27
1-31-79
240719
UC-20

UCID. 18017

Lawrence Livermore Laboratory

PROGRESS REPORT ON THE NEUTRAL BEAM RADIATION 'HARDENING' STUDY

J. D. Lee, R. H. Condit, C. L. Hoenig, T. P. Wilcox, & J. Erickson*

October 1, 1978

* on loan from Grumman

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

CONTENTS

NOTICE
This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Department of Energy, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights.

Page

1. Introduction.....	1
2. Outline of FY'78 Program.....	2
3. Monte Carlo Evaluation of Radiation Environment (T. Wilcox).....	8
3.1 Case 1, Helium-3 Beam Line.....	8
3.2 Case 2, TMR Beam Line.....	16
3.3 Case 3, Standard Hybrid Beam Line.....	35
4. Criteria for Setting Allowable Neutron and Gamma Heating of Cryopanel (J. Erickson).....	36
5. Material Limitations (R. Condit and C. Hoening).....	41
Components & Materials.....	44
Radiation Effects.....	45
Flux effects on bulk conductivity.....	45
Flux effects on dielectric breakdown.....	49
Fluence effects.....	49
The beam dump.....	55
General remarks about materials problems.....	57
6. Summary of Results and FY 79 plans.....	60
Appendix A Units and Conversion Factors.....	64
B Fission vs. Fusion Neutrons.....	67
C Damage in Steels.....	75
D Damage in Al_2O_3	78
E Damage in Epoxy Materials.....	83

1. INTRODUCTION

A neutral beam injector as presently conceived directly views the plasma it is sustaining.* In turn the injector is exposed to the primary fusion neutrons plus secondary neutrons and gammas streaming back up the neutral beam duct.

The intent of this work is to examine representative beam lines to see how performance and lifetimes could be affected by this radiation environment and to determine how unacceptable effects could be alleviated.

Potential radiation induced problems addressed in this report have been limited to:

- overheating of cryopanel and insulators
- gamma flux induced electrical conductivity increase of insulators
- neutron and gamma fluence induced damage to insulator materials.

* A different approach from ours to neutral beam design is to employ bending magnets to get the ion source, accelerator structure and direct converter out of the direct radiation from the fusing plasma. We did not use bending magnets because of the difficulty of achieving well directed beams and because of the difficulty of packing many beams in a small space.

2. Outline of FY'78 Program

Two beam lines were examined, available information on neutron and gamma flux and fluence limitations in candidate materials was compiled, and preliminary criteria for setting neutron and gamma heating limits in cryopanel were established.

The two beam lines examined are modeled from on-going beam line or reactor system studies. The first is loosely based on the general layout of a 120 keV ^3He beam line (Ref. 1).^{*} The second is the 1 MeV ^2H beam line for the Tandem Mirror Reactor (TMR)². A third beam line, a 125 keV ^2H (and 185 keV ^3H) beam line developed for the standard mirror hybrid reactor conceptual design³, is being examined at the present time.

The first case considers a simplistic beam line enclosure, shielded duct and plasma enclosure. The objective for the second case is to realistically treat the geometry of one TMR plug plasma, shielding and beam duct thru the shielding. Unlike other designs, the 1 MeV design does not use insulator columns to support elements at high voltage. It is an attempt to design a beam line with an electrically insulated wall. A fiberglass reinforced epoxy covered with alumina tiles was selected for the wall.

For the third case, a complete beam line system is being analyzed. Principle beam line internals are included to see how they perturb fluxes. In-line shielding is included to determine its effectiveness.

Layouts of these three beam lines are shown in Figures 2.1, 2 and 3. Table 2.1 lists candidate material for beam line components.

^{*} References are listed at the end of each section.

Geometry for Case 1
Figure 2.1

Geometry for Case 2
Figure 2.2

Geometry for Case 3
Figure 2.3

Table 2.1

CANDIDATE MATERIALS FOR NEUTRAL BEAM INJECTOR COMPONENTS

- MAIN VACUUM ENCLOSURE - The walls of the vessel can be 316 stainless steel. It behaves well in a deuterium environment. Support structure external to the walls themselves can be made of 304 stainless steel. All vacuum seals will be either metal (Cu) gaskets for removable connections or welded seams and joints.
- CRYOPANELS - Stainless steel can be used for welded panel casings; Ti 6-4 for supports; chevrons can be of copper; and liquid nitrogen and liquid helium will also be used.
- H.V. INSULATION - The high voltage insulation system will extend from very near the beam centerline to the chamber walls, seeing a total range of radiation levels. The insulating elements will be alumina, epoxy (S-glass, and cast) and sulfur hexafluoride.
- GRID SUPPORTS and ELECTROSTATIC SHIELDS - can be made of titanium, stainless steel, and copper, to extend from the insulating walls toward the beam centerline.
- ION SOURCE - The present source concept used lanthanum hexaboride as an electron emitter. The probable materials also include tungsten, molybdenum, copper, and stainless steel. Alumina will be used for insulators.
- EXTRACTION GRIDS - The grids will be tungsten, molybdenum, or a combination of the two with alumina insulators.
- CESIUM CELLS - Cesium cells (DCX and neutralizing) will use cesium and sodium as working fluids, freon and liquid helium as coolants, copper for heat transfer, 304 or 316 stainless steels for structure, and perhaps an alloy found to be very resistant to Na corrosion at high temperatures. T-222, a hafnium-tungsten alloy, is one such candidate.
- ACCEL GRIDS - Molybdenum or TZM with alumina supports.
- BEAM DUMP - The bending magnet and beam dump will be made of soft magnet iron and water-cooled copper and stainless steel.
- DIRECT CONVERTER - Molybdenum or TZM with alumina supports.

REFERENCES FOR SECTION 2

- 1 A.S. Blum et al., "120 keV Helium-3 Neutral Beam Injector: Design Considerations," UCRL 80904, March 1978.
- 2 R.W. Moir et al., "Preliminary Design Study of the Tandem Mirror Reactor (TMR)," UCRL 52303, July 1977.
- 3 D. J. Bender et al., "Reference Design for the Standard Mirror Hybrid Reactor," UCRL 52478, May 1978.

3. Monte Carlo Evaluation of Radiation Environment

3.1 Case I. Neutron and Gamma-Ray Distributions in a Helium-3 Neutral Ion Injector Beam Line

An analysis has been made of the neutron and gamma-ray flux and heating distributions within that portion of the machine which would contain a Helium-3 neutral ion injector. This work indicates the general levels that would be present. The specific details of the injector itself were not included in the analysis. It is felt that they would perturb the calculated values by only small amounts in the regions of interest.

The reactor was modeled by a five meter diameter sphere surrounded by a three meter thick spherical shield. The shield was assumed to be composed of 20 v/o borated water and 80 v/o stainless steel. A rectangular slit, 20 cm x 100 cm, is cut through this shield to connect to the neutral ion injector. The injector is located in a box 160 x 160 x 500 cm in size, the walls of which are composed of an inner liner of boral (an aluminum matrix containing boron) followed by steel plates and then by a lead loaded cement. Figure 3.1.1 illustrates this geometry and defines the (x,y,z) coordinate system used in presenting the calculated results.

Two different neutron sources were used to bracket the range of possible plasma sizes within the reactor. The first located all fusion neutrons at the sphere's center, a point source. The second distributed the fusions uniformly per unit volume throughout the center five meter diameter sphere. All neutrons were assumed to be born with 14.1 Mev of energy. High energy neutrons were used so the results of the study would be more directly applicable to D-T fusion reactor situations. Both sources were normalized so that the uncollided neutron energy current on the inner wall of the shield (first wall) would be one megawatt per square meter..

Calculations were performed using the MORSE-L¹ Monte Carlo code with a cross section library which contained 30 neutron groups and 15 gamma groups.² Figures 3.1.2 and 3.1.3 present maps of the "fast" neutron flux (neutrons with energies above 0.1 MeV) for the two different

Figure 3.1.1 Calculational geometry.

Figure 3.1.2 Fast neutron fluxes. Point source, neutron/cm²-s.
Dimensions in cm.

sources. The distributed source produces lower results since most of the source cannot be "seen" by the injector. Figures 3.1.4 and 3.1.5 similarly present the total heating in stainless steel while Figures 3.1.6 and 3.1.7 indicate the gamma heating. The gamma heating rate based on a per gram (or Rad) basis, is for all practical purposes, the same for stainless steel as for aluminum oxide. The heating per gram in carbon is about 10% greater than that listed for steel.

Figure 3.1.8 indicates the spectrum of neutrons and gamma-rays in the beam at the point closest to the plasma within the injector's box. Analysis of the problem's results indicate that the average gamma energy within the beam is about 2.6 MeV. Outside of the beam it is about 1.8 MeV. About 60% of the total heating within the beam is caused by neutrons, outside the beam this drops to about 10%.

Several changes could be made to the system in order to decrease the fluxes and heating rates:

Since the neutrons hitting the back wall of the injector box give rise to significant numbers of both neutrons and gamma-rays, levels could be reduced by moving the back wall further away from the injector and the source or by creation of a "get-lost hole" in this back wall. This hole would allow neutrons to travel into a collimated area and would prevent the scattered particles from traveling into the box except along the same beam by which they entered the hole.

Some neutrons are able to hit the sides of the box. Making the box bigger will reduce this.

Altering the composition of the box's walls can reduce the amount of backscatter into the box.

Note that the typical neutron wall loading in a $D-^3He$ reactor is 0.05 MW/m^2 of 2.5 MeV neutrons. Because neutron and gamma effects tend to fall rapidly when outside the region from which one has line of sight to the reactor interior, the sensitive elements are generally located outside this region. Outside, where we have a degraded neutron spectrum, it is probable that simply reducing the exposures listed in the accompanying diagrams by a factor of 20 (to correspond to 0.05 MW/m^2 neutron wall loading) will, to the first order, correctly represent the neutron and gamma effects, despite the fact that the $D-^3He$ neutrons are at 2.5 MeV.

Figure 3.1.3 Fast neutron fluxes. Distributed source, neutrons/cm²-s.

Figure 3.1.4 Total heating rate in stainless steel. Point source, W/kg.
Dimensions in cm.

Figure 3.1.5 Total heating rate in stainless steel. Distributed source, W/kg. Dimensions in cm.

Figure 3.1.6 Gamma heating rate in stainless steel. Point source, Rad/s. Dimensions in cm.

Figure 3.1.7 Gamma heating rate in Al_2O_3 . Distributed source, Rad/s.
Dimensions in cm.

Figure 3.1.8 Neutron and gamma-ray energy-dependent current at $x = 0$, $\gamma = 0$, $z = 550$ cm.

REFERENCES FOR SECTION 3.1

- 1 T.P. Wilcox, "MORSE-L, A Special Version of the MORSE Program Designed to Solve Neutron, Gamma, and Coupled Neutron-Gamma Penetration Problems," UCID-11680, September, 1972.
- 2 T.P. Wilcox, "MORSE Cross Section Library Tapes," UCID-16683, July, 1973.

3.2 Case 2: Neutron and Gamma-Ray Distributions in the TMR 1 MeV Ion Injector

In this case, great detail was given to the end plug plasma region and the shielding around it. But all detail of the injector and its surrounding shield were left as void. The calculated quantities then represent the beam-line values - i.e., the results at places which can "see" directly into the reactor. These will be the highest values within the system and will give "ball-park" numbers to evaluate radiation damage and heating.

Figure 2.2 is a mechanical layout of a TMR beam line.

Figures 3.2.1 through 3.2.6 indicate the geometry used for the calculations. Every part which is not shield is considered to be a vacuum. No definition of materials other than the shield was made since the design was not firm at the time and the shield materials represent the materials that would be used throughout the entire structure. Future work should evaluate different structural materials in an attempt to find those which reduce the radiation environment at the injectors.

Derivation and modeling of the TMR plug neutron source geometry and intensity.

Plasma boundary: The total plasma length is 280 cm. At the axial mid-plane the cross section is circular with a 48 cm radius. At either end the cross section is elliptical, with radii of 24 cm and 44 cm. The two end ellipses are rotated 90° with respect to one another, in the normal Yin Yang fashion. For the present purpose the axial variation of cross section can be taken to be linear, like so:

Figure 3.2.1 Geometrical Description of the 1 MeV Injector

INJECTOR PROBLEM
U 04/07/76 11:01:51

dimensions in cm.

Figure 3.2.2 Geometrical description of the 1 MeV Injector

Figure 3.2.3 Geometrical description of the 1 MeV Injector

INJECTOR PROBLEM
U 04/07/73 11:01:51

Section A-A

Figure 3.2.4 Geometrical description of the 1 MeV Injector

INJECTOR PROBLEM
U 04/07/73 11:01:51

Section B-B

Figure 3.2.5 Geometrical description of the 1 MeV Injector

INJECTOR PROBLEM
U 04/07/78 11:01:51

Section C-C

Figure 3.2.6 Geometrical description of the 1 Mol Injector

INJECTOR PROBLEM
U 04-07-78 11:01:51

Section D-D

Radial variation of source strength: We assumed radially uniform plasma density in TMR; so there is no radial variation of source strength out to the plasma boundary.

Axial variation of source strength: This one is complicated and deserves some explanation. There are four ion populations in the plug: the high energy D injected into the plug (D_{hot}), the 1 MeV T resulting from DD fusion (T_{hot}), and the lower energy D and T intruding from the central cell (D_{warm} and T_{warm}). Thus there are three different DD reactions and four different DT reactions to be considered.

First consider the axial density variations for the four ion populations. D_{hot} has a maximum value of $8.57 \times 10^{14} \text{ cm}^{-3}$ at the plug midplane and falls off to zero at the mirror points at $Z = 140 \text{ cm}$. T_{hot} has a maximum value of $9.51 \times 10^{12} \text{ cm}^{-3}$ at the plug midplane and falls off to zero at the mirror points at $Z = 140 \text{ cm}$. (This calculated density follows from consideration of the $D_{hot}D_{hot} \rightarrow T_{hot}$ reaction and assumption that T_{hot} is contained for the same time as D_{hot} .) D_{warm} and T_{warm} have maximum values of $5.38 \times 10^{13} \text{ cm}^{-3}$ at the $Z = 140 \text{ cm}$ mirror and fall off to zero at the plug midplane. There is no D_{warm} or T_{warm} at negative Z . For the present purpose the axial variation of all 4 of these ion populations can be taken to be linear.

Now consider the reaction rates of the seven different neutron producing reactions. Here we used σv values reported by Miley.¹

Reaction	E used for evaluation	σv
$D_{hot}D_{hot} \rightarrow n(2.45 \text{ MeV})$	878 keV (plug ion energy)	$1.1 \times 10^{-16} \text{ cm}^3/\text{s}$
$D_{hot}D_{warm} \rightarrow n(2.45)$	460 $((878 + 45)/2)$	6.5×10^{-17}
$D_{warm}D_{warm} \rightarrow n(2.45)$	45 (cent. cell ion ener.)	5.4×10^{-18}
$D_{hot}T_{hot} \rightarrow n(14.1 \text{ MeV})$	878	3.7×10^{-16}
$D_{hot}T_{warm} \rightarrow n(14.1)$	460	5.0×10^{-16}
$D_{warm}T_{hot} \rightarrow n(14.1)$	460	5.0×10^{-16}
$D_{warm}T_{warm} \rightarrow n(14.1)$	45	6.6×10^{-16}

¹ G. M. Miley, H. Towner, and N. Ivich, "Fusion Cross Sections and Reactivities," COO-2218-17, University of Illinois (1974).

Multiplying the appropriate density products by the appropriate $\overline{\sigma v}$ values gives the following axial distribution of 2.45 MeV and 14.1 MeV neutron production per unit volume:

Z	2.45 MeV DD Neutrons (n/cm ³ -s)	14.1 MeV DT Neutrons (n/cm ³ -s)	Plasma Cross Section Area (cm ²)
-140 cm	0	0	3320
-126	4.0E11	3.0E10	3680
-112	1.6E12	1.2E11	4050
-98	3.6E12	2.7E11	4440
-84	6.4E12	4.8E11	4810
-70	1.0E13	7.5E11	5200
-56	1.4E13	1.1E12	5600
-42	2.0E13	1.5E12	6000
-28	2.6E13	1.9E12	6410
-14	3.2E13	2.4E12	6820
0	4.0E13	3.0E12	7240
14	3.2E13	4.5E12	6820
28	2.6E13	5.7E12	6410
42	2.0E13	6.5E12	6000
56	1.5E13	6.9E12	5600
70	1.1E13	7.0E12	5200
84	7.1E12	6.7E12	4810
98	4.2E12	6.0E12	4440
112	2.1E12	5.0E12	4050
126	6.8E11	3.6E12	3680
140	1.6E10	1.9E12	3320

Multiplying the neutrons per unit volume by the plasma area and the length increment (7 cm for the end increments, 14 cm for all the others) yields the following values for neutron and power production:

Z Increment	2.45 MeV DD Neutrons Neutrons/s	14.1 MeV DT Neutrons Neutrons/s
-140 to -133 cm	0	1.5E15
-133 to -119	2.1E16	6.8E15
-119 to -105	9.1E16	6.8E15
-105 to -91	2.2E17	1.7E16
-91 to -77	4.3E17	3.2E16
-77 to -63	7.3E17	5.5E16
-63 to -49	1.1E18	8.6E16
-49 to -35	1.7E18	1.3E17
-35 to -21	2.3E18	1.7E17
-21 to -7	3.1E18	2.3E17
-7 to 7	4.1E18	3.0E17
7 to 21	3.1E18	4.3E17
21 to 35	2.3E18	5.1E17
35 to 49	1.7E18	5.5E17
49 to 63	1.2E18	5.4E17
63 to 77	8.0E17	5.1E17
77 to 91	4.8E17	4.5E17
91 to 105	2.6E17	3.7E17
105 to 119	1.2E17	2.8E17
119 to 133	3.5E16	1.9E17
133 to 140	<u>3.7E14</u>	<u>4.4E16</u>
TOTAL	2.4E19	4.9E18

The total DD neutron power is

$$(2.4 \times 10^{19} \text{ s}^{-1}) (2.45 \text{ MeV}) (1.6 \times 10^{-19} \frac{\text{MJ}}{\text{MeV}}) = 9.4 \text{ MW}$$

The total DT neutron power is

$$(4.9 \times 10^{18} \text{ s}^{-1}) (14.1 \text{ MeV}) (1.6 \times 10^{-19} \frac{\text{MJ}}{\text{MeV}}) = 11 \text{ MW.}$$

First wall neutron loadings are estimated by dividing the total neutron power by the surface area of a cylinder large enough to enclose the plasma:

$$\bar{\Gamma}_n = \frac{(9.4 + 11) \text{ MW}}{2\pi (0.48) (2.8) \text{ m}^2} = 2.4 \frac{\text{MW}}{\text{m}^2} .$$

Dividing the neutron power in the central 14 cm of the plug by the area of that cylindrical segment:

$$\begin{aligned} (\Gamma_n)_{\text{max}(\text{?})} &= \frac{(4.1 \times 10^{18})(2.45) + (3.0 \times 10^{17})(14.1) (1.6 \times 10^{19})}{2\pi(0.48) (0.14)} \\ &= 5.4 \frac{\text{MW}}{\text{m}^2} . \end{aligned}$$

The complicated geometrical variation in the source was worked up with a series of boxes of various sizes and shapes and each with a different neutron source strength and energy distribution. Fig. 3.2.7 indicates the axial position of these boxes while Fig. 3.2.8 indicates the outer radial shape for axial segments near both ends and the middle. On Fig. 3.2.8, the solid lines indicate the derived shape.

Fig. 3.2.9 indicates the positions of three detectors used to evaluate the neutron and gamma fluxes. All three are located on the center line of the hole through the shielding. Table 3.2.1 indicates the calculated quantities at these positions. (It should be noted that the energy depositions are those quantities which would exist at the detector positions if a very small quantity of the listed material was placed at the detector. Larger amounts of material would perturb the flux values from those calculated.) Figures 3.2.10 and 3.2.11 indicate the neutron and gamma spectra at detector 2. Similar spectra for the other detectors have the same shapes.

Figure 3.2.7 Axial shape of neutron source

Figure 3.2.8 Radial shape of neutron source at three axial positions

Figure 3.2.9 Detector locations

detector locations

Figure 3.2.10

INJECTOR PROBLEM
ENERGY DEPENDENT FLUENCE FOR DETECTOR NUMBER 2
NEUTRON

Figure 3.2.11

INJECTOR PROBLEM

ENERGY DEPENDENT FLUENCE FOR DETECTOR NUMBER 2

GAMMA

Table 3.2.1 indicates the same quantities as Table 3.2.2 with the uncollided (those particles which came directly from the source without having a collision) contributions subtracted. Table II results would then be the approximate solutions for a detector which was moved sideways far enough so as to not be able to "see" the plasma source directly.

As in the previous case, all calculations were performed using the Morse-L Monte Carlo code with a cross section library which contained 30 neutron groups and 15 gamma groups.

TABLE 3.2.1 - Detector Responses

	Detector Position		
	<u>1</u>	<u>2</u>	<u>3</u>
Distance to center of plasma, M.	6.6	10.6	16.6
Fast (>100 KeV) neutrons, N/CM ² -sec	5.4x10 ¹²	1.7x10 ¹²	6.1x10 ¹¹
Total neutrons, N/CM ² -sec	6.5x10 ¹²	2.0x10 ¹²	7.2x10 ¹¹
Gamma energy flux, MeV/CM ² -sec	2.3x10 ¹²	5.5x10 ¹¹	1.7x10 ¹¹
Gamma flux, γ/CM ² -sec	1.1x10 ¹²	2.5x10 ¹¹	7.8x10 ¹⁰
Energy Deposition in SS304, ergs/cc-sec	1.1x10 ⁶	2.7x10 ⁵	9.0x10 ⁴
Ditto, caused by only gammas	0.8x10 ⁶	1.9x10 ⁵	5.8x10 ⁴
Energy deposition in SS304, ergs/gm-sec	1.4x10 ⁵	3.4x10 ⁴	1.1x10 ⁴
Ditto, caused by only gammas	1.0x10 ⁵	2.4x10 ⁴	7.3x10 ³
Energy deposition in carbon, rad/sec	3.9x10 ³	1.1x10 ³	4.2x10 ²
Ditto, caused by only gammas	0.93x10 ³	0.21x10 ³	6.7x10 ¹
Energy deposition in Silicon, rad/sec	2.3x10 ³	6.6x10 ²	2.4x10 ²
Ditto, caused by only gammas	0.99x10 ³	2.3x10 ²	7.1x10 ¹
Energy deposition in Aluminum, rad/sec	1.7x10 ³	4.7x10 ²	1.6x10 ²
Ditto, caused by only gammas	0.95x10 ³	2.2x10 ²	6.8x10 ¹
Energy deposition in Al ₂ O ₃ , rad/sec	2.2x10 ³	6.0x10 ²	2.1x10 ²
Ditto, caused by only gammas	0.95x10 ³	2.2x10 ²	6.8x10 ¹
Energy deposition in air, rad/sec	3.5x10 ³	1.0x10 ³	3.7x10 ²
Ditto, caused by only gammas	0.94x10 ³	2.1x10 ²	6.7x10 ¹

Table 3.2.2

Detector Responses Without Uncollided Contributions

	Detector Position		
	<u>1</u>	<u>2</u>	<u>3</u>
Fast (>100 keV) neutrons, N/CM ²	2.1x10 ¹²	4.0x10 ¹¹	1.6x10 ¹¹
Total neutrons, N/CM ² -sec	3.2x10 ¹²	8.0x10 ¹¹	2.7x10 ¹¹
Gamma energy flux, MeV/CM ² -sec	2.3x10 ¹²	5.5x10 ¹¹	1.7x10 ¹¹
Gamma flux, γ/CM ² -sec	1.1x10 ¹²	2.5x10 ¹¹	7.8x10 ¹⁰
Energy deposition in SS304, ergs/cc-sec	8.6x10 ⁵	2.0x10 ⁵	6.2x10 ⁴
Ditto, caused by only gammas	8.0x10 ⁵	1.9x10 ⁵	5.8x10 ⁴
Energy deposition in SS304, ergs/gm-sec	1.1x10 ⁵	2.5x10 ⁴	7.8x10 ³
Ditto, caused by only gammas	1.0x10 ⁵	2.4x10 ⁴	7.3x10 ³
Energy deposition in carbon, rad/sec	1.7x10 ³	3.7x10 ²	1.2x10 ²
Ditto, caused by only gammas	0.93x10 ³	0.21x10 ³	6.7x10 ¹
Energy deposition in Silicon, rad/sec	1.2x10 ³	2.8x10 ²	8.6x10 ¹
Ditto, caused by only gammas	0.99x10 ³	2.3x10 ²	7.1x10 ¹
Energy deposition in Aluminum, rad/sec	1.1x10 ³	2.6x10 ²	8.0x10 ¹
Ditto, caused by only gammas	0.95x10 ³	2.2x10 ²	6.8x10 ¹
Energy deposition in Al ₂ O ₃ , rad/sec	1.2x10 ³	2.8x10 ²	8.9x10 ¹
Ditto, caused by only gammas	0.95x10 ³	2.2x10 ²	6.8x10 ¹
Energy deposition in air, rad/sec	1.5x10 ³	3.5x10 ²	1.1x10 ²
Ditto, caused by only gammas	0.94x10 ³	2.1x10 ²	6.7x10 ¹

3.3 Case 3: Beam Line for the Standard Mirror Hybrid Reactor

The objective of Case 3 is to analyze in some detail a complete system composed of the plasma neutron source, structure surrounding the plasma, the neutral beam duct through this structure, beam line walls, beam line components and beam line internal shielding. Figure 2.3 shows mechanical layout of this beam line. This beam line is sized to produce 2600 amps equivalent of 125 keV D^0 or 188 keV T^0 .

The standard mirror hybrid reactor is a 400 MW fusion device producing 600 MWe net plus 2000 kg ^{239}Pu net per year. A complete description of this reactor, including its beam lines, is contained in Ref. 2.3.

The objectives for this case are to address the following questions:

- Is the design adequate as is?
- Are there better positions for the internal shielding?
- What are the effects of using better shielding materials?
- What effects do the beam line walls have?
- What wall materials and wall shape can reduce internal fluxes.
- What effects do the materials which line the duct through the reactor shield have on the results?
- If the present design is not adequate what can be done to make it so?
- Are the results obtained without internal beam line components (like Cases 1 and 2) meaningful?

Work on Case 3 is underway, results are not yet available.

4. Criteria for Setting Allowable Neutron and Gamma Heating of Cryopanels

The overall thermal load on a liquid-helium-cooled cryopanel is the result of several factors:

- radiation heating from exposed higher temperature surface
- enthalpy of pumped gas
- conduction losses through hangers and supports
- neutron and gamma heating
- radioactive decay of pumped gas
- eddy current heating in pulsed magnetic field environments
- heat used to defrost the panels.

Although defrosting characteristics affect the overall LHe cooling system and cryopump performance, it is not a factor when the panel is actually pumping. Thus, heat required for defrosting will not be considered an operating load. Neutron and gamma radiation may even help the defrosting process.

Overall heat load can vary tremendously from design to design. In general, conventional cryocondensing panels (such as used on HVT3, TFTR, MFTF) are expected to remove $\approx 1 \text{ watt m}^{-2}$.¹ However, a conceptual design for a cryotrapping panel² considered heat loads from 12 to 21 watts m^{-2} . These panels are essentially the same as conventional panels, except that argon atoms must be condensed to trap each helium atom. The enthalpy of pumped gas is therefore 30 times higher.

A glance at the radiation heating results for the ^3He and 1-MeV injectors shows that it will be necessary to design future cryopanels for higher thermal loads. Total heating rates in stainless steel for regions in the ^3He injector (Case 1) without line-of-sight to the plasma range from .001 W/kg for the distributed source to 1.5 W/kg for the point source. Resultant heat loads using .75 mm and 1.5 mm stainless sheets are:

$$\begin{aligned} 1000 \text{ W/kg} \times 8000 \text{ kg/m}^3 \times .00225 \text{ m}^3/\text{m}^2 &= .018 \text{ W/m}^2 \\ \text{and } 1.5 \text{ W/kg} \times 18 \text{ kg/m}^2 &= 27 \text{ W/m}^2. \end{aligned}$$

Average off-axis values for the ^3He injector are somewhat lower. Results for the 1-MeV injector (case 2) are higher (14 W/kg just in front of beam duct opening) and indicating that cryopanel in such a design may experience still higher loads.

There are ways to reduce the effect of neutron and gamma fluxes on cryopanel. From the equation above, we see that the thermal load is proportional to the sheet thickness. Cryopanel made of two .25 mm stainless steel sheets have been proposed, and would reduce the 27 W/m^2 above to 6 W/m^2 , with an attendant loss in structural integrity and, possibly, reliability. Shielding can also be used to locally reduce radiation flux, although none of the studies yet completed have actually demonstrated the effectiveness of local shielding. The modeling of the injector for the hybrid mirror reactor will take local shielding into account and give a clue to our ability to tailor flux patterns with local shielding.

There are basically four factors that could limit the allowable thermal load on a cryopanel. First, the transition from heat transfer by nucleate boiling of He to film boiling is believed to begin at $\sim 10^4 \text{ W/m}^2$, and would produce burnouts in the stainless steel sheets. Second, excessive temperature rise of the pumping surface, i.e., $\geq 1/2^\circ\text{K}$, would reduce the pumping speed. Third, for the thinner sheets, the pressure in the helium passages must be kept below the forming pressure of the quilted panels. Finally, the factor which will undoubtedly prove most significant will be the relative costs of installing local shielding (to the extent it works), reducing flux in other ways (such as adjusting the enclosure shape), and paying to refrigerate more helium.

The effect of n and γ heating on the pumping surface temperature is dependent upon the thickness of the stainless steel sheet on the pumping side of the panel.³

$$\Delta T = \frac{\dot{q}''' a^2}{2k}$$

where $\dot{q}'''$ = volumetric heat generation in W/m^3

a = thickness of pumping panel

k = thermal conductivity of SS at 4.2°K = $.23 \text{ W/m}^\circ\text{K}$

Therefore $\dot{q}'''_{\text{max}} = \frac{\Delta T_{\text{max}} \cdot 2k}{a^2}$

for ' a ' = 0.75 mm

$$\dot{q}'''_{\text{max}} = \frac{1/2 \cdot 2 \cdot 0.23}{.00075^2} = 4.1 \times 10^5 \text{ W/m}^3$$

= 52 W/kg .

Total heating must be limited to levels where the cost in both capital and energy needed for refrigeration are acceptable. There is an economic trade off between providing refrigeration and reducing heat loads by shielding or other means. To get an idea of what heating might be acceptable assume that an amount of electrical power equal to 1% of neutral beam power is used to remove the nuclear heat deposited in the

cryopanel. The 1.0 MeV injector (case 2) produces 25 amps equivalent and has approximately 70 m^2 of cryopanel. Therefore if refrigeration efficiency is $1/300$, $0.01 \times (1.0 \times 25)/(300 \times 70 \text{ m}^2) = 12 \text{ W/m}^2$ of average cryopanel nuclear heating is acceptable. So for a cryopanel using a 0.75 mm sheet on the pumping side and 1.5 mm sheet as a backing sheet the volumetric heating to produce 12 W/m^2 is 0.7 W/kg . Therefore 0.7 W/kg is assumed to be an acceptable average volumetric nuclear heating rate in SS cryopanel.

REFERENCES FOR SECTION 4

1. Lee C. Pittenger, Large-Scale Cryopumping for Controlled Fusion, LLL Report No. UCRL-79381 (July 25, 1977).
2. A.S. Blum, et al., 120-keV Helium-3 Neutral Beam Injector: Design Considerations, LLL Report No. UCRL-80904 (March 15, 1978).
3. Lee C. Pittenger, Time Dependent Temperature Rise on Cryopanel Due to Uniformly Deposited Heat Load from TFTR, LLL Engineering Note 76-16 (5/25/76).

5. MATERIAL LIMITATIONS

This section reviews some of the materials problems which we should anticipate in the operation of neutral beam injectors. It is an attempt to scope out only the most basic expectations on materials behavior in the designed environments. However, there is some discussion of possible subtleties of materials behavior. As has been pointed out in a recent survey of special purpose materials needs for fusion reactors¹, there is often insufficient basic data available to make materials choices for certain applications.

Principal attention is directed to effects of radiation on electrical insulators. The discussion is based on two previous reviews of insulator problems in fusion reactors.^{2,3} Radiation effects on structural elements are not considered in detail, because they are expected to be less susceptible to damage than insulators and also, they have received considerable attention elsewhere for applications as first wall materials where the irradiation will be greater than in the injector.⁴ We are concerned that the surface of the beam dump may be seriously degraded and the discussion here is based on a previous analysis.⁵

Where specific examples are used the radiation environment is taken to be that estimated for the 1 MeV injector (case 2), Fig. 5.1. The expected neutron and gamma exposures (case 2) are listed in Table 5.1. The possible x-ray fluxes have not been estimated even though they may have a significant effect on organic materials, on some surfaces, and on secondary electron emission.

The units used for radiation flux, irradiation energy deposited in materials, and other parameters such as electrical conductivity are often a source of confusion, and Appendix A lists those units which are used here. Another problem in discussing radiation effects is our need to extrapolate from available data on fission neutron damage to possible effects of fusion neutrons. In the first case, energies peak around 1 MeV whereas in the second, they peak at 14 MeV. Appendix B summarizes available data on scaling relationships.

Table 5.1: Expected Neutron and Gamma Ray Exposures at Components in the 1 MeV neutral beam injector

Component	Neutrons		Gamma Radiation		
	Flux $\text{cm}^{-2}\text{s}^{-1}$ $\times 10^{-1}$	Fluence - 1 yr* $\text{cm}^{-2} \times 10^{-20}$	Flux W cm^{-2} $\times 10^{+2}$	R s^{-1} ** $\times 10^{-2}$	Fluence - 1 yr* $\text{R} \times 10^{-9}$
1. Vacuum Enclosure	60, max.	2, max.			
2. Cryopanel	3 to 60	2, max.	60	20	50
3. Beam Dump	3 to 10	0.3	5	1	4
4. Bending Magnet	9	0.3	4	1	3
5. Cesium Cells	7	0.2	3	0.8	2
6. Accel Grids					
a. Alumina	6	0.2	3	0.8	2
b. Epoxy	3 to 8	.09 to 0.2	3 to 4	0.8 to 1	2 to 3
7. Ion Source	6	0.2	3	0.8	2

*Fluences are calculated assuming one year of uninterrupted operation.

**The conversion of W cm^{-2} to R s^{-1} has assumed that all materials are like alumina for which $1 \text{ W cm}^{-2} = 2.6 \times 10^{+3} \text{ R s}^{-1}$. This approximation does not introduce an error greater than a factor of 2.

The 1 MeV neutral beam injector, side and top views showing neutron fluences $\text{cm}^{-2} \text{s}^{-1}$. Components are: (1) vacuum enclosure, (2) cryopanel, (3) beam dump, (4) bending magnet not shown but located at the beam dump, (5) tesium cells, (6) accelerator grids, and (7) ion source.

1 MeV NEUTRAL BEAM INJECTOR
GENERAL ARRANGEMENT

Figure 5.1

Components and Materials

Components of the 1 MeV injector are listed below and the candidate materials are mentioned for each. Numbers for the components correspond with those listed in Figure 5.1 and Table 5.1.

1. Main vacuum enclosure The walls of the vacuum vessel can be 316 stainless steel. The radiation behavior of this material is discussed in Appendix C. It behaves well in a deuterium environment. Support structures external to the walls themselves can be made of 304 stainless steel. All vacuum seals will be either metal (copper) gaskets for removable connections or welded joints and seams.

2. Cryopanels These will condense stray atoms in the vacuum chamber. Stainless steel can be used for welded panel casings; Ti 6-4 for supports; copper for chevrons; and liquid nitrogen and liquid helium for heat transfer fluids.

3. Beam dump The beam dump will be made of water-cooled copper and stainless steel.

4. Bending magnets The bending magnets which provide the field to divert the unneutralized atoms from the beam will be made from water-cooled copper, stainless steel, soft magnet iron, and ceramic insulators, perhaps alumina.

5. Cesium cells Cesium cells (double charge exchange and neutralizing) will both use cesium and sodium as working fluids, freon and liquid helium as coolants, copper for heat transfer, 304 or 316 stainless steel for structure, and perhaps an alloy needed for its resistance to corrosion by sodium at high temperatures. A hafnium-tungsten alloy, T-222, is one such candidate.

6. Accel grids Molybdenum or TZM will be used with alumina supports. The radiation behavior of alumina is discussed in Appendix D. The high voltage insulation system will extend from very near the beam centerline to the chamber walls, seeing a total range of radiation levels. From the point of view of fabrication it would be convenient to use epoxy (S-glass and cast) for the primary insulating elements. The irradiation behavior of epoxy materials is discussed in Appendix E. Sulfur hexafluoride would be used outside the chamber walls. Grid supports and electrostatic shields can be made of titanium, stainless steel, and copper, and will extend from the walls toward the beam centerline.

7. Ion source The present source concept uses lanthanum hexaboride as an electron emitter. The probable materials also include tungsten, molybdenum, copper, and stainless steel. Alumina will be used for insulators. Extraction grids will be tungsten, molybdenum, or a combination of the two with alumina insulators.

Radiation effects

Radiation effects are of two types, immediate (flux dependent) and cumulative (fluence dependent). In insulators the immediate effect of irradiation is to create electron-hole pairs. The mobility of these charge carriers contributes to an enhanced electrical conductivity. The cumulative effects of atomic displacements are likely to be complex. Their alterations in the properties of 316 stainless steel, alumina, and epoxy are discussed in Appendices C, D, and E.

Flux effects on bulk conductivity

The increase in electrical conductivity resulting from irradiation is

$$\Delta\sigma = \sigma - \sigma_0, \quad (1)$$

where σ_0 is the intrinsic conductivity and σ is the conductivity during irradiation. It is often taken ⁷⁻¹⁰ to be a linear function of the gamma flux, ϕ ,

$$\Delta\sigma = g\phi, \quad (2)$$

where g is a constant for which typical values are given in Table 5.2. The value of g for each material varies with temperature (but not strongly), material purity, and gamma ray energy, and our tabulated values can only serve as a general guide to likely ranges of values.

Table 5.2:

RADIATION-INDUCED CONDUCTIVITY IN SELECTED INSULATORS

Class	Material	Temp	$g = \sigma/\dot{\phi}$	$\frac{\text{mho/cm}}{\text{rads/sec}}$	Ref.
Crystalline Inorganic	Al ₂ O ₃ (sapphire)	25	7×10^{-14}	8, 11	
	MgO	25	5×10^{-15}	8,12,13	
		800	8×10^{-15}		
	SiO ₂	25	2×10^{-16}	8, 14	

Polycrystalline Inorganic	SiO ₂ (Suprasil)	25	6×10^{-14}	8, 15	
	SiO ₂ (std. grade)	25	2×10^{-16}	8, 14	
	Al ₂ O ₃ (Lucalox)	25	6×10^{-17}	8, 14	
	Glass(Corning 0211)	25	1×10^{-17}	8, 14	
	BaTiO ₃ (ceramic)	25	1×10^{-18}	8, 16	

Organic Polymer	PTFE (Teflon)	25	2×10^{-18}	8,16-18	
		70	5×10^{-18}	9, 10	
	Teflon	38	1×10^{-16}		
	H-film	25	4×10^{-19}	8	
		38	6×10^{-18}		
	Mylar	25	6×10^{-19}	8	
	Epoxy 1478-1*	38	3×10^{-17}	9, 10	
		49	3×10^{-17}	9, 10	
		60	4×10^{-17}	9, 10	

* In these studies on the epoxy, there was no measurable conductivity below:
 = 1.7 rad/s (38 c), - 9 rad/s (48 c),
 and = 75 rad/s (60 c)

For the gamma fluxes of interest to us, the increase in conductivity is very much larger than the intrinsic conductivity so we may write

$$\sigma = \Delta\sigma = g\phi. \quad (3)$$

This increased conductivity in a material subjected to an applied voltage will result in ohmic heating of the material. The usual electrical power-voltage equation, $P = V^2/R$ takes the form

$$P = \sigma \left(\frac{\partial V}{\partial x} \right)^2 \quad (4)$$

where P is the rate of energy input into a unit volume of material under a voltage gradient, $\partial V/\partial x$.

This can be related to the temperature in a specimen subjected to both irradiation and applied voltage. Taking the one dimensional case of a slab insulator cooled on its surfaces and conducting current homogeneously, parallel with the surfaces,

$$P = \frac{\partial J}{\partial x} = \frac{k \partial^2 T}{\partial x^2} \quad (5)$$

where $\partial J/\partial x$ is the gradient in the energy flux, k is the thermal conductivity, and $\partial^2 T/\partial x^2$ is the second derivative of temperature with respect to distance. Integration of this equation gives

$$\frac{\partial T}{\partial x} \Big|_x = \frac{\ell}{2} = \left[\frac{g\phi}{k} \left(\frac{\partial V}{\partial x} \right)^2 \right] \frac{\ell}{2}, \quad (6)$$

the temperature gradient at the surface of the slab, and

$$T_x = 0 = \frac{1}{2} \left[\frac{g\phi}{k} \left(\frac{\partial V}{\partial x} \right)^2 \right] \left(\frac{\ell}{2} \right)^2, \quad (7)$$

the temperature rise in the center of the slab, where the thickness of the slab is taken to be λ , and the center of the slab is taken to be at $x = 0$.

These equations can be used to estimate the effects of irradiation in an insulator. The maximum gamma flux in an insulator taken from Table 1, position 5b, is 100 R s^{-1} . Neutrons will also contribute to the formation of charge carriers, although a major fraction of their energy is dissipated in producing atomic displacements and thermal energy. We are not aware of any literature on the effect of neutrons alone on increasing transient electrical conductivity, but for the moment we will assume as do others³ that it will not add greatly to the effect of gamma irradiation. If we take the highest value of g found anywhere in Table 2, $7 \times 10^{-14} \text{ mho cm}^{-1} \text{ R s}^{-1}$, the conductivity becomes $7 \times 10^{-12} \text{ mho cm}^{-1}$. Typical conductivities for epoxy materials in the absence of irradiation are between 10^{-13} and $10^{-16} \text{ mho cm}^{-1}$.

The above calculation assumed the greatest value for g of any in Table 2. Most values are less by two to three orders of magnitude. It appears that the increased conductivity will likely be smaller than the $7 \times 10^{-12} \text{ mho/cm}$ we have estimated. Even so, we can insert this result in equations 6 and 7 to estimate temperature rises in the center of a flat slab and temperature gradients at the surface of the slab. For a slab 20 mm thick, $\lambda = 10 \text{ mm}$, and a thermal conductivity of $2 \times 10^{-1} \text{ W m}^{-1} \text{ K}^{-1}$ for epoxy it follows that a voltage gradient of 10^6 V m^{-1} gives a power input of 700 W m^{-3} , a temperature rise in the center of about 0.5K, and a surface temperature gradient of 50 K m^{-1} . If the epoxy were used with an alumina filler which has a greater thermal conductivity or if alumina were used, the temperature rise would be less. It appears that the limitation on the use of insulators in a radiation field may be the temperature rises and temperature gradients resulting during irradiation. The temperature rise will increase the intrinsic electrical conductivity and if this rises above the range of 500°C in alumina, for example, it may lead to runaway electrical conductivity with consequent electrical breakdown. Steep temperature gradients generate stress gradients which will result in cracking of insulator parts or separation from their metal contacts.

The interplay between these factors is suggested by equations 6 and 7, but calculations are actually very design specific, material dependent, and complex because thermal and electrical conductivity depend on temperature. Aside from material problems, the device efficiency will be diminished if appreciable fractions of the power consumption are expended in heating electrical resistors.

Flux Effects on Dielectric Breakdown

Direct effects of radiation on dielectric breakdown have not been studied adequately. The only data which we have found that is remotely relevant is shown in Fig. 5.2 for beryllium oxide exposed to 0.6 R s^{-1} . In that range there appears to be no effect. To a first approximation, there is no particular reason to expect trouble at 100 R s^{-1} . However, the combined effects of irradiation, reducing conditions (chamber vacuum and free atomic hydrogen isotopes), and conduction heating are an obvious invitation for growth of instabilities in material behavior, especially organic insulator materials. Growth of any conducting filaments (metal or carbon) would produce steep local fields and a cascading of breakdown processes.

Fluence Effects

The magnitude of the cumulative radiation damage resulting from neutron and gamma irradiation is suggested in Figs. 5.3 (a general, broad guide to radiation effects), 5.4 (damage in plastic materials), and 5.5 (damage in ceramics). The effects of radiation are complex. Changes in thermal conductivity may alter the ability of an insulator to dissipate heat. Increased brittleness may make them more susceptible to thermal shock. Swelling may cause fractures or separation between different materials such as ceramic-metal joints. These may all be more important than any direct effect on electrical conductivity or dielectric breakdown strength. In view of these complexities, we will not attempt a detailed analysis at this time but will merely quote data on generally accepted ranges of practical radiation sensitivity.

Figure 5.2 Electrical breakdown field strength as a function of temperature for BeO insulators with and without an impressed radiation field.

Figure 5.3 Relative effects of radiation on various materials.

Figure 5.4 Relative radiation resistance of organic insulating materials based upon changes in physical properties.

Figure 5.5 Relative radiation resistance of inorganic insulating materials based upon changes in physical properties. The neutron fluence exposures in the neutral beam injector will be around $1 \times 10^{19} \text{ n cm}^{-2}$ for one year of continuous operation. The gamma exposure will be between 10^9 and 10^{10} R .

* Unsatisfactory at $10^{16} \text{ n cm}^{-2}$

A problem in applying available data on neutron damage in fusion reactors is that the neutron energies differ from those in fission reactors for which more data is available. Some scaling factors are suggested in Appendix B and for insulator materials seem to range between 2 and 3 to 1 for the greater damaging efficiency of 14 MeV fusion neutrons.

The behavior of epoxy materials is discussed in detail in Appendix E. In summary, it appears that damage begins to become serious at 10^9 R or 4×10^{17} n cm⁻² (fission), but may not become overwhelming until exposures an order of magnitude higher are experienced. From Table 1 we see that the highest anticipated exposures of epoxy materials (position 6b) during a year of continuous operation of the neutral beam injector would fall below the midpoint of this range for gamma sensitivity but above this range for neutrons. The fact that fusion neutrons are more damaging by a factor two increases our concern about the use of epoxy materials for this application.

The behavior of aluminum oxide is discussed in detail in Appendix D. In summary it appears that damage begins to become serious at 3×10^{10} R or 1×10^{19} neutrons cm⁻². Even under exposures an order of magnitude greater, the damage may still be acceptable. From Table 1 we see that exposures during one year of continuous operation (position 6a) are entering the lower part of this sensitivity span for the accelerator grids. The bending magnets experience a somewhat higher fluence (position 4), because they see the direct, uncollided neutrons from the plasma. However, we conclude that in both positions alumina insulation should be satisfactory.

The foregoing remarks apply to the irradiation behavior of materials at room temperature. At higher temperatures the sensitivity tends to increase, because the irradiation produced defects are no longer frozen in place within the material but can migrate and lead to precipitation of vacancies (formation of voids and swelling of the material) and agglomeration of impurities generated by nuclear transmutations. This tends to be more true of a temperature sensitive material such as epoxy than it does of a ceramic.

If it were to turn out that even aluminum oxide were unsatisfactory, it appears that yet other choices are available. Both silica and yttrium oxide, for example, have lower radiation sensitivity³. At the same time, some "convenience" ceramics such as the machinable glasses would be much less acceptable than alumina, because they contain large amounts of mica. It swells at radiation doses about two orders of magnitude less than does alumina².

The behavior of 316 stainless steel is discussed in Appendix E. The exposures in the neutral beam injector during a year are likely to be well below those which can cause serious material degradation, even at the worst temperatures.

The Beam Dump

A review of materials problems in neutral beam dumps for the fusion energy program is available elsewhere⁴. This study points out that the material flux density in that unneutralized portion of the beam which is diverted to the collector surfaces is high enough to saturate the near surface region with hydrogen isotope atoms in a short span of operation. Since the fluxes will be higher than the range for which experimental data are now available on beam-surface interactions, we can only speculate on the nature of possible problems. A recent survey of special purpose materials for fusion reactor environments¹ reaches this same conclusion, particularly with regard to beam dumps.

Since the energy densities in a dumped beam are high, it is planned to catch the beam at a grazing angle on the collector surface, thereby spreading out the energy over a large surface. However, while this will reduce the energy flux, it will introduce other problems: (1) some reflection of the beam from the first point of impact to a second may deposit energy on unexpected points, (2) sputtering of material tends to be relatively high when the angle between the beam and the surface is small, and (3) hydrogen isotope implantation of the unscattered portion of the beam may disrupt the surface. The dump surface will experience these problems in a environment which includes other factors typical of other components of a fusion reactor interior, i.e. irradiation damage, and steep temperature gradients.

It appears likely that:

1. Stresses in the irradiated surfaces which result from thermal cycling and temperature gradients may well combine synergistically with effects of hydrogen impregnation to produce more rapid deterioration than would either effect alone.
2. Large amounts of hydrogen will be implanted into exposed surfaces and rapid re-emission will be essential to avoid warping, blistering, and spalling.
3. Sputter etching of surfaces may be serious.
4. Scattering of the beam onto other surfaces may deposit moderately augmented energies at those secondary impact points.

We recommended that the material to be directly exposed to the beam should have a good thermal conductivity, be refractory (not vaporize at the occasional hot spots which may occur at spots such as blisters), should probably be roughened to avoid specular beam reflection, should be a metal through which hydrogen diffuses rapidly, and should be a metal which does not form a stable hydride since the concentration of hydrogen may be so high. Low solubility of hydrogen would be desirable to avoid buildup of the ^3He decay product from the tritium isotope. While copper does not meet all these criteria perfectly, it appears to be a best choice.

A principal concern is the ease of re-emission of hydrogen from the impregnated surface. Hydrogen within metals is usually present as atomic hydrogen. For it to escape, these atoms must combine into hydrogen molecules, H_2 , at the surface. This combination is often inhibited on surfaces poisoned with impurities such as oxygen, sulfur, carbon, halogens, and many other elements. We do not know if irradiation will alter surfaces in ways which will inhibit hydrogen atom recombination and hence hydrogen escape. The fact that the surfaces will be hot and the rate of all chemical reactions is enhanced at high temperatures may help the escape of hydrogen.

General Remarks about Materials Problems

The earlier review of electrical insulator requirements for mirror fusion reactors² discusses a broad range of questions which will not be elaborated further here. Perhaps the most immediately important of these is the behavior of insulator surfaces under the environmental conditions which are proposed: irradiation by neutrons, gamma rays and x-rays; attack by stray hydrogen atoms and cesium atoms; possible modification of charge buildup and emission of electrons; and sputter transport of material away from and onto surfaces. These processes may at worst result in insulator breakdown. Short of such catastrophic deterioration, unplanned electric charge buildups may distort the fields within the beam trajectory and reduce the quality of the beam focusing.

A second source of uncertainties about the behavior of materials under the unusual circumstances of fusion reactors are the unknown consequences of combining steep temperature gradients, electric fields, junctions between different materials, and differential responses to irradiation. There has been very little basic research in such areas.

REFERENCES FOR SECTION 5

1. The Fusion Reactor Materials Program Plan, Section IV," Special Purpose Materials," DoE/ET - 0032/4, July 1978.
2. R. H. Condit and R. A. van Konynenburg, "Electrical Insulator Requirements for Mirror Fusion Reactors", UCRL 52098, October 1977
3. A. A. Bauer and J. L. Bates, "An Evaluation of Electrical Insulators for Fusion Reactors", Battelle Columbus Laboratories, BMI 1930, (1974)
4. See, for example, J.L. Scott, "Materials for Fusion Reactors", Metal Progress, July 1978, p.40-49.
5. R. H. Condit and R. G. Meisenheimer, "Materials Problems in Neutral Beam Dumps for Fusion Energy Machines", UCID in preparation.
6. V. A. J. van Lind, J. M. Bunch, and T. M. Flanagan, "The Effects of Ionizing Radiation on Electrical Properties of Refractory Insulators" in Radiation Effects and Tritium Technology for Fusion Reactors, J. S. Watson and F. W. Wiffner, Eds., Oak Ridge National Laboratory Conf-750989, vol II., p. 531 - 547, March 1976.
7. C.L. Hanks and D. J. Hamman, "Radiation Effects Design Handbook, Section 3: Electrical Insulating Materials and Capacitors", National Aeronautics and Space Administration, Washington, D.C., NASA-CR-1787, July, 1971.
8. S.E. Harrison, F.N. Coppage, and A. W. Snyder, "Gamma-Ray and Neutron- Induced Conductivity in INSulating Materials", IEEE Trans., Nuclear Science, NS-10(5), p. 159-167, (Nov. 1963), BRC 22639.
9. J. W. Harrity, "Range of Secondary Electrons in Sapphire", IEEE Trans. Nucl. Sci. NS-17, No. 6, 200 (1970).

10. V. A. J. van Lint, J. W. Harrity, and T. M. Flanagan, "Scaling Laws for Ionization Effects in Insulators", IEEE Trans. Nucl. Sci. NS-15, No. 6, 194 (1968).
11. V. A. J. van Lint and T. M. Flanagan, "Ionization-Induced Conductivity in MgO, Proc. International Conf. on Solid-State Physics Research with Accelerators, Upton, NY, September 25-28, 1967, p. 420 (1968).
12. J. W. Harrity, "Radiation Effects in Dielectric Materials," Final Report on DA 28-043 AMC---2446(E), ECOM report ECOM-02446-F, February 1969.
13. R. C. Hugues, "Transient X-Ray-Induced Conductivity in Single-Crystal and Fused Quartz," Proc. 27th Annual Symp. Frequency Control, June 1973, p. 128.
14. T. M. Flanagan and J. W. Harrity, "Study of Effects of Ionizing Radiation in Capacitors," Final Report on contract DASA-01-C-0014, April 20, 1970.
15. W. H. Sullivan and R. L. Ewing, "A Method for the Routine Measurement of Dielectric Photoconductivity," IEEE Trans. Nucl. Sci. NS-18, December 1971, p 370.
16. R. H. Bartlett, G. A. Fulk, and R.C. Weingart, "Temperature Dependence of X-Ray-Induced Photoconductivity in Kapton and Teflon," to be published in IEEE Trans. Nucl. Sci., December 1975.
17. Monte V. David, "The Effect of Environment on Ceramic Insulators for Nuclear Thermionic Applications" in Proc. Conf. Nuclear Applications of Non fissionable Ceramics, A. Boltax and J. H. Handwerk, Eds. Amer. Ceramic Soc. and Amer. Nuclear Soc., 1967.
18. R. O. Bolt and J. G. Carroll, Radiation Effects on Organic Materials Academic Press, New York, 1963, p. 2.

6. SUMMARY OF RESULTS

Based on the evaluation done to date we think neutral beam injection systems can be designed to operate for reasonable times in the radiation environment they are exposed to.

The radiation effects we think are of prime importance are direct heating of insulators and cryopanel, gamma induced conductivity increase of insulators, and fast neutron and gamma fluence induced damage to insulator materials. Our estimate of design limits for these radiation sensitive components are summarized in Table 6.1.

The local cryopanel heating limit limits temperature rise of panel surface to 0.2°K while the average heating limit is dictated by limiting refrigerator power requirements to 1% of the beam power. These limits are design dependent, especially the average heating limit. The 1 MeV design is used here and is hopefully representative.

Insulator heating is caused by both direct neutron and gamma heating and by V^2/R heating caused by a gamma flux induced decrease in electrical resistance. Over heating of insulators can cause conductivity runaway and ultimately thermal stress failure. We should be able to design insulators to operate successfully at radiation levels encountered in neutral beam lines.

Damage to insulator materials by fast neutrons and gammas are cumulative thus fluence limits must be imposed. Keeping damage to acceptable levels will limit lifetime. Reducing fast neutron and gamma flux in critical components by placement, shielding or both can extend operating lifetime. The tradeoff between lifetime and flux reductions can be used to arrive at optimum designs.

The dose rate limits for insulators listed in Table 6.1 are based on the requirement to limit temperature and temperature gradients in representative insulator configurations to acceptable levels. The temperature and temperature gradient limits used are 500°C and $25^{\circ}\text{C}/\text{cm}$ for alumina and 200°C and $400^{\circ}\text{C}/\text{cm}$ for epoxy. These insulator dose rate limits listed in Table 6.1 are conservative for they assume neutrons are as effective as gammas in reducing electrical resistance.

Results of the Monte Carlo neutron and gamma analyses of the two beam lines considered are summarized in Table 6.2. For case one the values listed are at locations where components are located and are normalized to a source of 14 MeV neutrons resulting in a first wall loading of 1.0 MW/m^2 . Results are expected to scale about linearly with wall loading. For the second case values listed are center line values minus the uncollided contribution. No critical components (insulators or cryopanel) directly view the plasma. Critical components will be located off axis where values are expected to be less.

Comparing the design limits for radiation sensitive component (Table 6.1) to the calculated values for the two beam lines considered (Table 6.2) indicates that these beam lines can operate as exposed to the fusion neutron source. But the estimated minimum lifetime of the epoxy in Case 2 is too short (0.04 - 0.4 years). Useful beam line lifetimes should be achievable without greatly complicating beam line design. Providing some shadow shielding or replacing the epoxy with alumina should increase the endurance of the IMR beam line to an adequate level.

While results thus far are encouraging much more detailed work is needed to substantiate our conclusion that our conceptual linear neutral beam systems can operate in the radiation environment they will be exposed to.

Work for the next fiscal year (FY'79) will concentrate on analyzing more complete designs of beam lines than those available in FY'78. The beam line for the Standard Mirror Hybrid (Case 3) is an example of a beam line design where all major elements (neutron source, duct, beam line enclosure, and beam line components) are specified.

Table 6.1

Preliminary Design Limits for Radiation Sensitive Components:

Cryopanel neutron and gamma heating rate design limits:

Peak 17 w/kg

Average 0.7 w/kg

Insulator gamma plus neutron dose rate design limits:

Epoxy 5000 rad/sec

Alumina 1000 rad/sec

Insulator gamma fluence design limits:

Epoxy 3×10^9 to 3×10^{10} rads

Alumina 3×10^{11} to 3×10^{12} rads

Insulator fast neutron fluence design limits:

Epoxy 10^{18} to 10^{19} n/cm²

Alumina 10^{20} to 10^{21} n/cm²

(Damage caused by a neutron fluence of 4×10^8 n/cm² is approximately equal to damage caused by 1 rad of gamma.)

TABLE 6.2

SUMMARY OF MONTE CARLO RESULTS

	<u>Case I*</u>		<u>Case II**</u>
	Point Source	Distributed Source	
(SS 304)			
Cryopanel heating:			
Peak (w/kg)	1.5	.5	18
Average (w/kg)	.75	.01	3.5
Insulator Peak Dose			
Rates (Rads/sec.):			
Epoxy	not used	not used	330***
Alumina	not calculated	25 (gamma only)	240
Insulator Peak Fast Neutron			
Fluence per year			
(neut./cm ²)			
Epoxy	not used	not used	7.7x10 ¹⁸
Alumina	1.3 X 10 ²⁰	4.8 X 10 ¹⁸	2.2x10 ¹⁹
Insulator Peak Gamma			
Fluence per year (Rads)			
Epoxy	not used	not used	3.5x10 ⁹
Alumina	not calculated	7.9 X 10 ⁸	2.4x10 ⁹

* Normalized to 1.0 MW/m² of 14 MeV neutron through first wall.

** Numbers quoted are center line values minus uncollided contribution, off axis values are expected to be less.

*** Energy deposition in epoxy is assumed equal to energy dep in alumina.

Appendix A: Units and Conversion Factors

Energy deposited by radiation:

$$\begin{aligned}1 \text{ Gray (Gy)} &= 1 \text{ Joule (J)/kg} \\1 \text{ Rad (R)} &= 100 \text{ ergs/g} \\1 \text{ Gy} &= 10^2 \text{ R} \\1 \text{ Gy} &= 6.24 \times 10^{15} \text{ eV/g}\end{aligned}$$

$$\text{Energy:} \quad 1 \text{ eV} = 1.6 \times 10^{-19} \text{ J}$$

Electrical

$$\text{Conductivity} \quad 1 \text{ Siemen(S)} = 1 \text{ mho} = 1/\text{ohm}$$

It may be convenient to relate (approximately) the neutron flux and energy deposition. This depends on the material, neutron energy, and energy distribution. Nevertheless, we have used:

$$\begin{aligned}1 \text{ Gy} &= 4 \times 10^{14} \text{ n m}^{-2} \text{ for fission neutrons}^1 \\1 \text{ Gy} &= 6 \times 10^{15} \text{ n m}^{-2} \text{ for fusion neutrons}^2 \\1 \text{ Gy} &= 1 \times 10^{15} \text{ n m}^{-2} \text{ for fission neutrons in} \\&\quad \text{hydrocarbon materials}^3 \\&\quad \text{see Fig. A1.}\end{aligned}$$

Figure A1. Relationship of rads and reactor radiation in hydrocarbons taken from Ref. A3, 4, and 5. Reactor slow neutrons are subcadmium values monitored with cobalt; gamma rays and higher energy neutrons were also present. Oak Ridge Graphite Reactor hole 19, Brookhaven Research Reactor hole 25, and a British Experimental Pile hole (unknown) are holes in graphite-moderated air-cooled reactors fueled with natural uranium. Materials Testing Reactor hole VH-3 is in a water-moderated and -cooled reactor fueled with enriched uranium.

Appendix A: References

1. C. L. Hanks and D.J. Hamman, "Radiation Effects Design Handbook, Section 3: Electrical Insulating Materials and Capacitors", NASA-CR-1787, July 1971.
2. T. P. Wilcox, "Preliminary Neutron and Gamma-Ray Distributions in the 1 MeV Ion Injector; Memorandum CPG-78-179, September 28, 1978, Lawrence Livermore Laboratory.
3. J. G. Carroll and R. O. Bolt, "Radiation Effects on Organic Materials" Academic Press, New York, 1963, p. 4.
4. V. W. David and R. Irving, "Effects of Nuclear Radiation on Hydrocarbon Oils, Greases, and some Synthetic Fluids", in Conference on Lubrication and Wear, proceedings, Inst. Mech. Eng., London, 1957.
5. H. P. Smith, Ed., "Compilation of Organic Moderator and Coolant Technology", USAEC Report TID-7007, (Pt. 2, Del), Jan. 1957, p. 361.

Appendix B: Comparisons between damage of fission and fusion neutrons.

Data on radiation damage measure in various ways is shown in accompanying figures which compare effects of 14 MeV neutrons with fission neutrons. These results are summarized roughly in Table B1. The ratio in ceramic materials seems to be around 2 except for BK7 glass. In this case, boron has a large cross-section for the $^{10}\text{B}(n, \alpha)^7\text{Li}$ reaction at low neutron energies, and this reaction causes heavy alpha-ray damage in fission reactors but less damage in fusion reactor neutron spectra.

Figure B1. Irradiations of Niobium. Increase in 0.2%-offset yield stress vs. neutron fluence for RTNS, Be(D,n), and LPTR irradiations of niobium.

Figure B2. Irradiations of Copper: Increase in 0.2%-offset yield stress vs neutron fluence for RTNS, Be (D,n), and LPTR irradiations of copper.¹

Figure B3. Irradiation of Aluminum Oxide: Optical Absorption of Sapphire Irradiated and Measured at Room Temperature.²

Figure B4. Irradiations of Magnesium Oxide: Concentration of anion vacancies and absorption coefficient of the 4.95 eV band resulting from neutron irradiations in the RTNS and ORR as a function of dose.³

Figure B5. Irradiations of Pure Silica and Glass: Pure Silica (solid line) and BK7 glass (dashed line) were irradiated in the Livermore LPTR thermal column and core and in the Livermore RTNS for 14 MeV irradiation.⁴ Electron spin resonance was used to measure free electron densities, i.e., spins cm^{-3} .

Table B1: Ratio of 14 MeV neutron damage to fission reactor neutron damage per neutron:

<u>Material</u>	<u>Effect</u>	<u>Ratio</u>	<u>Refs</u>
Nb	Offset yield stress	15	1
Nb-1% Zr	Offset yield stress	9	1
Cu	Offset yield stress	18	1
V	Offset yield stress	17	1
Al ₂ O ₃	Optical absorption	2	2
MgO		2	3
SiO ₂	Free electron density	2	4
BK7 Glass	Free electron density	0.1	4

Appendix B: References

1. J. B. Mitchell, "Exploratory Experiments Comparing Damage Effects of High-Energy Neutrons and Fission-Reactor Neutrons in Metals", Lawrence Livermore Laboratory UCRL 52388, January, 1978.
2. D. W. Muir and J. M. Bunch, "High Energy Proton Simulation of 14 MeV Neutron Damage in Al_2O_3 ", in Radiation Effects and Tritium Technology for Fusion Reactors, J. S. Watson and F. W. Wiffen, Eds. Oak Ridge National Laboratory, ERDA, Conf. 750989, vol II, p. 517-530.
3. Y. Chen, M. M. Abraham, M. T. Robinson, J. B. Mitchell, and R. A. Van Konynenburg, "Production of Point Defects in 14.8 MeV Neutron-Irradiated MgO ", in Radiation Effects and Tritium Technology for Fusion Reactors, J. S. Watson and F. W. Wiffen, Eds. Oak Ridge National Laboratory, ERDA, Conf-750989, vol. II, p. 492-497.
4. M. J. Moran, "Investigation in Silicate Glasses: I Radiation Damage; II Optical Nonlinearity", Lawrence Livermore Laboratory UCRL 52251, Nov. 1976.

Appendix C: Damage in Steels

There is considerable data on radiation damage in steels¹. Data presented here for 316 stainless steel is collected from a review in Ref. 1, for example. The principal points are (1) that swelling sensitivity appears to be greatest in the temperature range 500 - 600° C, Fig. C1, and (2) even in this range the swelling under a fluence of 10^{26} n/m² causes swelling less than 0.1%, Fig C2. This fluence is nearly two orders of magnitude greater than the vacuum chamber walls should experience in a year.

Figure C1. Temperature dependence of swelling in ion bombarded and neutron irradiated type 316 stainless steel.²

Figure C2. In-reactor and ion-produced swellings in Type 316.

Appendix C: References

1. Radiation Damage in Metals, N. L. Peterson and S. D. Harkness, Eds.
American Soc. for Metals, Metals Park, Ohio, 1975.
2. W. G. Johnston, T. Lauritzen, J. H. Rosolowski, and A. M. Turkalo,
"The Effect of Metallurgical Variables on Void Swelling", ref. 1
p. 227.

Appendix D: Damage in Al_2O_3

There are extensive reviews^{1,2} and discussions³ of radiation damage in aluminum oxide and other ceramics. There provide guidance for the data in Table D1. Swelling is shown in Fig. D1. Electrical effects are taken from Ref. 1 and shown in Table D2. From these measurements it appears that serious dielectric deterioration should not occur at our $2 \times 10^{23} \text{ n m}^{-2}$ ($2 \times 10^{19} \text{ n cm}^{-2}$) during a year's operation.

Figure D1. Predicted volume expansion as a function of dose, dose rate and irradiation temperature in Al_2O_3 .

Table D1:

Percent Decrease in Thermal Conductivity of Aluminum Oxide Following
Neutron Irradiation⁴

Fluence n/cm^2 -s	(Temp)	Sapphire	Sintered Alumina
6×10^{20}	(50°C)	-67%	
4×10^{20}	(50°C)		-78%
6×10^{19}	(50°C)	-50%	
3×10^{19}	(50°C)		-42%

Table D2: Electrical Properties of Aluminum Oxide Under Neutron Irradiation.

Type of Oxide	Un-irradiated Oxide	Change Due to Irradiation, Ω/n	Integrated Flux	Temp. of Irrad. $^{\circ}\text{C}$	Remarks	Ref.
Polycrystalline Al_2O_3 in powder and disk form	10^{10} ohms	Decrease to 10^8 ohms during irradiation	$3.5 \times 10^{18} \text{ n cm}^{-2}$	400	Al_2O_3 becomes so highly conducting in a nuclear reactor field that the resistance is essentially independent of temperature. Radiation as well as elevated temperature causes gradual increase in resistivity with time.	6
Highly purified Al_2O_3 (polycrystalline)		Loss of some of the insulation properties	$5.2 \times 10^{20} \text{ n cm}^{-2}$	400	A high instantaneous neutron flux has no significant effect on insulation properties. Some loss in insulation develops during irradiation occurring rapidly at first, then leveling off to a value still high enough not to effect the operation of the thermocouple.	7
Polycrystalline Al_2O_3		No Change	$3.5 \times 10^{11} \text{ n cm}^{-2}$	75	No change found in dielectric constant, dissipation factor, d-c volume resistivity and surface resistivity.	8

Appendix D: References

1. R. A. Wallaert, R. J. Burian, J. B. Melehan, M. Kangilaski, and J. E. Gates, "Effects of Radiation on Ceramic Materials" in Effects of Radiation on Materials and Components, J. F. Kircher and R. E. Bowman, Eds., Reinhold Corp., New York, 1964, p. 277 - 402.
2. A. A. Bauer and J. L. Bates, An Evaluation of Electrical Insulators for Fusion Reactors, Battelle Columbus Laboratories, BMI-1930, 1974.
3. F. W. Clinard, J. M. Bunch, and W. A. Ranken, "Neutron Irradiation Damage in Al_2O_3 and Y_2O_3 ", in Radiation Effects and Tritium Technology for Fusion Reactors, J.S. Watson and F.W. Wiffen, Eds., Oak Ridge Nat. Lab and ERDS Conf-750989m v1k 2p. 498-516.
4. J. H. Crawford and M. C. Wottels, "Radiation Stability of Nonmetals and Ceramics", Second United Nations Int. Conf. on Peaceful Uses of Atomic Energy, Geneva, Switzerland 5:300 (1958).
5. R. S. Wilks, "Neutron Induced Damage in BeO , Al_2O_3 and MgO - A Review", in Nuclear Materials 26: 137-173 (1968).
6. G. W. Monk, "Refractory Dielectrics in a Reactor" WADC-TR-57-599 (1957)
7. N. J. Palladino, WAPD-RES-13 (1954).
8. E. G. Luiden and A.L. Long, USASRDL - TR - 2080 (1951).

Appendix E: Damage in Epoxy Materials

The following discussion of epoxy materials is lifted directly from the review by Hanks and Hamman¹. Comparing the values mentioned here with our exposures suggests that we might be able to use epoxy materials, but we will be pushing the material to do so.

Several, but not necessarily all epoxy resin encapsulants have shown a radiation resistance that is above average for plastics. Polyfunctional epoxy resin and polyfunctional epoxy novolac resin with anhydride or aromatic amine hardeners appear to be the most resistant. Epoxies have withstood neutron and gamma doses up to 1.1×10^{16} n/cm² (E 0.5 MeV) and 1×10^9 rads (c) from a reactor source without serious deterioration.

Information concerning the degradation of an epoxy encapsulant's physical properties indicate that a noticeable darkening in color and a slight loss in weight occurs when these materials are irradiated. Other changes that have also been reported for radiation doses in the ranges up to 10^{16} n cm⁻² and gamma doses up to 10^9 rads include increases in hardness (2 percent), stiffness in flexure (4 percent), and tensile strength (8 percent), stiffness in flexure elongation (6 percent). These changes in physical properties should not be of serious concern in the use of epoxies as encapsulants for electronic components and equipment. However, gamma doses from a cobalt-60 source in excess of 10^8 rads may result in failure in some applications where the epoxy is under stress. The ultimate tensile strength of epoxies has decreased to between 43 and 24 percent of the initial value at 10^9 rads. Compressive strength was essentially unaffected to 10^{10} rads.

¹ C. L. Hanks and D. J. Hamman, "Radiation Effects Design Handbook, Section 3: Electrical Insulating Materials and Capacitors", NASA-CR-1787, July 1971.

The electrical properties of epoxy encapsulants show some variation in radiation tolerance but are generally of adequate stability for use in most electronic circuits at doses up to 10^{16} n cm⁻² and gammas to 10^9 rads. The insulation resistance has decreased by as much as two orders of magnitude during irradiation with a minimum of 1.7×10^{10} ohms being reported. Recovery to near initial value normally occurs within 2 to 4 hours after the irradiation is terminated. Changes in dielectric constant, capacitance, and dissipation factor are insignificant; the latter shows the greatest sensitivity to radiation by increasing approximately one order of magnitude. Polyfunctional epoxy resin and polyfunctional epoxy novolac resin with anhydride or aromatic amine hardeners have retained 90 percent of their initial dielectric strength at 1×10^9 rads. Diglycidylethers of bisphenol A araldite epoxy with an aliphatic amine hardener retained 84 percent of its initial dielectric strength at 6.8×10^8 rads, but was severely damaged physically at 10^9 rads so that the dielectric strength could not be measured.

The above information is representative of the radiation resistance of several epoxy encapsulants, but the reader should be cautioned that one type of epoxy (358-G) was considered as unsuitable following a test because it exhibited large variations in volume resistivity during exposure. The extent of these variations is unknown and this information is included only as precautionary information.

Epoxy-glass laminates have shown little or no degradation in mechanical and/or electrical properties from reactor irradiation to 2×10^{13} n/cm² 1×10^7 rads. Variations in breakdown voltage have been observed at gamma doses below this level, however, with decreases of 15 to 30 percent from preirradiation values at 1×10^7 rads and 50 to 70 percent at 1×10^9 rads. Other electrical properties such as resistivity, dielectric constant, and dissipation factor are not degraded significantly at this total dose.

The tensile strength of epoxy-glass laminates remains unchanged to a gamma dose of 1×10^7 rads but decreases to less than 20 percent of the initial value at 1×10^9 rads. Compressive strength does not change significantly. A total gamma dose of 10^{10} rads will cause an epoxy-glass laminate to become brittle and weak or to lose most of the resin binder.