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General Electric Company
KNOLLS ATOMIC POWER LABORATORY
Schenectady, New York

KAPL-708(DEL)

KAPL STUDY OF HETEROGENEOUS PLUTONIUM POWER REACTORS

R. Ehrlich
E. A. Luebke

May 1, 1953

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KAPL STUDY OF HETEROGENEOUS PLUTONIUM POWER REACTORS

R. Ehrlich and E. A. Luebke

I. INTRODUCTION

During 1952 a large number of calculations were performed to explore the neutron properties of fast, heterogeneous, sodium-cooled, U-235-fueled, power-converter reactors. Previous studies reported in KAPL-ADM-611* involved the use of plutonium as the fuel in a breeder cycle. The change of cycle to be studied was based on economic considerations that with the present raw material market and expected plant construction and operating costs, the converter reactor should produce cheaper plutonium than the breeder. The purpose of this work was to provide a foundation on which to base an engineering design, reasonable cost estimates, and a preliminary pile assembly (PPA) program to firm up the conclusions of the calculations. Since this program has been terminated at KAPL in favor of continued efforts on submarine reactors, the economic, engineering, and PPA programs have not been undertaken. Therefore, this report will be confined to the results of the nuclear investigations of possible reactor types with some recommendations for continued calculations and a supporting zero power assembly program.

Bare-reactor calculations which are relatively easy to perform were done to explore the core behavior as a function of composition. Unfortunately, such calculations do not always predict the behavior of reflected assemblies because of the different energy distribution of neutrons returned to the core by a reflector. Reflected and blanketed core calculations were performed to test the effect of various possible types of reflector and blanket, to check the performance of a number of cores in the best region indicated by the bare reactor study, and to compare the calculated results against experimental results for existing fast assemblies, such as the PPA-11 and the EBR. Summaries of all bare-reactor and reflected-assembly calculations together with an indication of the cross sections used are given in Appendix A and Appendix B. During the course of this work, cross-section changes were made to keep pace with the cross-section measurement program at Los Alamos which was very active during this period. The greater part of the work was done with a set of cross sections entitled the May 1952 set which are tabulated in Appendix C. The cross-section choices will be discussed in Section IX.

II. FUEL-ELEMENT CONFIGURATIONS

Some arrangements of reactor structure which were used in the calculation of nuclear properties are described below. It was not intended that these configurations have complete technical and engineering feasibility. The purpose

*KAPL-ADM-611, "The Plutonium Power Breeder Reactor," by T. M. Snyder, J. E. Burke, and J. F. Flagg, November 13, 1952.

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was to establish some degree of correlation between the nuclear physics results and these configurations. Consideration was given to plate-type, as well as pin-or wire-type, fuel structures. The uranium was used as the metal and as the oxide.

Uranium Oxide in Stainless Steel Pins

In the simple case, one may consider the reactor core as made up of a uniform array of pin-type fuel elements with sodium coolant flowing between them.

<u>Material</u>	<u>Volume %</u>
Na	66
Steel	8.2
U-235 oxide, U-238 oxide, and void	25.8

Because of the large sodium fraction and dilute form of the fuel (density of the oxide taken as 10.9 g/cm^3), such a reactor has a large critical mass. Reference to Case I, Appendix A shows that only 2 mols of U-238 per mol of U-235 can be accommodated without exceeding 400 kg in critical mass. Reactors with so little U-238 have too low an internal conversion ratio* to maintain a reasonably constant reactivity for large fuel burnup.

Critical masses of 400 to 500 kg can be obtained with the oxide pin and with a U-238-to-U-235-atom ratio of 4 or more by reducing the pin spacing to about 0.010 inch. The material composition of a core with this arrangement is

<u>Material</u>	<u>Volume %</u>
Na	28.2
Steel	17.3
U-235 oxide, U-238 oxide, and void	54.5

If the pins are assembled into a simplified form of rod structure consisting of a uniform density of pins with 0.010-in. spacing in a 2-in.-diameter tube with 0.014-in.-wall thickness, then the core composition is

*The internal conversion ratio is the ratio of plutonium produced in the core to U-235 destroyed in the core. The external conversion ratio is the ratio of plutonium produced in the blanket to U-235 destroyed in the core. The conversion ratio is the sum of both the internal and external conversion ratios.

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<u>Material</u>	<u>Volume %</u>
Na	
Flowing	25
Stagnant	<u>9.2</u>
Total	34.2
Steel	
Pin wall	15.3
Tube wall	<u>2.5</u>
Total	17.8
Inside Pin	
U-235 oxide	6.24
U-238 oxide	25.0
Void	<u>16.8</u>
Total	48.0

In this composition, no provision has been made for the small support members required to hold the pin elements in place in the tube. The results of the calculation for this composition are given as Case 08303, Table I, Appendix B. When corrections are made for later cross-section revision and for criticality* ($k \neq 1$), the critical mass is estimated to be 450 kilograms.

Uranium Metal in Clad Plates

It was apparent from previous considerations that a configuration allowing a more concentrated form of fuel is desirable. The dilution of U-235 associated with uranium in the oxide form is increased further by the 4 to 8 atoms of U-238 which may be associated with each U-235 atom in the fuel structure. To increase the attainable U-235 concentration, consideration was given to a plate-type arrangement with alternating corrugated and flat plates.

In such an alloy plate structure uranium is used as the metal. A particular configuration with fuel plates 0.075 in. thick including 0.010-in. steel cladding, with flat plates on 0.3-in. centers, separated by corrugated fuel plates, resulted in the material composition:

*If the reactivity k calculated for a reactor of a given size and composition is different from unity, the size of the reactor core with the same composition which will be critical inside a reflector and blanket of the original composition and thickness is estimated, assuming $\Delta M/M = 5(\Delta k/k)$.

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<u>Material</u>	<u>Volume %</u>	<u>Relative Atoms</u>		
Na	46.2	5.4		
U-235	4	1		
U-238	16	4		
		<u>Fe</u>	<u>V</u>	<u>Zr</u>
Diluent	20.2	9	7.5	4.5
Cladding	13.5	<u>6</u>	<u>5</u>	<u>3</u>
		15	12.5	7.5

For purposes of calculation, vanadium was selected as having typical space-filling properties for possible low-capture alloying elements. The results of the calculations for this composition are given in IBM Cases 08304 through 08306, Table I, Appendix B. The three cases are for no reflector, a 24-cm-thick lead reflector, and 18-cm-thick iron reflector, respectively, inside a uranium blanket. When corrections are made for later cross-section revision and for criticality (k_{eff}), the critical masses are about 600 kg with no reflector and 450 kg with a reflector. On the basis that 1% of the total alloy atoms can be fissioned, an alloy element composed of four U-238 and 14 diluent atoms per U-235 atom is capable of 20% fission. By changing the vanadium-to-U-235-atom ratio in a particular configuration from 14 to 33 (Case 8, Table I, Appendix A), the critical mass for a bare core is increased by a factor of 2.2, and the conversion ratio is decreased from 1.2 to 0.8.

In order to reduce the sodium fraction with a plate-type fuel element, a structure of flat or fluted plates separated by spacers was considered. With 0.075-in.-thick plates including 0.010-in. cladding and separated by 0.045 in., the material composition is

<u>Material</u>	<u>Volume %</u>
Na	34.2
U-235	4
U-238	16
Diluent	25.8
Clad and separators	20

By using the critical mass versus sodium-to-U-235-atom ratio curves given in Figure 1, this reactor will have 15% less critical mass, namely, about 400 kg for a reflected reactor.

Uranium Metal in Stainless Steel Pins or Wires

A structure was considered which had fuel elements of 0.080-in. OD, 0.010-in. cladding thickness, 0.020-in. spacing, assembled in 2 in.-diameter tubes with 0.014-in.-wall thickness; the material composition of such an arrangement is

<u>Material</u>	<u>Volume %</u>
Na	
Flowing	37.2
Stagnant	<u>9.2</u>
Total	46.4
Steel	
Pin wall	12.3
Tube wall	<u>2.5</u>
Total	14.8
Inside pin	
U-235 metal	4.5
U-238 metal	18
Diluent metal	<u>16.3</u>
Total	38.8

This composition is approximately that of the many cases in Table II, Appendix B, having a core molar composition of 1: 4: 10: 5 for U-235: U-238: steel: sodium. The critical mass with the thick lead or iron reflector is about 370 kilograms.

In order to make an intelligent choice between these different possible fuel-element configurations, additional engineering feasibility and radiation-stability studies are necessary. Since it is desirable for economic reasons to fission a large fraction of the fuel in a fuel element between reprocessing operations, the radiation-stability tests should be given high priority in the future program.

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III. CORE COMPOSITION STUDIES*

The effect of changing the U-238-to-U-235-and sodium-to-U-235-atom ratios in the core has been investigated both for bare reactors and for reactors with a 12-cm molten-lead reflector surrounded by a U-238 breeding blanket. One calculation was done with lead as a coolant to compare it with sodium.

Effect of U-238 Content

The variation of critical mass and conversion ratio with U-238-to-U-235-atom ratio is shown in Figure 2. The data are taken from Table I which summarizes the pertinent bare-reactor results and from Table II in Section VI which summarizes the pertinent reflected reactor results. It is to be noted that for an atom ratio of U-238-to-U-235 greater than 4, the critical mass rises sharply, and the total conversion ratio is relatively insensitive to increases in the ratio.

The conversion ratio C.R. for the critical reactor size** (reactivity= $k=1$) is obtained from that given by the multigroup procedure, C.R.', for the noncritical calculated reactor size (reactivity= $k \neq 1$) using $C.R. = k' (1 + C.R.)^{-1}$. This expression is rigorous if all neutrons which escape from the core are converted in the blanket on a 1-to-1 basis, if there is no capture by core materials other than uranium, if core U-238 fissions are neglected, and if the core average α for U-235 is constant. Examination of the neutron balances indicates that the multiplication by fission in the blanket of the neutrons escaping from the core is of such a magnitude that the number of neutrons captured in U-238 in the blanket is approximately equal to the number escaping from the core. The capture in structural material and the fission in U-238 in the core are relatively small, and the core spectrum is insensitive to k for values in the neighborhood of unity. For the reactors studied, the conversion ratios calculated by this expression have been independent of core size for the cases in which more than one core size was calculated. Since the core spectrum is reasonably constant, the internal conversion ratio is independent of k . The difference between C.R. and C.R.' results from the change in the external conversion ratio due to the fact that the leakage from the core into the blanket and the total capture in the core depend sensitively on the assumed core size. For bare reactors, the conversion ratio is computed on the basis of conversion of all escaping neutrons.

When the multigroup calculation gives $k \neq 1$, the critical mass for $k = 1$ is estimated using $\Delta M/M = \gamma \Delta k/k$, with γ varying from 4 to 10, the value for a given type of reactor being determined empirically from cases where calculations were performed for more than one core size.

*Reactor calculations discussed in this and subsequent sections were performed using the May 1952 cross sections of Appendix C except where otherwise stated.

**The reactor is made critical by changing this core radius and by keeping all compositions and reflector and blanket thicknesses unchanged.

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CRITICAL MASS AND CONVERSION RATIOS vs SODIUM to U235 RATIO

U238 to U235 Atom Ratios

12 cm. Lead Reflector

Bare Core

x

o

□

+

5

6

8

21

Conversion Ratio for x
Conversion Ratio for x=5

3

2

1

0

Critical Mass for x
Critical Mass for x=5

Figure 1

x = Sodium to U235 Atom Ratio

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CRITICAL MASS AND CONVERSION RATIOS vs U238 to U235 ATOM RATIO

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TABLE I

BARE-REACTOR CALCULATION RESULTSNormalization to May 1952 Cross Sections and 0% Void

Case Number	Atom Composition					Cross Sections	Calculated Corrected to May σ's				
	U-235	U-238	Fe	Na	% Void		C.R.	M _c	C.R.	M _c	M _c (no void)
12**	1	4	5.1	4.2	3.8	May	1.38	1098	1.38	1098	1015
11	1	4	5.1	8.4	3.8	March	1.39	1840	1.34	1840	1700
25	1	4	5.1	12	0.0	April	1.35	2250	1.33	2500	2500
37	1	4	5.1	1.56	5.25	April	1.45	608	1.43	670	600
13	1	2	5.1	4.2	5	March	1.38	1086	1.33	1086	980
19	1	6	5.1	4.2	2.6	March	1.47	1200	1.42	1200	1140
17	1	8	5.1	4.2	2.03	March	1.50	1643	1.45	1643	1580

Bare-reactor calculations indicate that the critical mass is independent of the iron-to-U-235-atom ratio for ratios below 10. The conversion ratio, however, falls by about 0.05 when the iron content goes from 5 to 10 atoms per U-235 atom. In the reflected-reactor calculations which were performed primarily to evaluate a series of possibilities that may be realized in engineering, both the U-238 and iron were varied relative to U-235. It was estimated that it requires about 2 iron atoms per uranium atom to construct a realistic pin core structure. Thus, 2 iron atoms were added for each U-238 atom added. The difference between the behavior of the conversion ratio for the bare and reflected cases shown in Figure 2 is largely attributable to the change in iron content in the reflected case. The internal conversion ratio (the ratio of the core-produced plutonium to the U-235 destroyed) is proportional to the U-238-to-U-235-atom ratio as one would expect (Table II, Section VI). The proportionality factor is about 0.12. Reflected reactor studies were made only with a U-238-to-U-235-atom ratio of 4 or more. The internal conversion ratio with less U-235 is small enough that large reactivity variations with burnout are to be expected making the control problem difficult. In addition, the bare-reactor calculations indicated that the critical mass does not increase with increasing U-238-to-U-235-atom ratio until the ratio exceeds 4, although subsequent reflected reactor calculations show that this conclusion was not entirely valid. Figure 2 mass variation curves show that, because of the slower spectrum of neutrons returned to the core by the reflector, the critical mass is much more sensitive to U-238 content in reflected reactors. Thus, it is probably also more sensitive to iron content, and part of the larger mass variation in the reflected curves is probably a result of the changes in iron content. The reason for this greater sensitivity in the reflected case is the slower core spectrum resulting from inelastic degradation in the reflector coupled with the fact that the U-238- and iron-capture cross sections increase more rapidly than the fission cross section in going to lower energies. This is illustrated in Figure 3 in which the fissions for a bare and reflected core together with the relative cross sections are compared as a function of lethargy u . (In multigroup calculations, it is convenient to use the lethargy $u = \ln 10^7/E$ instead of the energy E in ev.)

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Effect of Sodium Content

The variation of critical mass and conversion ratio with the sodium-to-U-235-atom ratio is shown in Figure 1. The data are taken from Table I and from Table II, Section VI. The critical mass increases quite rapidly with the sodium content, as one would expect, because of the dilution of the fuel and the structural material. The reflected-reactor mass, however, does not increase as rapidly as the bare-reactor mass since with increasing dilution by sodium in the core, the ratio of the transport cross section of the reflector to that of the core increases, and thus the reflector effectiveness increases. The conversion ratio falls slowly as the sodium content increases.

These illustrations point up the fact that studies of fast reactor behavior based on the core properties alone as determined by bare-reactor calculations can be somewhat misleading when applied to reflected reactors. Thus, bare-reactor calculations should be used only for very general investigations, such as deciding what is the best area for a reflected reactor study.

Lead as Coolant

A bare-reactor calculation (No. 16, Appendix A) was done in which 4.2 atoms of sodium per U-235 atom were replaced by 5.51 atoms of lead per U-235 atom (occupying the same fractional volume as the sodium). The critical mass was reduced by 30 per cent. Thus from the point of view of critical mass, lead is a better coolant than sodium.

IV. REFLECTOR COMPOSITION STUDIES

Iron and liquid lead or lead-bismuth reflectors between the core and blanket were considered both as possible reactor-control elements and as a means of reducing the critical mass. The use of such a reflector lowers the spectrum of neutrons escaping from the core to the breeding blanket as a result of inelastic scattering. This results in a reduced fast fission effect and thus a somewhat lower blanket conversion ratio which must be weighed against any gains due to the smaller core size. In addition, there will be some nonproductive capture in the reflector. A series of IBM calculations was done in which a 45-cm-radius core containing 323 kg of U-235 with a relative composition of 1 mol of U-235, 4 mols of U-238, 10 mols of iron, 5 mols of sodium, and 3.8 volume % void was surrounded by different thicknesses of iron and lead reflectors (Table II, Appendix B). The core plus reflector were surrounded by 45 cm of depleted-uranium blanket.

The iron reflector contains 77% iron and 23% sodium by volume. The lead reflector contains 60% lead (molten), 23% iron, and 17% sodium by volume. The sizable sodium content of the reflector is in part for cooling but is primarily to allow channels for getting the sodium between the reactor core and the primary heat exchangers.

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In order to scope the blanket composition, an array of 1-in.-diameter rods with 0.020-in. spacing between rods and clad with 0.015-in. steel was considered. On a unit cell basis, this has a material composition of

<u>Material</u>	<u>Volume %</u>
Na	12.4
Steel	5.2
U	82.4

for the side blanket.

At the ends of the core, the composition will be different in order to allow space for the sodium to enter and leave the core. If the end blanket is considered as a continuation of the pin fuel elements filled with normal uranium or U-238 oxide, the material composition is

<u>Material</u>	<u>Volume %</u>
Na	34
Steel	18
U	48

for the case with pins spaced 0.010 in. apart and assembled in 2-in.-diameter tubes.

The average blanket composition used in the calculations was

<u>Material</u>	<u>Volume %</u>
Na	23
Steel	10
Depleted U	67

The reactivity k and conversion ratio are plotted against the thickness of reflector inserted between the core and the blanket in Figure 4 (1113731). The conversion ratio drops rapidly as the thickness of the reflector increases. The decrease in conversion ratio is much slower for the lead reflector due to the smaller lead capture cross section. For 24-cm thickness, the iron reflector absorbs 5% of the core source neutrons compared to 1.8% for the lead reflector. The reactivity curves show that the reflecting properties of iron and lead are quite similar. Only at large thicknesses of 20 cm or more does the lead reflector yield a greater reactivity increase than the iron reflector. For both materials, the change in reactivity is small until more than 5 cm of reflector is added. Then the reactivity gain rises rapidly with increasing reflector thickness and eventually begins to level off beyond 30-cm thickness.

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At this point, k has increased by about 0.1. The spherical critical mass is 560 kg with no reflector and 320 kg for a 30-cm lead reflector.

The curves indicate that the use of a 12-cm lead reflector is perhaps a good compromise. The reactivity is increased by about \$5 over the reactivity of the uranium reflected reactor which reduces the critical mass of a 500- or 600-kg reactor by about 100 kilograms. It should provide some \$2 or \$3 for control. The conversion ratio is reduced by 0.08. The 12-cm lead reflector was used in the last set of multigroup calculations which was done for the purpose of finding the best design region for the core (Section VI).

V. BLANKET COMPOSITION STUDIES

Two multigroup calculations were performed with a natural-uranium blanket replacing the depleted-uranium blanket to scope the neutron economy changes resulting from the use of the relatively cheap K-25 waste instead of natural uranium for blanket material. The change was made in the blanket surrounding the 30-cm lead reflected case described in Section IV and in the blanket surrounding an otherwise unreflected core with relative composition 1 mol of U-235, 6.5 mols of U-238, 15 mols of iron, and 8 mols of sodium (Cases 08327 vs 08313 and 08313 and 08323 vs 08322, respectively, Table II, Appendix B). For the reflected reactor, changing to natural uranium raises the conversion ratio from 1.19 to 1.21; for the unreflected reactor, from 1.30 to 1.33. The whole increase in both cases is in the external conversion ratio (that plutonium which is produced in the blanket) and results from the increased U-235 fission source in the blanket.* In the first reactor, the blanket fission source rises from 2 to 5% of the core source, while the leakage source in the blanket (neutrons escaping from core) is 31% of the core source. In the second reactor, the blanket fission source rises from 4 to 7% of the core source, while the leakage source in the blanket is 25% of the core source. Thus, even though the blanket fission source is roughly proportional to the U-235 concentration, the total effect on the blanket of changes in U-235 concentration is small because most of the blanket source is due to neutrons escaping from the core. Furthermore, the build-up of plutonium, after the reactor has been operating for awhile, will decrease differences between the two blankets. Therefore, it seems reasonable to construct the breeding blanket from depleted rather than natural uranium.

VI. CONVERSION RATIO AND POWER COMPARISON OF COMPOSITION STUDIED

The amount of heat which can be generated by the reactor has been calculated in the following fashion. The reactor is assumed to have right cylindrical geometry with the height equal to the diameter. The critical mass for this geometry (and hence the core volume) is taken to be 10% greater than that calculated for spheres of the same composition. The sodium is assumed to flow through the reactor core at an average speed of 30 ft/sec with none of the sodium stagnant. Preliminary engineering studies**indicate that the pumping loss with the sodium flowing at

*If the destruction of U-235 in the blanket were included in the conversion ratio, the conversion ratio would actually decrease by about 0.01 when depleted uranium is replaced by natural uranium.

**Private communication from L. R. Dahl, March 25, 1952.

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REACTIVITY and CONVERSION RATIO vs REFLECTOR THICKNESS

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30 ft/sec is about 8% of the total reactor power for a core with 0.010-in.-diameter pins. The temperature rise across the reactor core is assumed to be 300°F, and the heat capacity of sodium is taken to be 0.3 cal/g/deg/second.

Heat production in the blanket and reflector is ignored. These assumptions lead to the following relationships which were used in the calculations:

$$\text{S.P.} = \text{Specific power (watts/g)} = \frac{0.458 \times 10^3 \times (\text{Core volume } \% \text{ of Na})}{M_c^{1/3} \times (\text{Volume } \% \text{ of U-235})^{2/3}} \quad (1)$$

$$H = \text{Total core heat generation} = \text{S.P.} \times M_c \times 10^{-3} \text{ megawatts}$$

where M_c is the cylindrical mass in kilograms. From the power and breeding point of view, the reactor value V is in some sense proportional to the product of the specific power and the conversion ratio, since this product is proportional to the watts delivered per unit of fuel invested and to the plutonium produced per unit of fuel destroyed. In Table II, the results are summarized in regard to H , S.P., and $V = 10^{-3} \times \text{C.R.} \times \text{S. P.}$

In Figure 5, the value V is plotted against the sodium-to-U-235-atom ratio for reactors with a 12-cm lead reflector for three different U-238-to-U-235-atom ratios. In addition, contours for constant total reactor power are plotted. Contours of constant critical mass are not plotted but roughly parallel those for constant power. Figure 5 indicates that V is increased by removing U-238 or by adding sodium in the area of reactors covered. In this area (reactors under 2000 megawatts, 2000 kg of U-235), the changes in the conversion ratio with composition are relatively small. The main variations in V arise from changes in the specific power. Equation (1) can be rewritten as

$$\text{S.P.} \sim \frac{\text{Na atoms}}{\text{U-235 atoms}} \frac{1}{L(\rho_{25})^{1/3}}$$

where ρ_{25} is the density of U-235 metal (18.6 g/cm³), and L is a characteristic dimension proportional to the cube root of the core volume. The addition of U-238 decreases the specific power (and also V), even though the sodium-to-U-235-atom ratio is unchanged because the quantity L increases as a result of the increased critical size. The addition of sodium increases the specific power (and also V) because the increase in the sodium-to-U-235-atom ratio more than offsets the increase in L . If the U-238-to-U-235-atom ratio is 4 or greater, then Figure 5 indicates that the reactor should be designed to generate at least 1000 megawatts in order to produce the best specific power and conversion ratio. If the U-238-to-U-235-atom ratio falls below 4, there will be large reactivity losses associated with large fuel burnup. A U-238-to-U-235-atom ratio greater than 4 may be justified if they are necessary to maintain relatively constant reactivity with time. In other words, the study shows that the best reactor from the specific power and conversion ratio point of view is obtained by using as low a U-238 and as large a sodium content in the core as is consistent with

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other requirements (such as fuel inventory, controllability, and total power limitations). This analysis has not included provisions for reactor control, fission-product poisoning, or change of composition with burnup.

A comparison of Cases 08316 and 08343 shows that it is desirable to keep the iron content of the core as low as possible. These two cases differ principally in core iron content, and the conversion ratio and V are much larger for the low iron content core (Case 08316).

On the basis of this very preliminary analysis of the calculations, a reasonable design appears to have about 4 U-238 atoms per U-235 atom in the core. A first reactor might have 5 sodium atoms per U-235 atom and produce 500 megawatts with a critical mass of about 500 kg (Case 08329). Dilution with sodium to 9 atoms per U-235 atom results in a more effective reactor with higher specific power generating 1000 megawatts and having a critical mass of 700 kg (Case 08345). The conversion ratio for these two reactors will be reduced by about 0.1 to 1.17 and 1.15, respectively, due to cross-section changes which are discussed in Section IX.

Some of the more detailed properties of the 1000-megawatt reactor are shown in Figures 6, 7, and 8. The data are taken from Case 08345 which has a core composition of 1 mol of U-235, 4 mols of U-238, 10 mols of iron, and 9 mols of sodium. The composition is critical with 700 kg of U-235 when surrounded by a 12-cm lead reflector and a 45-cm depleted-uranium blanket. In Figure 6, the energy dependence of the uranium captures and fissions is shown both in the core and blanket. It is interesting to note that the captures take place at an effectively lower energy than the fissions. In Figure 7, the flux at three positions in the core is plotted as a function of lethargy u . The peak around 150 kev is due to inelastic scattering degrading neutrons from higher energies into this energy region coupled with the small absorption cross section at these lower energies. (At 150 kev, the inelastic cross section has disappeared and capture cross sections have not yet become very large.) Figure 8 shows the spatial distribution of the fission power distribution in both the core and the blanket. The core power curve is seen to be reasonably flat. Table III gives the neutron balance indicating in which energy group, in which region, and in what material the source neutrons produced by the core and blanket fissions are utilized. The numbers are normalized to one core source neutron.

VII. COMPARISON OF THEORY WITH THE PPA-11 AND THE EBR

Fast spectrum critical assemblies which are as large as the proposed power converter do not yet exist in this country; therefore, experimental verification of the theory for this type of reactor lies in the future. The theory has proved very valuable in predicting the behavior of intermediate spectrum reactors such as the SIR or the series of KAPL critical assemblies (PPA-1 through PPA-11) used in designing the WMA reactor.* In this section, the results of the calculations

*For discussions of PPA-2, -3 and -5, refer to KAPL-346, "Multigroup and Adjoint Calculations for SAPL-5," by R. Ehrlich, May 10, 1950 and to KAPL-347, "Multigroup and Adjoint Calculations for SAPL-2 and SAPL-3," by R. Ehrlich, July 17, 1950.

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TABLE II
POWER-CONVERSION PROPERTIES⁽¹⁾ OF FAST REACTORS SURVEYED
May 1952 Cross Sections

IBM Case	Core						Kg of U-235 Cylinder	Reflector		I.C.R.	C.R.	H mega- watts	S.P. watts/g	V
	U-235	U-238	Fe	Na	% Void	Cylinder Radius (cm)		Material	Thickness (cm)					
08329	1	4	10	5	3.80	44.9	485	Pb	12	0.464	1.27	497	1025	1.30
08344	1	4	10	7	3.19	50.8	585	Pb	12	0.468	1.26	745	1270	1.60
08345	1	4	10	9	2.74	56.6	700	Pb	12	0.471	1.25	1025	1465	1.83
08340	1	6	14	5	3.19	58.1	880	Pb	12	0.708	1.29	700	797	1.03
08338	1	6	14	7	2.74	65.5	1080	Pb	12	0.713	1.27	1068	987	1.25
08339	1	6	14	9	2.41	73.6	1350	Pb	12	0.719	1.25	1522	1127	1.41
08347	1	8	18	5	2.75	89.0	2700	Pb	12	0.962	1.28	1405	520	0.667
08343	1	8	18	9	2.15	110	4045	Pb	12	0.972	1.24	3040	754	0.935
08313	1	4	10	5	3.8	40.5	350	Pb	30	0.47	1.19	400	1140	1.35
08315 ⁽²⁾	1	4	10	5	3.8	40.8	360	Fe	30	0.46	1.05	415	1135	1.19
08324	1	4	10	5	3.8	48.9	620	None		0.46	1.27	590	945	1.29
08328	1	4	6.3	3.1	3.8	38.4	425	None		0.45	1.37	320	750	1.03
08322	1	6.5	15	8	0	81.2	1900	None		0.77	1.30	1730	910	1.18
08316	1	8	8	8	0	79.9	1960	Fe	18	0.95	1.29	1810	925	1.19

(1) Masses, ratio, and conversion ratio extrapolated to $k = 1$. All reactors have 45-cm depleted-uranium blanket.

(2) Reflector thickness extrapolated from 24 to 30 cm.

ENERGY DISTRIBUTION OF FISSIONS AND CAPTURES IN CASE 08345

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Figure 8

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TABLE III
ABSORPTION BREAKDOWN FOR CASE 08345⁽¹⁾

Group	u	Core							Reflector			Blanket						
		F 235	F 238	C 235	C 238	Other Captures	In- Elastic	Escape	Captures	In- Elastic	Escape	F 235	F 238	C 235	C 238	Other Captures	In- Elastic	Escape
1	0.5-1	0.0038	0.0073	0.0002	0.0002	0.0000		0.0195	0.0000		0.0053	0.0000	0.0015	0.0000	0.0000	0.0000	0.0000	0.0000
2	1-2	0.0196	0.0253	0.0011	0.0033	0.0002	0.0163	0.0846	0.0000	0.0040	0.0319	0.0000	0.0058	0.0000	0.0008	0.0000	0.0013	0.0001
3	2-3	0.0262	0.0021	0.0016	0.0106	0.0003	0.1627	0.00672	0.0001	0.0358	0.0501	0.0001	0.00007	0.0000	0.0039	0.0000	0.0177	0.0001
4	3-3.75	0.0453		0.0038	0.0202	0.0012	0.3080	0.0526	0.0007	0.0616	0.0706	0.0004		0.0000	0.0131	0.0000	0.0523	0.0003
5	3.75-4.5	0.0800		0.0097	0.0401	0.0030	0.2323	0.0612	0.0016	0.0055	0.0452	0.0016		0.0002	0.0664	0.0005	0.1070	0.0033
6	4.5-5.5	0.1070		0.0188	0.0643	0.0044	0.0863	0.0464	0.0025		0.0547	0.0027		0.0005	0.1338	0.0010	0.0332	0.0052
7	5.5-7	0.0726		0.0186	0.0542	0.0024		0.0005	0.0019		0.0563	0.0016		0.0004	0.0970	0.0004		0.0013
8	7-10	0.0120		0.0045	0.0070	0.0003		-0.0033	0.0002		0.0072	0.0002		0.0000	0.0119	0.0000		0.0000
<u>Total</u>		0.3665	0.0347	0.0582	0.1999	0.0118	0.8056	0.3287	0.0070	0.1069	0.3212	0.0066	0.0080	0.0011	0.3269	0.0019	0.2115	0.0103

⁽¹⁾ Under columns entitled Core and Blanket, F = fissions and C = captures.

of the theory to the EBR and the PPA-11 are discussed. The EBR has even a faster spectrum than the proposed converter because of the high density of U-235 in the core. But, as a result of this high density, the core is very small (corresponding to a sphere of under 10-cm radius), thus the question of the validity of diffusion theory arises. The PPA-11 is larger, (24-cm-radius core with 114 kg of U-235) but it has a beryllium reflector and a small amount of beryllium in the core, resulting in a somewhat lower spectrum than the proposed converter. The calculated spectrum of core fissions in the PPA-11, the EBR, and Case 08345 (Table II, Appendix B) are compared in Figure 9. In both these instances, the theory gives reasonable agreement with experiment, and hence it is reasonable to assume that the properties predicted for the larger power converters are, in general, correct. Relative differences from one composition to the next are probably more accurate than the over-all normalization which depends on cross sections which are still rather uncertain (Section IX).

The composition of the PPA-11 is given in Table IV.

TABLE IV
COMPOSITION OF THE PPA-11

Constituent	Core		Reflector	Hot Breeder	Cold Breeder
	Relative Mols	Volume %	Volume %	Volume %	Volume %
U-235	1	10.56	-	4.04	100 % U _{nat}
U-238	0.053	0.56	-	55.81	
Be	1.74	7.10	65.56	4.71	
Na	2.03	40.00	13.86	16.18	
Fe	4.74	28.00	8.55	10.58	
Al	0.77	6.42	6.50	6.58	
Void		7.37	5.53	5.60	

The assembly was critical with 123.7 kg of U-235 contained in a core structure consisting of 43.4-cm-long rods filling the first five rings of a 9-ring hexagonal matrix having 2-in. cross-flats unit cells. The core was surrounded by 1 1/2 rings of reflector rods (about 7 cm thick) and 2 1/2 rings of hot breeder rods (about 11 cm thick). The end reflector was 9.7 cm thick and the end hot breeder was 7.7 cm thick. Outside the 9-ring matrix, there were about 30 cm (26,000 kg) of natural-uranium blanket. For the purpose of calculation, this reactor was replaced by a spherical core with a 24-cm radius containing 114 kg of U-235 surrounded by 8-cm-thick shells of reflector and hot breeder. The natural-uranium blanket was omitted since the IBM machines were programmed to handle only up to three-spatial regions. This does not appreciably affect the over-all criticality, but it does make a comparison of theory with experiment for the hot breeder region difficult. The calculation (Case No. 08320, Table III, Appendix B) gave $k = 1.003$. Allowing 10% in mass for the difference between the spherical and hexagonal geometries gives a calculated mass for the PPA-11 of 124 kg in excellent agreement with experiment. The

U-238 cross sections are not sensitively involved in this calculation since there is very little U-238 in the core. Unfortunately, the PPA-11 had to be disassembled before any extensive experimental studies of its detailed properties could be carried out. Preliminary measurements were made on the U-235 and U-238 fission rates in the blanket.* The measured ratio of U-238-to-U-235 fissions in the hot breeder was 1.44. The calculated ratio is 1.53. The calculated total fission rate in the hot breeder is lower than the measured value by about a factor of 2 1/2 because of the absence of the cold breeder in the calculation (also the actual hot breeder is somewhat thicker than was assumed in the calculations).

The composition of the EBR was taken from ANL-WHZ-295** with 10 kg of fuel added to the core at the expense of NaK to correspond to the observed critical mass of 50 kilograms. (The actual fuel loading of the EBR is 51.7 kg of U-235, but the concentration is higher in the outside of the core, and 50 kg is assumed to be the actual mass for uniform fuel loading.) For the spherical model, the EBR core composition was put in a 10% smaller volume (9.8-cm-radius sphere). This core was surrounded by a 14-cm reflector (NaK-cooled breeding blanket) and a 10-cm blanket (air-cooled breeding blanket). The composition used is given in Table V.

TABLE V

COMPOSITION OF SPHERICAL MODEL OF THE EBR

<u>Material</u>	<u>Core Volume %</u>	<u>Reflector Volume %</u>	<u>Blanket Volume %</u>
U-235	61.3		
U-238	4.2	49.1 ⁽¹⁾	56 ⁽²⁾
NaK	20.1	33.4	18
Steel	14.4	17.5	15
Void	0.0	0.0	11

(1) Natural uranium.

(2) The top and bottom NaK-cooled reflector more than 14 cm from the core was considered to be in the blanket.

By using the ordinary diffusion theory, the calculation gave $k = 0.99$ (Case No. 08325, Table III, Appendix B). The calculated conversion ratio is 0.59 (0.006 in the core, 0.40 in the reflector, and 0.186 in the blanket). There are 0.44 neutrons per core source neutron escaping from the blanket. In the actual EBR, some of these neutrons are reflected to the blanket by the graphite regions and by the thicker parts of the uranium blanket which were not included in the spherical mode. If all the escaping neutrons are assumed to be

*Private communication from J. A. Bistline.

**Private communication from W. H. Zinn to T. M. Snyder, April 20, 1951.

CORE AVERAGE FISSION SPECTRUM FOR THE EBR , PPA-11 ,and PFR Case 08345

reflected to the blanket and captured in U-238, the conversion ratio becomes 1.60. This is clearly an upper limit. The ratio of U-238-to-U-235 fissions in the core on an equal atom basis falls from 0.115 at the center to 0.067 at the core reflector interface. Measured values of 0.146 at the center and 0.094 at the edge are reported in ANL-4811,* indicating that the U-235-inelastic-to-fission-cross-section ratio used in the calculation may be somewhat too large. In the reflector, the calculated ratio of U-238-to-U-235 fission power is 4.18; ANL-4811 reports 4.07 for the inner blanket. The calculation gives 86% of the fissions in the core; ANL-4811 reports 82 per cent. Thus, the theory and experiment appear to be in general agreement except that the reported measurements of the average α for U-235 in the core indicate that this quantity is perhaps twice what we have assumed in our calculations. This will be discussed further in Section IX.

A second calculation was done in which a crude attempt was made to apply integral theory corrections (Case No. 08330, Table III, Appendix B). This gave $k = 0.94$. Corrections were applied to the diffusion coefficients based on some rough calculations reported by H. Hummel in ANL-MH-57**. The decrease in reactivity resulting from integral theory is surprising. It results from the fact that although the corrections to the core diffusion coefficients were negative, the corrections used were positive and large in the reflector and blanket. (The integral-theory correction to the diffusion coefficient is positive in a nonmultiplying medium.) The loss of reflection more than offsets the smaller leakage from the core. The diffusion-theory boundary conditions (continuity of flux and current) were used in this calculation. Hummel points out that the Serber boundary condition on the flux is necessary in the high-energy groups in order to get a correct result because the diffusion boundary condition overestimates the leakage. We have not tried to adapt the Serber boundary condition to the multigroup methods. Further investigation is required to determine the actual modification of the diffusion theory which would result from a consistent application of integral theory. Nevertheless, since it appears that a consistent application of the integral theory would raise the calculated reactivity above 0.94, the difference between integral theory and diffusion theory is relatively small even in the case of the EBR. Thus integral theory corrections should be unimportant for larger assemblies such as the PPA-11 and the power converters discussed in this report.

Thus, in general, theory and experiment appear to be in agreement. A proposed schedule for testing the theory for larger assemblies with bigger critical masses will be discussed in Section VIII.

*ANL-4811, "Fission Rate Measurements in the Experimental Breeder Reactor," by S. Untermyer, April 18, 1952.

**Private communication from H. Hummel to H. Hurwitz, Jr., July 11, 1952.

VIII. PPA PROGRAM

The PPA program envisaged by KAPL to firm up the multigroup-design calculations for the full-scale 500- or 1000-megawatt reactor would begin with the last fast assembly of the WMA power-breeder program, namely, the PPA-11. The PPA-11 had a sodium-cooled, enriched-uranium core with a small amount of beryllium as described in Section VII. The next step would be to replace the core sodium and beryllium by U-238 to enlarge the core to six rings, and to replace the beryllium reflector by a lead or iron reflector. This would mock up the larger proposed converter with the sodium left out. The density transformation achieved by omitting sodium would permit an assembly with less than 200 kg of U-235 before going to the proposed larger mass converters. The chief neutron behavior features of the larger assembly would be present. An IBM calculation (Case No. 08321, Table III, Appendix B) indicates that the critical mass would be 190 kg of U-235 with a core composition of 4 mols of U-238 per U-235 and with 40% steel and 5% void (interstices) by volume. An extrapolation from the PPA-10 and the PPA-11 composition to the proposed PPA composition on the basis of the PPA-10 and the PPA-11 danger coefficients indicates an expected critical mass of about 190 kg for the new PPA in agreement with the multigroup result*. On the basis of the check on theory obtained from and with the use of extrapolations based on danger coefficients made in this intermediate PPA it should be possible to undertake successfully a zero power mockup of a full-scale 500- or 1000-megawatt power converter. Although some information could be obtained from exponential experiments, it is felt that critical experiments would provide more information with less effort. In particular the danger coefficient technique in critical experiments has been found to be invaluable at KAPL in the investigation of epi-thermal reactors.

IX. DISCUSSION OF CROSS SECTIONS

The cross sections used in these calculations have evolved on the basis of results obtained by the Los Alamos experimental groups which have been investigating fission and inelastic scattering cross sections intensively during the past year. The principal cross-section changes are summarized in Table III, Appendix C. The bulk of the calculations were done with the May 1952 cross sections (tabulated in Table II, Appendix C) which were in good agreement with the PPA-11 and the EBR experiments. Additional Los Alamos and Oak Ridge results reported at the Oak Ridge Reactor Conference held in September 1952 indicate some revisions in the May cross sections; these will be referred to as the September 1952 cross sections. The May cross sections differed from earlier sets primarily in a 10% reduction in fission cross sections, in higher inelastic cross sections for iron and lead, and in a somewhat smaller inelastic cross section for U-238 which extended, however, to lower energies. This behavior of the U-238 inelastic scattering cross section is in agreement with the three-group transport theory calculation results of Carlson for bare oralloy spheres reported in LA-1272.**

*Private communication from H. B. Stewart.

**LA-1272, "Three-Velocity Neutron Diffusion Calculations for an Untamped Oralloy Sphere," by B. Carlson, October 13, 1951.

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It is also in accord with the results of a comparison of a six-group diffusion theory calculation on the equilibrium behavior in natural uranium with the results of the Snell experiment. The calculations are reported in KAPL-741.* H. A. Bethe now feels that this high inelastic cross section for U-238 at low energies (several hundred kilovolts) is not in agreement with Los Alamos measurements. Hence, the September inelastic cross section for U-238 does not extend below 500 kev. However, this raises the problem of correlating theory and experiment for the Snell experiment. The problem is to find a process in the 100-kv region which will absorb the inelastically scattered neutrons strongly enough to give as small a calculated diffusion length as was observed in the Snell experiment. The September 1952 cross sections take into account theoretical corrections to the inelastic-scattering measurements which give values some 15% less than the May values. The September U-238 fission cross section has a fission-spectrum average of 0.295 barns and is thus higher than the May value which averaged 0.26 barns. In addition, it has a flat plateau at 0.54 barns from $u = 0.75$ to $u = 1.5$ (2 to 5 Mev). The September U-235 fission cross section is somewhat higher than the May cross section above 1.25 Mev (1.25 as against 1.20 barns) and is reduced by 15% instead of 10% from the old standard value below 0.9 Mev. In addition, measurements of the average α of U-235 in the core of the EBR give $\alpha = 0.185$.** The IBM calculation 08325 gave $\alpha = 0.094$ or just one half of the measured value. This indicates that α for U-235 does not decrease as rapidly above 500 kev as we have assumed. In the September 1952 cross sections, α is raised at high energies to give $\alpha = 0.185$ in the EBR spectrum.

Calculations 08329 and 08339 were repeated as Cases 08349 and 08350, respectively, using the September instead of the May cross sections. A comparison of the results is given in Table VI. Use of the September 1952 cross sections results in a reactivity loss $\Delta k = 0.07$. All calculations were done with a 12-cm lead reflector and a 45-cm depleted-uranium blanket. Cases 08329 and 08349 have a relative core composition of 1 mol of U-235, 4 mols of U-238, 10 mols of iron, and 5 mols of sodium. Cases 08339 and 08350 have a relative core composition of 1 mol of U-235, 6 mols of U-238, 14 mols of iron, and 9 mols of sodium. In Figure 10, the energy distribution of the core fissions and escapes and of the U-238 captures in the blanket are compared for the two sets of calculations. Table VI shows that a large fraction of the reactivity lost in changing from the May to September cross sections is due to the increased leakage from the core. This increased leakage is concentrated in the energy group extending from 250 to 500 kev. This results from the reduction in inelastic-scattering cross sections particularly in the group from 250 to 500 kev where the U-238 inelastic cross section is reduced to zero. The cross-section changes which will bring the September 1952 data calculations into agreement with the Snell experiment

*KAPL-741, "Fast Neutron Cross Sections of U-235 and U-238," by R. Ehrlich and S. E. Russell, May 19, 1952.

**Private communication from S. Untermeyer.

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TABLE VI

COMPARISON OF RESULTS WITH MAY 1952 AND SEPTEMBER 1952 CROSS SECTIONS

<u>Property</u>		<u>Case 08329</u>	<u>Case 08349</u>	<u>Case 08339</u>	<u>Case 08350</u>
Cross sections		May	September	May	September
k		1.018	0.941	1.010	0.936
C.R. (1)		1.27	1.17	1.25	1.17
I.C.R.		0.464	0.471	0.719	0.741
α		0.149	0.214	0.164	0.225
Core (2)	Escape	0.326	0.353	0.219	0.241
	U-235 captures	0.056	0.071	0.059	0.073
	U-235 fissions	0.372	0.332	0.361	0.325
	U-238 fissions	0.036	0.043	0.041	0.049
Blanket (2)	U-235 fissions	0.007	0.006	0.004	0.004
	U-238 fissions	0.008	0.011	0.005	0.006
	U-238 captures	0.325	0.339	0.217	0.230

(1) Corrected to $k = 1$.

(2) Normalized per core source neutron.

results will reduce the diffusion length in this energy group and will thus probably also raise the calculated PFR reactivities above those given by the September 1952 data calculations. The part of the reactivity loss between May and September due to increased U-235 capture is approximately cancelled by the reactivity gain due to the increased U-238 core fissions. At present, the best compromise appears to be the use of the most recent cross-section results except that the inelastic scattering of U-238 should be extended down to 250 kev as in the May 1952 cross-section set. This appears to be the most obvious way of forcing agreement with the Snell experiment results. This cross-section set should yield comparable reactivities and somewhat lower conversion ratios (by 0.05 to 0.1) than the May 1952 set on which the bulk of the present calculations are based. Thus, data given in Section VI on conversion ratios should be reduced by about 0.1 to bring it in line with more recent cross-section information.

X. BRIEF DESCRIPTION OF METHOD OF CALCULATION

The calculations were performed by the multigroup method developed at KAPL and described in KAPL-39,* using the modified Scheme (a) described on page 4. In this scheme, the average slowing-down density q is related to the slowing-down

*KAPL-39, "Further Developments of Multigroup Method for Intermediate Piles," by H. Hurwitz, Jr. and R. Ehrlich, March 1, 1948.

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density at the low-energy end of the energy group, q^+ by $\bar{q} = f q^+$. The parameter f is chosen, and the group is calculated until approximate consistency is obtained. The method is based on the age-diffusion equation in which the diffusion coefficient D is equated to $1/3\Sigma_{\text{transport}}$. Diffusion-theory boundary conditions on the continuity of flux and current are used between spatial regions. The spatial integration of the equations was performed numerically with spatial intervals Δr of 3 centimeters. The assumed fission neutron source was distributed over the first five energy groups (Table I, Appendix C). A fission source was assumed both in the core and in the breeding blanket, although criticality was based on the reactivity given by the ratio of calculated to assumed source strength in the core alone. Small errors in the blanket source have little effect on the reactivity. The slowing down in sodium due to elastic collisions is taken into account by the term $\partial q / \partial u$ in the age-diffusion equation. Slowing down in the heavier materials was ignored. The lowest energy group considered in the calculations extended from $u = 7$ to $u = 10$ (0.45 to 9 kev). The moderation by sodium below 450 ev is negligible in the reactors studied. The inelastic scattering by the heavier materials was included, and the distribution $\chi_{i \rightarrow j}$ to lower energy groups j from a given group i is tabulated in Table I, Appendix C. Inelastic scattering was not considered to degrade neutrons below 41 kev.

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APPENDIX APage

Table I PPR Bare-Reactor Calculation Summary	45-46
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TABLE I
PFR BARE-REACTOR CALCULATION SUMMARY (1)

Case Number	Composition (volume %)									Relative Atoms					Cross Sections	Critical Radius (cm)	Critical Mass (kg)	I.C.R.	E.C.R.	C.R.
	U-235	U-238	Fe	Na	(U-235)O ₂	(U-238)O ₂	V	Void	Other	U-235	U-238	Fe	Na	V						
1			10	67	5	10		8		1.0	2.0	6.9	12.1		Feb	96	1785	0.21	0.98	1.19
2			17.8	34.2	10.4	20.8		16.8		1.0	2.0	5.9	2.97		Feb	57.2	788	0.21	1.1	1.29
3			17.8	34.2	7.8	23.4		16.8		1.0	3.0	7.8	3.95		Feb	68	990	0.321	0.958	1.279
4			17.8	34.2	6.25	25.0		16.8		1.0	4.0	9.8	4.96		Feb	78.3	1192	0.436	0.837	1.273
7(2)	4	16		50			30			1.0	4.0	5.87	11.13		Feb	75	1320	0.414	0.889	1.303
7(2)	4	16		50			30			1.0	4.0	5.87	11.13		Feb	75	1320	0.424	0.827	1.251
8	2	4		50			44			1.0	2.0		11.74	32.67	Feb	120	2705	0.217	0.589	0.806
10	4	8		50			38			1.0	2.0		5.87	14.1	Feb	73.5	1248	0.203	0.995	1.198
11	3.7	14.8	10.7	67.0				3.8		1.0	4.0	5.1	8.4		Mar	86	1840	0.43	0.96	1.39
12	5.73	22.92	16.34	51.21				3.8		1.0	4.0	5.1	4.2		Mar	62	1068	0.419	1.012	1.431
12*	5.73	22.92	16.34	51.21				3.8		1.0	4.0	5.1	4.2		Apr	60.5	992	0.431	0.965	1.396
12**	5.73	22.92	16.34	51.21				3.8		1.0	4.0	5.1	4.2		May	62.4	1098	0.441	0.941	1.382
13	6.42	12.84	18.32	57.41				5.0		1.0	2.0	5.1	4.2		Mar	60	1086	0.201	1.18	1.38
14	4.9	19.6	28.3	44.2				3.0		1.0	4.0	10.2	4.2		Mar	65	1052	0.426	0.954	1.38
14*	4.9	19.6	28.3	44.2				3.0		1.0	4.0	10.2	4.2		May	65.4	1055	0.451	0.872	1.32
15	4.75	19	27.38	42.87				6.0		1.0	4.0	10.2	4.2		Mar	67	1118	0.425	0.977	1.402
16	5.73	22.92	16.34					3.8	51.21(2x)	1.0	4.0	5.1		5.51(Pb)	Mar	54.6	732	0.412	1.037	1.449
17	4.71	37.68	13.45	42.13				2.03		1.0	8.0	5.1	4.2		Mar	76.3	1643	0.868	0.633	1.501
18(3)	4.94	19.76	28.17	44.13				3.0		1.0	4.0	10.2	4.2		Mar	65	1052	0.428	0.948	1.376
19	5.2	31.2	14.8	46.2				2.6		1.0	6.0	5.1	4.2		Mar	66.6	1200	0.64	0.83	1.47
20			16.2	21.1	51.6			8.0	3.1(Pa)	1.0(Pb)	6.0	14.0	9.0		Apr	78.8	1040	0.654	1.122	1.776
21	3.28	26.23	14.67	55.82						1.0	8.0	8.0	8.0		Apr	105.3	3000	0.936	0.491	1.427
22	0.75	3.75		16					79.5(2x)	1.0	5.0		10.0	94.0(2x)	Apr	136	1480	0.58	0.37	0.95
23	0.74	5.19		15.81					78.26(2x)	1.0	7.0		10.0	94.0(2x)	Apr	240	8000	0.816	0.15	0.966
24	3.45	22.42	15.43	58.7						1.0	6.5	8.0	8.0		Apr	92.3	2128	0.752	0.66	1.412
25	3.0	12.0	8.5	76.5						1.0	4.0	5.1	12.0		Apr	98.56	2250	0.460	0.891	1.351
26	3.55	23.08	12.91	60.46						1.0	6.5	6.5	8.0		Apr	91.66	2140	0.756	0.651	1.407
27	3.29	13.15	27.58	55.98						1.0	4.0	15.0	8.0		Apr	84.5	1556	0.463	0.84	1.303
28	3.04	19.75	25.49	51.72						1.0	6.5	15.0	8.0		Apr	98.3	2264	0.762	0.587	1.349
32	9.0	36	20	29.8				5.2		1.0	4.0	4.0	1.56		Apr	44.2	608	0.44	1.01	1.45
35	4.53	18.12	25.34	48.21				3.8		1.0	4.0	10.0	5.0		May	70.6	1245	0.453	0.887	1.32

TABLE I (continued)

Case Number	$\bar{\alpha}$	Fissions				Captures					Inelastic Scattering			
		Leakage	U-235	U-238	R ⁽⁴⁾	U-235	U-238	Fe	Na	V	U-235	U-238	Fe	V
1	0.18	0.431	0.372	0.028	0.075	0.066	0.094	0.007	0.002		0.023	0.168	0.131	
2	0.14	0.455	0.366	0.034	0.093	0.052	0.086	0.0063	0.0003		0.024	0.1647	0.1175	
3	0.163	0.3993	0.3583	0.0417	0.116	0.0586	0.1337	0.0077	0.0005		0.0224	0.236	0.1456	
4	0.178	0.3453	0.3537	0.0463	0.131	0.063	0.1816	0.0091	0.0008		0.0283	0.268	0.150	
7	0.134	0.3518	0.3492	0.0508	0.145	0.0467	0.164		0.0004	0.037	0.02	0.324		0.187
7(2)	0.142	0.3361	0.3557	0.0443	0.125	0.0505	0.1722		0.0005	0.0406	0.018	0.421		0.163
8	0.185	0.2671	0.3832	0.0168	0.044	0.0705	0.0984		0.0029	0.1612	0.0131	0.1348		0.358
10	0.131	0.4176	0.3708	0.0292	0.079	0.0487	0.0853		0.0006	0.0478	0.021	0.288		0.251
11	0.14	0.376	0.345	0.055		0.049	0.169	0.005	0.001		0.022	0.480	0.046	
12	0.126	0.3898	0.342	0.058	0.170	0.0431	0.1613	0.0055	0.0003		0.024	0.523	0.049	
12*	0.140	0.3777	0.3431	0.0569	0.166	0.0481	0.1687	0.0052	0.0003		0.024	0.699	0.048	
12**	0.126	0.3741	0.3531	0.0469	0.133	0.0445	0.1755	0.0057	0.0003		0.056	0.632	0.151	
13	0.116	0.4714	0.3592	0.0408	0.114	0.0418	0.0805	0.0060	0.0003		0.033	0.360	0.067	
14	0.133	0.375	0.3468	0.0532	0.153	0.0462	0.1676	0.0109	0.0003		0.021	0.502	0.087	
14*	0.132	0.3557	0.3603	0.0397	0.110	0.0476	0.1841	0.0123	0.0003		0.0504	0.5790	0.264	
15	0.133	0.3804	0.3438	0.0534	0.155	0.0456	0.1656	0.0109	0.0003		0.021	0.495	0.088	
16	0.118	0.3946	0.3404	0.0596	0.175	0.0401	0.1567	0.0056	0.003(Pb)		0.024	0.55	0.049	0.026(Pb)
17	0.134	0.2325	0.3241	0.0759	0.234	0.0434	0.3189	0.0050	0.0002		0.015	0.684	0.031	
18(3)	0.134	0.3737	0.3475	0.0525	0.151	0.0465	0.1685	0.0114	0.0003		0.021	0.502	0.087	
19	0.130	0.3122	0.3309	0.0691	0.209	0.0431	0.2392	0.0052	0.0003		0.076	0.330	0.263	
20	0.143	0.3776	0.2943	0.0527	0.079	0.0421(Pu)	0.2201	0.0125	0.0007		0.015(Pu)	0.673	0.082	
21	0.148	0.1863	0.3303	0.0897	0.211	0.0489	0.3552	0.0091	0.0005		0.038	0.855	0.046	
22	0.28	0.183	0.388	0.012	0.031	0.109	0.289		0.009	0.014(Zr)	0.011	0.174		0.738(Zr)
23	0.277	0.0739	0.3844	0.0156	0.041	0.1066	0.4007		0.0051	0.0137(Zr)	0.011	0.236		0.714(Zr)
24	0.143	0.2536	0.336	0.064	0.190	0.048	0.2887	0.0092	0.0005		0.044	0.792	0.052	
25	0.148	0.3569	0.3491	0.0505	0.146	0.0517	0.1845	0.0058	0.0011		0.054	0.585	0.043	
26	0.146	0.2502	0.3355	0.0645	0.192	0.049	0.2907	0.0095	0.0006		0.044	0.793	0.054	
27	0.148	0.3414	0.3541	0.0459	0.130	0.0524	0.1882	0.0173	0.0007		0.054	0.593	0.112	
28	0.153	0.2305	0.3406	0.0594	0.174	0.0522	0.2993	0.0173	0.0007		0.041	0.754	0.089	
32	0.12	0.386	0.340	0.060		0.41	0.169	0.004	0.000		0.066	0.716	0.039	
35	0.134	0.3542	0.3602	0.0398	0.110	0.0483	0.1851	0.0121	0.0003		0.050	0.572	0.257	

(1) Balance sheet normalized to 1 core source neutron

(2) Larger $\bar{\alpha}$ used for vanadium(3) Modified $\bar{\alpha}$ used for iron

(4) R is the ratio of U-238-to-U-235 fissions

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APPENDIX B

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TABLE I

SUMMARY OF RESULTS WITH APRIL 1952 CROSS SECTIONS FOR REFLECTED REACTORS

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Case Number	Core Volume %					U-235 (lbs)	Reflector (Volume %)				Blanket (Volume %)				Outer Radius (cm)			k	Conversion Ratio (2)		
															Core	Ref	Blkt		L.C.R.	R.C.R.	S.R.
Oxide Pin Fuel Element																					
	(U-235)O ₂	(U-238)O ₂	Fe	Na	Void	Pb	Na	Fe	UO ₂	U	Na	Fe									
08301	5	10	10	67	8	375	60	23	17	8	53	29	10	57	81	126	1.000	0.226	0.935	1.161	
08302	10.4	20.8	17.8	34.2	16.8	250	60	12	28	16	53	18	13	39	63	108	1.067	0.223	0.987	1.210	
08303	6.2	25	17.8	34.2	16.8	275	0	18	82		67	23	10	48	63	108	0.903	0.462	0.715	1.177	
Vanadium Alloy Fuel Element																					
	U-235	U-238	Na	V		Pb	Na	Fe	U	Na	Fe										
08304	4	16	50	30		286	—	none	—	67	23	10		45	—	90	0.795	0.436	0.783	1.219	
08305	4	16	50	30		286	60	17	23	67	23	10		45	69	114	0.910	0.442	0.768	1.210	
08306	4	16	50	30		286	0	23	77	67	23	10		45	63	108	0.900	0.443	0.740	1.183	
Metal Pin Fuel Element																					
	U-235	U-238	Fe	Na	Void	Pb	Na	Fe	U	Na	Fe										
08308	4	16	34	43	3.8	347	0	23	77	67	23	10		48	66	111	0.961	0.462	0.765	1.227	
08309	4.5	18	25	48	3.8	323	60	17	23	67	23	10		45	69	114	0.995	0.457	0.833	1.290	

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TABLE I (continued)

Case Number	Absorption Breakdown (1),(3)															
	Core					Reflector					Blanket					
	F 235	F 238	G 235	G 238	Other	In	E	G	In	E	F 235	F 238	G 238	Other	In	E
					C									C		
Oxide Pin Fuel Element																
08301	0.372	0.023	0.069	0.100	0.008	0.331	0.428	0.017	0.098	0.411	0.008	0.012	0.412	0.004	0.162	0.023
08302	0.398	0.029	0.063	0.103	0.007	0.399	0.400	0.020	0.114	0.380	0.008	0.010	0.392	0.005	0.146	0.016
08303	0.331	0.035	0.060	0.180	0.009	0.451	0.384	0.025	0.111	0.360	0.008	0.008	0.371	0.003	0.137	0.019
Vanadium Alloy Fuel Element																
08304	0.284	0.038	0.038	0.142	0.022	0.584	0.466	—	none	—	0.009	0.028	0.436	0.004	0.333	0.038
08305	0.328	0.041	0.047	0.166	0.040	0.579	0.378	0.016	0.080	0.362	0.008	0.010	0.370	0.003	0.150	0.021
08306	0.325	0.039	0.048	0.165	0.040	0.565	0.381	0.027	0.105	0.354	0.008	0.007	0.366	0.003	0.129	0.019
Metal Pin Fuel Element																
08308	0.344	0.040	0.055	0.184	0.016	0.746	0.361	0.032	0.068	0.329	0.007	0.008	0.341	0.004	0.211	0.011
08309	0.353	0.045	0.061	0.186	0.015	0.759	0.352	0.017	0.054	0.335	0.007	0.010	0.343	0.003	0.233	0.011

- (1) Normalized to one core source neutron
- (2) Corrected to $k = 1$
- (3) F = fissions
C = captures
In = inelastic scatterings
E = escapes

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TABLE II

SUMMARY OF RESULTS WITH MAY AND SEPTEMBER 1952 CROSS SECTIONS
FOR REFLECTED REACTORS

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Case Number	Core						Reflector (Volume %)			Blanket (Volume %)			Outer Radius (cm)			k	Conversion Ratios ⁽⁵⁾		
	Atom Ratio					U-235 (kg)													
	U-235	U-238	Fe	Na	% Void														
Reflector Studies																			
08308 ⁽¹⁾	1	4	15	5	3.8	347	0	23	77	67	23	10	48	66	111	0.961	0.462	0.765	1.227
08318	1	4	15	5	3.8	416	0	23	77	67	23	10	51	69	114	0.988	0.469	0.715	1.184
08326	1	4	10	5	3.8	323	none			67	23	10	45	-	90	0.892	0.458	0.898	1.356
08311	1	4	10	5	0	392	none			67	23	10	48	-	93	0.930	0.459	0.909	1.368
08324	1	4	10	5	3.8	471	none			67	23	10	51	-	96	0.964	0.460	0.883	1.343
08314	1	4	10	5	3.8	323	0	23	77	67	23	10	45	51	96	0.909	0.460	0.806	1.266
08310	1	4	10	5	3.8	323	0	23	77	67	23	10	45	63	108	0.966	0.463	0.739	1.202
08315	1	4	10	5	3.8	323	0	23	77	67	23	10	45	69	114	0.984	0.461	0.676	1.137
08312 ⁽¹⁾	1	4	10	5	3.8	323	60	17	23	67	23	10	45	57	102	0.940	0.462	0.807	1.269
08309 ⁽¹⁾	1	4	10	5	3.8	323	60	17	23	67	23	10	45	69	114	0.995	0.457	0.833	1.290
08319	1	4	10	5	3.8	323	60	17	23	67	23	10	45	69	114	0.986	0.464	0.776	1.240
08313 ⁽³⁾	1	4	10	5	3.8	323	60	17	23	67	23	10	45	75	120	1.004	0.465	0.733	1.198
08327 ⁽³⁾	1	4	10	5	3.8	323	60	17	23	67	23	10	45	75	120	1.003	0.463	0.750	1.213
Core Studies																			
08328	1	4	6.26	3.13	3.8	453	none			67	23	10	45	-	90	1.034	0.451	0.922	1.373
08322	1	6.5	15	8	0	1266	none			67	23	10	81	-	123	0.943	0.775	0.523	1.298
08323 ⁽³⁾	1	6.5	15	8	0	1266	none			67	23	10	81	-	123	0.947	0.774	0.544	1.318
08316	1	8	18	8	0	1365	0	23	77	67	23	10	81	99	144	0.977	0.946	0.356	1.302
08329 ⁽²⁾	1	4	10	5	3.8	471	60	17	23	67	23	10	51	63	108	1.018	0.464	0.804	1.268
08349 ⁽²⁾	1	4	10	5	3.8	471	60	17	23	67	23	10	51	63	108	0.941	0.471	0.695	1.166
08344	1	4	10	7	3.19	551	60	17	23	67	23	10	57	69	114	1.006	0.468	0.764	1.262
08336	1	4	10	7	3.19	643	60	17	23	67	23	10	60	72	117	1.034	0.468	0.787	1.255
08334	1	4	10	7	3.19	856	60	17	23	67	23	10	66	78	123	1.080	0.469	0.780	1.249
08345	1	4	10	9	2.74	640	60	17	23	67	23	10	63	75	120	1.001	0.471	0.776	1.247
08337	1	4	10	9	2.74	956	60	17	23	67	23	10	72	84	129	1.068	0.472	0.763	1.235
08335	1	4	10	9	2.74	1361	60	17	23	67	23	10	81	93	138	1.119	0.473	0.756	1.229
08340	1	6	14	5	3.19	744	60	17	23	67	23	10	63	75	120	0.991	0.708	0.581	1.289
08338	1	6	14	7	2.74	841	60	17	23	67	23	10	69	81	126	0.979	0.713	0.555	1.268
08339 ⁽²⁾	1	6	14	9	2.41	1332	60	17	23	67	23	10	84	96	141	1.010	0.719	0.528	1.247
08350 ⁽²⁾	1	6	14	9	0	1332	60	17	23	67	23	10	84	96	141	0.936	0.741	0.429	1.170
08341	1	8	18	5	2.75	1080	60	17	23	67	23	10	75	87	132	0.939	0.956	0.337	1.293
08347	1	8	18	5	2.75	2060	60	17	23	67	23	10	93	105	150	0.984	0.960	0.326	1.286
08342	1	8	18	9	2.15	1189	60	17	23	67	23	10	84	96	141	0.912	0.964	0.316	1.280
08343	1	8	18	9	2.15	1946	60	17	23	67	23	10	99	111	156	0.952	0.968	0.298	1.266

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TABLE II (continued)

(4), (6)

Absorption Breakdown

Case Number	Core										Reflector			Blanket					
	F 235	F 238	C 235	C 238	C Fe	C Na	In 235	In 238	In Fe	E	C	In	E	F 235	F 238	C 238	Other C	In	E
Reflector Studies																			
08308 ⁽¹⁾	0.344	0.040	0.055	0.184	0.015	0.0005	0.057	0.588	0.101	0.361	0.032	0.068	0.329	0.007	0.008	0.341	0.004	0.211	0.011
08318	0.365	0.030	0.059	0.199	0.017	0.0005	0.045	0.500	0.315	0.330	0.033	0.171	0.297	0.006	0.004	0.314	0.003	0.124	0.008
08326	0.323	0.034	0.046	0.169	0.010	0.0003	0.049	0.519	0.229	0.417	none	none	none	0.009	0.024	0.437	0.004	0.466	0.023
08311	0.337	0.035	0.049	0.177	0.011	0.0003	0.050	0.533	0.234	0.391	none	none	none	0.009	0.023	0.419	0.004	0.448	0.021
08324	0.350	0.035	0.051	0.184	0.011	0.0003	0.051	0.545	0.239	0.368	none	none	none	0.008	0.021	0.389	0.004	0.409	0.019
08314	0.330	0.034	0.049	0.174	0.010	0.0004	0.049	0.525	0.230	0.403	0.009	0.103	0.394	0.008	0.011	0.391	0.004	0.290	0.015
08310	0.352	0.034	0.055	0.189	0.011	0.0004	0.050	0.532	0.231	0.359	0.034	0.197	0.325	0.007	0.004	0.333	0.003	0.128	0.009
08315	0.359	0.034	0.058	0.192	0.011	0.0005	0.050	0.535	0.231	0.345	0.051	0.214	0.294	0.006	0.002	0.297	0.003	0.079	0.006
08312	0.342	0.034	0.053	0.181	0.011	0.0004	0.060	0.532	0.233	0.381	0.008	0.122	0.374	0.008	0.009	0.374	0.004	0.249	0.014
08309 ⁽¹⁾	0.353	0.045	0.061	0.186	0.011	0.0004	0.061	0.622	0.076	0.352	0.017	0.054	0.335	0.007	0.010	0.343	0.003	0.233	0.011
08319	0.361	0.034	0.056	0.194	0.011	0.0004	0.051	0.541	0.234	0.345	0.018	0.184	0.326	0.007	0.005	0.337	0.003	0.150	0.010
08313	0.367	0.035	0.058	0.197	0.011	0.0004	0.050	0.538	0.234	0.331	0.025	0.199	0.307	0.006	0.003	0.308	0.003	0.101	0.007
08327 ⁽³⁾	0.367	0.035	0.058	0.197	0.011	0.0005	0.050	0.537	0.234	0.333	0.025	0.205	0.308	0.015	0.004	0.315	0.005	0.125	0.008
Core Studies																			
08328	0.372	0.042	0.051	0.190	0.007	0.0002	0.059	0.621	0.174	0.338	none	none	none	0.008	0.020	0.356	0.004	0.393	0.020
08322	0.336	0.042	0.053	0.301	0.016	0.0005	0.037	0.666	0.260	0.252	none	none	none	0.005	0.012	0.258	0.003	0.244	0.013
08323 ⁽³⁾	0.337	0.042	0.054	0.303	0.016	0.0005	0.037	0.670	0.262	0.248	none	none	none	0.013	0.013	0.263	0.004	0.264	0.014
08316	0.337	0.054	0.054	0.370	0.009	0.0006	0.038	0.825	0.142	0.244	0.019	0.092	0.157	0.003	0.002	0.161	0.002	0.058	0.003
08329	0.372	0.036	0.056	0.198	0.012	0.0004	0.052	0.561	0.244	0.326	0.007	0.103	0.320	0.007	0.008	0.325	0.003	0.213	0.011
08349 ⁽²⁾	0.332	0.043	0.071	0.189	0.011	0.0004	0.039	0.327	0.281	0.353	0.007	0.120	0.346	0.006	0.011	0.339	0.004	0.120	0.022
08344	0.367	0.035	0.056	0.198	0.012	0.0005	0.051	0.544	0.238	0.331	0.007	0.108	0.323	0.007	0.008	0.330	0.003	0.217	0.011
08336	0.378	0.036	0.059	0.204	0.012	0.0005	0.052	0.553	0.242	0.311	0.007	0.100	0.304	0.006	0.008	0.310	0.003	0.203	0.010
08334	0.395	0.037	0.061	0.214	0.012	0.0006	0.053	0.566	0.247	0.281	0.006	0.092	0.274	0.006	0.007	0.282	0.003	0.184	0.009
08345	0.366	0.035	0.058	0.200	0.012	0.0007	0.049	0.526	0.231	0.329	0.007	0.107	0.321	0.007	0.008	0.327	0.003	0.211	0.010
08337	0.392	0.036	0.063	0.215	0.012	0.0008	0.051	0.551	0.241	0.282	0.006	0.092	0.276	0.006	0.007	0.282	0.003	0.181	0.009
08335	0.410	0.036	0.066	0.225	0.013	0.0008	0.053	0.565	0.246	0.249	0.006	0.085	0.243	0.005	0.006	0.250	0.003	0.161	0.008
08340	0.353	0.042	0.055	0.289	0.016	0.0004	0.041	0.686	0.267	0.246	0.006	0.071	0.240	0.005	0.006	0.249	0.003	0.153	0.008
08338	0.349	0.041	0.055	0.289	0.016	0.0005	0.040	0.663	0.260	0.249	0.006	0.073	0.244	0.005	0.005	0.247	0.003	0.147	0.007
08339	0.361	0.041	0.059	0.302	0.016	0.0007	0.040	0.657	0.260	0.219	0.005	0.065	0.214	0.004	0.005	0.217	0.002	0.127	0.006
08350	0.325	0.049	0.073	0.295	0.016	0.0007	0.030	0.376	0.168	0.241	0.005	0.077	0.235	0.004	0.006	0.230	0.003	0.070	0.012
08341	0.327	0.045	0.051	0.361	0.019	0.0003	0.033	0.756	0.277	0.197	0.005	0.056	0.192	0.004	0.004	0.195	0.002	0.111	0.006
08347	0.344	0.046	0.055	0.383	0.020	0.0003	0.034	0.777	0.284	0.152	0.004	0.045	0.148	0.003	0.004	0.154	0.002	0.091	0.004
08342	0.319	0.044	0.053	0.358	0.019	0.0006	0.032	0.715	0.268	0.208	0.005	0.058	0.203	0.004	0.004	0.205	0.002	0.114	0.006
08343	0.333	0.044	0.056	0.376	0.019	0.0007	0.033	0.731	0.273	0.170	0.004	0.094	0.004	0.003	0.003	0.168	0.002	0.049	0.166

- (1) April cross sections
- (2) September cross sections
- (3) Blanket natural uranium
- (4) Normalised to one core source neutron
- (5) Normalised to $k = 1$
- (6) F = fissions
C = captures
In = inelastic scatterings
E = escapes

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TABLE III

SUMMARY OF RESULTS WITH MAY 1952 CROSS SECTIONS FOR THE FPA'S AND THE EBR

Case Number	Core								Reflector					Blanket							Outer Radius			k	Conversion Ratios (3)				
	Volume %							U-235 (kg)	Volume %				Volume %						(cm)			Ratio (3)							
	U-235	U-238	Be	Na	Fe	Al	Void		Be	Na	Fe	Al	Void	U-235	U-238	Be	Na	Fe	Al	Void	Core	Ref1	Blkt		I.C.R.	E.C.R.	C.R.		
FPA-11																													
08320	10.56	0.56	7.1	40	28	6.4	7.4	114	65.6	13.9	8.5	6.5	5.5	0.4	55.8	4.7	16.3	10.6	6.6	5.7	24	32	40	1.003	0.006	0.335	0.341		
Proposed Intermediate FPA																													
08321	10.24	40.97	0	0	43.54	0	5.25	180	0	0	94.75	0	5.25	0.625	86.4	0	0	3.5	0	9.5	28.2	42.29	84.99	1.030	0.446	0.920	1.366		
EBR Calculations																													
	U-235	U-238	Fe	NaK	U-235 (kg)						U	Fe	NaK					U	Fe	NaK	Void	Core	Ref1	Blkt	k	I.C.R.	Ref1 E.C.R.	Blkt E.C.R.	C.R.
08325	61.3	4.2	14.4	20.1	45.2						49.1	17.6	33.4					56	15	18	11	9.8	23.8	33.6	1.004	0.006	0.404	0.188	0.598
08330 ⁽¹⁾	61.3	4.2	14.4	20.1	45.2						49.1	17.6	33.4					56	15	18	11	9.8	23.8	33.6	0.946	0.006	0.301	0.136	0.443

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TABLE III (continued)

Case Number	Absorption Breakdown (2), (4)																
	Core							Reflector			Blanket						
	F 235	F 238	G 235	G 238	Other C	In	E	E	In	E	F 235	F 238	G 235	G 238	Other C	In	E
FPA-11																	
08320	0.300	0.001	0.074	0.003	0.005	0.319	0.517	0.002	0.030	0.515	0.009	0.013	0.003	0.157	0.001	0.177	0.433
Proposed Intermediate FPA																	
08321	0.373	0.040	0.049	0.188	0.009	0.908	0.342	0.024	0.182	0.318	0.018	0.009	0.003	0.358	0.001	0.230	0.014
NBR Calculations																	
08325 08330 (1)		<u>F 235</u>	<u>F 238</u>	<u>G 235</u>	<u>G 238</u>	<u>Other C</u>	<u>In</u>	<u>E</u>									
	Core	0.399	0.003	0.037	0.003	0.0004	0.252	0.558									
	Ref1	0.010	0.042	0.001	0.175	0.0036	0.712	0.447									
	Blkt	0.004	0.008	0.001	0.081	0.0011	0.165	0.442									
	Core	0.376	0.002	0.035	0.003	0.0004	0.246	0.584									
	Ref1	0.009	0.039	0.001	0.147	0.0030	0.657	0.536									
	Blkt	0.004	0.009	0.001	0.066	0.0009	0.173	0.516									

- (1) Rough integral theory corrections applied to transport cross section
- (2) Normalized to one core source neutron
- (3) Normalized to $k = 1$
- (4) F = fissions
C = captures
In = inelastic scatterings
E = escapes

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APPENDIX C

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TABLE I

ENERGY GROUPS AND FISSION AND INELASTIC
SCATTERING SPECTRA USED IN CALCULATIONS(1)

<u>Group Number</u>	<u>Range in $u = \ln(10^7/E)$</u>	<u>Range in Energy (Mev)</u>	<u>X_F = Fraction of Fission Neutrons in Group</u>
1	0.5 to 1	6.09 to 3.70	0.1344
2	1 to 2	3.70 to 1.35	0.4522
3	2 to 3	1.35 to 0.50	0.2954
4	3 to 3.75	0.50 to 0.24	0.0807
5	3.75 to 4.5	0.24 to 0.113	0.0373
6	4.5 to 5.5	0.113 to 0.041	-
7	5.5 to 7	0.041 to 0.0091	-
8	7 to 10	0.0091 to 0.00045	-

Inelastic Spectrum $\chi_i \rightarrow j$

$\begin{smallmatrix} i \\ j \end{smallmatrix}$	<u>U-238</u>				<u>U-235</u>			<u>Fe, V, and K</u>			<u>Pb and Bi</u>		
	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>1</u>	<u>2</u>	<u>3</u>	<u>1</u>	<u>2</u>	<u>3</u>	<u>1</u>	<u>2</u>	<u>3</u>
2	0.15	-	-	-	0.10	-	-	0.4	-	-	0.4	-	-
3	0.35	0.35	-	-	0.20	0.20	-	0.4	0.6	-	0.3	0.6	-
4	0.24	0.33	0.51	-	0.30	0.33	0.51	0.2	0.4	1.0	0.2	0.3	0.9
5	0.16	0.22	0.34	0.80	0.20	0.22	0.34				0.1	0.1	0.1
6	0.10	0.10	0.15	0.20	0.20	0.25	0.15						

(1) Multigroup calculations were done by Scheme (a) which is described in TID-2009, "Multigroup Methods for Neutron Diffusion Problems," by H. Hurwitz, Jr. and R. Ehrlich (to be issued in June 1953).

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TABLE IIa⁽¹⁾

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MAY 1952 U-235 CROSS SECTIONS

<u>u</u>	<u>σ_{tr}</u>	<u>σ_c</u>	<u>σ_f</u>	<u>σ_{in}</u>
	4.00	0.065	1.20	1.80
0.25	4.25	0.065	1.20	1.80
0.50	4.35	0.065	1.20	1.80
0.75	4.50	0.065	1.20	1.80
1.00	4.75	0.065	1.20	1.80
1.25	4.90	0.065	1.20	1.50
1.50	5.20	0.065	1.20	1.50
1.75	5.40	0.065	1.20	1.50
2.00	5.75	0.065	1.20	1.20
2.25	6.00	0.067	1.20	1.20
2.50	6.30	0.072	1.21	1.10
2.75	6.60	0.081	1.25	0.60
3.00	7.00	0.091	1.28	0.00
3.25	7.46	0.108	1.39	0.00
3.50	7.50	0.130	1.47	0.00
3.75	8.00	0.156	1.58	0.00
4.00	8.40	0.185	1.70	0.00
4.25	8.80	0.222	1.79	0.00
4.50	9.40	0.274	1.90	0.00
4.75	9.80	0.320	2.00	0.00
5.00	10.25	0.372	2.10	0.00
5.25	10.70	0.431	2.21	0.00
5.50	11.20	0.483	2.30	0.00
5.75	11.75	0.554	2.45	0.00
6.00	12.20	0.627	2.60	0.00
6.25	12.70	0.714	2.79	0.00
6.50	13.00	0.804	3.00	0.00
6.75	13.50	0.897	3.18	0.00
7.00	14.10	1.036	3.50	0.00
7.25	14.60	1.124	3.65	0.00
7.50	15.20	1.280	4.00	0.00
7.75	15.80	1.454	4.38	0.00
8.00	16.40	1.598	4.70	0.00
8.25	16.80	1.822	5.19	0.00
8.50	17.75	2.034	5.65	0.00
8.75	18.30	2.331	6.30	0.00
9.00	19.00	2.729	7.20	0.00
9.25	20.20	3.026	7.90	0.00
9.50	22.00	3.764	9.70	0.00
9.75	23.50	4.485	11.50	0.00
10.00	25.00	5.214	13.10	0.00

(1) The following notations apply to Tables IIa through IIj.

All cross sections are in barns per atom.

$u = \ln[10^7/E(\text{ev})]$.

σ_{tr} = transport cross section.

σ_c = capture cross section.

σ_f = fission cross section.

σ_{in} = inelastic scattering cross section.

σ_s = elastic scattering cross section.

TABLE IIb

MAY 1952 U-238 CROSS SECTIONS

<u>u</u>	<u>σ_{tr}</u>	<u>σ_c</u>	<u>σ_f</u>	<u>σ_{in}</u>
	4.00	0.013	0.990	3.00
0.25	4.00	0.013	0.909	3.00
0.50	4.00	0.013	0.648	3.00
0.75	4.00	0.013	0.576	2.75
1.00	4.00	0.020	0.522	2.50
1.25	4.05	0.030	0.477	2.50
1.50	4.10	0.044	0.441	2.50
1.75	4.23	0.071	0.338	2.50
2.00	4.36	0.092	0.099	2.50
2.25	4.56	0.116	0.014	2.30
2.50	4.85	0.133	0.004	2.10
2.75	5.16	0.143	0.001	1.90
3.00	5.57	0.148	0.000	1.70
3.25	6.03	0.152	0.000	1.50
3.50	6.55	0.162	0.000	1.00
3.75	7.10	0.180	0.000	0.00
4.00	7.70	0.205	0.000	0.00
4.25	8.32	0.228	0.000	0.00
4.50	9.05	0.258	0.000	0.00
4.75	9.80	0.283	0.000	0.00
5.00	10.65	0.317	0.000	0.00
5.25	11.47	0.350	0.000	0.00
5.50	12.53	0.390	0.000	0.00
5.75	13.21	0.430	0.000	0.00
6.00	13.78	0.472	0.000	0.00
6.25	14.00	0.516	0.000	0.00
6.50	14.15	0.572	0.000	0.00
6.75	14.32	0.637	0.000	0.00
7.00	14.40	0.690	0.000	0.00
7.25	14.50	0.740	0.000	0.00
7.50	14.55	0.790	0.000	0.00
7.75	14.60	0.840	0.000	0.00
8.00	14.65	0.900	0.000	0.00
8.25	14.70	0.950	0.000	0.00
8.50	14.70	1.000	0.000	0.00
8.75	14.70	1.060	0.000	0.00
9.00	14.70	1.120	0.000	0.00
9.25	14.70	1.160	0.000	0.00
9.50	14.70	1.200	0.000	0.00
9.75	14.70	1.250	0.000	0.00
10.00	14.70	1.290	0.000	0.00

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TABLE IIc

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MAY 1952 SODIUM CROSS SECTIONS

<u>u</u>	<u>σ_{tr}</u>	<u>σ_c</u>	<u>σ_s</u>
	1.40	0.00029	2.00
0.25	1.47	0.00029	2.08
0.50	1.53	0.00029	2.16
0.75	1.59	0.00029	2.23
1.00	1.67	0.00029	2.30
1.25	1.74	0.00029	2.36
1.50	1.81	0.00029	2.40
1.75	1.90	0.00029	2.45
2.00	2.00	0.00029	2.50
2.25	2.80	0.00029	3.38
2.50	4.93	0.00029	5.76
2.75	6.25	0.00029	7.02
3.00	2.54	0.00029	2.61
3.25	4.53	0.00029	4.66
3.50	3.33	0.00029	3.43
3.75	4.39	0.00029	4.52
4.00	4.86	0.00029	5.00
4.25	3.33	0.00029	3.43
4.50	3.19	0.00029	3.28
4.75	3.34	0.00029	3.44
5.00	3.98	0.00029	4.10
5.25	5.58	0.00029	5.75
5.50	3.75	0.00033	3.86
5.75	3.64	0.00037	3.75
6.00	4.32	0.00042	4.45
6.25	4.25	0.00048	4.38
6.50	4.27	0.00053	4.40
6.75	4.31	0.00062	4.43
7.00	4.37	0.00069	4.50
7.25	4.56	0.00080	4.85
7.50	5.22	0.00089	5.37
7.75	5.65	0.00102	5.80
8.00	23.69	0.12391	24.40
8.25	31.75	0.12412	32.70
8.50	6.07	0.00149	6.25
8.75	5.35	0.00170	5.50
9.00	4.86	0.00194	5.00
9.25	4.68	0.00217	3.94
9.50	4.37	0.00249	3.35
9.75	3.87	0.00280	3.28
10.00	3.19	0.00325	3.28

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TABLE IIa

MAY 1952 IRON CROSS SECTIONS

<u>u</u>	<u>σ_{tr}</u>	<u>σ_c</u>	<u>σ_{in}</u>	<u>σ_s</u>
	1.94	0.0010	0.90	3.21
0.25	1.97	0.0010	0.85	3.18
0.50	1.98	0.0010	0.85	3.09
0.75	2.01	0.0010	0.85	3.00
1.00	2.05	0.0010	0.80	2.97
1.25	2.04	0.0010	0.80	2.89
1.50	2.00	0.0010	0.80	2.78
1.75	1.94	0.0010	0.80	2.23
2.00	1.90	0.0010	0.70	1.80
2.25	2.02	0.0011	0.50	2.06
2.50	2.50	0.0012	0.30	2.70
2.75	2.02	0.0014	0.10	2.42
3.00	2.43	0.0025	0.00	2.90
3.25	4.85	0.0037	0.00	5.55
3.50	2.53	0.0045	0.00	2.80
3.75	2.96	0.0050	0.00	3.25
4.00	3.60	0.0060	0.00	4.00
4.25	3.30	0.0066	0.00	3.55
4.50	2.60	0.0072	0.00	2.81
4.75	4.90	0.0080	0.00	4.98
5.00	3.22	0.0090	0.00	3.21
5.25	4.21	0.0090	0.00	4.20
5.50	5.36	0.0090	0.00	5.35
5.75	7.66	0.0090	0.00	7.65
6.00	2.40	0.0090	0.00	2.39
6.25	3.05	0.0090	0.00	3.00
6.50	4.31	0.0090	0.00	4.30
6.75	5.75	0.0090	0.00	6.25
7.00	8.01	0.0090	0.00	8.00
7.25	11.05	0.0090	0.00	11.50
7.50	11.71	0.0090	0.00	11.80
7.75	11.74	0.0090	0.00	11.83
8.00	11.71	0.0095	0.00	11.80
8.25	11.69	0.0100	0.00	11.76
8.50	11.71	0.0110	0.00	11.70
8.75	11.55	0.0120	0.00	11.59
9.00	11.46	0.0130	0.00	11.45
9.25	11.30	0.0140	0.00	11.25
9.50	11.20	0.0150	0.00	11.00
9.75	10.99	0.0160	0.00	10.94
10.00	10.96	0.0170	0.00	10.94

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TABLE IIe

APRIL 1952⁽¹⁾ VANADIUM CROSS SECTION

<u>u</u>	<u>σ_{tr}</u>	<u>σ_c</u>	<u>σ_{in}</u>	<u>σ_s</u>
	2.10	0.0022	1.20	1.30
0.25	2.10	0.0022	1.10	1.45
0.50	2.10	0.0022	1.00	1.60
0.75	2.15	0.0022	0.90	1.75
1.00	2.20	0.0022	0.80	1.90
1.25	2.25	0.0022	0.80	1.95
1.50	2.30	0.0022	0.80	2.00
1.75	2.30	0.0022	0.40	2.45
2.00	2.30	0.0022	0.00	2.90
2.25	2.70	0.0022	0.00	3.30
2.50	2.60	0.0022	0.00	3.00
2.75	2.60	0.0032	0.00	2.90
3.00	2.40	0.0045	0.00	2.60
3.25	4.60	0.0070	0.00	4.80
3.50	4.20	0.0095	0.00	4.20
3.75	4.00	0.0120	0.00	4.00
4.00	6.00	0.0150	0.00	6.00
4.25	5.00	0.0170	0.00	5.00
4.50	8.00	0.0200	0.00	8.00
4.75	2.50	0.0240	0.00	2.50
5.00	9.50	0.0300	0.00	9.50
5.25	2.60	0.0350	0.00	2.60
5.50	3.30	0.0430	0.00	3.30
5.75	5.50	0.0580	0.00	5.50
6.00	8.50	0.0650	0.00	8.50
6.25	10.00	0.0780	0.00	10.00
6.50	10.90	0.0900	0.00	10.90
6.75	10.90	0.1100	0.00	10.90
7.00	10.90	0.1400	0.00	10.90
7.25	10.80	0.1700	0.00	10.80
7.50	10.80	0.1700	0.00	10.80
7.75	10.80	0.1700	0.00	10.80
8.00	130.00	0.2000	0.00	130.00
8.25	420.00	0.5000	0.00	420.00
8.50	130.00	0.1500	0.00	130.00
8.75	80.00	0.1000	0.00	80.00
9.00	12.00	0.0700	0.00	12.00
9.25	12.00	0.0640	0.00	12.00
9.50	12.00	0.0600	0.00	12.00
9.75	12.00	0.0570	0.00	12.00
10.00	12.00	0.0540	0.00	12.00

(1) Not used in calculations after April 1952.

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TABLE II f

MAY 1952 LEAD CROSS SECTIONS

<u>u</u>	<u>σ_{tr}</u>	<u>σ_c</u>	<u>σ_{in}</u>	<u>σ_s</u>
	3.57	0.0025	1.00	4.00
0.25	3.61	0.0025	0.95	4.05
0.50	3.65	0.0025	0.95	4.05
0.75	3.70	0.0025	0.95	4.05
1.00	3.76	0.0025	0.90	4.10
1.25	3.84	0.0025	0.90	4.10
1.50	3.93	0.0025	0.90	4.10
1.75	4.06	0.0025	0.90	4.10
2.00	4.20	0.0025	0.80	4.20
2.25	4.38	0.0025	0.60	4.40
2.50	4.70	0.0025	0.40	4.85
2.75	5.40	0.0025	0.20	5.70
3.00	4.30	0.0025	0.00	4.70
3.25	5.80	0.0025	0.00	6.20
3.50	7.20	0.0025	0.00	7.50
3.75	7.50	0.0025	0.00	7.70
4.00	8.20	0.0025	0.00	8.40
4.25	9.20	0.0025	0.00	9.48
4.50	10.10	0.0025	0.00	10.40
4.75	11.01	0.0025	0.00	11.05
5.00	11.56	0.0025	0.00	11.60
5.25	11.92	0.0025	0.00	11.96
5.50	12.19	0.0025	0.00	12.23
5.75	12.40	0.0025	0.00	12.44
6.00	12.56	0.0025	0.00	12.60
6.25	12.71	0.0025	0.00	12.75
6.50	12.80	0.0025	0.00	12.84
6.75	12.89	0.0025	0.00	12.93
7.00	12.94	0.0025	0.00	12.98
7.25	12.96	0.0025	0.00	13.00
7.50	12.96	0.0025	0.00	13.00
7.75	12.96	0.0025	0.00	13.00
8.00	12.96	0.0025	0.00	13.00
8.25	12.96	0.0025	0.00	13.00
8.50	12.96	0.0025	0.00	13.00
8.75	12.96	0.0025	0.00	13.00
9.00	12.96	0.0025	0.00	13.00
9.25	12.96	0.0025	0.00	13.00
9.50	12.96	0.0025	0.00	13.00
9.75	12.96	0.0025	0.00	13.00
10.00	12.96	0.0025	0.00	13.00

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TABLE IIg

MAY 1952 OXYGEN CROSS SECTIONS

<u>v</u>	<u>σ_{tr}</u>	<u>σ_c</u>	<u>σ_s</u>
	0.84	0.00012	1.20
0.25	0.71	0.00012	1.00
0.50	0.78	0.00012	1.10
0.75	1.07	0.00012	1.50
1.00	2.03	0.00012	2.80
1.25	0.96	0.00012	1.30
1.50	0.92	0.00012	1.20
1.75	1.55	0.00012	2.00
2.00	2.40	0.00012	3.00
2.25	5.80	0.00012	7.00
2.50	2.99	0.00012	3.50
2.75	2.67	0.00012	3.00
3.00	3.71	0.00012	4.00
3.25	11.56	0.00012	12.00
3.50	3.83	0.00012	4.00
3.75	3.83	0.00012	4.00
4.00	3.83	0.00012	4.00
4.25	3.83	0.00012	4.00
4.50	3.83	0.00012	4.00
4.75	3.83	0.00012	4.00
5.00	3.83	0.00012	4.00
5.25	3.83	0.00012	4.00
5.50	3.83	0.00012	4.00
5.75	3.83	0.00012	4.00
6.00	3.83	0.00012	4.00
6.25	3.83	0.00012	4.00
6.50	3.83	0.00012	4.00
6.75	3.83	0.00012	4.00
7.00	3.83	0.00012	4.00
7.25	3.83	0.00012	4.00
7.50	3.83	0.00012	4.00
7.75	3.83	0.00012	4.00
8.00	3.83	0.00012	4.00
8.25	3.83	0.00012	4.00
8.50	3.83	0.00012	4.00
8.75	3.83	0.00012	4.00
9.00	3.83	0.00012	4.00
9.25	3.83	0.00012	4.00
9.50	3.83	0.00012	4.00
9.75	3.83	0.00012	4.00
10.00	3.83	0.00012	4.00

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TABLE IIh

MAY 1952 POTASSIUM CROSS SECTIONS

<u>u</u>	<u>σ_{tr}</u>	<u>σ_c</u>	<u>σ_{in}</u>	<u>σ_s</u>
	2.80	0.0010	0.60	2.80
0.25	2.82	0.0010	0.60	2.82
0.50	2.84	0.0010	0.40	2.84
0.75	2.86	0.0010	0.40	2.86
1.00	2.90	0.0010	0.40	2.90
1.25	2.77	0.0010	0.40	2.77
1.50	2.64	0.0010	0.35	2.64
1.75	2.53	0.0010	0.30	2.53
2.00	2.40	0.0010	0.00	2.40
2.25	2.20	0.0011	0.00	2.20
2.50	1.99	0.0012	0.00	1.99
2.75	1.78	0.0014	0.00	1.78
3.00	1.86	0.0025	0.00	1.86
3.25	2.50	0.0037	0.00	2.50
3.50	2.50	0.0045	0.00	2.50
3.75	1.50	0.0050	0.00	1.50
4.00	1.70	0.0060	0.00	1.70
4.25	1.90	0.0066	0.00	1.90
4.50	2.20	0.0072	0.00	2.20
4.75	1.80	0.0080	0.00	1.80
5.00	3.00	0.0090	0.00	3.00
5.25	1.80	0.0090	0.00	1.80
5.50	1.50	0.0090	0.00	1.50
5.75	1.70	0.0090	0.00	1.70
6.00	1.80	0.0090	0.00	1.80
6.25	1.90	0.0090	0.00	1.90
6.50	2.00	0.0090	0.00	2.00
6.75	2.00	0.0090	0.00	2.00
7.00	2.00	0.0090	0.00	2.00
7.25	2.00	0.0090	0.00	2.00
7.50	2.00	0.0090	0.00	2.00
7.75	2.00	0.0090	0.00	2.00
8.00	2.00	0.0095	0.00	2.00
8.25	2.00	0.0100	0.00	2.00
8.50	2.00	0.0110	0.00	2.00
8.75	2.00	0.0120	0.00	2.00
9.00	2.00	0.0130	0.00	2.00
9.25	2.00	0.0140	0.00	2.00
9.50	2.00	0.0150	0.00	2.00
9.75	2.00	0.0160	0.00	2.00
10.00	2.00	0.0170	0.00	2.00

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TABLE III

MAY 1952 BERYLLIUM CROSS SECTIONS

<u>u</u>	<u>σ_{tr}</u>	<u>σ_c</u>	<u>σ_s</u>
	0.55	0.00002	0.55
0.25	0.64	0.00002	0.64
0.50	0.80	0.00002	0.80
0.75	1.02	0.00002	1.02
1.00	1.49	0.00002	1.49
1.25	1.83	0.00002	1.83
1.50	1.69	0.00002	1.69
1.75	1.13	0.00002	1.13
2.00	2.02	0.00002	2.02
2.25	2.71	0.00002	2.71
2.50	3.34	0.00002	3.34
2.75	4.01	0.00002	4.00
3.00	3.39	0.00002	3.39
3.25	3.66	0.00002	3.66
3.50	4.00	0.00002	4.00
3.75	3.93	0.00002	4.24
4.00	4.17	0.00002	4.50
4.25	4.41	0.00002	4.76
4.50	4.63	0.00002	5.00
4.75	4.82	0.00002	5.21
5.00	5.00	0.00002	5.40
5.25	5.07	0.00002	5.48
5.50	5.09	0.00002	5.50
5.75	5.10	0.00002	5.51
6.00	5.14	0.00002	5.55
6.25	5.17	0.00002	5.62
6.50	5.23	0.00002	5.65
6.75	5.27	0.00002	5.70
7.00	5.30	0.00002	5.72
7.25	5.33	0.00002	5.74
7.50	5.34	0.00002	5.77
7.75	5.37	0.00002	5.80
8.00	5.39	0.00003	5.82
8.25	5.40	0.00003	5.85
8.50	5.43	0.00003	5.86
8.75	5.45	0.00004	5.88
9.00	5.46	0.00004	5.90
9.25	5.48	0.00005	5.90
9.50	5.50	0.00005	5.90
9.75	5.50	0.00006	5.94
10.00	5.51	0.00007	5.95

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TABLE IIj

MAY 1952 ALUMINUM CROSS SECTIONS

<u>u</u>	<u>σ_{tr}</u>	<u>σ_c</u>	<u>σ_s</u>
	1.61	0.00040	2.30
0.25	1.66	0.00040	2.35
0.50	1.70	0.00040	2.40
0.75	1.75	0.00040	2.45
1.00	1.81	0.00040	2.50
1.25	1.85	0.00040	2.50
1.50	1.89	0.00040	2.50
1.75	2.33	0.00040	3.00
2.00	2.40	0.00040	3.00
2.25	2.48	0.00040	3.00
2.50	3.42	0.00040	4.00
2.75	3.56	0.00040	4.00
3.00	3.71	0.00040	4.00
3.25	3.85	0.00040	4.00
3.50	3.90	0.00040	4.00
3.75	3.90	0.00040	4.00
4.00	3.90	0.00040	4.00
4.25	5.44	0.00040	6.60
4.50	2.63	0.00040	2.70
4.75	9.27	0.00040	9.50
5.00	1.46	0.00040	1.50
5.25	3.32	0.00040	3.40
5.50	5.56	0.00040	5.70
5.75	2.34	0.00040	2.40
6.00	1.22	0.00040	1.25
6.25	0.83	0.00040	0.85
6.50	0.84	0.00040	0.86
6.75	0.86	0.00040	0.88
7.00	1.06	0.00040	1.09
7.25	1.20	0.00045	1.23
7.50	1.37	0.00050	1.40
7.75	1.51	0.00058	1.55
8.00	1.61	0.00066	1.65
8.25	1.66	0.00074	1.70
8.50	1.76	0.00084	1.80
8.75	1.71	0.00095	1.75
9.00	1.59	0.00108	1.63
9.25	1.51	0.00120	1.55
9.50	1.51	0.00140	1.55
9.75	1.51	0.00156	1.55
10.00	1.51	0.00179	1.55

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CROSS-SECTION DEVIATIONS FROM TABLE II

Cross- Section Set Designation	U-238		U-235		Fe		Pb	V
	σ_{in}	σ_f	σ_{in}	σ_f	σ_{in}	σ_c	σ_{in}	σ_{in}
February	1.6 0 in Group 4	LAMS-777	0.9 0 in Group 3	LA-994	0.9	0.005	1.0	0.9
March	2.5 0 in Group 4	LAMS-777	0.9 0 in Group 3	Table II Appendix C	0.3	0.005	0.2	-
April	Table II Appendix C	LAMS-777	Table II Appendix C	Table II Appendix C	0.3	Table II Appendix C	0.2	-
May	Table II Appendix C	Table II Appendix C $\bar{\sigma} = 0.26$	Table II Appendix C	Table II Appendix C	Table II Appendix C	Table II Appendix C	Table II Appendix C	-
September	2.1 0 in Group 4	$\bar{\sigma} = 0.295$ ORNL Conf (1)	1.0	ORNL Conf (1)	Table II Appendix C	Table II Appendix C	Table II Appendix C	-

(1) As reported at Oak Ridge Reactor Conference, September 10 through 12, 1952. See Section IX for further details.