FISSION REACTOR EXPERIMENTS FOR SOLID BREEDER BLANKETS

P.J. GIERSZEWSKI[a] and M.A. ABDOU
University of California at Los Angeles
Los Angeles, CA 90024
(213) 206-0501

R. PUIGH
Hanford Engineering Development Laboratory
Richland, WA 99352
(509) 376-3766

ABSTRACT

The testing needs for solid breeder blanket development are different from those for liquid breeder blankets. In particular, a reasonable number of moderate volume test sites in a neutron environment are needed. Existing fission reactors are shown to be able to provide this environment with reasonable simulation of many important blanket conditions. Three major additional fission reactor tests are identified beyond those presently underway. These are thermal behavior, advanced in-situ tritium recovery and nuclear submodule experiments.

INTRODUCTION

Solid breeder blanket testing needs have been characterized in the FINESSE study.[1] These testing needs have unique characteristics, especially in comparison to liquid breeder blankets. First, there are a large number of potential breeder materials and material variables. Second, the influence of radiation on the primary uncertainties is large, but the influence of geometry is not. Finally, much of the important functional behavior of the solid breeder is not described by classical equations, but rather the controlling phenomena must be quantified by experiments. These factors suggest that a reasonable number of moderate volume test sites in a neutron environment are needed, and that the test conditions should match fusion reactor blanket conditions as closely as possible.

The test conditions or parameters that could influence the breeder behavior include the materials, tritium production, temperature profile, radiation environment, purge conditions and geometry.

Table 1 summarizes anticipated conditions in two representative solid breeder blankets and in present experiments.[2-5] These represent an unmultiplied blanket with natural enrichment Li_2O, and a multiplied blanket with beryllium and $LiAlO_2$ at 90% ^6Li enrichment. Significant differences between these examples are the peak burnup (due to the higher thermal flux in a multiplied blanket just behind the multiplier), and the differences in operating temperature (due to differences in breeder materials). The experiment examples include recent closed (FUBR-1B) and open (LISA, CRITIC) tests.

CONDITIONS IN FISSION REACTOR EXPERIMENTS

Materials

Clearly, the tests should be and can be conducted with reactor-relevant breeder and multiplier materials. In general, classes of materials are anticipated to have similar general behavior (e.g., low hydrogen solubility in the ternary ceramics). However, quantifying the behavior requires measurements with each specific material. Material variables also include the density, grain size, open/closed porosity, form, impurity content and other microstructural parameters.

Tritium Production

In-situ tritium generation through neutron reactions with lithium is necessary. Other methods of tritium production cannot provide an appropriate tritium source profile and rate. Reactor tritium production rates are desirable to reproduce the local chemistry and microstructural conditions during irradiation, and to provide the correct amount of tritium which, at equilibrium, may be small compared to impurities. Measurements with implanted or adsorbed hydrogen can best provide complementary information about specific phenomena such as surface adsorption isotherms. As Table 1 indicates, it is possible to achieve suitable tritium production rates in present fission reactors.

Temperature Profile

Uniform bulk heating at reactor levels is necessary to achieve correct temperature profiles, particularly the temperature

---

[a] Canadian Fusion Fuels Technology Project

Table 1. Representative reactor and test conditions for Li_2O and $LiAlO_2$ breeder material

	Li_2O			$LiAlO_2$		
Parameter	Reactor[2]	FUBR-1B[3]	CRITIC[4]	Reactor[2]	FUBR-1B[3]	LISA[5]
Form	Sintered	Sintered	Sintered	SpherePac	Sintered	Sintered
Density (% TD)	85	80	90	85	80	78
Grain size (x 10^{-6} m)	3	–	60	0.2	–	0.38
Metal impurities (wppm)	200	2000	–	200	2000	–
^{6}Li enrichment (%)	7.5	56	0.3	90	95	7.5
Tritium generation (T/Li-yr)	0.01	0.017	0.008	0.5	0.03	0.014
Peak burnup (% Li)	10	3.4	0.15	80	5.7	0.24
Breeder displacements (dpa/yr)	–	–	–	30	–	–
Time at temperature (yr)	4	2	0.5	4	2	0.17
Peak heating rate (MW/m^3)	50	45	50	50	28	10
Temperature (°C)	410-800	520-1000	400-950	350-1000	600-1100	450-730
Breeder thickness (cm)	1	2.4	1	0.5-5	2.3	0.8
Temperature gradient (°C/cm)	800	420	50	1300	430	under 100
Purge flow/breeder (mL/s-g)	0.03	NA	0.02	0.03	NA	0.09
Purge pressure (MPa)	0.1	0.24-1.0	under 0.4	0.6	0.23/0.4	–
Purge composition	He+0.1%H_2	He	He+1%H_2	He+0.1%H_2	He	He+0.1%H_2
Purge T content (Pa / Ci/m^3)	2 / 10	30 / 125	6 / 25	2 / 10	30 / 125	0.6 / 2.5
Clad material	HT9,PCA	Nickel	Inconel	HT9,PCA	Nickel	St. Steel
Breeder surface area (m^2/g)	1	(1)	(0.1)	1-10	(1)	(2)
Clad area/breeder surface area	10^{-4}	10^{-4}	10^{-3}	10^{-4}	10^{-4}	10^{-3}
Breeder mass (g)	10^9	34	92	10^9	46	7.7

gradients which may affect mechanical behavior and vapor phase transport. Gamma heating accounts for roughly 10% of the breeder heating in a fusion reactor, with the remainder from the neutrons and the ^{6}Li reaction. However, much useful materials data can be obtained at uniform temperatures which can be achieved at lower heating rates.

The ability of thermal reactors (ETR, ORR) and fast reactors (FFTF) to match fusion tritium and heat generation rates is shown in Figure 1 for Li_2O.[1] By appropriate matching of ^{6}Li enrichment in the breeder material with the reactor neutron energy spectrum, it is possible to simulate fusion tritium generation and heating rates within a factor of two. For multiplied blankets, the local tritium production and heating rates can also be reproduced in both fast and thermal reactors, although the match is generally better in a thermal reactor due to the characteristic thermal spectra of a multiplied blanket. Figure 1 is based on microscopic reaction cross-sections. Tests with high ^{6}Li enrichments may not be able to duplicate reactor conditions over moderate-sized (4 cm) specimens due to self-shielding of the neutrons. This is expected to be more significant for thermal reactors, where most of the heating and tritium production could occur near the pellet surface.

Practical limits in existing fission

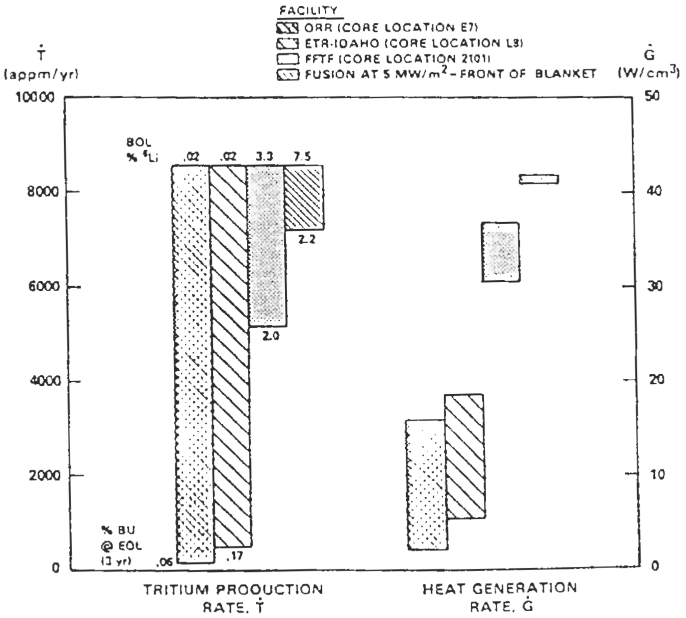

Figure 1. Beginning-of-life (BOL) and end-of-life (EOL) tritium production and heating rates in Li_2O in thermal (ETR, ORR), fast (FFTF) and fusion reactors.

reactors test sites due to core criticality or test cell diameter can constrain the thickness of the specimen and the ^6Li enrichment level. The core coolant temperature limits the minimum achievable temperature unless a separately-cooled test assembly is possible. The maximum temperature is also constrained by structural materials limits. However, Table 1 indicates that present experiments are able to explore a wide range of relevant temperatures, although the present temperature gradients are somewhat less than those possible under peak fusion conditions.

Radiation Environment

Radiation damage may be important for tritium behavior in the breeder, and is certainly important with respect to mechanical behavior. Local concentrations of bubbles, vacancies, interstitials, dislocations and so on may produce a microstructure that enhances or impedes tritium movement.

Reproducing the breeder tritium production rate will also reproduce the helium generation rate. Damage associated with atomic displacements will include that due to scattering of fast neutrons in addition to elastic recoil of the lithium fission products. In principle, a fusion neutron has 14 MeV available for fast neutron scattering, much more than typical neutron energies in fission reactors. However, the ^6Li fission reaction releases 4.8 MeV in triton and alpha particle kinetic energy which should lead to somewhat more localized but comparable total amounts of displacements per total tritium production. These fissions would be distributed over the neutron mean free path, as would the fast neutron scattering sites. The additional contribution to the total displacement from neutron scattering is probably only significant for unmultiplied Li_2O blankets.

In addition to the transient or rate-dependent effects of radiation, the cumulative effects of the damage may also be significant. Certainly, total helium production affects the amount of swelling.[6] The lithium burnup may also be important. Figure 2 illustrates part of the available experimental database for $LiAlO_2$ with respect to temperature and lithium burnup. It is possible that the changes in breeder chemistry at high burnup will lead to significant effects. At present, the planned FUBR-1B tests will approach 10% burnup levels but only for some materials and in closed capsule tests. All in-situ recovery tests have much less than 1% peak lithium burnup. Larger burnups are possible, at least in small diameter specimens, but will require very long irradiation times in present reactors (e.g., 3% lithium atom burnup per year in FFTF). An alternate approach might be to use pre-irradiated or artificially fabricated 'high-burnup' specimens in purged tests.

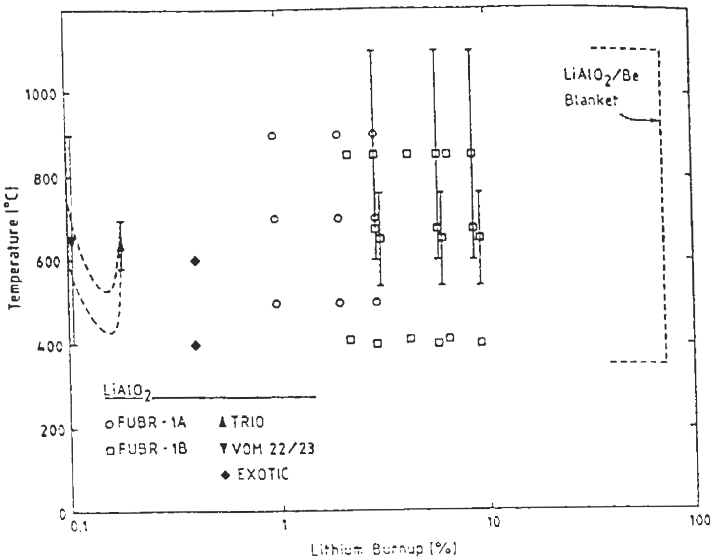

Figure 2. Part of the experimental database for $LiAlO_2$ indicating the achieved lithium burnup and temperatures. Solid symbols are purged tests, open are closed capsule tests.

Purge environment

Tests have been conducted with both closed and open (or purged or vented or swept) capsules. The amount of tritium retained in similar materials is usually different, although there are similar trends.[2] Since the blanket geometry is most likely to have an open or purged breeder, the latter tests are expected to be more representative of reactor behavior. Variables include the flow rate and the purge/clad chemical environment. Obviously, the presence of additives or impurities in the purge stream can affect the behavior. It has been well-established that the addition of oxygen to the purge tends to decrease the tritium recovery and promote the recovery of more HTO or T_2O.[5,7] The addition of hydrogen increases tritium recovery, primarily as HT.[5,7,8] Reactors with vented test capabilities can provide direct simulation of the purge chemistry conditions by appropriate inlet chemistry control. Many reactors, but not all, have the capability for purged tests.

The effect of the cladding and piping on the tritium recovery is also significant.

Fusion and fission reactors have a long length of purge piping between the irradiation site and the tritium analysis system. Experiments have shown that the nature of the irradiation capsule and its piping can affect the form of the recovered tritium.[5,8] Although possibly reactor-relevant if the appropriate materials and temperatures are used, interactions between piping and tritium complicate the interpretation of the tritium release data with respect to behavior at the breeder itself.

The local tritium release conditions in the breeder are affected by the cladding.[8] Most tests have specifically used non-reactor cladding (i.e., quartz or nickel) in order to specifically avoid effects due to tritium interactions with the cladding. However, comparable amounts of clad and breeder could be achieved in most fission reactor tests if desired (see Table 1).

Geometry

Although there is enough space in many fission test reactors for local geometrical details, these details are not considered to be particularly important at present except as needed to reproduce temperature gradients. The effects of the larger breeder geometry (e.g., mechanical strength, temperature profile, purge channel stability) may be significant once more definitive blanket designs are available.

Multiplier

Finally, the ability of fission reactor tests to address multiplier issues is important since a multiplier is required in many blanket concepts. The best multiplier with respect to neutronics, and possibly mechanics, is beryllium. Tritium recovery and mechanical behavior (swelling/creep/helium retention) are its most significant issues. The test conditions and previous comments with respect to the breeder material generally apply to the multiplier also. The important (n,2n) reaction cross-section has a 2.7 MeV threshold energy. Tritium production occurs by Be(n,T) and Be(n,alpha)^{6}Li(n,T) reactions, with 10 MeV and a few hundred keV threshold energies, respectively.[2] The latter accounts for almost 40% of the multiplier tritium production in some fusion conditions, and most of the tritium production in fission reactors. Table 2 summarizes the achievable helium, displacement, and tritium production capabilities of fission reactors in comparison with fusion blanket conditions. These conditions can reasonably be reproduced in fission reactors.

Table 2. Gas production in beryllium[1]

Source	Gas production rates		
	appm He/ dpa	appm He/ yr	appm T/ yr
Fusion breeder	420	20,000	–
Fusion electric	660	14,000	100
ORR thermal reactor	1300	21,000	–
EBR-II fast reactor	260	9,400	–
FFTF fast reactor	260	24,000	50-800
He beam	1000	–	NA

EXPERIMENTS IN FISSION REACTORS

In general, fission reactors are able to provide a reasonable simulation of many important solid breeder blanket conditions. Consequently, a number of closed and open capsule irradiations are underway or have been completed. These tests are exploring a range of temperatures, temperature gradients, materials, material characteristics, container materials, burnups, and purge gas compositions and flow rates. Closed capsule experiments do not have the flowing gas environment, but are cheaper and have proved useful for providing scoping data and irradiated specimens for subsequent properties measurement. The purged tests have provided important information on tritium recovery and the effect of purge composition.

However, as summarized in Table 3, existing experiments do not fully use the capabilities of fission reactors. For example, present open capsule tests expose breeder specimens to less than 0.5% lithium burnup, a few months irradiation, and 100 $^{\circ}$C/cm temperature gradients. Consequently, present tests primarily indicate the importance of material variables and some environmental conditions, and allow important processes to be quantified. Of course,

Table 3. Reactor-relevance of conditions in present and possible fission reactor tests

Parameter	Present		Possible		
	Closed	Open	Closed	Open	Module
Material	XX	XX	XX	XX	XX
Tritium prod.	XX	XX	XX	XX	XX
Heating rate	XX	XX	XX	XX	XX
Temperature	XX	XX	XX	XX	XX
Helium prod.	XX	XX	XX	XX	XX
Burnup	X	–	XX	X	X
Time at temp.	XX	–	XX	XX	XX
Temp. profile	X	–	XX	XX	XX
Rad. damage	X	X	X	X	X
Purge flow	–	XX	–	XX	XX
Purge comp.	–	XX	–	XX	XX
Clad material	–	–	XX	XX	XX
Geometry	–	–	X	–	XX

Degree of simulation: – Poor; X Fair; XX Good definitive data will await a fusion test device, but it is possible to achieve better simulation of important variables in fission reactor tests. These additional tests are indicated in Table 3 and summarized below.

Thermal Behavior Tests

Additional experiments could be performed to investigate local breeder thermal behavior, specifically chemical stability (corrosion, breeder/multiplier compatibility, thermal sintering, and vapor phase transport), mechanical stability (thermal expansion, thermal cycling, cracking, settling, gap conductance), and radiation stability (swelling, creep, radiation-induced sintering). These tests would not need to monitor tritium recovery, so could be simpler (i.e., closed capsules) with additional temperature instrumentation. These tests should use appropriate cladding materials and thicknesses, and include temperature gradients. Since these are simpler tests, they would provide cost-effective data at very high burnups and long test times.

Advanced In-situ Tritium Recovery Tests

Present tests also do not address the combination of moderate-to-high burnup with a flowing purge gas under temperature gradients and breeder/cladding interactions. Although these effects will be considered separately to some degree, synergistic effects and modeling inadequacies will make extrapolation to reactor-relevant combinations uncertain. Advanced in-situ tritium recovery experiments could address these interactions with relatively small capsules (about 1-5 cm diameter). The importance of achieving significant burnup while limiting self-shielding in a fission reactor neutron spectrum leads to relatively long irradiation times and a preference for fast reactors.

Nuclear Submodule Tests

Finally, nuclear test assemblies designed for fission reactors can provide the maximum concept verification possible in non-fusion devices. A full-blanket module test would need about 1 m^3 of test volume, require extensive modifications to any operating fission reactor core, but still only achieve the equivalent of (at most) a 1 MW/m^2 heating rate in any existing reactor.[1] In-core assemblies could be placed in existing fission reactors like FFTF at reactor-relevant heating rates (2-5 MW/m^2), but would be limited to about 10 cm diameter. These test assemblies would provide fairly realistic simulation of fusion conditions, with complete coolant and purge flow systems.

SUMMARY

The near-term emphasis for solid breeder blankets requires resolving materials-related uncertainties and identifying attractive engineering breeder/multiplier materials. This testing requires a reasonable number of moderate volume test sites in a neutron environment. Existing fission reactors are being used, and are providing a reasonable simulation of most important breeder related parameters. However, not all conditions, or important combinations of conditions, are being met in present experiments. Major additional tests can also be accomplished in fission reactors, particularly advanced in-situ tritium recovery tests, thermal behavior tests, and nuclear submodule tests.

ACKNOWLEDGEMENTS

This work was performed as part of the FINESSE project under U.S. Department of Energy Contract DE-AS03-84ER52105. One of the authors was partially supported by the Canadian Fusion Fuels Technology Project.

REFERENCES

1. M.A. ABDOU et al., "Technical Issues and Requirements of Experiments and Facilities for Fusion Nuclear Technology", PPG-909 (also UCLA/ENG-85-39), University of California at Los Angeles (1985).
2. D.L. SMITH et al, "Blanket Comparison and Selection Study, Final Report", ANL/FPP -84-1, Argonne National Laboratory (1984).
3. G.W. HOLLENBERG, Hanford Engineering Development Laboratory, private communication, 1984.
4. I.J. HASTINGS, Chalk River Nuclear Laboratory, private communication, 1986.
5. "The LISA experiment", 2nd Inter. Conf. on Fusion Reactor Materials, April 13-17, Chicago IL, 1986.
6. D.L. BALDWIN and G.W. HOLLENBERG, "Measurements of Tritium and Helium in Fast Neutron Irradiated Lithium Ceramics using High Temperature Vacuum Extraction", 2nd Inter. Conf. on Fusion Reactor Materials, April 13-17, Chicago IL, 1986.
7. R.G. CLEMMER et al, "The TRIO Experiment", ANL/FPP-84-55, Argonne National Laboratory (1984).
8. I.J. HASTINGS et al, "Irradiation of Lithium-Based Ceramics for Fusion Blanket Applications", 7th Annual Conf. of Canadian Nuclear Society, June 9-11, Toronto Canada, 1986.