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LABORATORY MICROFUSION FACILITY

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NEUTRONICS ANALYSIS OF THE LABORATORY MICROFUSION FACILITY

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ABSTRACT

The radiological safety hazards of the experimental area (EA) for the proposed Inertial Confinement Fusion (ICF) Laboratory Microfusion Facility (LMF) have been examined. The EA includes those structures required to establish the proper pre-shot environment, point the beams, contain the pellet yield, and measure many different facets of the experiments. The radiation dose rates from neutron activation of representative target chamber materials, the laser beam tubes and the argon gas they contain, the air surrounding the chamber, and the concrete walls of the experimental area are given. Combining these results with the allowable dose rates for workers, we show how radiological considerations affect access to the inside of the target chamber and to the diagnostic platform area located outside the chamber. Waste disposal and tritium containment issues are summarized. Other neutronics issues, such as radiation damage to the final optics and neutron heating of materials placed close to the target, are also addressed.

INTRODUCTION

The Laboratory Microfusion Facility will be designed to contain target fusion reaction yields up to a nominal 1000 MJ. This corresponds to the production of 3.6×10^{20} D-T neutrons per shot peaked at 14.1 MeV. The radiological hazards, resulting from this burst of high energy neutrons and from the surrounding materials they activate, present a significant challenge to the facility designers. Neutrons must be moderated and absorbed, along with the secondary and decay gamma rays, in such a manner that the facility meets the U.S. Department of Energy (DOE) radiological siting requirements,¹ the International Commission on Radiological Protection (ICRP) (and DOE) public dose limits,² the DOE occupational dose limits,³ and the more stringent LLNL as low as reasonably achievable (ALARA) guidelines.⁴

The primary goal of the facility is to conduct some 1400 target experiments over a five-year period to demonstrate high gain in the laboratory. Projected yields for these experiments range from <10 MJ to ~1000 MJ. After high gain has been achieved, target yields above 100 MJ will be used to conduct weapons physics and weapons effects experiments. High gain experiments are expected to be conducted once per week. This will extend the facility lifetime to as much as 30 years.

We have set two design goals that minimize the affect of radiation hazards on the operation of the facility. The first is that LMF workers will have access to the diagnostic platforms outside the target chamber within 24 hours after a high-yield shot. This is required to retrieve data and set up for the next shot. Although it is possible that routine interior chamber operations will be done remotely, workers may occasionally require access to the inside of the chamber for nonroutine operations and maintenance. Therefore, the second design goal is that workers can safely enter the chamber for a short time 7 days after a high-yield shot. As quantitative measures of these two design goals, we have established dose rate limits of 2 mrem/hr outside the chamber 24 hours after a shot and 100 mrem/hr inside the chamber 7 days after a shot.

Several conceptual designs are currently being proposed for the LMF target chamber. All the designs include an inner first wall (aluminum, carbon, Kevlar epoxy, or fiberglass) that is protected from x rays with some other material (frozen nitrogen, a layer of ice or porous alumina, or a carbon carpet), a neutron moderator and absorber (water or borated graphite), and an outer structural wall (aluminum or steel). In this paper, we describe the results for a representative target chamber that consists of an inner 4-cm-thick aluminum wall, a 2-m-thick region of water, and an outer 4-cm-thick structural wall

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also made of aluminum, and a concrete building as shown in Fig. 1. While this may not be the design chosen for the LMF, it serves as a useful example for discussing radiological issues, and it serves as a point of comparison for the other concepts being proposed. Where possible, we point out the anticipated impact of design modifications.

Note that although aluminum is generally considered a low-activation material, the ^{24}Na produced by (n,α) reactions with ^{27}Al is particularly troublesome for the first few days after a shot. The ^{24}Na has a half life of 15 hours and its decay yields two high energy gammas (2.75 and 1.37 MeV).

Our analysis includes: effects of leakage neutrons (from some 72 beam ports), materials impurities, nuclide buildup, and neutron scattering from concrete and beam tubes. Our estimate of the 24-hour dose rate at a diagnostic platform includes contributions from the activation of air, concrete, beam tubes, vacuum chamber, and interior chamber materials.

Two crucial design issues are the post-shot tritium recovery and radiation damage to final optics. We include data concerning the amount of tritiated waste and discuss means of handling and disposing of it. We also show that, at the projected radius of 25 meters, neutron damage to the final optics (degradation of transmission) will not impede LMF operations.

GENERAL NEUTRONICS ISSUES

The calculations performed for this paper were done using the computer codes TART,⁵ FORIG,⁶ and ORLIB.⁷ The neutron-photon transport code TART uses a Monte Carlo technique to determine the spatially- and energy-dependent fluences of neutrons and photons. ORLIB combines the neutron spectra from TART with our most current neutron cross section library to produce one energy group, spatially averaged, neutron activation cross sections. This information is then used by the nuclide generation-depletion code FORIG. Generally, TART was also used to determine the dose rates near activated surfaces using photon transport calculations and water to simulate tissue, as tissue is 71% water. Nitrogen, oxygen-like in absorbing decay radiation, is another 23%. Therefore water is an adequate model for 94% of the body.

The buildup of nuclides over time was considered in all the analyses using the equation

$$N = N_0 [(1 - e^{-\lambda t n}) / (1 - e^{-\lambda t})],$$

where N is the number of curies of a given nuclide after n shots spaced t time units apart, N_0 is the amount of the nuclide produced on a single shot, and λ is the decay constant for the particular nuclide. The nuclide buildup approaches secular equilibrium

Fig. 1. Final optics protection and biological shield dimensions for activation analysis.

(where the amount that decays between shots is equal to the amount produced in a single shot). For example, at a shot rate of one per week, the buildup of ^{24}Na produced in Al after 500 shots is 1.00043 times the quantity produced in the first shot. For ^{54}Mn , which is produced from Fe and has a half-life of 312 days, the buildup is 64.8 times the quantity produced in the first shot.

The pr of the target (the density of the fuel region times its thickness) has a moderate effect on the activation of interior-chamber and first wall materials since it changes the neutron energy spectrum due to scattering in the fuel. The greater the pr of the fuel, the greater the neutron energy deposited in the fuel itself before the neutrons escape. A pr of 3 g/cm² will be the average fuel density achieved in LMF although some targets may achieve a pr of 5 g/cm². The effect of the target's pr on material activation is expected to be greatest for those neutron reactions with thresholds (i.e., minimum neutron energy required to make the reaction energetically possible). For example, about 10% less ^{24}Na is produced in an Al first wall for a pr of 5 g/cm² target than for a pr of 3 g/cm² target, assuming the yield is the same in both cases. Implications of a range of pr values are being considered in the LMF design.

EXPERIMENT AREA DESIGNS

Two issues dominate the design of the EA: how to prevent or deal with the x-ray ablation of the first surface, and how to limit activation levels after high-yield shots to allow adequate accessibility.

The x rays will be dealt with in one or more of the following ways: with an x-ray and debris shield made of 15 cm of solid nitrogen or 0.25 cm of polyethylene at a radius of 20 cm (Ref. 8); with a water frost layer condensed on an aluminum first wall⁹; with a sacrificial layer at the first wall made of alumina; with a carbon-carpet inner surface; or with pyramid-like carbon tiles.¹⁰

Limiting activation levels is done through careful material selection (high purity, low activating) and by providing adequate shielding to moderate and absorb the neutrons sufficiently to retard activation. The choice of low-activating aluminum allows re-entry into the chamber at the 7-day point (100 mrem/hr). For such a design, sufficient shielding must back this first Al wall to do two things: prevent activation of the second Al wall outside the shielding and attenuate the decay radiation from the first wall. This can be accomplished with 2 m of water. (see Fig 1.)

The 15-cm-thick x-ray and debris shield made of solid nitrogen provides not only a sink

for soft x rays and debris but also moderates and absorbs enough neutrons through (n,γ) captures to reduce the activation in an aluminum first wall by about 1/3 from the case with no debris shield. This reduces the time required for re-entry by half a day. The carbon-14 and tritium production from the nitrogen is insignificant.

Aluminum will be the material of choice over stainless steel for the frost condensing surface. Calculations using 3-mm-thick panels indicate that SS 304 would give an interior chamber dose rate 25 times greater than Al 5083 6 days after 500 full-yield shots (one-week spacing). This Al 5083 dose rate (40 mrem/hr) could be cut in half if a low-activation Al with similar properties to Al 5083 were used. The constituents of this low-activation aluminum are listed in Table 1 (Ref. 11).

The frost concept currently includes the idea of using fiberglass (located behind the aluminum condensing surface) as the vacuum vessel and as the container of the water used as the main neutron shield. The choice of fiberglass would result in an extremely low residual level of activation. Access to both the outer wall and the interior would be much less inhibited than with any other scheme proposed thus far. Detailed analysis of the performance of fiberglass in a radiation field is necessary. However, epoxy phenolics can generally withstand about 10⁹ rad before experiencing significant weakening.¹² A 5-cm first wall of fiberglass would receive on the order of that dose during the LMF lifetime.

Proponents of the carbon-carpet first wall concept have also proposed using a Kevlar epoxy first wall, thus eliminating most of the Al (except for some support structures) from the high energy flux. This too would result in very low activation and good access to the interior of the chamber.

ACTIVATION OF ARGON GAS USED TO PROTECT FINAL OPTICS

Final optics must be protected from x rays and debris as well as shrapnel. Shown in Fig. 1 is

Table 1. Aluminum alloy with properties similar to Al 5083 but with lower impurity and activation levels.¹¹

94.999 wt%	Al
4.5	Mg
0.5	Si
0.001	Fe

a conceptual design of such a protection system.¹³ Of interest for neutronics analysis is the 24 Torr-m of argon intended to protect the fused-silica from soft x rays. A potential concern is the production of ^{41}Ar , both in the beam tube and in the surrounding ambient air, by the leakage neutrons. The concentrations of ^{41}Ar in and out of the beam tubes were calculated. The ^{41}Ar concentration in a beam tube after a shot is at most $4 \times 10^{-3} \text{ Ci/m}^3$. Although this exceeds the allowable concentration for continuous exposure to workers, the concentration decays to $4.5 \times 10^{-7} \text{ Ci/m}^3$ at the 24-hour re-entry point, about an order of magnitude less than the allowable limit. Likewise, the concentration in the ambient air of the target chamber (due to the activation of the argon in the air) is $5 \times 10^{-5} \text{ Ci/m}^3$ after a shot and $6 \times 10^{-9} \text{ Ci/m}^3$ at the 24-hour point.

ACTIVATION OF BEAM TUBES

The activation of the aluminum beam tubes is also a major concern that could prevent re-entry into the EA within 24 hours after a maximum yield shot. Estimated to be 2 cm thick, the tubes are modeled as cylindrical tubes encasing each final focusing optic at the concrete wall and extending to the discrete port for beam entry at the chamber outer wall. This design, a cylindrical instead of tapered cone tube, uses the overlap around the actual beam hole as a shadow shield that effectively protects the beam tube from direct-shine neutrons. The actual beam hole at the chamber outer wall may be only 50 cm in diameter. The tube diameter is 1.4 m at the final optic, changing to 0.7 m at the second wall (see Fig. 1). The smaller radius is required at the chamber wall to provide adequate spacing for all 36 beam tubes in the designated solid angle of the sphere. Calculations indicate that the dose rate from the aluminum tube 24 hours after the five hundredth 1000-MJ shot is only 1.2 mrem/hr. A further step to minimize beam tube activation would be to line the interior of the tube with a thin layer (1 cm) of CH_2 . It is expected that those neutrons with a near grazing incidence to the beam tube material will see an effective thickness of moderator that will serve to further reduce neutrons to energies below the threshold for the (n,α) reaction in Al that nets the troublesome nuclide, ^{24}Na . Further work is required to establish the merit of this proposal and to see if normal Al 5083 could be used.

NEUTRON DAMAGE TO FINAL OPTICS

Final focusing lenses should be made of a material that is resistant to radiation damage and produces insignificant residual radiation. The leading material of choice is fused-silica. We have conducted a study of the

effects of 14-MeV neutrons in fused-silica when exposed to fluences that are predicted for the LMF final focusing lenses, which are currently set at 25 m from the target. The experimental fluence was $1.5 \times 10^{16} \text{ neutrons/cm}^2 \pm 20\%$. This is equivalent to ~3300 full-yield shots. This far exceeds the projected number of full-yield shots for the LMF lifetime covering high-gain development (100 full-yield shots), weapons physics experiments (1000 full-yield shots), and weapons effects experiments (100 full-yield shots). Including moderate- and low-yield shots, the equivalent number of lifetime full-yield shots will be 1600 (Ref. 14).

In this more recent experiment, a sample of fused-silica, 2.7 cm in diameter and ~1 cm thick, was irradiated using the Rotating Target Neutron Source (RTNS II) facility over a two-week period in May 1987. Spectrophotometry measurements made before and after the exposure indicated a change in optical density primarily below 400 nm with the peak at 215 nm. The baseline transmittance at 1ω (1.06- μm wavelength light) and 2ω (0.53 μm) was not altered. However, there was some degradation at 3ω (0.33 μm) and substantial reduction in transmittance at 4ω (0.26 μm). An earlier study showed no degradation in transmission at any wavelength for an experimental fluence level of $1.5 \times 10^{15} \text{ neutrons/cm}^2$. The final optical density measurements as a function of wavelength are given in Fig. 2.

The neutron-induced effects were largely reversible. Annealing of the sample at 400°C for two hours significantly recovered the transmittance. An additional 4-hour annealing had significantly less impact, suggesting a much longer required time to completely reverse the effects. Gamma spectrometry analysis indicated that there is a small amount of impurities in fused-silica, namely, Zn, Mn, Ce, Na, and Y. However, the resultant activation made no significant contribution to the residual radiation levels.

The degradation in transmission is assumed to be due to atomic displacements by neutrons. However, 5.5% of the total dose (1 Mrad out of 18.7 Mrad) to the silica is from secondary gamma radiation. There are indications that the gamma radiation threshold for damage is greater than 10 Mrad (Ref. 15). The degree to which electronic collisions may be reducing transmission will be explored later when a sample will be exposed to a comparable gamma dose and more spectrophotometry measurements are made.

AIR ACTIVATION

The activation of the ambient air in the concrete enclosed dome must not prevent access to the chamber at the 24-hour point for a full-yield shot. Calculations indicate that a

day after a 1000 MJ shot, a 30-m-radius dome will contain 0.1 μCi of ^{41}Ar produced from the neutrons that get through the 2-m-thick shield. The contribution from the direct shine neutrons that escape through the 2 to 3% open solid angle and scatter off the concrete is 60 mCi of ^{41}Ar at this same time. The Derived Air Concentration (the allowable concentration for someone to continually breathe and not receive more than 5 rem in one year) is $3 \times 10^{-6} \text{ Ci/m}^3$ and is an order of magnitude greater than the actual concentration at 24 hours.

BIOLOGICAL SHIELD

Personnel must be protected from receiving excessive prompt doses of radiation from a high-yield shot, as well as from the residual radiation. Our results indicate that a 1.5-m-thick concrete wall enclosing the experimental area will reduce the prompt exposure immediately outside the EA building to less than 1 mrem/shot, or 50 mrem/yr, 1% of the allowable annual dose. An additional 0.7 m of

concrete (2.2 m total) will be required behind the final optics at the end of the beam ports (see Fig. 1).

The activation of the concrete presents a source of residual radiation that might constrain access to the diagnostic ports at the 24-hour point. The 72 beam tubes constitute clear paths for unmoderated neutrons to create "hot spots" in the concrete wall. Assuming that the inner radius of the concrete is 30 meters, the dose at the 7-m diagnostic ports at 24 hours is estimated to be about 10 mrem/hr. The overwhelming contribution to this dose rate is ^{24}Na production from (n,γ) reactions with ^{23}Na in the concrete. The addition of boron to the concrete to suppress this reaction will be required, as well as some inelastic scattering shielding in front of or imbedded in the concrete. Adding 5 at.% boron to the concrete is estimated to reduce the activation by an order of magnitude. Further work is ahead to tailor the concrete shield to the specific LMF needs. We may also consider the use of "mobile shielding" to shield the diagnostic platforms from the concrete.

Final optics are set at 25 m from the target
Transmission (T) = 10^{-D} , D is optical density

Fig. 2. Neutrons and neutron-induced gamma rays cause insignificant transmission loss at 3λ in fused-silica for proposed LMF lifetime fluences ($7 \times 10^{15} \text{ n/cm}^2$).

NEUTRON HEATING CLOSE TO THE TARGET

We made a number of neutronics calculations to determine the neutron heating in materials positioned close to the target. These might include target-support structures, shrouds, irradiation samples, and perhaps even diagnostics. Calculations were done for Al, Fe, Cu, and SiC.

Our results show that all the materials will melt, sublime, or dissociate if placed within 10 to 20 cm of a 1000-MJ-yield target. Carbon and silicon carbide have comparable minimum inner radii at about 13 cm, while the three metals considered have minimum radii closer to 20 cm. Steel is slightly better than Al and Cu. Even if we can protect materials from x rays and debris, anything this close to the target must be considered sacrificial since it will be destroyed by neutron heating. The innermost 10 to 20 cm of the target support structure, for example, will have to be replaced for every shot.

INTERIOR DOSE RATE

The cumulative interior dose rate goal is <100 mrem/hr about seven days after a full-yield shot. The contributions to this dose rate come from the following: the target shroud and support mechanism, the ablated and recondensed first wall, the remaining tritium (internal dose), the sacrificial diagnostics, the first wall, and any materials introduced into the chamber as part of an experiment (e.g., military applications). The tritium will be rapidly pumped from the chamber within minutes, depending on the first wall construction. The target support components will be constructed from composite materials such as carbon-carbon or Kevlar to avoid the activation problem here. Sacrificial diagnostics will also be made of nonactivating materials. The first wall, if solid nitrogen, will not present any significant radiological hazard, nor will polyethylene or frost. The carbon will not have persistent activation products, as long as it is kept pure. A first wall, made of 4 cm of low activating Al (see Table 1) at a 5-m radius, will give 100 mrem/hr about seven days after five hundred 1000-MJ shots spaced one week apart, when the solid nitrogen shield is used (and a value one-third higher when it is not). For a 3-m-radius first wall (2-cm thick), 100 mrem/hr is attained at 7.5 days after the five-hundredth 1000-MJ shot. The longer time is required due to $1/r^2$ effects and the less self-attenuation of decay gamma rays in the thinner aluminum wall. Therefore, we feel that a dose rate of 100 mrem/hr at about the seven-day point is attainable. This rate will allow limited personnel access into the chamber at this time.

EXTERIOR DOSE RATE

Contributions to the exterior dose rate are: the first wall decay, the second wall decay, decay of impurities in the water that are activated, the concrete, and the beam tubes. As mentioned earlier, the dose rate from the first wall (5-m radius) outside the second wall (after 2 m of water) is 1 mrem/hr. The dose rate at this same point from the activated concrete (due to "through-the-shield" neutrons) is only 3 μ mrem/hr. However, the "hot spots" created by the direct shine (leakage) neutrons will contribute ~1 mrem/hr for a 5 at-% boron concentration in the concrete backing the final optics. Our water shield is not borated, both because a saturated solution of boric acid in the water (the simplest way to borate the water) does not introduce enough boron in the water to make a significant difference and because the boric acid or other boron compounds would introduce impurities into the water, most notably sodium. The contribution from activated beam tubes to the overall dose rate at the equator of the sphere is less than 0.1 mrem/hr. Since the goal is to limit the exterior dose rate to ~2 mrem/hr, a conceptual design with the features listed above can feasibly achieve the goal dose rate.

WASTE HANDLING

The expected tritium (liquid (<5 mCi/yr) and gaseous (<100 Ci/yr)) effluents will be many orders of magnitude below the maximum permissible concentrations and total quantities as given in DOE 5480.11. The design objective is to limit the facility's controlled releases to <100 Ci/yr. This value was easily met by the RTNS II facility, which had a somewhat higher tritium throughput. A typical high gain target will contain 10 to 50 mg of tritium. An estimated annual throughput then is 25,000 Ci. For an aluminum first wall, the residual tritium diffusing from the wall after the chamber is pumped out will not present a personnel hazard. For a frost first wall, the Sulzer process for tritium recovery would be used.¹⁶ This can involve cryogenic distillation, combined electrolysis catalytic exchange (CECE), water distillation, electrolysis, gassing, gas storage, and fixation. Here, the tritium could be recycled as it is currently worth \$20,000 per gram. The Sulzer process increases the tritium concentration in one stream while greatly reducing (by up to six orders of magnitude) the concentration in the other stream. After 20 years operation, the total volume for the highly concentrated tritiated water could be only 300 liters. Since this water can be easily doubly contained in certified stainless steel vessels, it presents no significant hazard. Any discharges will be many

orders-of-magnitude below the pertinent standards. At concentrations >1000 Ci per container, the waste must be handled as Class B, with the appropriate containers, etc. We estimate the total cost for waste disposal for the first ten years of operation to be roughly \$2M. By keeping the site tritium inventory to <30 g could ease siting requirements.

CONCLUSIONS

Facility and experimental goals for the LMF can be achieved through careful design of the experimental area. Our calculations show that careful design and material selection can minimize activation and that sufficient shielding can be employed to protect materials prone to activation. For a chamber that uses an Al first wall, ^{24}Na produced by (n,α) reactions with ^{27}Al is the design-limiting radiological hazard. In the case we examined, a 2-m-thick water shield was needed to reduce the dose rate at the diagnostic platform at the 24-hour point to an acceptable level. Reducing the amount of Al (or eliminating it) from the high energy flux would reduce the shielding requirements and allow earlier access to the inside of the chamber. Interestingly, ^{24}Na produced by (n,γ) reactions with ^{23}Na in the concrete walls of the experiment area also present some difficulties. This radiation hazard can be reduced by adding boron to the concrete. Neutron damage to final optics appears to be manageable and the effects of neutrons leaking through beam ports can be mitigated. The waste produced by the facility, predominantly tritium, can be safely and economically managed.

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REFERENCES

1. "General Design Criteria Manual," U.S. DOE, DOE Order 6430.1A (Dec. 7, 1987).
2. "Principles for Limiting Exposure of the Public to Natural Sources of Radiation," International Commission of Radiological Protection, Pub. 39, *Ann. of ICRP* (1983).
3. "Radiation Protection for Occupational Workers," U.S. DOE, DOE Order 5480.11 (draft 4/5/88; effective Jan. 1, 1989).
4. "LLNL ALARA Program," Health and Safety Manual Supplement 33.011, Lawrence Livermore National Laboratory (1986).

5. E. F. PLECHATY and J. R. KIMLINGER, "TARTNP: A Coupled Neutron-Photon Monte Carlo Transport Code," *UCRL-50400 Vol. 14*, Lawrence Livermore National Laboratory (1976).
6. J. A. BLINK, "FORIG: A Computer Code for Calculating Radionuclide Generation and Depletion in Fusion and Fission Reactors," *UCRL-53633*, Lawrence Livermore National Laboratory (1985).
7. J. A. BLINK, R. E. OYE, and J. R. KIMLINGER, "ORLIB: A Computer Code that Produces One-Energy Group, Time- and Spatially-Averaged Neutron Cross Sections," *UCRL-53262*, Lawrence Livermore National Laboratory (1981).
8. J. H. PITTS et al., "Preventing Vaporization and Destructive Shock Waves in ICF Target-Chamber First Walls," these proceedings.
9. C. D. ORTH, "Frost as a First Wall for the ICF Laboratory Microfusion Facility," these proceedings.
10. M. J. MONSLER and W. R. MEIER, "A Carbon-Fiber Brush-Like First Wall for the Laboratory Microfusion Facility," these proceedings.
11. J. H. PITTS and P. R. LONDON, Lawrence Livermore National Laboratory, private communication (1988).
12. A. E. PROFIO, *Radiation Shielding and Dosimetry*, John Wiley and Sons, Inc., New York, NY (1979).
13. D. G. NILSON and J. G. WOODWORTH, "Final Optic Protection Designs for ICF Containment Chambers," these proceedings.
14. M. T. TOBIN, et al., "DOE Requirements for the LMF Experiment Area," *Laser Program Annual Report 87*, *UCRL-50021-87*, Lawrence Livermore National Laboratory, to be published.
15. L. G. DESHAZER, L. E. GORRE and R. C. PASTOR, "Water Free Fused Silica for Laser Windows Made by the Reactive Atmosphere Process," *J. Non-Cryst. Solids* **89**, 189 (1986).
16. C. BOODY, Sulzer Canada, Inc., Toronto, Canada, private communication (June 1988).