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SUBJECT: HIGH-FLUX REACTOR - MACHINE CALCULATION

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ABSTRACT

By means of 2-group, 3-region calculations the following features are investigated for a flux-trap reactor: finite fuel-layer thickness, moderation in the fuel layer, diffusion constant of the fuel layer, reflector thickness, and flux-trap radius.

A 31-group calculation was used to study simultaneously finite fuel-layer thickness, moderation in the fuel layer, and epithermal absorptions and fissions. The results obtained are not very much different from those obtained for the idealized case in ORNL CF-58-1-4.

HIGH-FLUX REACTOR - MACHINE CALCULATION

In a series of previous memoranda,[1] the problem of high-flux reactors was treated in a fundamental and highly idealized manner. The purpose of the present memorandum is to introduce realistic features, and to see how much the result of the idealized calculation is modified.

Two series of calculations were performed on the ORACLE. The first series used a 2-group, 3-region reactor code obtained from the corresponding 3-group code by a slight modification. The results of this series are tabulated in Tables 1-4. The reactors had spherical symmetry. They consisted of the central flux trap, a fuel shell, and a reflector. The flux trap and the reflector were of the same moderator material, and this moderator was, in successive calculations, assumed to be D_2O, Be, BeO, and graphite of density 2. The fuel shell was assumed in successive calculations to have one of three thicknesses (see below) and for each of the thicknesses three calculations were performed, assuming successively the shell to be non-moderating, to consist of a molten fluoride, and to consist of D_2O. The combination of three shell moderators with three shell thicknesses gives nine cases, each of which was studied for four moderators, a total of 36 calculations. These 36 calculations are discussed next.

The nuclear constants of the moderator material were taken from Table 1 of ORNL CF-57-12-100. For the fuel layer, we assumed throughout that the fuel did not absorb fast neutrons. The macroscopic thermal absorption cross section Σ_a was taken as 5 cm^{-1}, which was regarded as an approximation to a blackness of the shell for thermal neutrons. The approximation is very good, except for the "zero" thickness fuel layers (see below). For the non-moderating fuel we used, in any given case, the same diffusion constant as for the moderator in the flux trap and reflector. For the molten fluoride we assumed the following composition, in mole per cent:

1. W. K. Ergen, Flux Distribution in a Reactor Consisting of a Spherical Shell of Fuel in an Infinite Moderator, ORNL CF-57-12-100 (Dec. 24, 1957).
   Fluxes Obtainable in a Flux-Trap Reactor, ORNL CF-58-1-4 (Jan. 15, 1958).
   High Thermal-Neutron Flux from Fission-Oversimplified Cases, ORNL CF-58-2-127 (Feb. 26, 1958.
   Flux-Trap Reactor with Absorber in the Center, ORNL CF-58-3-27 (Mar. 4, 1958).
   Homogeneous High-Flux Reactor, ORNL CF-58-3-68 (Mar. 31, 1958).

NaF 11.5, KF 42.0, LiF 46.5. To these, of course, UF_4 would have to be added, but the effect of the UF_4 on nuclear properties, except the thermal absorption cross section, was regarded as negligible. Using for the scattering cross section the σ_{fa} values of BNL 325, and for the average cosine of the scattering angle the value appropriate for isotropic scattering, we obtained a microscopic transport cross section of 5.14 barns per average in molecule. The density of the molten fluoride at 700°C is 2.019 which gives the diffusion constant D = 2.193 cm*). For the transfer cross section from the fast to the slow group (microscopic slowing-down power/lethargy gain in from fission to thermal) the value of 0.001 cm^{-1} was used. For the D_2O in the fuel layer, the same diffusion constant and transfer cross section were used as for the D_2O in the flux trap and the reflector.

The radius of the flux trap was chosen as the one given in Table 2 of CF-58-1-4. This is the radius which gives the maximum central flux/power ratio in the idealized case of the infinitely thin, black fuel shell. The calculations were carried out for three different thicknesses of the fuel layers:

(1) "Zero" thickness. This was approximated by a thickness of 0.1 cm.

(2) A thickness which would give 1/10 of the power listed in Table 2 of CF-58-1-4 with a power density of 1 kw/cm^3.

(3) A thickness which would give the full power listed in Table 2 of CF-58-1-4 with 1 kw/cm^3.

The power listed was the power required to produce a flux of 10^{16} n/cm^2sec at the center of the flux trap under the idealized conditions of the infinitely thin, black fuel shell. Thus, were it not for the perturbations introduced into the flux by the thickness and the moderation of the fuel, the thickness given under (2) would make the volume of the fuel shell large enough to give 10^5 n/cm^2sec at the center with 1 kw/cm^3 in the fuel. The thickness given under (3) would, apart from the above perturbation, give the same flux with 100 w/cm^3, or 10^{16} n/cm^2sec at the center with 1 kw/cm^3.

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*) In the calculation for the D_2O moderator in the flux trap and reflector, and molten-fluoride fuel, erroneously a value of D = 2.179 was used. This error was too small to justify recalculation of the problem.

The thickness of the reflector was intended to be "infinite" and a 60-cm-thick reflector was taken as representing the infinite reflector sufficiently closely. However, in the case of Be and BeO, with the non-moderating fuel and the fuel-shell thickness selected as under (3) above, the machine failed to yield meaningful results with a 60-cm-reflector. A 40-cm-reflector was substituted.

In the calculations, all absorptions in the shell were taken as fissions, each leading to 2.48 fission neutrons. The calculations yielded $k > 1$. It was then assumed that the reactor would be made critical by reducing the thermal utilization in the shell to $1/k$. Note that the k given by the ORACLE is different from the "required multiplication factor" ηf used in the previous memos. The greater ηf the smaller k.

The ORACLE yields, among other information, the average thermal-neutron flux ϕ_{22av} in the fuel shell, the flux ϕ_c in the center of the flux trap and the multiplication constant k as discussed in the previous paragraph. The volume V_2 of the fuel layer is known as one of the inputs of the problem. The number of absorptions per sec and per cm^3 of the fuel shell is $\Sigma_a \phi_{22av} V_2$ or $5 \phi_{22av} V_2$. $1/k$ of these absorptions lead to fissions, and each fission gives 2.48 neutrons. Hence the neutrons born in the shell per second number $12.4 \phi_{22av} V_2/k$. The flux in the center of the flux trap per fission neutron emitted in the shell is thus given by $\phi_c k/12.4 \phi_{22av} V_2$.

Table 1 gives the case number, the moderator in the flux trap and the reflector, the material of the fuel layer, the radius of the flux trap, the thickness of the fuel shell, the thickness of the reflector, the average flux in the fuel shell ϕ_{22av}, and the center flux ϕ_c. ϕ_{22av} and ϕ_c are given in the same units. These units are arbitrary, but they cancel out when the ratio ϕ_c / ϕ_{22av} is formed. Table 1 also gives the multiplication constant k, the volume of the fuel shell V_2, and the center flux per fission neutron in the shell, computed as indicated above. In the last column the values obtained in CF-58-1-4 for the idealized case are recorded for each moderator.

It may be seen that the central fluxes computed in this memo for the "infinitely thin" (but not completely black) fuel shell are about the same as those for the idealized case. In fact, the fluxes of this memo are a little higher. Increasing the shell thickness, even without moderation, reduces the central flux. For the largest shell thickness used, the central flux is only about one half of what it is for the "infinitely thin" shell. Moderation in the shell further reduces the central flux. In the thickest shells used replacement of the non-moderating fuel by D_2O decreases the central flux by another factor of 2. Even the moderation in fluorides causes a noticeable reduction in the central flux. Nevertheless, even in the extreme cases the central flux remains of the same order of magnitude as for the idealized case of CF-58-1-4.

It appeared possible that the reduction in central flux caused by the replacement of the "non-moderating" fuel by fluoride was caused by the large diffusion constant of the fluoride. The fluoride fuel used in the above calculation was the one with the largest diffusion constant among the fluorides. For this reason, the cases using fluoride fuel in connection with the medium and large shell thickness were recomputed with D = 1.35 cm instead of the previous value of D = 2.193 cm. The results are shown in Table 2. The cases are identified by a case number of the corresponding case of Table 1. Each case in Table 2 has the same moderator, material of the fuel shell, radius of flux trap, thickness of fuel shell, and reflector thickness as the corresponding case of Table 1. It may be seen that the diffusion constant has very little influence on the result in particular the central flux per fission neutron emitted.

When it was noticed that the 60-cm reflector thickness gave meaningless results with non-moderating fuel and the thick fuel shell, an attempt was made to obtain acceptable results by reducing the reflector thickness. Table 3 shows the results for 30-cm reflector thickness. The radius of flux trap and thickness of the fuel shell were the same in cases BEF and BEG. These inputs were also equal in BEOF and BEOG. Comparison with the 40-cm reflector thickness (case BEG and BEOG) show that the difference is slight.

An attempt was made to retrieve some of the loss in the central flux by changing the radius of the flux-trap. This was carried out for the Be moderator, the 2.6-cm fuel thickness 60-cm reflector thickness and the non-moderating fuel (case No. beginning with BE) and fluoride fuel (case No. beginning with BB). The radius of the flux trap, in mm, varied between 182 and 207 and is indicated by the last 3 digits of the case number. As can be seen from Table 4, the central flux increases as the radius decreases within the limits used. At the smallest radius used the central flux is almost as great as in the idealized case of CF-58-1-4. However, the method of optimization which was used for convenience of computation was not quite fair. Since the fuel layer thickness remains constant, the fuel volume decreases with decreasing flux trap radius and if the total power remains the same the power density is greater for the reactor with the smallest radius which is the one which gives the higher flux.

Furthermore, a few fairly realistic spherical flux-trap reactors were calculated on the ORACLE on the basis of the 31-group code. Here, the moderator in the flux trap was beryllium, the fuel shell consisted of uranium fluoride dissoved in sodium-zirconium fluoride. The inner radius of the shell was varied from 17.2 to 21 cm. The shell thickness was 2.6 cm and the reflector was practically infinitely thick. This calculation took into account the finite shell thickness, moderation in the shell, as well as epithermal absorptions and fissions. The concentration of the uranium was varied to give a multiplication constant of 1.027 ± 0.001. The central flux turned out to be 10^{-3} n/cm^2, per fission neutron emitted by the shell, which corresponds very closely to the flux computed for the idealized case of 58-1-4.

TABLE 1

Case No.	Moderator	Material of the Fuel Shell	Radius of Flux Trap (cm)	Thickness of Fuel Shell (cm)	Thickness of Reflector (cm)	Average Flux in Fuel Shell $\phi_{22\ av}$	Central Flux ϕ_c	Multiplication Constant k	Volume V_2 of Fuel Shell (cm^3)	Central Flux (In Units of 10^{-4} n/cm^2 Per Fission n Emitted)	
										This Computation	Table 2, CF58-1-4
DO 1	D_2O	NM*	23.3	0.1	60	19327	66295	1.4978	682	6.0735	4.999
DO 2	"	F*	23.3	0.1	60	19385	66337	1.4983	"	6.0631	
DO 3	"	D_2O	23.3	0.1	60	19454	66422	1.5011	"	6.0589	
DO 4	"	NM	23.3	3.3	60	624	48638	1.6484	26200	3.9559	
DO 5	"	F	23.3	3.3	60	666	48118	1.6644	"	3.6988	
DO 6	"	D_2O	23.3	3.3	60	759	48757	1.7552	"	3.4686	
DO 7	"	NM	23.3	18.9	60	112	48693	1.6546	262000	2.2215	
DO 8	"	F	23.3	18.9	60	143	48087	1.7568	"	1.8232	
DO 9	"	D_2O	23.3	18.9	60	226	48702	2.0498	"	1.3593	
BE 1	Be	NM	19.2	0.1	60	10283	48129	1.2872	463	10.4904	9.485
BE 2	"	F	19.2	0.1	60	10326	48145	1.2869	"	10.4464	
BE 3	"	D_2O	19.2	0.1	60	10428	48242	1.2936	"	10.4183	
BE 4	"	NM	19.2	2.6	60	418	40240	1.4081	13800	7.9192	
BE 5	"	F	19.2	2.6	60	460	39712	1.4348	"	7.2399	
BE 6	"	D_2O	19.2	2.6	60	568	40061	1.5732	"	6.4839	
BEG	"	NM	19.2	15.0	40	66	40334	1.3393	138000	4.7540	
BE 8	"	F	19.2	15.0	60	102	39669	1.5333	"	3.4986	
BE 9	"	D_2O	19.2	15.0	60	197	40044	1.9429	"	2.3131	
BEO 1	BeO	NM	20.4	0.1	60	10855	51286	1.3555	523	9.8775	9.014
BEO 2	"	F	20.4	0.1	60	10886	51294	1.3491	"	9.8045	
BEO 3	"	D_2O	20.4	0.1	60	11011	51416	1.3616	"	9.8050	
BEO 4	"	NM	20.4	2.5	60	457	42579	1.4714	14500	7.6282	
BEO 5	"	F	20.4	2.5	60	498	42055	1.4952	"	7.0241	

* NM = non-moderating, F = Fluoride

TABLE 1 - Continued

Case No.	Moderator	Material of the Fuel Shell	Radius of Flux Trap (cm)	Thickness of Fuel Shell (cm)	Thickness of Reflector (cm)	Average Flux in Fuel Shell $\phi_{22\ av}$	Central Flux ϕ_c	Multiplication Constant k	Volume V_2 of Fuel Shell (cm^3)	Central Flux (In Units of 10^{-4} n/cm Per Fission n Emitted)	
										This Computation	Table 2, CF58-1-4
BEO 6	BeO	D_2O	20.4	2.5	60	606	42418	1.6216	"	6.3140	
BEOG	"	NM	20.4	14.7	40	72	42658	1.4040	145000	4.6181	
BEO 8	"	F	20.4	14.7	60	107	41997	1.5934	"	3.4688	
BEO 9	"	D_2O	20.4	14.7	60	205	42352	1.9771	"	2.2749	
C 1	C	NM	30.5	0.1	60	10641	52014	1.3445	1169	4.5340	4.169
C 2	"	F	30.5	0.1	60	10659	52017	1.3427	"	4.5205	
C 3	"	D_2O	30.5	0.1	60	10767	52127	1.3489	"	4.5053	
C 4	"	NM	30.5	2.5	60	450	43146	1.4567	31500	3.5767	
C 5	"	F	30.5	2.5	60	483	42807	1.4807	"	3.3629	
C 6	"	D_2O	30.5	2.5	60	591	43182	1.6119	"	3.0167	
C 7	"	NM	30.5	16.5	60	65	43185	1.4245	315000	2.4285	
C 8	"	F	30.5	16.5	60	92	42755	1.5878	"	1.8859	
C 9	"	D_2O	30.5	16.5	60	189	43111	2.0136	"	1.1742	

TABLE 2

Case No.	Average Flux in Fuel Shell	Central Flux	Multiplication Constant	Central Flux (in Units of 10^{-4} n/cm^2 per Fission n Emitted)
DO5D	656	48366	1.6643	3.7777
DO8D	137	48352	1.7546	1.9078
BE5D	456	39825	1.4344	7.3284
BE8D	96	39806	1.5185	3.6596
BEO5D	493	42172	1.4963	7.1125
BEO8D	103	42147	1.5842	3.6095
C5D	478	42929	1.4828	3.4091
C8D	89	42901	1.5879	1.9627

TABLE 3

BEF	68	40339	1.3281	4.6370
BEOF	70	42676	1.3649	4.5967

TABLE 4

BE182	412	37844	1.3836	8.230
BE187	415	39059	1.3962	8.096
BE197	421	41391	1.4197	7.808
BE202	423	42507	1.4306	7.664
BE207	425	43585	1.4411	7.519
BB182	455	37310	1.4110	7.497
BB187	456	38526	1.4233	7.385
BB197	462	40864	1.4463	7.140
BB202	465	41983	1.4570	7.017
BB207	467	43067	1.4672	6.891

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