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REPORT

NEUTRONIC CONSIDERATIONS
IN THE SELECTION OF A DRIVER FUEL
FOR THE FAST TEST REACTOR (FTR)

W. W. Little, Jr.
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December, 1965

BATTELLE-NORTHWEST

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DRIVER FUEL FOR THE FAST TEST REACTOR (FTR)

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SUMMARY

The neutronic characteristics of four possible driver fuels for the Fast Test Reactor (FTR) were studied using the Russian 26-group data compilation.⁽¹⁾ The fuels investigated were PuO_2 -SS, UO_2 -SS, PuO_2 - U^{28}O_2 , and UO_2 - U^{28}O_2 . Since a high flux is of paramount importance in a test reactor, low inventory plutonium fueled cores were found to be more suitable than uranium cores for use in the FTR. For core volumes of approximately 600 liters, it was found that cermet fuels appear preferable to ceramic fuels due to the small Doppler and perhaps unreliable thermal expansion coefficient of the ceramic fuel. If compromises in flux are acceptable, the Doppler coefficient in the ceramic fuel could be enhanced by using a BeO moderator.

I. INTRODUCTION

In order to expedite the development of fast reactors, the United States Atomic Energy Commission is considering the construction of a fast test reactor. A conceptual design of a fast test reactor, henceforth called the Fast Test Reactor (FTR), is being formulated by the Battelle-Northwest Laboratory under AEC contract. This paper describes some of the neutronic aspects involved in the selection of a driver fuel for the FTR.

The fuels chosen for study were $\text{PuO}_2\text{-SS}$, $\text{UO}_2\text{-SS}$, $\text{PuO}_2\text{-U}^{28}\text{O}_2$, and $\text{UO}_2\text{-U}^{28}\text{O}_2$. Other fast reactor fuels were judged unsuitable for a state of the art design. In particular, metal fuels, although attractive from a safety standpoint, were ruled out due to severe swelling problems at high burnup; carbide fuels were excluded since they have not been adequately tested.

Before proceeding, it is important to stress the difference between the objectives of a fast breeder reactor and a fast test reactor. In a fast breeder reactor, the objective is to produce economical electrical power and conserve nuclear fuel. A high specific power, high breeding ratio, and long reactivity life are therefore of prime importance. A fast test reactor, in contrast, is designed specifically to test fast reactor fuels and structural material. A prototypic spectrum and a high flux are of primary importance; there is very little economic incentive for a high breeding ratio or a high reactor power. Indeed, high power, per se, only adds to the reactor cost by increasing the required heat dissipation facilities.

These prefatory remarks indicate that the optimum fast breeder fuel is not necessarily the optimum fuel for a fast test reactor. Although

the relative neutronic merits of the various fast breeder fuels are fairly well known, the suitability of these fuels for use in a test reactor is not immediately obvious. The purpose of this paper is to show the principal neutronic advantages and disadvantages of four possible driver fuels for the Fast Test Reactor (FTR).

The first section of this paper gives the neutronic characteristics of the four reactors as a function of reactor size. A more detailed neutronic description of four 600-liter (volume of current FTR reference design) reactors is presented in the second section. The last section shows selected neutronic parameters in an 800-liter mixed-oxide reactor.

It should be stressed at the outset that the calculations shown herein were designed to show the relative merits of four fuels. For this comparison, many details, such as the effect of poison rods, are omitted. Consequently, the parameters listed in this report should not be considered as those corresponding to the final FTR reference design.

II. PARAMETRIC SURVEY

Introductory Remarks

Perhaps the single most important figure-of-merit in a fast test reactor is the total neutron flux. The principal objective of this section is to describe the dependence of neutron flux on reactor size in PuO_2 -SS, PuO_2 - U^{28}O_2 , UO_2 -SS, and UO_2 - U^{28}O_2 fueled reactors. The size dependence of intermediate variables, such as fuel density and fuel inventory, are also shown. For reference, all reactor compositions and material densities used in this study are tabulated in Appendix A.

In order to cover a wide range of core sizes without using excessive computer time, two size surveys were performed: one using a fundamental mode model and the other a transport model. The fundamental mode survey, shown first, demonstrates the essential features of the four reactor types over an extremely wide range of core compositions. The transport theory survey gives a more precise reactor description over a smaller range of fuel compositions. The cross sections for both surveys were obtained using a modified version of the Russian 26-group data compilation.^(1,2) A discussion of this compilation, and a brief comparison with other cross section sets, is given in Appendix B.

Size Survey Using Fundamental Mode Model

The total flux is related to the power density and fuel concentration by the simple expression

$$\Phi = \frac{P}{\epsilon N \bar{\sigma}_f}, \quad (2.1)$$

where: Φ = total flux (n/cm²-sec),

P = power density (MW/l),

N = fuel atom density (atoms/barn-cm),

$\bar{\sigma}_f$ = spectrum averaged fission cross section (barns),

ϵ = conversion factor (MeV/fission x 1.6×10^{-16}).

Hence, for a fixed power density, one would expect the flux to be roughly inversely proportional to the fuel concentration.

The results of 26-group PYRE⁽³⁾ fundamental mode calculations, shown in Fig. 2.1, indicate that this prediction is indeed correct. Note that for a specified fuel volume fraction, the flux in all four reactors is approximately equal. The slight inequality is caused by the interplay

FIGURE 2.1. TOTAL FLUX AS A FUNCTION OF FISSILE FUEL CONCENTRATION

of a number of second order factors, such as variations in $\bar{\sigma}_f$, fission in fertile fuel, and small differences in isotopic fission energy and atom density.

Needless to say, Eq. (2.1) and Fig. 2.1 give no information concerning the achievable flux for a given reactor size. To obtain this information, one needs to know the critical fuel concentration as a function of core size. This relationship is shown in Fig. 2.2 for the four fuels of interest. The measure of core size employed is the spherical radius, R_S , defined by the equation

$$R_S = \pi / \sqrt{\text{Buckling } (B^2)}. \quad (2.2)$$

The desired relationship, flux as a function of core size, is given in Fig. 2.3. For sizes of interest ($R_S \sim 70$ cm), note that the low inventory plutonium fueled reactors offer a significantly higher flux than the uranium fueled reactors. Furthermore, for either plutonium or uranium fuels, observe that cermet fuels yield a higher flux than oxide fuels.

Size Survey Using Transport Model

To obtain a more precise neutronics description, the size survey was repeated using the one-dimensional (radial) DTK⁽⁴⁾ transport code. For simplicity, the reactors were assumed to consist of a homogeneous core surrounded by a 30 cm inconel radial reflector. See Appendix A for nuclide densities. The axial leakage was simulated using a reflector savings of 11 cm (22 total) on each end.

FIGURE 2.2. VOLUME PERCENT FISSILE FUEL AS A FUNCTION OF CORE RADIUS

FIGURE 2.3. TOTAL FLUX AS A FUNCTION OF CORE RADIUS

A rather complex procedure was followed to obtain cross sections for this survey. First, 26-group cross sections were generated from the Russian data⁽²⁾ for each of the four fuels. To conserve computer time, these 26-group sets were collapsed to 12 groups using the CRUNCH code.⁽⁵⁾ These four 12-group sets were then used in all survey calculations.

The critical fissile fuel inventory as a function of core volume for the four fuel materials is shown in Fig. 2.4. For all cases, the core height is assumed to be 91.5 cm. Note that the inventory of the plutonium fueled reactors is significantly less than that of the uranium fueled reactors. This difference, of course, is caused by the high η in Pu²³⁹. Also, for a given core volume, cermet cores have a lower fissile fuel inventory than oxide cores. The large capture cross section in U²³⁸ is responsible for this behavior.

It is interesting to note that for small, hard spectrum cores (~ 200 l), U²³⁸ has little effect on critical mass; for large, soft spectrum cores, U²³⁸ is a significant poison.

Figs. 2.5 and 2.6 give the volume percent of fissile fuel (PuO₂ or UO₂) as a function of core volume and radius, respectively. The trends, obviously, are similar to those given by the fundamental mode model.

The core averaged flux as a function of core volume is shown in Fig. 2.7. For all compositions, the power density is assumed to be 2/3 MW/liter. Fig. 2.8 gives the center point flux as a function of volume for an axial peak-to-average ratio of 1.25. Observe that over a large range of core volumes, the flux in a PuO₂-SS fueled reactor is

FIGURE 2.4. FISSILE FUEL INVENTORY AS A FUNCTION OF CORE VOLUME FOR VARIOUS FUEL COMPOSITIONS

FIGURE 2.5. VOLUME PERCENT FISSILE FUEL AS A FUNCTION OF CORE VOLUME

FIGURE 2.6. VOLUME PERCENT FISSILE FUEL AS A FUNCTION OF CORE RADIUS

FIGURE 2.7. CORE AVERAGED FLUX AS A FUNCTION OF VOLUME

FIGURE 2.8. CENTER POINT FLUX FOR VARIOUS FUEL COMPOSITIONS

about 20% greater than that in a $\text{PuO}_2\text{-UO}_2$ core, and about 30% greater than that in a $\text{UO}_2\text{-SS}$ fueled reactor.

It should be mentioned at this point that the stated power density of 2/3 MW/liter is strictly an arbitrary choice to facilitate comparison. If some fuel materials can withstand higher power densities than others, the flux relationships shown in Figs. 2.7 and 2.8 should be multiplied by the appropriate factors.

III. NEUTRONIC CHARACTERISTICS OF 600-LITER CORES

Outline

In order to obtain a more complete description of the four fuels, a detailed study of 600 liter versions of each of the four reactors was performed. The composition of each reactor was taken from the results presented in Chapter II. The full 26 energy groups were employed in all calculations.

The first section describes the principal neutronic parameters in the four reference reactors. Further detail, such as reactivity life and flux spectrum, are shown in subsequent sections.

Summary of Nuclear Properties

The principal static characteristics of the four reference reactors are shown in Table 3.1. This table, in effect, is simply a synopsis of the material presented in Chapter II for 600-liter reactors.

Note the clear superiority of plutonium fuel in terms of both critical mass and total flux. In the two cermet reactors, for example, the mass ratio of U^{235} to Pu^{239} is about 1.6; in the oxide reactors, the ratio is roughly 1.5. Similarly, the flux in the $\text{PuO}_2\text{-SS}$ reactor is a factor of 1.37 greater than that in the $\text{UO}_2\text{-SS}$ reactor, and the $\text{PuO}_2\text{-UO}_2$ flux is a factor of 1.26 greater than the $\text{UO}_2\text{-UO}_2$ flux.

TABLE 3.1

STATIC CHARACTERISTICS OF 600-l REACTORS

<u>Item</u>	<u>PuO₂-SS</u>	<u>PuO₂-UO₂</u>	<u>UO₂-SS</u>	<u>UO₂-U²⁸O₂</u>
k _{eff}	1.042	1.049	1.044	1.048
v/o Fissile Fuel (PuO ₂ or UO ₂)	5.5	6.5	9.1	9.7
v/o Fertile Fuel (U ²⁸ O ₂)	0	23.5	0	20.3
Mass Pu ²³⁹ (kg)	300	355	0	0
Mass Pu ²⁴⁰	33.5	39.6	0	0
Mass U ²³⁵	0	0	489	522
Mass U ²³⁸	0	1362	37.3	1216
$\bar{\Phi}$ (core average) [†]	0.780×10^{16}	0.624×10^{16}	0.570×10^{16}	0.495×10^{16}
Φ_0 (center point) [†]	1.32×10^{16}	1.10×10^{16}	0.96×10^{16}	0.870×10^{16}
Φ (> .1 MeV)/ $\bar{\Phi}$	0.710	0.679	0.760	0.726

[†]Computed at 400 MWt.

Table 3.2 gives a summary of the important kinetic parameters in the four reference reactors. A number of these items warrant emphasis and elaboration. First, observe that the uranium fueled reactors have a much larger effective delayed neutron fraction than the plutonium reactors. Hence, for a given shutdown coefficient, the plutonium fueled reactors tend to be more susceptible to prompt critical excursions.

The sodium worth in the cermet cores, of course, is significantly larger (negative void coefficient) than that in the ceramic cores. This is especially true for the central sodium worth. For example, the center channel sodium worth in the PuO_2 -SS reactor is ten times larger than that in the PuO_2 - UO_2 reactor. It goes without saying that the autocatalytic central void accident, which is conceivable in a large breeder reactor, is totally inconceivable in a 600-liter cermet fueled test reactor.

The fuel expansion coefficient listed in Table 3.2 is actually the axial expansion coefficient of both stainless steel and fuel. Radial effects, if present, were not considered.

The last two lines in Table 3.2 give the Doppler coefficient with and without sodium. In the cermet reactors, the Doppler coefficient is small and positive — in fact, completely negligible in the UO_2 -SS reactor. The Doppler coefficient in the ceramic fueled reactors, of course, is negative, although rather small in magnitude.

TABLE 3.2

KINETICS CHARACTERISTICS OF 600-l REACTORS

<u>Item</u>	<u>PuO₂-SS</u>	<u>PuO₂-UO₂</u>	<u>UO₂-SS</u>	<u>UO₂- U²⁸O₂</u>
β_{eff}	0.0022	0.0032	0.0070	0.0072
Λ_0 (sec)	4.82×10^{-7}	3.55×10^{-7}	2.95×10^{-7}	2.59×10^{-7}
Na Worth (core)	0.07339	0.02044	0.07492	0.03734
Na Worth ($r \leq 5$ cm)	0.00114	0.00011	0.00120	0.00051
Na Expansion Coef. ($^{\circ}\text{K}^{-1}$)	-19.6×10^{-6}	-5.46×10^{-6}	-20.0×10^{-6}	-9.97×10^{-6}
Fuel Expansion Coef.	-8.5×10^{-6}	-	$\sim -7.0 \times 10^{-6}$	-
Doppler Coef. ($T=1000^{\circ}\text{K}$)	$+0.40 \times 10^{-6}$	-3.5×10^{-6}	$+0.056 \times 10^{-6}$	-1.8×10^{-6}
Doppler Coef. (no Na)	$+0.27 \times 10^{-6}$	-1.6×10^{-6}	$+0.023 \times 10^{-6}$	-0.94×10^{-6}

Effect of BeO on Mixed-Oxide Reactor

Perhaps the most unsatisfactory aspect of the 600 liter mixed-oxide reactor is its small temperature coefficient. Since the major contribution to the Doppler coefficient comes from the low energy flux, a moderator, such as BeO, could be employed to enhance the Doppler coefficient. This procedure would have the added advantage of decreasing the sensitivity of the Doppler coefficient to the sodium concentration.

A comparison of unmoderated and moderated mixed-oxide fueled reactors is shown in Table 3.3. As expected, the moderated core has a significantly larger Doppler coefficient. Note, however, that one must pay a penalty of about 10% in fast ($> .1$ MeV) flux in order to achieve this gain.

TABLE 3.3
COMPARISON OF UNMODERATED AND MODERATED PuO₂-UO₂ FUELED REACTORS

<u>Item</u>	<u>No BeO</u>	<u>10 v/o BeO[†]</u>
k_{eff}	1.049	1.040
v/o PuO ₂	6.5	5.8
v/o U ²⁸ O ₂	23.5	19.2
Φ^*	1.10×10^{16}	1.12×10^{16}
Φ ($> .8$ MeV)	0.213×10^{16}	0.213×10^{16}
Φ ($> .1$ MeV)/ Φ	0.679	0.598
Na Worth (core)	0.02044	0.02650
Na Worth ($r < 5$ cm)	0.00011	0.00031
Doppler Coef. ($T=1000^\circ K$)	-3.5×10^{-6}	-7.3×10^{-6}
Doppler Coef. (no Na)	-1.6×10^{-6}	-4.9×10^{-6}

† BeO assumed to displace 5 v/o SS and 5 v/o U²⁸O₂.

* Center point flux, normalized to 400 MWt.

Reactivity Life

In fast breeder reactors there is a strong economic incentive to achieve a long reactivity life. Since the FTR will not produce electrical power, the need for a long reactivity life is not as pressing. Moreover, the insertion and withdrawal of test samples will probably require reasonably short operating cycles in the FTR.

The above statements do not mean that a long reactivity life is of no value in the FTR. A long reactivity life is of value — especially from the viewpoint of reactor safety. For example, a long reactivity life might permit the use of low worth shim rods and, perhaps of more importance, reduce the chance of loading errors during refueling. The purpose of this section is to show the penalty brought on by using a non-breeding core.

A tabulation of the prominent burnup characteristics in the four reference reactors is given in Table 3.4. The effective multiplication constant as a function of time is shown in Fig. 3.1.

For a given δk , observe that the oxide fueled reactors produce a significantly longer reactivity life than the cermet reactors. This advantage, of course, is caused by the fissile fuel generation in U^{238} . Also, note that the uranium fueled reactors, owing to their large critical mass, exhibit a smaller reactivity change per MWD than the plutonium reactors.

TABLE 3.4

BURNUP PROPERTIES IN 600 LITER REACTORS OPERATED AT 400 MWT

<u>Item</u>	<u>PuO₂-SS</u>	<u>PuO₂-UO₂</u>	<u>UO₂-SS</u>	<u>UO₂-U²³⁸O₂</u>
Initial Fertile-Fissile Ratio	0.11	3.94	0.08	2.33
Initial Breeding Ratio	0.032	0.499	0.009	0.258
Reactivity Life (days, $\delta k = .04$)	42	75	105	135
δk (T = 100 days)	0.098	0.038	0.054	0.029

FIGURE 3.1. REACTIVITY AS A FUNCTION OF TIME IN 600 LITER REFERENCE CORES

Flux Spectra and Spectrum Averaged Cross Sections

The two primary neutronic characteristics of a fast test reactor are the flux magnitude and spectrum. The latter parameter draws its importance from the need to match the fuel and cladding damage rates with those encountered in fast breeder reactors.

Needless to say, the spectrum in a test reactor could never be made to precisely duplicate that of a given FBR. Furthermore, even if such a condition could be achieved, a single test reactor could not duplicate the variety of spectra exhibited by metal, carbide, and oxide fueled breeder reactors. Therefore, rather than designing a test reactor to duplicate the spectrum in a particular FBR, it appears reasonable to seek a test reactor spectrum which approximates those in the foreseeable fast breeder reactors.

Fig. 3.2 and Table 3.5 show the flux spectra and spectrum averaged cross sections, respectively, in the four 600-l reference reactor. The flux spectrum in typical metal, carbide, and oxide fueled breeder reactors is given in Fig. 3.3. Observe that the spectrum in the PuO_2 -SS fueled test reactor is harder than that in the oxide fueled FBR, softer than that in metal fueled FBR, and roughly equivalent to that in the carbide reactor.

FIGURE 3.2. FLUX SPECTRUM IN 600 LITER REFERENCE CORES

TABLE 3.5

SPECTRUM AVERAGED CROSS SECTIONS IN 600-LITER REFERENCE CORES

<u>Isotope</u>	<u>Cross Section</u>	<u>PuO₂-SS</u>	<u>PuO₂-U²⁸O₂</u>	<u>UO₂-SS</u>	<u>UO₂-U²⁸O₂</u>
Pu ²³⁹	σ_f	1.897	1.848	-	-
	σ_a	2.281	2.197	-	-
	$\nu\sigma_f$	5.576	5.440	-	-
Pu ²⁴⁰	σ_f	0.443	0.439	-	-
	σ_a	1.105	1.057	-	-
	$\nu\sigma_f$	1.324	1.320	-	-
U ²³⁵	σ_f	-	-	1.628	1.661
	σ_a	-	-	2.025	2.078
	$\nu\sigma_f$	-	-	4.032	4.115
U ²³⁸	σ_f	-	0.061	0.060	0.066
	σ_a	-	0.327	0.297	0.299
	$\nu\sigma_f$	-	0.170	0.168	0.187
O	σ_a	0.00118	0.00157	0.00139	0.00173
Cr	σ_a	0.00598	0.00659	0.00496	0.00569
Fe	σ_a	0.00924	0.01082	0.00825	0.00964
Ni	σ_a	0.02316	0.02528	0.02419	0.02584
Na	σ_a	0.00148	0.00153	0.00118	0.00129

FIGURE 3.3. FLUX SPECTRUM IN FTR AND VARIOUS BREEDER REACTORS

IV. NEUTRONIC CHARACTERISTICS OF AN 800-LITER MIXED-OXIDE REACTOR

Introduction

One would expect oxide fuels to become more attractive with increasing size owing to the larger Doppler coefficient and reactivity life; whereas cermet fuels would become less attractive due to the smaller axial expansion coefficient. The purpose of this section is simply to show what can be achieved neutronically by going to a larger, and probably more expensive, mixed-oxide reactor.

Calculations and Results

Selected physics characteristics of an 800-liter $\text{PuO}_2\text{-UO}_2$ fueled FTR are given in Table 4.1. The core composition is described in Appendix A.

It is clear from the values presented that the 800-l oxide reactor is an improvement over the 600-l oxide reactor. In particular, note that the Doppler coefficient in the 800-l core is 20% larger than in the 600-l core. Substantial gains are also present in the flux and breeding ratio.

To avoid misunderstanding, it should be emphasized that the sodium worth values listed in Table 4.1 were obtained using a 1-D (radial) transport model. A 2-D (R-Z) model, of course, would yield slightly different results. It is also important to note that the central sodium worth, measured on a per unit mass basis, would be expected to be more negative than the axial average value given in Table 4.1.

TABLE 4.1

NEUTRONIC PROPERTIES OF AN 800-LITER PuO₂-UO₂ FUELED REACTOR

<u>Parameter</u>	<u>Numerical Value</u>
k_{eff}	1.045
$\nu/\text{o PuO}_2$	6.0
Mass Pu ²³⁹ (kg)	436
Fertile-Fissile Mass Ratio	4.36
ϕ_0 (P = 533 MWt)	1.22×10^{16}
$\bar{\phi}$	0.669×10^{16}
ϕ (> .1 MeV)/ $\bar{\phi}$	0.664
Na Worth (core)	0.0182
Na Worth (r < 5 cm)	~ 0.0
Doppler $\left(\frac{dk}{dT} \quad T = 1000^\circ\text{K} \right)$	- 4.3×10^{-6}
β_{eff}	0.0032
Λ (sec)	4.0×10^{-7}
Breeding Ratio	0.559

V. CONCLUSIONS

It is obvious from the preceding chapters that no single fuel appears best in all areas of comparison. Compromises must be made. If a high flux is of paramount importance, then, for reactors of about 600 liters, a PuO_2 -SS cermet is clearly the most appropriate fuel. If, on the other hand, a 20-30% lower flux is acceptable, UO_2 -SS, PuO_2 - UO_2 , or PuO_2 - UO_2 -BeO fuel could be employed. A uranium cermet offers a large delayed neutron fraction and a reliable expansion coefficient. A mixed oxide fuel offers a number of attractive features, combined with a rather marginal Doppler shutdown coefficient. The magnitude of this coefficient can be increased by going to a larger reactor, but this road soon leads to an undesirable positive sodium void coefficient. A better scheme for increasing the Doppler coefficient is to add BeO to the core, although this procedure slightly decreases the fast flux.

APPENDIX A

REACTOR COMPOSITION AND NUCLIDE DENSITIES

TABLE A 1

REACTOR COMPOSITION FOR SIZE SURVEY

<u>Parameter</u>	<u>Cermet</u>	<u>Ceramic</u>
Core		
v/o Fissile Fuel (PuO ₂ or UO ₂)	X	X
v/o Fertile Fuel (U ²⁸ O ₂)	0	30-X
v/o SS	55-X	25
v/o Na	40	40
v/o Void*	5	5
Radial Reflector		
v/o Inconel	80	80
v/o Na	20	20

*Balance term to account for materials used at less
than theoretical density.

TABLE A 2

MASS AND ATOM DENSITY OF VARIOUS MATERIALS

<u>Material</u>	<u>gm/cc</u>	<u>Atoms/(barn-cm)</u>
Na	0.872	0.0228440
304 SS	7.92	-
Cr (18 w/o)	-	0.0165091
Fe (74 w/o)	-	0.0632043
Ni (8 w/o)	-	0.0065000
Inconel	8.43	-
Cr (18 w/o)	-	0.0175722
Fe (10 w/o)	-	0.0090911
Ni (72 w/o)	-	0.0622674
$U^{238}O_2$	10.96	-
U^{238}	-	0.0244403
O	-	0.0488806
UO_2 (93% 25, 7% 28)	10.96	-
U^{235}	-	0.0229715
U^{238}	-	0.0017290
O	-	0.0494010
PuO_2 (90% 49, 10% 40)	11.46	-
Pu^{239}	-	0.0229103
Pu^{240}	-	0.0025456
O	-	0.0509118
BeO	2.86	-
Be	-	0.0688673
O	-	0.0688673

APPENDIX B

CROSS SECTION COMPARISON

Introduction

As mentioned in the summary, all computations shown in this report were performed using a slightly modified version of the 26-group Russian cross section compilation. The Russian fission cross sections in Pu^{239} and U^{235} were modified using recent data from Perkin⁽⁶⁾ and White;⁽⁷⁾ the capture cross sections in iron were updated using information published by Macklin.⁽⁸⁾ The objective of this brief appendix is to show how the modified Russian set compared with the well-known ANL-224⁽⁹⁾ and Hansen-Roach⁽¹⁰⁾ sets.

Results

Table B2 presents selected neutronic characteristics obtained from fundamental mode calculations using the above cross section compilations. The atom densities were taken from the 5.5 v/o PuO_2 -SS fueled reference core.

Although the three sets agree reasonably well in the calculation of k_{eff} , the wide variation in the spectrum averaged cross sections indicates that this agreement is somewhat fortuitous. Results similar to those shown in Table B2 were obtained for the other three fuels.

TABLE B 2

CROSS SECTION COMPARISON FOR A PuO₂-SS FUELED REACTOR

<u>Item</u>	<u>Modified Russian 26-Group</u>	<u>ANL-224 22-Group</u>	<u>Hansen-Roach 16-Group</u>
k_{eff}	1.050	1.058	1.065
$B^2(\times 10^3)$	1.75	1.75	1.75
$\nu/o \text{ PuO}_2$	5.5	5.5	5.5
$\Phi(> .1 \text{ MeV})/\Phi$	0.710	0.721	0.722
$\Phi(> .01 \text{ MeV})/\Phi$	0.938	0.945	0.926
$\bar{\sigma}_f^{239}$	1.8969	1.8041	1.9805
$\bar{\nu}\sigma_f^{239}$	5.5755	5.3573	5.7996
$\bar{\sigma}_a^{239}$	2.2809	2.1177	2.3198
$\bar{\sigma}_f^{240}$	0.4428	0.4416	0.4819
$\bar{\nu}\sigma_f^{240}$	1.3239	1.3550	1.6170
$\bar{\sigma}_a^{240}$	1.1053	0.8773	0.8947
$\bar{\sigma}_a^0$	0.0012	0.0010	0.0014
$\bar{\sigma}_a^{Cr}$	0.0060	0.0103	-
$\bar{\sigma}_a^{Fe}$	0.0092	0.0106	0.0097
$\bar{\sigma}_a^{Ni}$	0.0232	0.0228	0.0149
$\bar{\sigma}_a^{Na}$	0.0015	0.0014	0.0017

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