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BNWL-281

UC-80 Reactor Technology

NEUTRONICS CHARACTERISTICS OF POSSIBLE
DRIVER FUELS FOR THE FAST TEST REACTOR (FTR)

by

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ABSTRACT

This report describes the principal neutronics characteristics of 14 possible driver fuels for the Fast Test Reactor (FTR). The fuels considered are PuO_2 -SS cermet, $(\text{Pu-U})\text{O}_2$ and $(\text{Pu-U})\text{C}$ ceramics, and Pu-U-Zr metals. For each fuel, core volumes of 600 and 800 liters are studied. The cermet fueled reactors are found to produce the highest flux for a given power density. Unmoderated 600 and 800 liter ceramic fueled reactors appear to have only a marginal Doppler coefficient. The Doppler coefficient, however, can be significantly increased by adding BeO to the core.

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I. INTRODUCTION

One of the important problems in the design of the Fast Test Reactor (FTR) is the selection of a suitable driver fuel. This selection process touches many areas, such as irradiation stability, fuel cost, and neutronics. The discussion in the present document is confined to the neutronics aspects of the problem.

Fourteen reactors, or fuel compositions, are examined. These 14 reactors are described in Table 1.1. The detailed compositions are given in Appendix A. Four basic fuels are included: stainless steel cermet, oxides, carbides, and metals. For each fuel, 600 and 800 liter versions are studied. The primary neutronics parameters computed for each reactor are critical mass, flux, flux spectrum, reactivity life, β coefficient, and Doppler coefficient. Many other parameters of secondary importance are also described. Since the main purpose of this paper is simply to compare the neutronics characteristics of various possible driver fuels, the effect of many important reactor details, such as fuel zoning and poison rods, are omitted.

Considerable work on cermet and oxide driver fuels has been reported in BNWL-197, "Neutronics Considerations in the Selection of a Driver Fuel for the Fast Test Reactor (FTR)."⁽¹⁾ For the sake of completeness and to facilitate comparisons, some of the information in BNWL-197 has been reported in the present report.

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FUEL	TYPE	MODERATOR	CORE VOLUME (LITERS)	CODE NAME
PuO ₂ - Stainless Steel	Cermet	----	600	PuO ₂ - SS - 600
PuO ₂ - Stainless Steel	Cermet	----	800	PuO ₂ - SS - 800
PuO ₂ - U ²⁸ O ₂	Ceramic	----	600	(Pu-U)O ₂ - 600
PuO ₂ - U ²⁸ O ₂	Ceramic	----	800	(Pu-U)O ₂ - 800
PuO ₂ - U ²⁸ O ₂	Ceramic	BeO	600	(Pu-U)O ₂ - BeO - 600
PuO ₂ - U ²⁸ O ₂	Ceramic	BeO	800	(Pu-U)O ₂ - BeO - 800
PuC - U ²⁸ C	Ceramic	----	600	(Pu-U)C - 600
PuC - U ²⁸ C	Ceramic	----	800	(Pu-U)C - 800
PuC - U ²⁸ C	Ceramic	BeO	600	(Pu-U)C - BeO - 600
PuC - U ²⁸ C	Ceramic	BeO	800	(Pu-U)C - BeO - 800
Pu - U - Zr	Metal	----	600	Pu - U - Zr - 600
Pu - U - Zr	Metal	----	800	Pu - U - Zr - 800
Pu - U - Zr	Metal	BeO	600	Pu - U - Zr - BeO - 600
Pu - U - Zr	Metal	BeO	800	Pu - U - Zr - BeO - 800

TABLE 1.1

Description of Reference Reactors

II. RESULTS OF CRITICALITY CALCULATIONS

Table 2.1 gives the critical mass of the 14 reference reactors. All calculations were performed using the 1-D (radial) DTF-IV⁽²⁾ transport code with 8 energy groups. Axial leakage was simulated by using a reflector saving of 11 cm (22 total). The 8-group cross sections for each fuel composition were obtained by collapsing the 26-group Russian (modified) cross sections.^(3,4) A comparison of the modified Russian cross section sets with other sets is given in Appendix C.

The results shown in Table 2.1 indicate that U^{238} behaves like a mild poison. In other words, for a given core size, the reactors containing the most fertile fuel also contain the most fissile fuel. Further, since

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

where: ϕ = total flux ($n\text{-cm}^{-2}\text{-sec}^{-1}$),

P = power density ($\text{MW}\text{-}\ell^{-1}$),

N = fuel density ($\text{atoms-barn}^{-1}\text{-cm}^{-1}$),

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

ϵ = conversion factor ($\text{MeV-fission}^{-1} \times 1.6 \times 10^{-16}$),

the reactors containing the most fertile fuel exhibit the lowest flux (for a specified power density).

The above arguments can be illustrated more clearly by graphs. Figs. 2.1 and 2.2 (from BNWL-197) show the fissile inventory and flux, respectively, in oxide and cermet fueled reactors. Observe that the cermet fueled reactors have a lower fissile inventory and a higher flux over a wide range of core volumes.

	Volume Percent Fissile Fuel † (PuO ₂ , PuC, or Pu-Zr)	Mass Pu ²³⁹ (Kg)	Mass Pu ²⁴⁰ (Kg)	Mass U ²³⁸ (Kg)	Fertile-Fissile Ratio	ϕ (n-cm ⁻² -sec ⁻¹) ($\bar{P}$ = 1 Mw/l)	$\phi_o/\bar{\phi}$ ‡
PuO ₂ - SS - 600	5.433	296	33.1	0	0.11	1.173 x 10 ¹⁶	1.65
PuO ₂ - SS - 800	4.860	354	39.4	0	0.11	1.297	1.71
(Pu-U)O ₂ - 600	6.396	349	38.9	1368	4.05	0.939	1.75
(Pu-U)O ₂ - 800	5.956	433	48.3	1858	4.42	1.003	1.80
(Pu-U)O ₂ - BeO - 600	6.053	330	36.9	1098	3.45	0.951	1.75
(Pu-U)O ₂ - BeO - 800	5.627	409	45.7	1497	3.78	1.014	1.81
(Pu-U)C - 600	5.360	375	41.8	1914	5.24	0.865	1.76
(Pu-U)C - 800	5.017	468	52.2	2588	5.67	0.919	1.83
(Pu-U)C - BeO - 600	5.085	355	39.6	1547	4.48	0.884	1.78
(Pu-U)C - BeO - 800	4.752	443	49.4	2097	4.87	0.939	1.84
Pu - U - Zr - 600	5.260	358	40.0	1489	4.28	0.958	1.73
Pu - U - Zr - 800	4.872	443	49.4	2024	4.70	1.031	1.79
Pu - U - Zr - BeO - 600	4.990	340	38.0	1132	3.45	0.973	1.74
Pu - U - Zr - BeO - 800	4.629	421	46.9	1546	3.80	1.042	1.80

* K_{eff} = 1.05

† See Appendix A

‡ Assuming Axial Peak-to-Average of 1.25

TABLE 2.1

Results of Criticality Calculations*
(BeO, if Present, Occupies 5 v/o of Core)

FIGURE 2.1

Fissile Fuel Inventory as a Function of Core Volume

FIGURE 2.2

Center Point Flux as a Function of Core Volume

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In passing, it should be mentioned that U^{235} fueled reactors were not included in this study because they have a larger critical mass and thus a lower flux than Pu^{239} fueled reactors. Previous investigations⁽¹⁾ indicate that $M^{235}/M^{239} \approx 1.5$, and $\phi^{235}/\phi^{239} \approx .77$.

III. FLUX SPECTRA

The FTR will have a flux spectrum similar to that anticipated in large breeder reactors. However, since it is easier to soften spectra than to harden spectra, it might be preferable to start with a hard flux reactor. This seems all the more reasonable if one wishes to perform accelerated testing of structural elements.

The flux spectra in 600-liter cermet, oxide, carbide, and metal fueled reactors are shown in Figure 3.1. Tabulated values of the integrated flux in all 14 reactors are given in Table 3.1. All calculations were performed using a fundamental mode model with the Russian 26-group cross sections. In general, metal fuels produce the hardest flux — followed by carbide, cermet, and oxide. Note that BeO has a marked effect on the low energy flux, but does not significantly alter the high energy flux.

The fundamental mode neutron importance, or adjoint flux, is shown in Figure 3.2. Again, only the 600-liter cermet, oxide, carbide, and metal fuels are included. Other core sizes, with or without BeO, exhibit similar behavior.

Observe that the adjoint flux increases with increasing energy (in the range .01 MeV to 10 MeV) in cores containing fertile fuel, and decreases with increasing energy in the cermet fueled core. This trend has two direct implications. First, any low energy absorber, such as B^{10} , would tend to be worth more in a cermet fueled core than in a core containing fertile fuel. Second, the presence of fertile fuel causes the Na void coefficient to become less negative. More will be said on this point in Section V.

FIGURE 3.1

Fundamental Mode Flux as a Function of Energy

	$\phi (> 1.4)/\phi$ (ENERGY IN MeV)	$\phi (> 0.1)/\phi$	$\phi (> 0.01)/\phi$	ϕ_{FM} (n-cm ⁻² - sec ⁻¹) ($\bar{P} = 1$ Mw/l)	$\bar{\phi}_{(1-D)}$ (n-cm ⁻² - sec ⁻¹) ($\bar{P} = 1$ Mw/l)
PuO ₂ - SS - 600	0.089	0.708	0.938	1.240 x 10 ¹⁶	1.173 x 10 ¹⁶
PuO ₂ - SS - 800	0.083	0.690	0.929	1.360	1.297
(Pu-U)O ₂ - 600	0.105	0.676	0.940	0.971	0.939
(Pu-U)O ₂ - 800	0.100	0.663	0.935	1.031	1.003
(Pu-U)O ₂ - BeO - 600	0.111	0.634	0.908	0.987	0.951
(Pu-U)O ₂ - BeO - 800	0.101	0.621	0.902	1.046	1.014
(Pu-U)C - 600	0.101	0.702	0.948	0.890	0.865
(Pu-U)C - 800	0.096	0.691	0.944	0.942	0.919
(Pu-U)C - BeO - 600	0.103	0.658	0.919	0.914	0.884
(Pu-U)C - BeO - 800	0.098	0.646	0.914	0.965	0.939
Pu - U - Zr - 600	0.098	0.762	0.973	0.987	0.958
Pu - U - Zr - 800	0.092	0.750	0.970	1.058	1.031
Pu - U - Zr - BeO - 600	0.100	0.689	0.938	1.009	0.973
Pu - U - Zr - BeO - 800	0.095	0.676	0.933	1.076	1.042

TABLE 3.1

Spectrum Characteristics in Reference Reactors

FIGURE 3.2

Fundamental Mode Adjoint Flux as a Function of Energy

IV. BURNUP CALCULATIONS

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.

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 since it can directly contribute to greater operating efficiency and indirectly contribute to 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.

A tabulation of the prominent burnup characteristics in the 14 reference reactors is given in Table 4.1. The effective multiplication constant as a function of time in selected 600-M reactors is shown in Fig. 4.1. All calculations were performed using a O-D model with 26-energy groups.

For a given δk , observe that the ceramic and metal fueled reactors produce a significantly longer reactivity life than the cermet reactors. This advantage, of course, is caused by the presence of fertile fuel in the ceramic and metal fueled reactors.

	FERTILE-FISSILE ATOM RATIO	INITIAL BREEDING RATIO	^{239}Pu LOSS RATE (gm/MWD)	REACTIVITY LIFE (Days, $\delta K = -0.04$ $\bar{P} = 2/3 \text{ Mw/l}$)	δK ($\delta T = 40 \text{ Days}$ $\bar{P} = 2/3 \text{ Mw/l}$)
PuO_2 - SS - 600	0.11	0.032	1.202	41	-0.039
PuO_2 - SS - 800	0.11	0.034	1.219	36	-0.045
$(\text{Pu-U})\text{O}_2$ - 600	4.05	0.512	0.553	104	-0.015
$(\text{Pu-U})\text{O}_2$ - 800	4.42	0.565	0.499	110	-0.015
$(\text{Pu-U})\text{O}_2$ - BeO - 600	3.45	0.464	0.643	85	-0.019
$(\text{Pu-U})\text{O}_2$ - BeO - 800	3.78	0.512	0.593	86	-0.019
$(\text{Pu-U})\text{C}$ - 600	5.24	0.632	0.407	159	-0.010
$(\text{Pu-U})\text{C}$ - 800	5.67	0.692	0.348	174	-0.009
$(\text{Pu-U})\text{C}$ - BeO - 600	4.48	0.575	0.496	121	-0.013
$(\text{Pu-U})\text{C}$ - BeO - 800	4.87	0.629	0.440	127	-0.013
Pu-U - Zr - 600	4.28	0.483	0.552	110	-0.015
Pu-U - Zr - 800	4.70	0.540	0.496	113	-0.014
Pu-U - Zr - BeO - 600	3.45	0.439	0.645	87	-0.018
Pu-U - Zr - BeO - 800	3.80	0.488	0.594	88	-0.018

TABLE 4.1

Burnup Characteristics in Reference Cores

FIGURE 4.1

K_{eff} versus Time in 600-Liter Reactors

V. KINETICS PARAMETERS

An adequate safety evaluation requires detailed knowledge of the reactor kinetics parameters. In this section, the principal kinetics parameters of each reactor will be discussed. From these parameters, one can obtain at least a rough indication of the safety characteristics of the reactors studied.

The effective delayed neutron fraction, Na worth, and reactivity coefficients in each of the 14 reactors are listed in Table 5.1. A few of these items warrant elaboration. First, the Na worth was computed using a 1-D (radial) transport model with 8 energy groups. No correction was made for the variation in the axial reflector savings since the composition of this region is not yet firm. The Na reactivity expansion coefficient was obtained by simply multiplying the Na volume expansion coefficient ($2.67 \times 10^{-4}/^{\circ}\text{K}$) by the total Na worth. If account is taken of changes in reflector savings, the magnitude of the negative sodium coefficient would decrease.

The Doppler coefficient in each reactor was computed using the FCC⁽³⁾ code with 26 energy groups. Heterogeneity effects — if present — were not considered. (A comparison of the Na void and Doppler calculations with those of other laboratories is described in Appendix C.)

A few observations should be made on the numerical values listed in Table 5.1. Since the cermet fueled reactors do not contain fertile fuel, they exhibit a low delay neutron fraction ($\sim .0022$) and a large negative Na void coefficient. The 600-l ceramic fueled reactors have a rather small Doppler coefficient, but this can be improved by increasing the core size and/or adding BeO to the core. The Doppler coefficient in the metal fueled reactors appears to be quite small — especially in a Na voided core. One might, however, be able to rely on fuel expansion for reactivity shutdown in a metal fueled reactor.

β_{eff}	Na WORTH (Core)	Na WORTH ($r \leq 5$ cm)	Na EXPANSION COEF. ($^{\circ}\text{K}^{-1}$)	FUEL EXPANSION COEF. ($^{\circ}\text{K}^{-1}$)	DOPPLER COEF. ($^{\circ}\text{K}^{-1}$) ($T = 1000$ $^{\circ}\text{K}$)	DOPPLER COEF. ($^{\circ}\text{K}^{-1}$) (No Na)
PuO_2 - SS - 600	0.07185	0.00108	-19.2×10^{-6}	-8.5×10^{-6}	$+0.41 \times 10^{-6}$	0.27×10^{-6}
PuO_2 - SS - 800	0.07701	0.00094	-20.6	~ -8.5	+0.46	0.30
(Pu-U)O ₂ - 600	0.01903	0.00008	-5.08	-----	-3.70	-1.74
(Pu-U)O ₂ - 800	0.01655	0.00004	-4.42	-----	-4.42	-2.19
(Pu-U)O ₂ - BeO - 600	0.02311	0.00019	-6.17	-----	-5.24	-2.97
(Pu-U)O ₂ - BeO - 800	0.02107	0.00013	-5.63	-----	-6.20	-3.70
(Pu-U)C - 600	0.01207	-0.00006	-3.22	-----	-4.00	-2.10
(Pu-U)C - 800	0.00922	-0.00008	-2.46	-----	-4.68	-2.56
(Pu-U)C - BeO - 600	0.01609	0.00005	-4.30	-----	-5.70	-3.46
(Pu-U)C - BeO - 800	0.01363	0.00001	-3.64	-----	-6.60	-4.17
Pu - U - Zr - 600	0.01979	0.00004	-5.28	-----	-1.66	-0.32
Pu - U - Zr - 800	0.01674	~ 0.0	-4.47	-----	-2.06	-0.43
Pu - U - Zr - BeO - 600	0.02434	0.00018	-6.50	-----	-3.42	-1.22
Pu - U - Zr - BeO - 800	0.02192	0.00012	-5.85	-----	-4.16	-1.60

TABLE 5.1

Kinetics Parameters in Reference Reactors

VI. CONCLUSIONS

The neutronics characteristics of the 14 reactors described in this report can be summarized as follows. The PuO_2 -SS fueled reactors have a large negative fuel expansion coefficient, a negative Na void coefficient, and a high flux to power ratio. Since the PuO_2 -SS reactors contain no fertile fuel, they exhibit a small delayed neutron fraction, short reactivity life, and no appreciable Doppler coefficient.

The unmoderated ceramic and metal fueled reactors show a moderate reactivity life and a marginal Doppler coefficient. The Doppler coefficient, however, can be enhanced by adding BeO to the core. Ceramic and metal fuels have the disadvantage of a low flux to power ratio and (perhaps) a central positive Na void coefficient. Owing to their higher density, carbide fuels have a more attractive reactivity life, Doppler coefficient, and flux spectrum (hardness) than oxide fuels, and a less attractive Na void coefficient.

<u>PARAMETER</u>	<u>CERMET</u>	<u>CERAMIC</u>	<u>METAL</u>
Core			
v/o Fissile Fuel (PuO ₂ , PuC, or Pu-Zr)	X	X	X
v/o Fertile Fuel Plus Moderator (U ²⁸ O ₂ , U ²⁸ C, U-Zr, Plus BeO) †	0	30-X	25-X
v/o SS	55-X	25	25
v/o Na	40	40	50
v/o Void *	5	5	0
Radial Reflector			
v/o Inconel	80	80	80
v/o Na	20	20	20

† BeO, If Included, Occupies 5 v/o

* Balance Term to Account for Materials used at less than Theoretical Density.

APPENDIX A

Reactor Composition and Nuclide Densities

TABLE A-1

Reactor Composition for Size Survey

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<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
PuC	13.6	
Pu^{239}	----	0.0293523
Pu^{240}	----	0.0032614
C	----	0.0326136
UC	13.6	
U^{238}	----	0.0327571
C	----	0.0327571
Pu-U-Zr	14.8	
Pu-U (85 w/o)	----	0.0318000
Zr (15 w/o)	----	0.0147000

TABLE A-2

Mass and Atom Density of Various Materials

	Pu-239	Pu-240	U-238	O	C	Be	Zr	Na	Cr	Fe	Ni
PuO_2 - SS - 600	0.0012447	0.0001383	0.0	0.0027660	0.0	0.0	0.0	0.0091376	0.0081831	0.0313285	0.0032219
PuO_2 - SS - 800	0.0011134	0.0001237	0.0	0.0024743	0.0	0.0	0.0	0.0091376	0.0082777	0.0316906	0.0032591
$(\text{Pu-U})\text{O}_2$ - 600	0.0014653	0.0001628	0.0057689	0.0147941	0.0	0.0	0.0	0.0091376	0.0041273	0.0158011	0.0016250
$(\text{Pu-U})\text{O}_2$ - 800	0.0013645	0.0001516	0.0058764	0.0147852	0.0	0.0	0.0	0.0091376	0.0041273	0.0158011	0.0016250
$(\text{Pu-U})\text{O}_2$ - BeO - 600	0.0013868	0.0001541	0.0046307	0.0157865	0.0	0.0034434	0.0	0.0091376	0.0041273	0.0158011	0.0016250
$(\text{Pu-U})\text{O}_2$ - BeO - 800	0.0012892	0.0001432	0.0047348	0.0157778	0.0	0.0034434	0.0	0.0091376	0.0041273	0.0158011	0.0016250
$(\text{Pu-U})\text{C}$ - 600	0.0015733	0.0001748	0.0080713	0.0	0.0098194	0.0	0.0	0.0091376	0.0041273	0.0158011	0.0016250
$(\text{Pu-U})\text{C}$ - 800	0.0014726	0.0001636	0.0081837	0.0	0.0098199	0.0	0.0	0.0091376	0.0041273	0.0158011	0.0016250
$(\text{Pu-U})\text{C}$ - BeO - 600	0.0014926	0.0001658	0.0065236	0.034434	0.0081820	0.0034434	0.0	0.0091376	0.0041273	0.0158011	0.0016250
$(\text{Pu-U})\text{C}$ - BeO - 800	0.0013948	0.0001550	0.0066327	0.0034434	0.0081825	0.0034434	0.0	0.0091376	0.0041273	0.0158011	0.0016250
$\text{Pu-U} - \text{Zr} - 600$	0.0015054	0.0001673	0.0062773	0.0	0.0	0.0	0.0036750	0.0114220	0.0041273	0.0158011	0.0016250
$\text{Pu-U} - \text{Zr} - 800$	0.0013944	0.0001549	0.0064007	0.0	0.0	0.0	0.0036750	0.0114220	0.0041273	0.0158011	0.0016250
$\text{Pu-U} - \text{Zr} - \text{BeO} - 600$	0.0014281	0.0001587	0.0047732	0.0034434	0.0	0.0034434	0.0029400	0.0114220	0.0041273	0.0158011	0.0016250
$\text{Pu-U} - \text{Zr} - \text{BeO} - 800$	0.0013248	0.0001472	0.0048880	0.034434	0.0	0.0034434	0.0029400	0.0114220	0.0041273	0.0158011	0.0016250

TABLE A-3

Atom Densities in Reference Reactors

APPENDIX B

MISCELLANEOUS FUNDAMENTAL MODE PARAMETERS

Selected spectrum averaged cross sections in the 14 reference reactors are listed in Table B-1. These values are computed using the Russian 26-group cross sections and the appropriate fundamental mode core flux spectrum.⁽⁴⁾

The fundamental mode buckling, generation time, total flux, and specific power for each reactor are given in Table B-2. The generation time, which is an excellent measure of the low energy flux, was computed using the algorithm

$$\Lambda = \frac{\sum_i \Phi_i \Phi_i^+ / v_i}{\sum_i (\nu \Sigma_f)_i \Phi_i \sum_j \psi_j \Phi_j^+} . \quad (B-1)$$

The specific power was computed using plutonium and uranium fission energies of 207 and 200 MeV, respectively.

	σ_f^{49}	σ_c^{49}	σ_f^{40}	σ_c^{40}	σ_f^{28}	σ_c^{28}	σ_c^{Na}	σ_c^{Cr}	σ_c^{Fe}	σ_c^{Ni}
PuO ₂ - SS - 600	1.9007	0.3881	0.4403	0.6682	0.0513	0.3511	0.0014	0.0068	0.0093	0.0230
PuO ₂ - SS - 800	1.9395	0.4292	0.4184	0.7249	0.0474	0.3864	0.0015	0.0071	0.0096	0.0225
(Pu-U)O ₂ - 600	1.8495	0.3522	0.4351	0.6231	0.0598	0.2687	0.0015	0.0072	0.0108	0.0251
(Pu-U)O ₂ - 800	1.8639	0.3706	0.4203	0.6495	0.0571	0.2767	0.0015	0.0073	0.0111	0.0247
(Pu-U)O ₂ - BeO - 600	1.9616	0.4578	0.4223	0.7691	0.0599	0.3108	0.0018	0.0082	0.0120	0.0257
(Pu-U)O ₂ - BeO - 800	1.9870	0.4850	0.4085	0.8068	0.0574	0.3205	0.0019	0.0084	0.0123	0.0254
(Pu-U)C - 600	1.8245	0.3231	0.4562	0.5813	0.0584	0.2520	0.0014	0.0069	0.0103	0.0248
(Pu-U)C - 800	1.8349	0.3375	0.4424	0.6021	0.0558	0.2584	0.0014	0.0071	0.0106	0.0244
(Pu-U)C - BeO - 600	1.9179	0.4143	0.4391	0.7086	0.0588	0.2889	0.0017	0.0078	0.0115	0.0254
(Pu-U)C - BeO - 800	1.9369	0.4357	0.4259	0.7386	0.0564	0.2967	0.0018	0.0080	0.0118	0.0251
Pu-U - Zr - 600	1.7605	0.2522	0.4695	0.4793	0.0567	0.2206	0.0011	0.0061	0.0089	0.0237
Pu-U - Zr - 800	1.7662	0.2638	0.4522	0.4967	0.0535	0.2269	0.0011	0.0062	0.0091	0.0232
Pu-U - Zr - BeO - 600	1.8608	0.3583	0.4438	0.6311	0.0575	0.2702	0.0014	0.0072	0.0106	0.0246
Pu-U - Zr - BeO - 800	1.8773	0.4282	0.4282	0.6601	0.0546	0.2788	0.0015	0.0074	0.0109	0.0242

TABLE B-1

Spectrum Averaged Cross Sections

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	B^2 (cm ⁻²) ($K_{\text{eff}} = 1.05$)	Λ (sec)	ϕ (n-cm ⁻² - sec ⁻¹) ($P = 1$ Mw/liter)	w/gm Pu-239 ($\phi = 10^{16}$)	w/gm Pu-240 ($\phi = 10^{16}$)	w/gm U-238 ($\phi = 10^{16}$)
PuO ₂ - SS - 600	1.72×10^{-3}	5.7×10^{-7}	1.240×10^{16}	1.588×10^3	0.366×10^3	0.042×10^3
PuO ₂ - SS - 800	1.52	6.9	1.360	1.621	0.348	0.038
(Pu-U)O ₂ - 600	1.91	3.2	0.971	1.545	0.362	0.048
(Pu-U)O ₂ - 800	1.67	3.5	1.031	1.557	0.350	0.046
(Pu-U)O ₂ - BeO - 600	1.93	4.3	0.987	1.639	0.351	0.049
(Pu-U)O ₂ - BeO - 800	1.69	4.8	1.046	1.660	0.340	0.047
(Pu-U)C - 600	1.94	2.6	0.890	1.525	0.380	0.047
(Pu-U)C - 800	1.70	2.8	0.942	1.533	0.368	0.045
(Pu-U)C - BeO - 600	1.95	3.5	0.914	1.603	0.365	0.048
(Pu-U)C - BeO - 800	1.71	3.8	0.965	1.618	0.354	0.046
Pu-U - Zr - 600	1.87	2.4	0.987	1.471	0.391	0.046
Pu-U - Zr - 800	1.64	2.6	1.058	1.476	0.376	0.043
Pu-U - Zr - BeO - 600	1.89	3.5	1.009	1.555	0.369	0.047
Pu-U - Zr - BeO - 800	1.65	3.8	1.076	1.569	0.356	0.044

TABLE B-2

Miscellaneous Fundamental Mode Parameters

APPENDIX C
CROSS SECTION COMPARISON

The Russian 26-group data compilation, with slight modifications, was used in all calculations discussed in this report. Modifications were made in the capture cross section of Fe,⁽⁵⁾ and the capture and fission cross sections of Pu²³⁹ and U²³⁵.^(6,7)

Table C-1 demonstrates how this modified set compares with other sets in a 5:1 oxide fueled reactor. The data in this table was taken from a paper presented by Okrent at the 1965 Argonne Conference on Safety, Fuels, and Core Design in Large Fast Power Reactors.⁽⁸⁾

Although the Russian set appears somewhat unreactive in this comparison, preliminary 2-D transport calculations on the SEFOR critical assembly indicate that, if anything, the modified Russian set is slightly overreactive.

	$B^2 \times 10^4$	% 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)	-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)	-0.0052	----	----	0.20
UK	~14.9	---	-0.0077	0.028	0.15	0.21
UNC	15.8	---	-----	0.032	0.14	0.16
BNWL (Russian)	15.2	-3.68 (900 ⁰)	-0.0054	0.032	0.14	0.21

TABLE C-1

5:1 Oxide Fueled Reactor

Cross Section Comparison⁽⁸⁾

No Fission Products or Pu-240

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