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DAMAGE FLUX IN CRITICAL ASSEMBLIES*

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ABSTRACT

An important limitation on the design of an imple thermionic reactor is the fast neutron irradiation damage to the insulators, which is expressed quantitatively by the energy weighted neutron flux, or damage flux.

The behavior of the damage flux has been investigated for a set of hypothetical critical assemblies containing different types of fuel and different amounts of moderator. It is shown that for a given fuel, the damage flux is reduced by moderating the neutron spectrum, and for a given amount of moderator the damage flux is reduced by increasing the amount of fuel in the fuel pin. Some effects of damage flux considerations on reactor design are also shown.

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DAMAGE FLUX IN CRITICAL ASSEMBLIES

1. INTRODUCTION

The amount of fast neutron irradiation damage to the components of plasma thermocouples is a matter of concern in the design of thermionic reactors. The component most susceptible to such damage is the insulator. The damage may be measured in terms of a quantity called the "damage flux."¹ It is the purpose of this work to explore the characteristics of critical assemblies, as a function of the amount of moderator, for three different fuel materials. Particular emphasis has been placed on the damage flux, since in any reactor there is some damage flux, and regardless of what other limitations may be placed on the reactor design, the limitation of a maximum allowable damage flux is always present.

The three fuels chosen for consideration, each with fully enriched uranium, are

- a. 40 v/o UO_2 , 60 v/o Mo
- b. 60 v/o UO_2 , 40 v/o Mo
- c. 100 v/o UO_2

The moderator material is Be, and the amount of moderator is expressed in terms of the Be-to- ^{235}U atom ratio, L.

2. MODEL

The critical assemblies whose properties are described below, are spheres of core material surrounded by 4" Be reflectors. This simple geometry was selected in order to avoid adding additional dimensions to the parameter study.

The core material consists of homogenized unit cells. The unit cell consists of two cylinders, end-to-end, containing a single plasma thermocouple and the insulators, cladding, coolant and moderator that would be associated with it, were the cells to be arranged in a hexagonal lattice.

The larger cylinder, 4 cm long, contains the fuel pin, and is referred to below as the "A-section" of the cell. The smaller cylinder, 1.2 cm long, contains no fuel, and is referred to as the "B-section".

The A- and B-sections each consist of several concentric zones. In the A-section, the three innermost zones are the hole in the fuel pin, the fuel, and the pin cladding, or emitter. The outer radius of the emitter has been held constant for all fuels, the inner radius has been varied to give thicknesses appropriate to each fuel. The innermost zone of the B-section is a homogeneous mixture of various hardware components.

In the unit cells containing no moderator ($L = 0$) the outermost zone of each section is the coolant. In the moderated cells, the outermost zones contain Be, the amounts determined by L .

For any particular L value, the amount of Be and the outer radius of the unit cell differ for each of the three fuels. The unit cells are described in Table I and in Figure 1.

The masses of ^{235}U in each of the three fuel pins are:

40 v/o UO_2	13.37 gm
60 v/o UO_2	20.05 gm
100 v/o UO_2	30.32 gm

3. CALCULATIONAL PROCEDURE

The properties of each core material were obtained as follows:

The DTF-IV² multigroup transport theory program was used for a cell calculation of the A-section.

The RCA-6 program was used, with the fluxes from the cell calculation, to produce flux-weighted cross sections for the A-section.

The RCA-6 program was again used to mix, homogeneously, the A-section and B-section and to produce a single set of core cross sections for use in the criticality calculations.

The DTF-IV program was used to find the radius of critical reflected spheres for each core material.

In the DTF calculations, the S⁴ approximation and 18 neutron energy groups were used. The cross sections used are given in Reference 3.

4. NORMALIZATION OF CALCULATED FLUXES

Whether it be in abstract critical assemblies, as used in this paper, or in a proposed reactor design, the fluxes, from which damage flux is obtained, must be calculated.

A calculation does not give true fluxes, but only a set of numbers proportional to them. A series of calculations gives not only flux values which are incorrect, relative to physical systems, but flux values which are, in general, incorrect relative to each other.

In order to obtain a meaningful comparison between the fluxes generated by different calculations it is necessary that the calculations be normalized in a reasonable way.

There exist a large number of normalization techniques. It is safe to say that, in comparing different fuel elements, some reasonable method can always be found which will yield any desired result.

In thermionic reactors, it is most reasonable to normalize to cell performance. In this work, each cell has been assumed to operate at 13% efficiency and to produce 10 electrical watts/square centimeter of emitter surface, which results in 154 electrical watts per fuel pin.

5. DAMAGE FLUX

The insulator damage flux is defined by

$$D = f \int_0^{\infty} E \phi(E) dE$$

where E is the energy (mev), $\phi(E)$ is the average neutron flux per unit energy in the core, and f is the ratio of the flux in the insulator to the flux in the unit cell.

The values of f , obtained from the DTF cell calculation, range from 1.0 in the unmoderated systems to about 1.3 in the more highly moderated ones.

Figure 2 shows the normalized damage fluxes for the three fuels, as functions of L .

a. The damage flux is extremely high for completely unmoderated systems ($L = 0$), but it can be reduced by factors of 2 or 3 by the addition of very small amounts of Be moderator.

b. There is a minimum value of the damage flux. This is the first flight flux, in which it is assumed that fission neutrons pass through the insulator only once, and are then slowed down in the moderator. The value, the same for all three fuels, is about 28×10^{12} n-mev/cm²-sec.

c. The differences in damage flux for the different fuels decrease as more moderator is added.

d. In a thermal reactor the damage flux will be at its minimum value and cannot be reduced by the addition of more moderator.

e. The damage flux decreases as the amount of ²³⁵U in the pin is increased. This apparently paradoxical result is caused by the fact that the normalization assumes the power level in the pin is not affected by the amount of fuel. The power is the product of the fission cross section and the flux. The additional fuel increases the fission cross section; the flux (and hence the damage flux) is reduced accordingly.

This effect is appreciable, but is not as great as the curves imply. Reduction of the amount of ²³⁵U causes an increase in L , hence in going, for example, from the 100 v/o UO₂ curve to the 40 v/o UO₂ curve, one does not follow a vertical path, but rather a diagonal path along a line of constant moderator thickness.

The behavior of the neutron spectrum as a function of L is shown in Fig. 3. The steady decrease of the fast flux peak is the reason for the steady reduction in the damage flux as L increases.

The damage flux is a function of position in a critical assembly. Figure 4 shows curves proportional to the power density, the fast flux, and the damage flux as a function of position for one of the critical assemblies. The shapes of the damage and fast fluxes are the same, with the maximum damage flux at the center of the assembly.

There has been no effort made to flatten the power in any of the calculated assemblies, but it is worth pointing out that power flattening by reduction of ^{235}U density at the center of the core does not solve the damage problem, since reduction of the ^{235}U in the fuel pins tends to increase the damage.

6. CHARACTERISTICS OF CRITICAL ASSEMBLIES

The calculated critical spheres represent minimum size critical assemblies for each of the three fuels as functions of L .

The properties of these systems as functions of damage flux are shown in Figures 5 through 9. The curves show that, given some conventional design parameter, such as core radius or power level, and the implicit parameter - maximum allowable damage flux - there are only certain fuels and L values which will yield a critical spherical reactor which meets both design requirements. These results apply, qualitatively at least, to more realistic geometries.

References

1. R. E. Dahl and H. H. Yoshikawa, Nucl. Sci. Eng. 21, 312-318 (1965).
2. K. D. Lathrop, "DTF-IV, a FORTRAN-IV Program for solving the Multi-group Transport Equation with Anisotropic Scattering," LA-3373, Los Alamos Scientific Laboratory (1965).
3. Lucie D. Connolly, "Los Alamos Group - Averaged Cross Sections," LAMS-2941, Los Alamos Scientific Laboratory (1963).

TABLE I

PROPERTIES OF UNIT CELLS

Section	Region	Material	Outer Radius (cm)
A	Hole	-	(.1000 ¹ (40, 60 v/o Fuel) (.1777 (100 v/o Fuel)
A	Fuel	(40 v/o UO_2 (60 v/o UO_2 (100 v/o UO_2	.6000 .6000 .5619
A	Cladding	W	.6127
A	Cs Gap	-	.6381
A	Collector	Nb	.7397
A	Insulator	Al_2O_3	.7651
B	Hardware	Nb, Al_2O_3 , W, Mo	.7651
A,B	Cladding	Nb	.7905
A,B	Coolant	Na	(.8834 (L = 0) (.8921 (L > 0)
A,B	Cladding	Nb	.9175 (L > 0)
A,B	Moderator	Be	variable (L > 0)

Figure 1

Unit Cell Radii vs L

Figure 2
Normalized Damage Flux
vs L

10¹⁵

10¹⁴

10¹³

Figure 43

Neutron Flux vs Lethargy for various L.
40 v/o UO₂ fuel

L	DAMAGE
0	465 x 10 ¹²
50	176
100	117
150	115

Figure 5
Fission neutron density
Damage flux
Fast neutron flux vs Radius
for 40 v/o UO₂ fuel, L = 100

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Figure 5
Critical Core Radius
vs Damage Flux

Figure 6
Critical Mass vs
Damage Flux

Figure 8
Electric Power
vs Damage Flux

Figure 9
Specific Mass vs
Damage Flux

3

2

1

SPECIFIC MASS
kg/kwe

100 % UO_2

60 % UO_2

40 % UO_2

DAMAGE FLUX

10^{13}

10^{14}

10^{12}