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**MULTIGROUP ANALYSIS
OF SELECTED FAST CRITICAL ASSEMBLIES**

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DECEMBER, 1966

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MULTIGROUP ANALYSIS
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By

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ABSTRACT

A modified version of the Russian data compilation is used to compute nuclear parameters for various fast critical assemblies. Calculated and experimental values of critical mass, fission ratios, and reactivity coefficients are compared. For the moderate volume plutonium and uranium assemblies analyzed, the data predicts k_{eff} within about 1.5%.

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I. INTRODUCTION

The objective of this report is to describe the applicability of a modified version of the Russian 26-group data compilation⁽¹⁾ to the analysis of moderate volume fast reactors. Since the modified Russian cross section set is currently being used in the nuclear design of the Fast Test Reactor (~800 liter, $\text{PuO}_2\text{-UO}_2$ fuel), emphasis is placed on the analysis of soft-spectrum critical assemblies with plutonium fuel.

A complete description of all modifications that have been made in the Russian data is tabulated in Appendix A. This modified set was constructed after examining numerous data sources. Particular emphasis was placed on the White⁽²⁾ and Petrel microscopic data,⁽³⁾ and the ZPR-III 47^(4,5) and 48⁽⁶⁾ integral data. A comparison of neutronics calculations using the modified Russian data to calculations performed at other laboratories is shown in Appendix B.

The cross section data is evaluated by comparing experimental and calculated central reaction rate ratios, central reactivity coefficients, and k_{eff} . In general, the first two tests are useful for detecting spectrum characteristics and cross section errors in individual elements; k_{eff} comparisons are of paramount importance in overall evaluation, but give limited information on error sources.

II. ASSEMBLIES ANALYZED

The characteristics of numerous critical assemblies can be found in the literature. Rather than attempting the extensive task of analyzing all assemblies, we decided to consider only the 16 listed in Table 2.1. Three of these--ZPR-III 47, 48, and ZEBRA 6A--are recent soft-spectrum plutonium fueled assemblies similar, in many ways, to the Fast Test Reactor (FTR). The other assemblies were selected to provide a wide range of flux spectra. A rough indication of the central flux spectrum in each assembly can be obtained from Table 2.2. The integrated fluxes and material bucklings listed in Table 2.2 were taken from 26-group FCC⁽⁷⁾ fundamental mode calculations.

The experimental data shown in this report was gathered from numerous sources. ZPR data were obtained from the Argonne National Laboratory Monthly Progress Reports, and the Geneva papers by F. W. Thalgott et al.⁽⁸⁾ and R. Avery et al.⁽⁹⁾ Information on ZEBRA 1, 2, and 3 was taken from the Geneva paper by R. D. Smith et al.⁽¹⁰⁾ ZEBRA 6 characteristics were obtained from a paper presented by A. R. Baker at the 1966 International Conference on Fast Critical Experiments and their Analysis.⁽¹¹⁾

III. REACTION RATE RATIOS

Reaction rate ratios--usually fission rate ratios--are widely used in cross section evaluation. If the fission cross section of one reference isotope, say U^{235} , is known accurately, the fission cross section of another less well known isotope can be evaluated by comparing the measured and calculated fission rate ratios. Although this approach has some practical drawbacks--due to inaccuracies in spectrum calculations and lack of a suitable reference fuel--it remains the most useful integral method for evaluating fission cross sections.

TABLE 2.1. Central Core Compositions

Assembly	Atoms/cm ³ x 10 ²²													
	U ²³⁵	U ²³⁸	Pu ²³⁹	Pu ²⁴⁰	Be	C	O	Na	Al	Cr	Fe	Ni	Mo	Cu
ZPR-III 14	0.4500	0.0350	-	-	-	5.3000	-	-	-	-	0.790	-	-	-
29	0.2390	0.4800	-	-	-	-	1.3900	-	1.470	-	2.110	-	-	-
34	0.2250	0.4950	-	-	-	0.7600	-	-	1.540	-	2.110	-	-	-
35 (FERMI B)	0.1954	0.0141	-	-	-	-	0.3994	0.7825	-	0.7556	3.180	0.2975	-	-
38	0.3638	3.5052	-	-	-	-	-	-	-	0.1507	0.572	0.0716	-	-
39	0.2792	0.4376	-	-	-	-	-	-	1.410	0.3969	1.507	0.1887	-	-
44 (RAPSODIE)	0.5034	0.3365	0.1558	0.0074	-	-	1.9570	-	1.031	0.1831	1.439	0.0908	-	-
45A	0.0015	0.7230	0.1070	0.0051	-	0.8330	-	1.0580	-	0.2030	0.757	0.1000	-	-
46B (FARET)	0.3392	0.5599	0.1565	0.0074	-	1.1080	-	0.8421	-	0.2856	1.064	0.1407	-	-
47 (SEFOR)	0.0015	0.7744	0.1480	0.0132	0.568	0.3360	1.5660	0.6630	0.699	0.2731	1.045	0.1080	0.0515	-
48	0.0016	0.7406	0.1654	0.0106	-	2.0765	-	0.6230	0.011	0.2658	0.990	0.1308	0.0206	-
ZPR-VI 2	0.2285	0.9853	-	-	-	1.2900	-	0.7910	-	0.2983	1.038	0.1390	-	-
ZEBRA 1	0.4700	3.4000	-	-	-	-	-	-	-	0.1100	0.410	0.0420	-	-
2	0.2400	1.5000	-	-	-	3.8000	-	-	-	0.1000	0.380	0.0390	-	-
3	0.0230	3.1000	0.3520	0.0180	-	-	-	-	-	0.1100	0.420	0.0430	-	0.4600
6A	0.0046	0.6353	0.1895	0.0144	-	2.9590	-	0.4474	0.2504	0.1270	0.4525	0.0443	-	0.0829

Table 2.2. Calculated Critical Assembly Characteristics

<u>Assembly</u>		<u>Percent of Flux Above Indicated Energy</u>			<u>Critical Buckling $B^2 \times 10^3$</u>
		<u>0.01 MeV</u>	<u>0.1 MeV</u>	<u>1.4 MeV</u>	
ZPR-III	14	94.4	74.7	22.2	5.58
	29	97.2	73.2	11.8	2.42
	34	97.2	74.5	11.3	2.01
	35 (FERMI B)	97.0	75.9	10.1	1.80
	38	99.9	85.0	8.9	3.22
	39	99.3	83.4	13.1	2.14
	44 (RAPSODIE)	99.5	87.1	22.1	8.55
	45A	93.2	67.2	10.0	1.07
	46B (FARET)	99.1	85.7	20.2	5.88
	47 (SEFOR)	89.3	61.0	10.3	2.40
	48	92.0	67.2	12.1	2.54
ZPR-VI	2	96.7	73.5	11.4	2.08
ZEBRA	1	99.9	87.1	10.8	5.62
	2	92.8	66.5	11.8	2.59
	3	99.8	87.3	12.0	8.29
	6A	90.9	67.1	14.6	3.27

The fission rate ratios of Pu^{239} -to- U^{235} and U^{233} -to- U^{235} are shown in Tables 3.1 and 3.2. All ratios were computed using the FCC fundamental mode code. Since these ratios are insensitive to flux spectra, the satisfactory agreement indicates that the relative fission cross sections for U^{233} , U^{235} , and Pu^{239} are roughly correct.

Fission cross section ratios for threshold fuels-- Pu^{240} , U^{234} , U^{236} , and U^{238} --are listed in Tables 3.3 through 3.6. Note that the Pu^{240} ratios are predicted reasonably well in the soft-spectrum plutonium fueled assemblies. In contrast, the calculated U^{238} fission ratios are consistently much too high. This implies that either the U^{238} fission cross section is too large or the flux spectrum is too hard. If the U^{238} fission cross sections are lowered, the agreement between the measured and calculated U^{238} reactivity coefficients (See Section IV) is lost. If the spectrum is too hard, it is difficult to explain the Pu^{240} fission ratios and the U^{238} reactivity coefficients. Thus, these anomalous U^{238} fission ratios are probably the most unsatisfactory aspect of this data compilation.

TABLE 3.1. Central Reaction Rate Ratios of Pu^{239} -to- U^{235}

<u>Assembly</u>	$\bar{\sigma}_f^{239}/\bar{\sigma}_f^{235}$		<u>C/E</u> ^(a)
	<u>Experimental</u>	<u>Calculated BNW</u>	
ZPR-III 14	1.050	1.089	1.037
29	1.060	1.080	1.019
34	1.067	1.088	1.020
35 (FERMI B)	1.090	1.084	0.994
38	1.170	1.085	1.013
39	1.180	1.181	1.001
44 (RAPSODIE)	1.186	1.235	1.041
45A	1.047	0.969	0.926
46 (FARET B)	-	1.212	-
47 (SEFOR)	0.916	0.900	0.983
48	0.976	0.959	0.983
ZPR-VI 2	1.090	1.074	0.985
ZEBRA 1	1.198	1.208	1.009
2	0.987	1.003	1.016
3	1.190	1.212	1.018
6A	0.961	0.949	0.988

(a)

Ratio of Calculated Value to Experimental Value

TABLE 3.2. Central Reaction Rate Ratios of U^{233} -to- U^{235}

<u>Assembly</u>		$\bar{\sigma}_f^{233}/\bar{\sigma}_f^{235}$		<u>C/E</u>
		<u>Experimental</u>	<u>Calculated BNW</u>	
ZPR-III	14	1.450	1.443	0.995
	29	1.470	1.429	0.972
	34	1.454	1.434	0.986
	35 (FERMI B)	1.409	1.459	1.035
	38	-	1.464	-
	39	-	1.453	-
	44 (RAPSODIE)	1.532	1.446	0.944
	45A	1.440	1.469	1.020
	46 (FARET B)	-	1.447	-
	47 (SEFOR)	1.451	1.480	1.020
	48	1.480	1.469	0.993
ZPR-VI	2	1.450	1.443	0.995
ZEBRA	1	1.550	1.467	0.946
	2	1.453	1.441	0.992
	3	1.542	1.466	0.951
	6A	1.420	1.470	1.035

TABLE 3.3. *Central Reaction Rate Ratios of Pu²⁴⁰-to-U²³⁵*

<u>Assembly</u>		$\bar{\sigma}_f^{240} / \bar{\sigma}_f^{235}$		<u>C/E</u>
		<u>Experimental</u>	<u>Calculated BNW</u>	
ZPR-III	14	-	0.396	-
	29	0.301	0.296	0.982
	34	0.282	0.307	1.089
	35 (FERMI B)	0.250	0.287	1.148
	38	-	0.330	-
	39	0.326	0.396	1.215
	44 (RAPSODIE)	0.490	0.512	1.045
	45A	0.218	0.219	1.005
	46 (FARET B)	-	0.482	-
	47 (SEFOR)	0.200	0.182	0.910
	48	0.243	0.239	0.984
ZPR-VI	2	0.254	0.291	1.146
ZEBRA	1	0.355	0.372	1.048
	2	0.237	0.248	1.046
	3	0.373	0.394	1.056
	6A	0.252	0.255	1.012

TABLE 3.4. Central Reaction Rate Ratios of U^{234} -to- U^{235}

Assembly		$\bar{\sigma}_f^{234} / \bar{\sigma}_f^{235}$		C/E
		Experimental	Calculated BNW	
ZPR-III	14	0.317	0.347	1.095
	29	0.269	0.263	0.978
	34	0.257	0.273	1.062
	35 (FERMI B)	0.241	0.256	1.062
	38	-	0.298	-
	39	-	0.352	-
	44 (RAPSODIE)	0.445	0.463	1.040
	45A	0.182	0.197	1.082
	46 (FARET B)	-	0.427	-
	47 (SEFOR)	-	-	-
	48	0.204	0.212	1.039
ZPR-VI	2	0.233	0.258	1.107
ZEBRA	1	-	0.334	-
	2	-	0.220	-
	3	-	0.356	-
	6A	0.209	0.226	1.081

TABLE 3.5. *Central Reaction Rate Ratios of U^{236} -to- U^{235}*

<u>Assembly</u>		$\bar{\sigma}_f^{236}/\bar{\sigma}_f^{235}$		<u>C/F</u>
		<u>Experimental</u>	<u>Calculated BNW</u>	
ZPR-III	14	-	0.139	-
	29	0.087	0.085	0.977
	34	0.081	0.087	1.074
	35 (FERMI B)	-	0.078	-
	38	-	0.081	-
	39	-	0.113	-
	44 (RAPSODIE)	0.175	0.174	0.994
	45A	0.063	0.067	1.063
	46 (FARET B)	-	0.154	-
	47 (SEFOR)	-	-	-
	48	0.067	0.072	1.075
ZPR-VI	2	-	0.084	-
ZEBRA	1	-	0.098	-
	2	-	0.075	-
	3	-	0.109	-
	6A	0.072	0.082	1.139

TABLE 3.6. Central Reaction Rate Ratios of U^{238} -to- U^{235}

Assembly		$\bar{\sigma}_f^{238} / \bar{\sigma}_f^{235}$		C/E
		Experimental	Calculated BNW	
ZPR-III	14	0.060	0.071	1.182
	29	0.038	0.040	1.053
	34	0.037	0.039	1.054
	35 (FERMI B)	0.031	0.034	1.100
	38	0.033	0.035	1.061
	39	0.048	0.050	1.042
	44 (RAPSODIE	0.082	0.086	1.049
	45A	0.028	0.029	1.036
	46 (FARET B)	-	0.078	-
	47 (SEFOR)	0.026	0.026	1.000
	48	0.031	0.034	1.097
ZPR-VI	2	0.036	0.038	1.056
ZEBRA	1	0.041	0.043	1.049
	2	0.032	0.035	1.094
	3	0.046	0.048	1.043
	6A	0.036	0.040	1.111

Finally, the U^{238} capture ratios, which give an indication of the error in breeding ratio and doubling time, are tabulated in Table 3.7. In general, the agreement seems satisfactory, although a few large discrepancies, such as in ZPR-III 45A, have not been resolved.

TABLE 3.7. *Central Reaction Rate Ratios of U^{238} -to- U^{235}*

<u>Assembly</u>	$\bar{\sigma}_f^{238}/\bar{\sigma}_f^{235}$		<u>C/E</u>
	<u>Experimental</u>	<u>Calculated BNW</u>	
ZPR-III 14	0.134	0.141	1.054
29	0.141	0.137	0.970
34	0.150	0.135	0.901
35 (FERMI B)	0.136	0.147	1.083
38	0.113	0.125	1.106
39	-	0.124	-
44 (RAPSODIE)	0.108	0.111	1.028
45A	0.152	0.133	0.875
46 (FARET B)	0.111	0.115	1.036
47 (SEFOR)	0.144	0.135	0.938
48	0.138	0.133	0.964
ZPR-VI 2	0.130	0.134	1.031
ZEBRA 1	-	0.120	-
2	0.136	0.133	0.978
3	0.111	0.119	1.072
6A	-	0.131	-

IV. CENTRAL REACTIVITY WORTHS

Central reactivity coefficients, which can often be measured with high accuracy, are an ideal test for evaluating the cross section of any particular isotope or mixture. A comparison of measured and calculated reactivity coefficients in ZPR-III 47 (SEFOR) is shown in Table 4.1. The experimental values were taken from Reference 5; the calculated values were obtained using the DTF-IV⁽¹²⁾ transport code in spherical geometry. Note that the Pu^{239} , U^{235} , and B_4C worths are accurately predicted, whereas the U^{238} worth is poorly predicted.

If small material samples are used, it is more convenient to express central worths in terms of the fundamental mode perturbation cross section. The perturbation cross section, σ_p , can be computed by the formula

$$\sigma_p = \frac{\sum_{i=1}^N \left\{ (\nu\sigma_f)^i \phi_i - \sigma_a^i \phi_i \phi_i^+ - \phi_i \sum_{j=i+1}^N \sigma_{i \rightarrow j} (\phi_i^+ - \phi_j^+) \right\}}{\sum_{i=1}^N \phi_i}, \quad (4.1)$$

where the adjoint flux, ϕ_i^+ , is normalized such that

$$\sum_{i=1}^N x_i \phi_i^+ = 1. \quad (4.2)$$

TABLE 4.1. ZPR-III 47 (SEFOR) Reactivity Coefficients

<u>Sample</u>		<u>Reactivity Worth (mk)</u>	
<u>Material</u>	<u>Mass (g)</u>	<u>Experimental</u>	<u>Calculated</u>
Pu-SS-Al	69.1	0.297	0.295
Enriched UA1	72.1	0.223	0.213
Depleted uranium	2460.4	-0.535	-0.719
B ₄ C (Enriched)	32.4	-1.445	-1.416
Sodium	91.2	-0.031	-0.043
Stainless Steel	1026.1	-0.114	-0.146
Iron	1028.0	-0.099	-0.141
Beryllium	240.7	0.082	0.036

A comparison between experimental⁽⁶⁾ and calculated σ_p values in ZPR-III 48 is shown in Table 4.2. For convenience, the experimental data have been normalized to the calculated U²³⁵ value. In general, the agreement is satisfactory, although large discrepancies are present in Pu²⁴⁰ and B¹⁰.

Experimental and calculated perturbation cross sections for ZEBRA assemblies 1, 2, and 3⁽¹⁰⁾ are given in Table 4.3. No renormalization was used in Table 4.3 since the published ZEBRA fuel values were very close to those given by Equation (4.1). Note that, as in ZPR-III assemblies 47 and 48, the sodium worth is predicted with moderate accuracy. This agreement is somewhat surprising in view of the sensitivity of the sodium coefficient to small cross section changes.

TABLE 4.2. *ZPR-III 48 Central Perturbation Cross Sections*

<u>Sample</u>	σ_p , barns	
	<u>Experimental⁽⁶⁾</u>	<u>Calculated</u>
U ²³⁵	2.385	2.385
Pu ²³⁹	3.230	3.368
Pu ²⁴⁰	0.550	0.282
U ²³⁸	-0.177	-0.169
B ¹⁰	-2.680	-2.432
Carbon	-0.003	-0.002
Sodium	-0.004	-0.005
Oxygen	-0.003	-0.005
Iron	-0.020	-0.024
Nickel	-0.032	-0.036
Chromium	-0.015	-0.022
Molybdenum	-0.125	-0.155
Tantalum	-0.890	-0.073

Table 4.3. Central Perturbation Cross Sections In Zebra Assembly

Sample	ZEBRA 1		ZEBRA 2		ZEBRA 3	
	Experimental	Calculated	Experimental	Calculated	Experimental	Calculated
U ²³⁵	2.043	2.070	2.282	2.293	2.049	2.088
Pu ²³⁹	3.343	3.391	3.236	3.330	3.299	3.420
U ²³⁸	-0.104	-0.111	-0.177	-0.170	-0.103	-0.110
B ¹⁰	-1.100	-0.846	-3.136	-2.178	-1.091	-0.805
Carbon	-0.012	-0.013	+0.010	+0.006	-0.041	-0.035
Sodium	-0.013	-0.015	+0.005	+0.004	-0.026	-0.029
Iron	-	-0.034	-0.019	-0.028	-	-0.040
Tantalum	-0.327	-0.307	-0.790	-0.734	-0.312	-0.318
Aluminum	-0.015	-0.022	-0.009	-0.013	-0.028	-0.034

As an additional check on the important sodium coefficient, the total-core sodium worth was computed (using a 2-D transport model) for ZEBRA assembly 6A. A comparison between the measured and calculated worth is given in Table 4.4. In light of all the uncertainties involved in comparisons of absolute worth, the agreement is within acceptable bounds.

TABLE 4.4. *Sodium Worth in Zebra 6A*

Experimental ⁽¹¹⁾	+ 0.0126 $\delta k/k$
Calculated	+ 0.0108 $\delta k/k$

V. CRITICAL MASS CALCULATIONS

An important criterion for evaluating a data set is its ability to predict critical size or critical mass. However, it is often difficult to decide whether a given discrepancy between calculated and experimental critical mass is caused by inaccurate data or improper models. Large errors, such as 4% in k_{eff} , can usually be attributed to inaccurate data; small errors, the order of 0.5% in k_{eff} , can be caused by numerous factors.

Table 5.1 gives the calculated effective multiplication constant for eight assemblies. All calculations were performed using the 2DF⁽¹³⁾ transport code (S_4 option, R-Z geometry) with five energy groups. The collapsed 5-group cross sections were computed in FCC by simply flux-weighting the initial 26-group data.

From Table 5.1, it can be seen that the error in k_{eff} is rather small for the three soft spectrum plutonium fueled assemblies. The corresponding percentage error in critical mass (added homogeneously to the core) is roughly twice the error in k_{eff} . The largest discrepancy is in the soft spectrum U^{235} fueled ZEBRA 2 assembly. This indicates, perhaps, that the low energy fission or capture cross sections for U^{235} are incorrect.

Table 5.1. *Calculated Effective Multiplication Constant*

			<u>k_{eff}</u>
ZPR-III	35	(FERMI B)	0.993
	44	(RAPSODIE)	1.004
	46B	(FARET)	0.981
	47	(SEFOR)	1.006
	48		1.002
ZEBRA	2		0.976
	3		0.991
	6A		0.998

CONCLUSIONS

As stated in Section I, the primary purpose of this study was to assess the accuracy of a modified version of the Russian data compilation for moderate volume fast assemblies. From the data presented, it is clear that errors of about 1% in k_{eff} could be anticipated, with an upper bound of about 2%. Reaction rate ratios are generally accurate to within 10%. Reactivity coefficients, especially those involving small differences of large terms, often exhibit larger percentage errors.

For the model considered, it is the authors' contention that no simple cross section change or changes will significantly improve the agreement between calculated and experimental values. To achieve better comparisons, it may be necessary to employ detailed spatial and energy models, together with more precise experimental techniques.

APPENDIX ATABULATION OF GROUP CROSS SECTIONS

For reference, the modified fission and capture cross sections are listed in Table A.1. The only other modification is an overall reduction of 1% in σ for Pu^{239} .

TABLE A.1. *Modified Cross Sections*

Group	E_L MeV	U^{233}		U^{234}		U^{235}		U^{236}		Pu^{239}		Pu^{241}		Fe
		σ_f	σ_c	σ_f	σ_c	σ_f	σ_c	σ_f	σ_c	σ_f	σ_c	σ_f	σ_c	σ_c
1	6.5	2.100	0.010	1.880	0.0	1.750	0.020	1.500	0.010	2.200	0.010	2.000	0.000	0.045
2	4.0	1.520	0.020	1.300	0.017	1.150	0.030	0.880	0.019	1.900	0.021	1.500	0.011	0.029
3	2.5	1.600	0.027	1.400	0.037	1.250	0.040	0.840	0.047	1.970	0.030	1.550	0.022	0.009
4	1.4	1.840	0.040	1.430	0.080	1.310	0.062	0.770	0.090	2.000	0.041	1.700	0.035	0.002
5	0.8	1.840	0.070	1.180	0.242	1.250	0.120	0.400	0.314	1.750	0.058	1.550	0.072	0.003
6	0.4	1.850	0.104	0.600	0.214	1.200	0.166	0.015	0.150	1.640	0.107	1.500	0.104	0.005
7	0.2	2.190	0.190	0.120	0.300	1.350	0.240	0.0	0.330	1.630	0.167	1.700	0.170	0.009
8	0.1	2.200	0.240	0.040	0.450	1.525	0.359	0.0	0.400	1.680	0.240	2.070	0.256	0.007
9	0.0465	2.300	0.276	0.030	0.550	1.850	0.528	0.0	0.600	1.700	0.289	2.370	0.330	0.009
10	0.0215	3.000	0.514	0.020	0.750	2.300	0.869	0.0	0.800	1.750	0.561	2.850	0.458	0.020
11	0.01	4.100	0.758	0.020	1.000	2.850	1.256	0.0	1.100	1.800	0.810	3.400	0.611	0.025
12	0.00465	4.600	0.976	0.020	1.400	4.000	1.910	0.0	1.600	2.500	1.250	4.070	0.814	0.062
13	0.00215	7.400	1.730	0.0	2.200	5.400	2.750	0.0	2.500	3.400	1.900	6.150	1.367	0.023
14	0.001	12.500	3.125	0.0	3.500	7.300	3.800	0.0	4.000	4.200	2.600	8.400	2.030	0.131
15	0.000465	21.100	6.154	0.0	5.500	11.000	6.300	0.0	6.000	7.600	5.000	12.400	3.100	0.027
16	0.000215	29.100	11.640	0.0	8.000	16.000	9.500	0.0	10.000	13.000	9.100	26.600	6.967	0.029
17	0.0001	41.000	16.400	0.0	21.000	22.000	13.500	0.0	14.500	19.000	14.000	28.200	7.520	0.031

APPENDIX BINTERCOMPARISON OF NEUTRONICS PARAMETERS IN A
5:1 OXIDE CORE

Table B.1 demonstrates how the modified 26-group cross section set described in this report compares with other sets in a 5:1 mixed-oxide fueled reactor. The data in Table B.1, excluding Battelle's entry, was taken from a paper presented by D. Okrent at the 1965 International Conference on Safety, Fuels, and Core design in Large Fast Power Reactors.⁽¹⁴⁾ It is clear that the modified Russian set is less reactive than all other sets shown.

TABLE B.1. Intercomparison of 5:1 Oxide Fueled Reactor

	$B^2 \times 10^4$	Percent δk for Removal of 50% Na	Doppler Effect, $\frac{Tdk}{dT}$	$\frac{\sigma_f^{U^{238}}}{\sigma_f^{Pu^{239}}}$	$\frac{\sigma_c^{U^{238}}}{\sigma_f^{Pu^{239}}}$	$\frac{\sigma_c^{Pu^{239}}}{\sigma_f^{Pu^{239}}}$
AI	18.2	-3.64	-0.0067	0.030	0.12	0.23
ANL	15.6	-3.65	-0.0049	0.027	0.14	0.21
APDA	16.6	-3.79	-0.0055	0.031	0.14	0.19
BNL	16.6	-3.46	-0.0009	0.032	0.13	0.17
B&W	17.9	-4.53	-0.0068	0.029	0.14	0.23
France	15.6	-3.4 (900°)	-0.0034	0.030	0.14	0.21
GE	16.5	-4.49	-0.0046	0.029	0.13	0.19
Karlsruhe	~15.0	-2.8 (900°)	-0.0052	---	---	0.20
UK	~14.9	---	-0.0077	0.028	0.15	0.21
UNC	15.8	---	---	0.032	0.14	0.16
BNW	14.3	-3.05	-0.0054	0.032	0.15	0.21

REFERENCES

1. I. I. Bondarenko (ed.). Group Constants for Nuclear Reactor Calculations, Consultants Bureau, New York, 1965.
2. P. H. White, J. G. Hodgkinson, and G. J. Wall. "Measurements of Fission Cross Sections for Neutrons of Energies in the Range 40-500 keV," IAEA Conference on the Physics and Chemistry of Fission, Salzburg, Austria, March 1965.
3. B. C. Diven. Use of Nuclear Explosions as a Pulsed Neutron Cross Section Technology, CONF-660303, March 1966.
4. Reactor Development Program Progress Report, January 1966, ANL-7152. Argonne National Laboratory, Lemont, Illinois, January 1966.
5. Reactor Development Program Progress Report, February 1966, ANL-7176. Argonne National Laboratory, Lemont, Illinois, February 1966.
6. W. G. Davey. "Intercomparison of Calculations for a Dilute Plutonium Fueled Fast Critical Assembly," International Conference on Fast Critical Experiments and Their Analysis, ANL-7320. Argonne National Laboratory, Lemont, Illinois, October 1966.
7. W. W. Little, Jr., and R. W. Hardie. FCC - A Fundamental Mode Code for Fast Reactor Analysis, BNWL-234. Pacific Northwest Laboratory, Richland, Washington, March 1966.
8. F. W. Thalgott, J. K. Long, W. G. Davey, W. Y. Kato, S. G. Carpenter, H. A. Morewitz, and G. H. Best, "Fast Critical Experiments and Their Analysis," Proceedings of Third United Nations Conference on Peaceful Uses of Atomic Energy, Geneva, August 1964, vol. 6, pp. 124-133. New York, 1964.
9. R. Avery, H. H. Hummel, R. N. Hwang, D. Meneghetti, P. A. Moldauer, A. B. Smith, P. Greebler, and J. B. Nims. "Physics of Fast Reactors," Proceedings of Third United Nations Conference on Peaceful Uses of Atomic Energy, Geneva 1964, vol. 6, pp. 112-123. New York, 1964.

10. R. D. Smith, J. L. Rowlands, A. R. Baker, D. C. Smith, E. P. Hicks, J. E. Mann, and J. Weale. "Fast Reactor Physics, Including Results from UK Zero Power Reactors," Proceedings of Third United Nations Conference on Peaceful Uses of Atomic Energy, Geneva, August 1964, vol. 6, pp. 90-99. New York 1964.
11. J. Adamson et al., "ZEBRA 6: A Dilute Plutonium Fuelled Assembly," International Conference on Fast Critical Experiments and Their Analysis, ANL-7320. Argonne National Laboratory, Lemont, Illinois, October 1966.
12. K. D. Lathrop. DTF-IV, A FORTRAN-IV Program for Solving the Multigroup Transport Equation with Anisotropic Scattering, LA-3373, Los Alamos Scientific Laboratory, Los Alamos, New Mexico, November 1965.
13. Two-dimensional Transport Code from the Los Alamos Scientific Laboratory, (unpublished).
14. D. Okrent. "Summary of Intercomparison Calculations," International Conference on Safety, Fuels, and Core Design in Large Fast Power Reactors, ANL-7120, Argonne National Laboratory, Lemont, Illinois, 1965.

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