A MODULE FOR TESTING A LITHIUM COOLED TOKAMAK BLANKET IN A TANDEM MIRROR TEST REACTOR[a]

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ABSTRACT

A preliminary conceptual design was made of a test module for testing the BCSS lithium cooled, vanadium structure blanket concept in a tandem mirror fusion test reactor. The test module is intended to run at the correct stresses and temperatures of a segment of the blanket, while operating at reduced neutron wall loadings. Although this approach ties the effort to a single design and its issues, it tends to reveal interactions that might not be apparent if a more general approach were taken, and can lead to some useful general conclusions.

INTRODUCTION

Fusion development will require integrated blanket testing prior to building a full scale power reactor. Exploring integrated test concepts now can illuminate problems and define the limitations of the types of tests studied. Integrated fusion blanket testing in fusion test devices is probably one of the most expensive, and therefore one of the most interesting, test catagories. These test devices may operate at lower neutron wall loading, lower first wall heat flux, and smaller size then a power reactor. An attempt was made to synthesize our current understanding of the effects of scaling these parameters into a test module design that can do integrated, act-alike testing of several important issues.

A specific design and limited set of issues were chosen in order to illuminate generic testing problems and their solutions. A preliminary design, which incorporates consideration of fluid flow, heat transfer, structural analysis, mechanical design and neutronics gives a realistic assessment of the limitations of scaled integrated testing. This work has been directed toward testing in tandem mirror test devices, and has concentrated primarily on the BCSS lithium cooled vanadium reference blanket design described in Reference 1. Thermal-mechanical issues have been emphasized because of their importance to the liquid metal designs. Also included is a qualititive evaluation of the test module performance in addressing the key issues of fusion nuclear technology, as identified in the FINESSE[2] study.

Thermal and pressure stresses are the primary causes of possible structural failure in the BCSS toroidal-poloidal flow lithium cooled blanket. To preserve thermal stresses, temperature changes throughout structural members must be preserved. If temperature differences can be preserved, absolute temperatures can generally also be maintained by controlling the coolant inlet and/or outlet temperature. Pressure stresses can also be made act-alike over a limited portion of the blanket by adjusting the inlet pressure, assuming that geometry has been preserved. The test module attempts to reproduce the BCSS blanket stresses and temperatures exactly over a limited region of the composite first wall. This "unit cell" approach is felt to be valid for this blanket design because the local stresses are due primarily to the local loading. Whole-blanket forces are generally supported by the external support structure. Global stresses generated by these forces will change the local stresses, but this is assumed to be a second order effect.

REFERENCE BLANKET DESIGN

The blanket design chosen for study is the BCSS toroidal/poloidal flow, lithium cooled, vanadium structure blanket.[1] Lithium

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[b] On assignment from University of Tokyo.
[c] On assignment from University of Kyoto.

entering the top of the inboard side of the blanket flows in large channels primarily in the poloidal direction, but with a slight angle so as to direct the entire flow to the side of the blanket where the toroidal feeder manifolds are located. Flow in the toroidal channels is nearly parallel to the magnetic field, allowing higher velocities to cool the first wall without incurring large MHD pressure drops. The neutron wall load is 5 MW/m^2, and the first wall heat flux is 1 MW/m^2. The inboard blanket sector is approximately 9 meters high and 3 meters wide.

REFERENCE DEVICE

The test device is a beam driven tandem mirror of the type described in references 2, 3, and 4. The area available for testing is 1 to 2 meters long, with a minimum first wall radius of 25 cm. The neutron wall loading at a radius of 25 cm is 2 MW/m^2.

CONFIGURATION OF THE TEST MODULE

The configuration of the lithium cooled test module is shown in Fig. 1. The design approach has been to define a minimum structural unit cell and attempt to attain act-alike temperatures, stresses, and velocity profiles within the cell. Lithium inlet and outlet piping through the shield may need to be large in the test device because of high MHD pressure drops in the vicinity of high field coils, which are needed to create the fusion plasma. Four test modules could surround the plasma circumferentially.

Lithium flows through the three large "poloidal" channels in the same direction (from right to left in Fig. 1). The two end channels provide the inlet and outlet to the first wall "toroidal" channels, while the middle channel is a dummy, providing only cooling to the second wall and some neutron reflection. Velocity profiles in the three large channels may not become the fully developed Hartmann flow expected in the blanket poloidal channels because of the short path length. The test module depth could be reduced to increase flow development in this region. Velocities will be scaled to achieve act-alike flow in the first wall channels, not the large "poloidal" channels which are felt to have a lesser impact on the thermal and structural response of the first wall. Flow in the first wall channels will be discussed below. As indicated in Fig. 1, it may be desirable to configure the test module inlet and outlet so that turn angles are minimized and velocity changes do not occur at the test module edge.

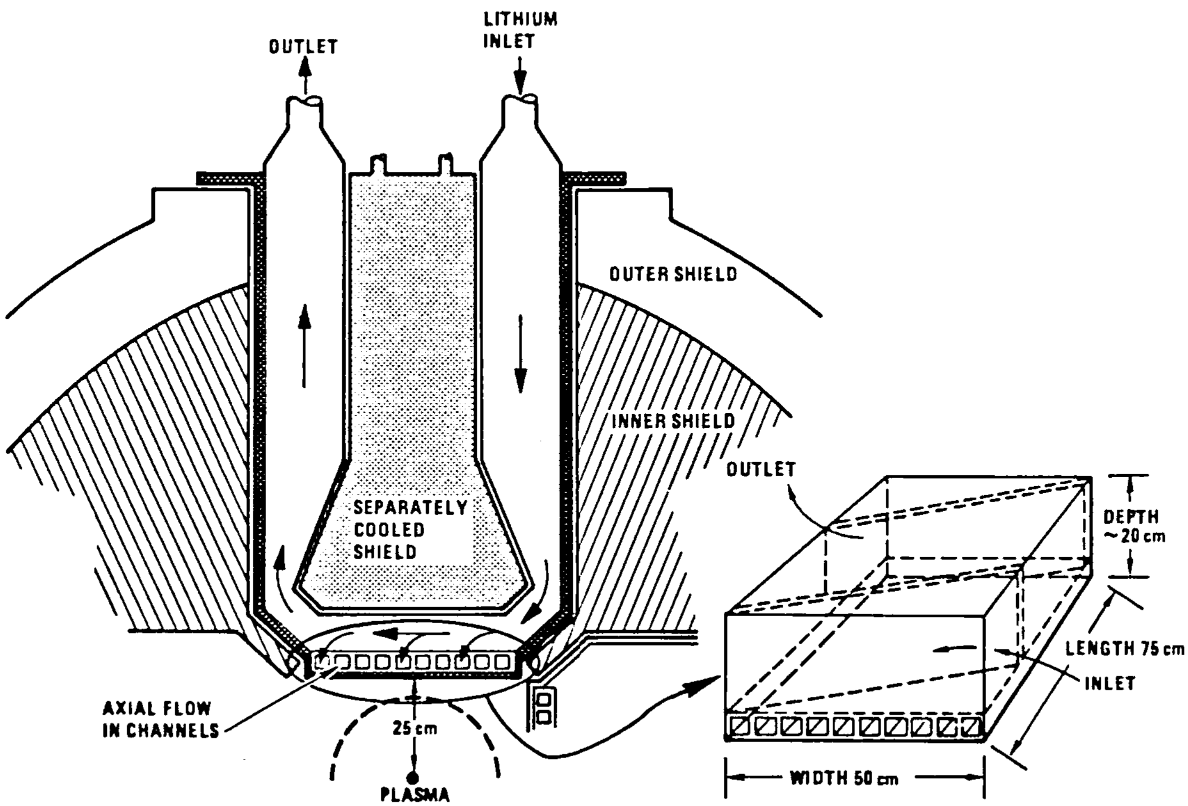

Figure 1. Lithium cooled vanadium tokamak blanket test module.

The test module first wall is flat to preserve geometry. Curving the wall to follow the plasma radius would result in more uniform neutron fluxes, but would also significantly stiffen the structure. Neutronics results[2] indicate a reasonably flat flux profile. Nuclear heating is approximately 24% lower at the sides of the module than in the middle. Axial neutron losses in the tandem mirror device are also significant. Neutronics calculations[2] for a 1 m test cell with a 2 m plasma indicate a neutron loss of approximately 25% and a heating rate loss of approximately 15%.

The first wall area is based on considerations of the space available and on a minimum structural unit cell. Minimizing the area required for testing is important because of cost. Increasing length in the axial direction is expensive because it requires increasing the power injected into the plasma. One span of a single poloidal channel is the absolute minimum structural unit cell. The module width is limited here by the requirement of fitting four modules as closely as possible to the plasma. The structural unit cell in this direction is one toroidal channel. Shear and bending boundary con ditions can be neglected in all but a few of the outermost channels because of structural damping. The test module first wall area has been determined somewhat qualitatively, but we believe it is reasonably close to a minimum structural unit cell. First wall area is very dependent on the particular blanket design being tested.

The dimensions of the test module can be scaled with the first wall heat flux and neutron wall loading in order to preserve temperatures. For example, if the first wall heat flux is lowered, the first wall should be thickened until the temperature rise through it (due to heat flux plus heat generation) is equal to the temperature rise in the reference design. The other channel dimensions are increased in direct proportion to the first wall to maintain act-alike structural response.

Maintaining tokamak first wall heat fluxes can reduce the required size of the test module and may improve the value of the test for reasons discussed below. It may be feasible to artificially heat the first wall in the TM test device with tungsten heating elements. Thin tungsten wires will have little impact on neutronics and can easily supply over 50 W/cm^2 to the first wall by radiation. It may be difficult to achieve 100 W/cm^2 or more by this means because of the required temperature of the heating elements.

STRUCTURAL ANALYSIS

Elastic Stresses

Calculated elastic stress values, primarily due to thermal stress, are above the yield stress in the BCSS lithium cooled vanadium blanket under consideration. This may be generic to tokamaks if both first wall heat fluxes and erosion rates are high. The secondary thermal stresses will be reduced when the material yields and, therefore, will never reach the values obtained using elastic analysis. However, since the local geometry (e.g., thickness-to-length ratios) of the power reactor blanket is preserved in the test module, if the calculated elastic stresses are matched, then the initial plastic stresses and strains should also be matched to a good approximation. Irradiation and thermal creep, and swelling, also play an important role in stress relaxation, stress buildup, and deformation, and will be discussed in the next section.

The surface heat flux is the largest contributor to the first wall stress in tokamak blankets. It has been demonstrated in the BCSS composite first wall structure,[2,3] that the thermal stress depends primarily on the temperature gradient across the first wall and the difference in temperature between the first and second walls. The exact shape of the temperature profiles is less important. If control of the coolant temperatures is possible, then the radial temperature profile (where "radial" is the direction away from the plasma) can be simulated well without bulk heating.

Initial elastic stresses can be made act-alike in the test module first wall region over a wide range of wall loadings and first wall heat fluxes as long as all temperatures, pressures, and aspect ratios are preserved. For example, if the first wall thickness must be increased to preserve temperatures and thermal stresses at reduced heating rates, then all other dimensions (e.g., first wall channel width and depth, second wall thickness, etc.) must be increased in direct proportion. However, neutron mean free paths do not scale and, therefore, damage gradients are altered if the blanket thicknesses are changed in the test module. This, together with the linear increase in test module size with decreasing first wall heat flux, indicates that it is desirable to preserve first wall heat fluxes by artificial means.

Irradiation Creep and Plastic Stress

Figure 2 shows stress relaxation of the composite first wall due to irradiation creep. The first wall surface is initially under very high compressive stresses, as indicated by the lowest curves at time zero. The back of the composite wall is in tension, as indicated by the upper curve. After about 6 dpa, the lines cross. The first wall surface goes into tension, and the back into compression as would be expected from the pressure stresses alone. Thermal stresses appear to relax nearly all the way out after about 15 dpa. Deflection calculations[5] indicate a constant rate of deflection after about 6 dpa.

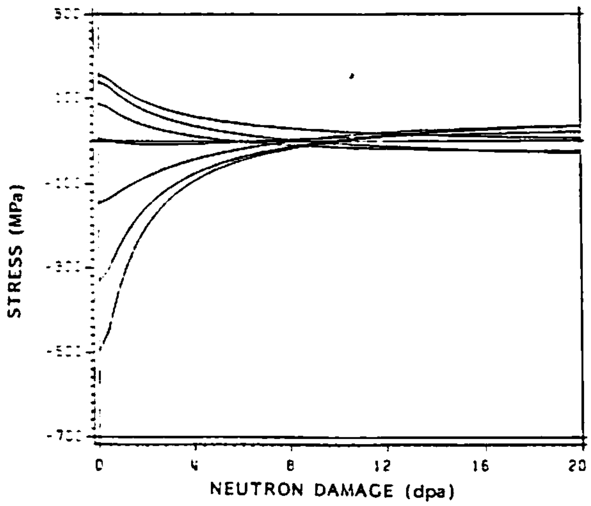

Figure 2. Composite first wall stress history due to irradiation creep including pressure and thermal stresses

Although the model has some important limitations,[5] three significant conclusions can be drawn: 1) relaxation of thermal stresses due to irradiation creep occurs over a period of a few months, 2) the displacement rate due to irradiation creep under primary (pressure) stresses appears to be constant, and 3) scaling the dimensions while preserving aspect ratios may be feasible. No irradiation creep data for V-15Cr-5Ti was available at the time of this writing, so HT-9 creep properties were used. The vanadium alloy may have a different steady-state creep rate and relaxation times may change somewhat, but we expect the above conclusions to hold.

A stress relaxation period of a few months can have significant impact on the requirements (and cost) of a fusion test test device due to ratcheting. If the reactor were shut off after irradiation creep relaxed most of the thermal stresses (after a few months of operation), the thermal stresses would reverse and might again enter the plastic regime. The stress distribution at shutdown would be dependent on how long the reactor had been operating and, if plastic deformation occurred, the stress distribution after the reactor was re-started would also depend on how long the reactor had been operated before shutdown. Thus, a unique stress distribution would exist for each start-up/shut-down history, requiring long periods (months) of continuous operation of a fusion test device if stresses are to be matched. Note that if stresses do not enter the plastic regime when the test device is shut down, the ratcheting behavior described above would not occur and stress matching could be done in a short duty cycle test device.

Calculations[2,5] indicate that creep relaxation stresses are preserved in different sized composite walls if all aspect ratios are preserved. However, failure modes and times to failure due to cracks may not be act-alike. If possible, the original size as well as the geometry of the power blanket should be preserved in the test to remove this uncertainty. This is another reason for preserving first wall heat fluxes.

THERMAL ANALYSIS

Temperatures in the reference BCSS toroidal/poloidal flow blanket structure are determined by heat generation in the structure, heat deposition on the first wall, and heat transfer to the coolant. It has been shown[2] that the heat transfer coefficient is dependent on the first wall heat flux, the heat generation in the coolant and structure, and the fluid velocity profile. Thus, the structural temperatures depend on these operating parameters as well. Unfortunately, the MHD-dominated velocity profiles are not well understood at this time, contributing to large uncertainties in the actual temperatures and stresses.

The effect of bulk heating on the first and second wall structural temperatures is indicated in Fig. 3. The analysis, as described in reference 2, uses parabolic and couette (a triangular profile with the maximum velocity occurring at the second wall) velocity profiles.

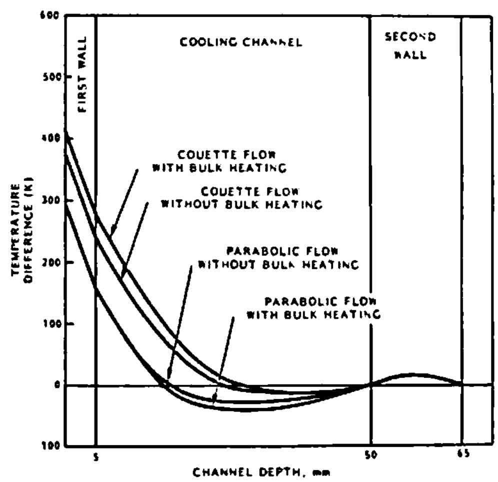

Figure 3. Effect of bulk heating on the temperature profile through the first wall.

If velocity profiles are assumed to be fully developed, then temperature profiles can be made act-alike by preserving the Fourier number, which generally requires slowing the flow rate in the test module. Preserving velocity profiles in shortened channels also requires slowing the flow, but magnetic field affects and end affects are not currently well enough understood to even assess the feasibility of the reference design, making analysis of act-alike flow testing impossible. Some control may be possible by slightly adjusting the orientation of the first wall channels to the magnetic field. If velocity profiles can be made act-alike, temperatures can be made very nearly act-alike over a wide range of operating conditions.[2]

It will be possible to control the second wall temperature profile in the test module to some extent by controlling the temperature in the central "poloidal" channel. This approach is based on the assumption that the first wall is more important and should be emphasized, and results in the loss of information in regions away from the first wall.

EVALUATION OF TEST MODULE PERFORMANCE WITH RESPECT TO MAJOR ISSUES

The test module discussed here emphasizes thermal-mechanical issues. The intent was to design a module that will test as many issues as possible in an integrated manner. When conflicts arise, stress and thermal-mechanical issues are given the highest priority. The next step would be to design another module or integrated test that would emphasize the issues on which this module performs poorly. As a beginning point, MHD testing in a non-nuclear reactor mock-up, to address some of the issues not covered by the test module, has been considered. Act-alike (or nearly act-alike) velocity profiles, pressure drops and temperature regimes (but not local tempera tures) were assumed. If one is optimistic about the performance of the test module and the MHD mock-up, they tend to address the issues (as currently understood) in a complementary fashion (see Table 1). The nine issues listed are a working list of critical and high priority issues developed for FINESSE. Cost effectiveness has not been considered.

If a less optimistic approach is taken, the test module and MHD mock-up may give useful information that can be used to benchmark codes and test theories. Note that the test module provides benchmark information for the MHD issues, but the MHD test provides no information for most of the stress issues. The test module can provide some information about all the issues listed because it embodies all the conditions and relevant components found in the power reactor.

Table 1. Test Module Performance Relative to the Nine Liquid Metal Blanket Issues Identified in FINESSE

Issue	Very Useful or Act-Alike	Benchmark	Little or No Information
1) DT fuel self-sufficiency		A	B
2) Irradiation effects on materials properties	A[1]		B
3) Structural response to environmental conditions	A	B	
4) MHD pressure drop and pressure stresses	B	A	
5) MHD flow distribution and heat transfer	B	A	
6) Corrosion	B	A	
7) Tritium control in the primary loop		A	B
8) Characterization of hear sources		A	B
9) Identification of failure modes	A	B	

A = test module, B = non-nuclear MHD mock-up.

1) Low-to-medium fluence only.

SUMMARY AND CONCLUSIONS

A minimum unit cell is a likely candidate for providing the most information at near act-alike values for testing the BCSS toroidal/poloidal blanket in a fusion reactor. Act-alike structural behavior and failure modes may be attainable in the unit cell test module. However, under realistic operating conditions in an affordable test device, it will be difficult to obtain the fluence that may be required to relax thermal stresses (~ 10-12 dpa), and thus adequately test ratcheting and other stress relaxation dependent failure modes. (Swelling has not been explored here. Fluences high enough to achieve significant swelling in HT-9 and V-15Cr-5Ti are not likely to be obtainable in a fusion test device). Most of the issues not thoroughly addressed in the structural tests are MHD related, and fusion test module performance is limited by the geometry and strength of the magnetic field in the test device.

Useful blanket tests with act-alike behavior parameters can be performed in a tandem mirror test device at reduced wall loadings. The ability to control first wall heat flux in the test device would be extremely useful and will be necessary to match temperatures in some cases. For testing high heat flux tokamak blankets, fluxes equal to those in the power reactor (50 - 100 W/cm^2) will greatly increase the value of the test, and can reduce the required size. Methods of controlling the first wall heat fluxes in test devices should receive further attention.

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