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NEUTRONICS CHARACTERISTICS
OF
SELECTED COMPACT FAST REACTORS

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NEUTRONICS CHARACTERISTICS
of
SELECTED COMPACT FAST REACTORS

by

W. W. Little, Jr. and R. W. Hardie

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ABSTRACT

This report describes the neutronics characteristics of $U^{235}O_2$ -W, $U^{233}O_2$ -W, and $Pu^{239}N$ -W fueled fast compact reactors. The principal items studied are reactor criticality and fuel burnup.

For the fuel compositions and void fractions considered, U^{233} and Pu^{239} fueled reactors require considerably less fuel than corresponding U^{235} fueled reactors. All reactors considered are rather insensitive to changes in the refractory metal inventory.

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NEUTRONICS CHARACTERISTICS
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SELECTED COMPACT FAST REACTORS

1.0 INTRODUCTION

Compact fast reactors – by virtue of their small volume, high fuel density, and compatibility with refractory materials – offer much promise as lightweight power sources for space applications.⁽¹⁻³⁾ Further, since Pu^{239} and U^{233} fueled fast reactors require less fuel than comparable U^{235} fueled reactors, these high worth fuels may be more suitable than U^{235} for use in compact space reactors.

This report portrays the principal neutronics characteristics of $\text{U}^{235}\text{O}_2\text{-W}$, $\text{U}^{233}\text{O}_2\text{-W}$, and $\text{Pu}^{239}\text{N-W}$ fueled fast reactors. A wide range of core sizes and compositions is considered. Emphasis is placed on neutronic-static characteristics, although selected kinetics parameters are also included. The results of the neutronics calculations are given in a form suitable for incorporation into preliminary engineering design studies.

2.0 CRITICAL SIZE SURVEY

This section describes the dependence of reactor size on void fraction and cermet composition [fuel to fuel-plus-metal ratio, designated $F/(F + M)$]. For each of the three fissile fuels, calculations are performed with void fractions of 0.2, 0.35, and 0.5 and with fuel to cermet ratios ($F/(F + M)$) of 0.3, 0.5, and 0.7. A more detailed description of the reactor compositions is given in Appendix A.

Two size surveys are shown. The first survey is made with the FCC⁽⁴⁾ fundamental mode code and a modified version of the Russian 26-group cross sections.⁽⁵⁾ The objective is to describe the basic neutronics-statics characteristics of bare $U^{235}O_2$ -W, $U^{233}O_2$ -W, and PuN-W fueled cores. If reflector savings are known, the fundamental mode calculations can be used to represent any simple geometry. The second criticality survey is performed with the DTF-1V⁽⁶⁾ one-dimensional transport code (S_4 option) with eight energy groups. All 1-D calculations are performed in spherical geometry with a 10 cm BeO reflector. For each fuel, 8-group cross sections are generated from the 26-group Russian data.

Figures 2.1 through 2.3 give the fundamental mode spherical radius ($R = \pi/\sqrt{B^2}$) as a function of void fraction and for fuel to cermet ratio for $U^{235}O_2$ -W, $U^{233}O_2$ -W, and PuN-W fueled cores, respectively. Figure 2.4 — a cross-plot from Figures 2.1 through 2.3 — shows the spherical radius of each fuel for a fuel to cermet ratio of 0.5.

The one-dimensional calculations of core volume as a function of void fraction are shown in Figures 2.5 through 2.7. A cross-plot of these graphs, for a cermet fraction of 0.5, is shown in Figure 2.8.

FIGURE 2.1. Spherical Radius Versus Void Fraction for U²³⁵O₂-W Fueled Cores

FIGURE 2.2. Spherical Radius Versus Void Fraction for U²³³O₂-W Fueled Cores

FIGURE 2.3. Spherical Radius Versus Void Fraction for PuN-W Fueled Cores

FIGURE 2.4. Spherical Radius Versus Void Fraction for a Fuel-Cermet Ratio of 0.5

FIGURE 2.5. Core Volume Versus Void Fraction for U²³⁵O₂-W Fueled Reactors

FIGURE 2.6. Core Volume Versus Void Fraction for U²³³O₂-W Fueled Reactors

FIGURE 2.7. Core Volume Versus Void Fraction for PuN-W Fueled Reactors

FIGURE 2.8. Core Volume Versus Void Fraction for a Fuel-Cermet Ratio of 0.5

Another comparison of the relative merits of U^{235} , U^{233} , and Pu^{239} can be obtained by comparing the critical masses for comparable reactor configurations. This information is shown in Figures 2.9 through 2.11. In these graphs, Pu^{239} and U^{233} fueled cores can be seen to require significantly less fuel than U^{235} fueled cores. This behavior, of course, can be traced to the high η , $\nu\sigma_f/\sigma_a$, of Pu^{239} and U^{233} .

FIGURE 2.9. Critical Mass Versus Void Fraction for a Fuel-Cermet Ratio of 0.3

FIGURE 2.10. Critical Mass Versus Void Fraction for a Fuel-Cermet Ratio of 0.5

FIGURE 2.11. Critical Mass Versus Void Fraction for a Fuel-Cermet Ratio of 0.7

3.0 REACTIVITY COEFFICIENT CALCULATIONS

In the compact fast reactors under study, materials other than fuels usually have small reactivity worths. These small reactivity coefficients are due, in part, to the extremely hard flux and, thus, low spectrum averaged cross sections. In this section, the reactivity worth of several elements in U^{235} , U^{233} , and Pu^{239} fueled cores is described. Both zero and one-dimensional models are employed.

In a bare reactor with energy-independent extrapolation lengths, material worths can be estimated using the perturbation relationship

$$\delta K \approx \sum_{i=1}^N \left\{ \phi_i \delta(\nu \Sigma^f)_i - \phi_i \phi_i^+ \delta(\Sigma^a)_i - \phi_i \phi_i^+ \delta \left(\frac{B^2}{3 \Sigma^{tr}} \right)_i - \phi_i \sum_{j=i+1}^N (\phi_i^+ - \phi_j^+) \delta \Sigma_{i \rightarrow j} \right\} \quad (3.1)$$

where: δK = reactivity change

ϕ_i = flux in group i

ϕ_i^+ = adjoint flux in group i

B^2 = buckling (cm^{-2})

$\delta \Sigma_i^x$ = variation of macroscopic cross section in group i (cm^{-1})

N - number of energy groups.

The flux and adjoint flux in Equation (3.1) are normalized by the expressions:

$$\sum_{i=1}^N (\nu \Sigma^f)_i \phi_i = \quad (3.2)$$

$$\text{and} \quad \sum_{i=1}^N x_i \phi_i^+ = 1 \quad (3.3)$$

Observe that the addition of inert material tends to increase reactivity by decreasing leakage and to decrease reactivity via parasitic capture. Depending on the adjoint flux spectrum, the reactivity change due to neutron moderation can be either positive or negative.

Table 3.1 gives the worth of various materials in "typical" U^{235} , U^{233} , and Pu^{239} fueled cores. These typical cores—or reference cores—are described in Table 3.2. All calculations were performed using a fundamental mode model with 26-energy groups.^(4,5) The associated flux, adjoint flux, and spectrum averaged cross sections are shown in Figures 3.1 and 3.2 and Table 3.3.

TABLE 3.1. Material Worths in Bare Reference Cores^(a)

Material	Worth, δK for $\delta N = 0.001$		
	$U^{235}O_2$ -W	$U^{233}O_2$ -W	PuN-W
U^{235}	0.0783	---	---
U^{233}	---	0.0828	---
Pu^{239}	---	---	0.0784
Beryllium	0.0078	0.0102	0.0094
Tantalum	0.0012	0.0118	0.0124
Tungsten	0.0069	0.0141	0.0143
Molybdenum	0.0088	0.0140	0.0139
Niobium	0.0083	0.0142	0.0144
Sodium	0.0067	0.0091	0.0092
Lithium	0.0014	0.0042	0.0044

^(a) Fundamental Mode Calculation
 Void Fraction = 0.35
 50 vol% Fuel in Cermet

TABLE 3.2. Characteristics of Reference Cores^(a)

Item	$U^{235}O_2$ -W	$U^{233}O_2$ -W	PuN
B^2 [K = 1.05]	6.80×10^{-3}	13.5×10^{-3}	19.1×10^{-3}
Spherical	24.27	14.72	12.18
Radius, cm			
Volume, liters	59.9	13.4	7.6
Mass Fissile	174.6	41.9	31.4
Fuel, kg			
Mass Tungsten, 376.		84.	48.
kg			

^(a) Void Fraction = .35
 50 vol% Fuel in Cermet

FIGURE 3.1. Flux Spectra in U^{235} , U^{233} , and Pu^{239} Fueled Reactors

FIGURE 3.2. Adjoint Flux Spectra in U^{235} , U^{233} , and Pu^{239} Fueled Reactors

TABLE 3.3. Spectrum Averaged Cross Sections in Reference Reactors^(a)

Element	$U^{235}O_2-W$	$U^{233}O_2-W$	PuN
$\sigma_a^{FF(b)}$	1.6498	2.2820	1.8776
σ_f^{FF}	1.4060	2.1418	1.7541
$\nu\sigma_f^{FF}$	3.5277	5.5101	5.3081
σ_a^{Nb}	0.0769	0.0600	0.0609
σ_a^{Mo}	0.0526	0.0442	0.0443
σ_a^W	0.1183	0.1021	0.1025
σ_a^{Ta}	0.2557	0.2127	0.2151

(a) Fundamental Mode Calculation

Void Fraction = 0.35

50 vol% Fuel in Cermet

(b) Fissile fuel, either U^{235} , U^{233} , or Pu^{239}

Reactivity worths in a reflected $U^{233}O_2-W$ fueled reactor are given in Table 3.4. These 1-D (spherical) calculations were performed using the DTF-1V⁽⁶⁾ transport code (S_4 option) with eight energy groups. The 8-group cross sections were obtained from FCC by collapsing 26-group data.

Of the structural elements considered, observe that tantalum is the most poisonous, followed by tungsten, niobium, and molybdenum. None of these elements, however, significantly penalizes the critical mass requirements. This is true for both bare and reflected cores.

TABLE 3.4. *Material Worth in Reflected $U^{233}O_2$ -W Reactor^(a)*

<u>Material</u>	<u>δM, kg/liter ($\delta N = 0.001$)</u>	<u>δK</u>
U^{233}	0.3869	0.0677
Beryllium	0.0150	0.0033
Tantalum	0.3004	-0.0031
Tungsten	0.3052	0.0010
Molybdenum	0.1593	0.0024
Niobium	0.1543	0.0017
Sodium	0.0380	0.0022
Lithium	0.0115	0.0003

^(a) *Spherical Reactor*

Void Fraction = 0.35

50 vol% Fuel in Cermet

10 cm BeO Reflector

4.0 BURNUP RELATIONSHIPS

In hard spectrum fast reactors containing no fertile fuel, such as those considered in this report, it is easy to estimate the reactivity variation during burnup. If the flux and adjoint flux are assumed to be independent of both space and energy, it follows that

$$\delta K \approx \left(\frac{\eta - 1}{\eta} \right) \frac{\delta M}{M}, \quad (4.1)$$

where:

η = eta for fissile isotope ($\nu\sigma_f/\sigma_a$)

M = fissile inventory

δM = change in fissile inventory

δK = reactivity variation.

The appropriate nuclear parameters for each fuel are listed in Table 4.1. Thus, for a given fuel, δK is proportional to the fuel burnup, or mass change, and inversely proportional to the fuel inventory.

TABLE 4.1. Fuel Burnup Parameters^(a)

<u>Fuel</u>	<u>η</u>	<u>$\frac{\eta - 1}{\eta}$</u>	<u>g/MWd</u>
U ²³⁵	2.14	0.53	1.23
U ²³³	2.41	0.59	1.11
Pu ²³⁹	2.83	0.65	1.08

^(a) From fundamental mode calculations using 26 groups

Void fraction = 0.35

50 vol% Fuel in cermet

Parenthetically, it should be mentioned that, for a specified energy release, a fast compact reactor has a much smaller reactivity variation than a thermal compact reactor. This difference is due, in part, to the large critical mass requirements in fast reactors.

Figure 4.1 shows the reactivity as a function of time in U^{235} , U^{233} , and Pu^{239} fueled reactors. These burnup calculations were performed using a fundamental mode model with 26 energy groups. Fission products were included, although they have little effect on reactivity. It can be seen that these numerical results agree reasonably well with predictions given by Equation (4.1).

FIGURE 4.1. *Effective Multiplication Constant as a Function of Time*

5.0 KINETICS CHARACTERISTICS

A detailed kinetics study of compact fast reactors is beyond the scope of this report. However, since the three fissile fuels under study do exhibit markedly different kinetic characteristics, it is of interest to at least identify the principal kinetics parameters.

The most significant kinetics parameter is the delayed neutron fraction, β . Its influence on reactor kinetics can be understood from the following arguments.

First, let us consider the lumped parameter kinetics equations

$$\frac{dP}{dt} = \frac{\rho - \beta}{\Lambda} P + \sum_i \lambda_i C_i \quad (5.1)$$

$$\frac{dC_i}{dt} = \frac{\beta_i P}{\Lambda} - \lambda_i C_i, \quad (5.2)$$

where $P(t)$ = power density, W/cm^3
 $\rho(t)$ = reactivity
 $C_i(t)$ = precursor density for group i , W/cm^3
 λ_i = decay constant for group i , sec^{-1}
 β_i = delay fraction for group i
 β = total delay fraction, $\sum_i \beta_i$
 Λ = generation time, sec.

If the analysis is restricted to reactivities less than β , the short-term power variation can be computed using the simple expression

$$P(t) \approx \frac{P_0}{1 - \frac{\rho(t)}{\beta}} \quad (5.3)$$

where P_0 is the initial power (at equilibrium).

For the sake of argument, let us now suppose that $\rho \ll \beta$.
From Equation (5.3),

$$P(t) \approx P_0 \left[1 + \frac{\rho(t)}{\beta} \right] \quad (5.4)$$

Thus, for a specified reactivity variation, $\rho(t)$, the change in reactor power is inversely proportional to the delayed neutron fraction, β . Since the delayed neutron fraction (see Table 5.1) for U^{233} and Pu^{239} is roughly 1/3 that of U^{235} , reactors fueled with U^{233} and Pu^{239} are more sensitive to small reactivity variations. For the same reason, U^{233} and Pu^{239} fueled reactors are—under most circumstances—more susceptible to prompt critical excursions.

The above arguments per se do not preclude, or in any sense restrict, the use of U^{233} or Pu^{239} fuels in compact fast reactors. The safety of any reactor can only be evaluated by a detailed study of the reactor, its objective, and its environment.

TABLE 5.1. *Approximate Delayed Neutron Fractions*

<u>Fuel</u>	<u>Delayed Neutrons Per Fission⁽⁷⁾</u>	<u>Delayed Neutrons Per Fission Neutron(β)</u>
U^{235}	0.0165	~ 0.0064
U^{233}	0.0070	~ 0.0026
Pu^{239}	0.0063	~ 0.0021

6.0 OBSERVATIONS ON RESULTS

Perhaps the most significant, and most obvious, characteristic discussed in this report is the low critical mass requirements in U^{233} and Pu^{239} fueled reactors. If these high worth fuels could be used in place of U^{235} , one could:

- Design a smaller reactor for a given void fraction and cermet loading
- Reduce the cermet (fuel to metal ratio) loading for a given core size and void fraction
- Increase the void fraction for a specified core size and cermet loading.

Thus, since power plant size and weight are of critical importance in space applications, it appears that Pu^{239} and U^{233} fuels are potentially more suitable for space employment than U^{235} fuels.

APPENDIX A

Material Densities

<u>Material</u>	<u>ρ, g/cm³</u>	<u>N, atoms/b-cm</u>
Molybdenum	10.2	0.06403
Niobium	8.4	0.05445
Tungsten	19.3	0.06322
Tantalum	16.6	0.05525
Sodium	0.872	0.02284
Lithium	0.534	0.04634
BeO	2.86	—
Be	—	0.06887
O	—	0.06887
PuN	14.2	—
Pu ²³⁹ , 95 wt%	—	0.03210
Pu ²⁴⁰ , 5 wt%	—	0.00169
N	—	0.03379
U ²³³ O ₂	10.97	—
U ²³³	—	0.02493
O	—	0.04986
U ²³⁵ O ₂	10.97	—
U ²³⁵ , 93 wt%	—	0.02299
U ²³⁸ , 7 wt%	—	0.00173
O	—	0.04945

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