conductivity of irradiated Li_2O and LiAlO_2 has been investigated in the range 373°K - 1173°K by Ethridge (39). Samples were irradiated at 773-1173°K to lithium burn-ups $\sim 11.5 \times 10^{20}$ captures/cm³. In general, the measured conductivity of the irradiated materials was quite similar to the unirradiated values. Reductions in thermal conductivity at low temperatures were observed for lithium oxide samples as a result of radiation-induced lattice damage whereas at higher temperatures the thermal conductivities approached values well within the error band of unirradiated lithium oxide. The thermal conductivity of irradiated lithium aluminate samples remained relatively constant with temperature for all irradiation times and temperatures and was only slightly lower than that of unirradiated LiAlO_2 at high temperatures (see Fig. 5).

V. TRITIUM TRANSPORT AND RELEASE

In-situ tritium recovery experiments provide a method of evaluating the feasibility of the lithium ceramics as tritium breeders. Many experiments (40-50) have been performed to determine the tritium release rate. For some materials the rates have shown significant variance. Further, differences in interpretation concerning the controlling mechanism for tritium release appear throughout the literature generally because of poor characterization of system variables e.g., grain size, extraction temperature, material impurity content, carrier gas composition during extraction.

Laboratory and in-situ tritium recovery experiments have focused on the release process with the objective of characterizing the rate-limiting phenomena. In-situ tritium recovery experiments have been conducted on Li_2O

(41-47) on LiAlO_2 (40, 45-52) and on $\text{Li}_2\text{SiO}_3/\text{Li}_4\text{SiO}_4$ (47, 50-54). A striking feature of these in-situ release experiments is the ease with which tritium is released from the solid. Characteristically, tritium is observed in the purge gas stream within hours after irradiation was initiated. Further, residual tritium content in the solid has been found to be very low, indicative of low tritium solubility, and therefore, a potentially low inventory. Studies continue in order to rigorously define release parameters.

For the TRIO experiment (40), tritium recovery was readily achieved to temperatures as low as 770 K. In this experiment, it was found that the addition of H_2 to the purge gas stream significantly enhanced tritium release. This implies that surface processes play a significant role in the overall release process. Throughout the course of the TRIO experiment, the predominant species evolved from the test capsule was the noncondensable tritium form (HT). In sealed capsule tests, where tritium is evolved by post-irradiation annealing, the HT fraction is generally <10% of the total. Recent in-situ tritium recovery tests on Li_2O [VOM (52), CRITIC (55)] also showed a very high fraction (~90%) of the condensable tritium species (HTO , T_2O). The LILA-1 experiment (48) in which LiAlO_2 was placed in quartz and stainless steel capsules showed that material in the quartz capsule released tritium predominantly in the condensable form, while material in the steel capsule released it in the noncondensable form. These data suggest the potential for immediate reduction of the released material on structural members.

Tritium diffusion data on various lithium ceramic breeder candidates have been tabulated from both laboratory and in-reactor release-type experiments (58). Much of the release data pertain to Li_2O (59-63). There is considerable spread in the apparent activation energy for tritium diffusion in lithium ceramics. The reason for this is not understood at present. Possible

causes may include a) different experimental conditions (e.g., wet or dry), b) different sample characteristics (e.g., single crystal versus polycrystal, pellets, powder forms), and c) effects of other mechanisms besides bulk diffusion contributing to the tritium release process. There is also a wide spread in the values of the derived diffusion coefficient D_T . For example, for studies on Li_2O , D_T (at 600°C) can range between a low value of $5 \times 10^{-11} \text{ cm}^2/\text{sec}$ (64) to a high value of $3.5 \times 10^{-8} \text{ cm}^2/\text{sec}$ (63); but, as discussed before, this could be attributed partly to uncertainty in the sample characteristics.

A computer program is being developed (65) to describe tritium behavior in ceramic breeder materials. As presently constructed, the program focuses on modeling diffusion in the solid and desorption from the surface. The program is based on the solution (66) of the differential equation

$$\frac{1}{D} \frac{\partial C}{\partial t} = \frac{1}{r^2} \frac{\partial}{\partial r} \left(r^2 \frac{\partial C}{\partial r} \right) + \frac{G}{D}$$

where D is the diffusivity, C is the tritium concentration, r is the grain radius, G is the tritium generation rate. The equation for tritium release R_t is given by

$$R_t = 1 - \frac{3K_d}{Ga} C_1 - \frac{Ga}{3} - 2h^2 a G \sum_{n=1}^{\infty} \frac{\exp(-Da_n^2 t)}{a_n^2 [a^2 a_n^2 + ah(ah-1)]} + K_d C_1$$

where

- a = grain radius
- K_d = effective first-order desorption rate constant
- C_1 = tritium concentration evaluated at $r=a$ prior to a temperature change
- h = K_d/D
- t = time

In this equation, a_n represents the roots of $aa \cot aa = 1-ah$. The release rate was calculated using the diffusion-desorption model, and the calculated

results were compared to the observed tritium release rate for a sample of Li_2SiO_3 from the LISA-1 experiment (50). A typical plot is shown in Fig. 6. The model predicts the initial rise in release rate and the release rate after 100,000 seconds quite well but does a poorer job of predicting the release rate between 10,000 and 75,000 seconds. The calculated tritium release profile is also in good agreement with the observed data for time periods following a large temperature increase or decrease. One shortcoming of the current model is that it uses a characteristic radius of the sample to calculate the release rate. The samples used in the LISA-1 experiments have a large distribution of grain sizes, and it is difficult to assign a characteristic grain size to the sample. Since the tritium release is dependent on the grain size, this could have a substantial effect on the release profile. To correct this shortcoming the model is being altered to allow for a distribution of grain sizes as input.

VI. CONCLUSIONS

The current data base for ceramic breeder materials does not exhibit any negative features as regards to thermophysical, mechanical, and irradiation behavior. All candidate materials show excellent stability for irradiation testing to 3% burnup. In-situ tritium recovery tests show very low tritium inventories for all candidates. Theoretical models are being developed to accurately predict real time release rates. Fabrication of kilogram quantities of materials has been achieved and technology is available for further scale-up.

REFERENCES

1. M. Tetenbaum, A. K. Fischer, and C. E. Johnson, "An Investigation of the Solubility of LiOH in Li₂O," *Fusion Technol.*, 7, 53 (1985).
2. J. H. Norman and G. R. Hightower, "Measurements of the Activity Coefficient of LiOH Dissolved in Li₂O(s) for Evaluation of Li₂O as a Tritium Breeder Material," *J. Nucl. Mater.*, 122&123, 913 (1984).
3. H. Katsuta, S. Konishi, and H. Yoshida, "Solubility and Diffusivity of Hydrogen in Li₂O," *J. Nucl. Mater.*, 116, 244 (1983).
4. H. Kudo, K. Okuno, and S. O'Hira, "Tritium Release Behavior of Ceramic Breeder Candidates for Fusion Reactors," *J. Nucl. Mater.*, (in press).
5. H. Yoshida, S. Konishi, H. Takeshita, T. Kurasawa, H. Watanabe, and Y. Naruse, "J. Nucl. Mater.", 122&123, 934 (1984).
6. K. Uchida, K. Noda, T. Tanifuji, S. Nasu, T. Kirihaara and A. Kikuchi, Optical Absorption Spectra of Neutron-Irradiated Li₂O, *Phys. Stat. Sol. (a)* 58 (1980) 557-566.
7. K. Noda, K. Uchida, T. Tanifuji and S. Nasu, Radiation Damage in Sintered Li₂O Pellets-ESR Study, *J. Nucl. Mater.* 91 (1980) 234-236.
8. K. Noda, K. Uchida, T. Tanifuji and S. Nasu, A Study of Radiation Damage in Li₂O by Means of Electron-Spin Resonance, *Phys. Rev. B* 24 (1981) 3736-3742.
9. K. Noda, T. Tanifuji, Y. Ishii, H. Matsui, M. Masaki, S. Nasu and H. Watanabe, Irradiation Effects on Lithium Oxide, *J. Nucl. Mater.* 122-123 (1984) 908-912.
10. K. Noda, Y. Ishii, H. Matsui and H. Watanabe, Irradiation Damage in Lithium Oxide, *J. Nucl. Mater.* 133-134 (1985) 205-208.
11. K. Noda, Y. Ishii, H. Matsui and H. Watanabe, Irradiation Damage in Lithium Oxide, *Radiat. Eff.* 97 (1986) 297-305.
12. H. Akiyama, K. Ando, and Y. Oishi, Self-Diffusion of Constituent Ions in Antifluorite-Cubic Lithium Oxide, *Solid State Ionics*, 3-4 (1981) 469-472.
13. H. Ohno, S. Konishi, T. Nagasaki, T. Kurasawa, H. Katsuta, and H. Watanabe, Correlation Behavior of Lithium and Tritium in Some Solid Breeder Materials, *J. Nucl. Mater.* 133-134 (1985) 181-185.
14. T. Matsuo, H. Ohno, K. Noda, S. Konishi, H. Yoshida, and H. Watanabe, Nuclear Magnetic Resonance Investigations of Lithium Diffusion in Li₂O, Li₂SiO₃ and LiAlO₂, *J. Chem. Soc. Faraday Trans. 2* (1983) 1205-1216.

15. T. Tanifuji, K. Noda, T. Takahashi, and H. Watanabe, Tritium Release from Neutron-Irradiated Li_2O : Diffusion in Single Crystal, J. Nucl. Mater. 149 (1987) 227-232.
16. T. Takahashi and T. Kikuchi, Porosity Dependence on Thermal Diffusivity and Thermal Conductivity of Lithium Oxide Li_2O from 200°C to 900°C, J. Nucl. Mater., 120 (1984) 213.
17. G. W. Hollenberg, Thermal Properties of Lithium Ceramics for Fusion Applications, presented at the 84th Annual Meeting of the Am. Cer. Soc. Cincinnati, May 3-5, 1982.
18. B. Schultz and H. Wedemeyer, Preparation, Characterization and Thermal Diffusivity of LiAlO_2 , J. Nucl. Mater. 139 (1986) 35.
19. B. Rasneur, Tritium Breeding Material: LiAlO_2 , Fusion Technology, Vol 8, July 1985.
20. S. W. Tam, L. Leibowitz, Y. Y. Liu, R. Blomquist, and C. E. Johnson, Saturation Phenomena and Percolative Transitions in the High Temperature Thermal Conductivity of LiAlO_2 , J. Nucl. Mater. 133 and 134 (1985) 234.
21. P. Kennedy, K. E. Gilchrist, D. E. Walker, and S. Broughton, The Preparation, Characterization and Properties of Lithium Oxide and Lithium Meta-Zirconate Specimens Irradiated in HFR Petten, in the Second and Third Exotic Experiments, Fusion Technology 2 (1986) 1013.
22. W. Dienst and H. Zimmermann, Investigation on Mechanical Properties of Ceramic Breeder Materials, presented at ICFRM 3 to be published in J. Nucl. Mat. 18.
23. F. Botter, F. Lefevre, B. Rasneur, M. Troabas, and E. Roth, Effects of Radiation on Lithium Aluminate Samples Properties, J. of Nucl. Mater. 141-143 (1986) 364.
24. T. Kurasawa, T. Takahashi, K. Noda, H. Kakeshita, S. Nasu, and H. Watanabe, Thermal Expansion of Lithium Oxide, J. Nucl. Mater 107 (1982) 334.
25. A. Skokan, H. Wedemeyer, D. Vollath, and E. Gunther, Thermal Properties and Application of Potential Lithium Silicate Breeder Materials, Fusion Technology, 1986, Vol. 2 p. 1255.
26. G. Hollenberg, private communication.
27. M. C. Billone, Y. Y. Liu, R. B. Poeppel, J. L. Routbort, K. C. Goretta, and D. S. Kupperman, Elastic and Creep Properties of Li_2O , J. Nucl. Mat. 141-143 (1986) 282.
28. P. Kennedy, private communication, results to be published.
29. C. Alvani, S. Casadio, C. Ciavola, F. Croce, and V. Violante, Physico-Chemical Properties of LiAlO_2 for a Ceramic Type Tritium Breeding Blanket. Proc. VI CIMTEC Milano 1986.

30. M. Barra Caracciolo, C. Alvani, and S. Casadio, Mechanical Properties of Ceramic Breeder Pellets, presented at ICFRM 3, to be published in J. Nucl. Mat.
31. F. Botter, B. Rasneur, and E. Roth, Irradiation Behavior of LiAlO_2 in Alice I and Alice 2 Experiments, presented at ICFRM 3, to be published in J. Nucl. Mat.
32. K. Noda, M. Arita, Y. Ishi, H. Saka, T. Imura, K. Kuroda, and H. Watanabe, Mechanical Properties of Lithium Oxide at High Temperatures, J. Nucl. Mat. 141-143 (1986) 353.
33. G. W. Hollenberg, Y. Y. Liu, and B. Arthur, Creep of Li_2O , J. Nucl. Mater. 133-134 (1985) 216.
34. G. W. Hollenberg, HEDL 7643 (1987).
35. G. W. Hollenberg, Proc. 89th Ann. Meet. Amer. Ceramic Soc., Pittsburgh, PA, USA, April 1987, Advances in Ceramics 1988 (in print).
36. G. W. Hollenberg and D. L. Baldwin, J. Nucl. Mater. 133&134 (1985) 242).
37. J. P. Bonal, Memoire du Conservatoire National des Arts et Metiers (1987).
38. B. Rasneur, F. Botter, and M. Briec, CEA Note 2536 (1987).
39. J. L. Ethridge and D. E. Baker, Effects of Fast Neutron Irradiation on Thermal Conductivity of Li_2O and LiAlO_2 . Pittsburg Conference, April 1987.
40. R. G. Clemmer, P. A. Finn, B. Misra, M. C. Billone, A. K. Fischer, S. W. Tam, C. E. Johnson, and A. E. Scandora, J. Nucl. Mater. 133&134 (1985) 171.
41. T. Kurasawa, H. Takeshita, W. Watanabe, H. Yoshida, Y. Naruse, T. Miyauchi, K. Mimura, and H. Umei, J. Nucl. Mater. 122&123 (1984) 902.
42. T. Kurasawa, H. Yoshida, H. Takeshita, W. Watanabe, T. Miyauchi, T. Matui, H. Umei, and Y. Naruse, J. Nucl. Mater. 133&134 (1985) 1986.
43. R. A. Verrall, J. M. Miller, and S. R. Bokwa, Proc. 89th Ann. Meet. Amer. Ceramic Soc., Pittsburgh, PA., USA, April 1987, Advances in Ceramics 1988 (in print).
44. S. Tanaka, T. Teraï, H. Mohri, and Y. Takahashi, J. Nucl. Mater. (This conference, paper C-20).
45. H. Kwast, R. Conrad, A. J. Flipot, P. Kennedy, and J. D. Elen. (This conference, paper C-25).
46. T. Kurasawa, H. Watanabe, G. W. Hollenberg, Y. Ishii, A. Nishimura, H. Yoshida, Y. Naruse, M. Aizawa, H. Ohno, and S. Konishi, J. Nucl. Mater. 141-143 (1986) 265.

47. H. Kwast, R. Conrad, P. Kennedy, and A. J. Flipot, J. D. Elen, J. Nucl. Mater. 141-143 (1986) 300.
48. E. Roth, J. J. Abassin, F. Botter, M. Briec, P. Chenebault, M. Masson, B. Rasneur, and N. Roux, J. Nucl. Mater. 133&134 (1985) 238.
49. M. Briec, F. Botter, J. J. Abassin, R. Benoit, P. Chenebault, M. Masson, B. Rasneur, P. Sciars, H. Werle, and E. Roth, J. Nucl. Mater. 141-143 (1986) 357.
50. H. Werle, J. J. Abassin, M. Briec, R. G. Clemmer, H. Elbel, H. E. Hafner, M. Masson, P. Sciars, and H. Wedemeyer, J. Nucl. Mater. 141-143 (1986) 321.
51. M. Briec, J. J. Abassin, M. Masson, E. Roth, P. Sciars, and H. Werle, J. Nucl. Mater. 154-156 (1988) 397.
52. T. Kurasawa, H. Watanabe, E. Roth, and D. Vollath (this conference, paper C-22).
53. R. G. Clemmer, H. Werle, and M. Briec, Proc. Int. Symp. on Fusion Reactor Blanket and Fuel Cycle Technology, Tokai-mura, Japan, Oct. 1986, 51.
54. H. Werle, M. Briec, R. G. Clemmer, H. Elbel, H. E. Hafner, M. Masson, G. Schumacher, and H. Wedemeyer, J. Nucl. Mater. (this conference, paper C-21).
55. R. A. Verrall, J. M. Miller, S. R. Bokwa, and C. E. Johnson, J. Nucl. Mater. (this conference, paper C-24).
56. T. Kurasawa, G. W. Hollenberg, and H. Watanabe, Proc. Int. Symp. on Fusion Reactor Blanket and Fuel Cycle Technology, Tokai-mura, Japan, Oct. 1986, 43.
57. H. Watanabe, T. Kurasawa, E. Roth, and D. Vollath, Ibid., 33.
58. J. Shearer, S. W. Tam, and C. E. Johnson, Trans. Amer. Nucl. Soc. 44 (1983) 139.
59. H. Kudo and K. Okuno, J. Nucl. Mater. 101 (1981) 38.
60. K. Okuno and H. Kudo, Proc. Fall Meeting of Atomic Energy Soc. of Japan, F-17, Kyushu, Japan, Oct. 1981.
61. S. Nasu, H. Kudo, K. Shiozawa, T. Takahashi, T. Kurasawa, M. Tachiki, and K. Tanaka, J. Nucl. Mater. 68 (1977) 261.
62. V. G. Vasiliev, S. R. Borisov, N. N. Riazantseva, and A. A. Vashman, US/USSR Workshop on Engineering and Economic Problems of ETF, Moscow and Leningrad, USSR, September 1979.
63. K. Okula and D. K. Sze, Proc. ANS Meeting on Tritium Technology in Fission, Fusion and Isotopic Applications, April 1980, Dayton, OH, USA, 286.

64. D. Guggi, H. R. Ihle, D. Bruning, U. Kurz, S. Nasau, K. Noda, and T. Tanifuji, J. Nucl. Mater. 18 (1983) 100.
65. J. P. Kopasz, S. W. Tam, and C. E. Johnson, J. Nucl. Mater. 154-156 (1988).
66. H. S. Carslaw and J. C. Jaeger, "Conduction of Heat in Solids," Second Edition, Oxford University Press, New York 1959.

Table 1. Estimation of Tritium Inventory (in Kg-T)*1

Temperature (°C)	T ₂ O Solution		T ₂ Solution	
	Tetenbaum	Norman	Kudo	Katsuta
1000	1.35	1.70	[0.18]*3	[0.013]
800	0.37	1.08	[0.11]	[0.010]
600	[0.06]*2	[0.57]	0.06	0.007
500	[0.02]	[0.36]	0.04	0.005
400	[0.003]	[0.20]	0.02	[0.003]

*1 Total weight of Li₂O: 10⁵ kg
T₂O partial pressure: 5x10⁻⁵ atm

*2 Values in blackets were obtained by extrapolation.

Table 2. Results of Compressive Strength and Bending Strength for 3 Types of Fabrications of LiAlO₂ (P, A.1, A.2).

Material Porosity		Compressive Test		Bend Test	
		Mean Strength (MPa)	Weibull modulus	Mean Strength (MPa)	Weibull modulus
P	0.19	322	5.27	60	7.25
A-1	0.19	173	6.52	48	25.74
A-2	0.19	58	3.03	-	-
P	0.37	-	-	29	10.41
P	0.29	-	-	40	7.61
P	0.10	-	-	91	7.13
P	0.06	-	-	105	11.42
A-1	0.25	-	-	32	7.57
A-1	0.14	-	-	68	13.47

Fig. 1. Location of F^+ Center in Crystal Lattice.

Fig. 2. Change in F^+ Center and Ion Intensity as a Function of Temperature.

Fig. 3. Thermal Conductivity of Ceramic Breeder Materials.

Fig. 4. Thermal Expansion as a Function of Temperature for Several Ceramics and Structural Materials.

THERMAL CONDUCTIVITY OF LiAlO_2 IRRADIATED TO 297 FPD

Fig. 5. Thermal Conductivity of Irradiated LiAlO_2 .

Fig. 6. Calculated and Observed Tritium Release from Lithium Metasilicate (P1).