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NUCLEAR PARAMETERS
AND PARAMETRIC STUDIES
FOR THE FAST TEST REACTOR (FTR)

June 1969

AEC RESEARCH &
DEVELOPMENT REPORT

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NUCLEAR PARAMETERS AND PARAMETRIC STUDIES
FOR THE FAST TEST REACTOR (FTR)

By

W. W. Little, Jr.
L. L. Maas

Reactor and Plant Technology Department
FFTF Division

June 1969

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NUCLEAR PARAMETERS AND PARAMETRIC STUDIES
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ABSTRACT

This report describes the principal neutronics characteristics of the vertical FTR reference reactor. Since the design has not been frozen, selected parametric studies are also included to show the effect of design changes.

NUCLEAR PARAMETERS AND PARAMETRIC STUDIES FOR THE FAST TEST REACTOR (FTR)

W. W. Little, Jr. and L. L. Maas

I. INTRODUCTION

This report gives a brief description of the neutronics characteristics of the FTR conceptual design. Parametric studies are also included to delineate the effects of possible design changes.

It should be understood that the values given in the report merely reflect the current FTR reference design. This is particularly true for those items that are sensitive to design changes, such as Na and Doppler coefficients.

All neutronics calculations were performed using an extensively modified version of the Russian 26-group data compilation. The results of an analysis of selected critical assemblies using this data set are given in Appendix A.

All 1-D and 2-D computations were performed using the 1DX⁽¹⁾ and 2DB⁽²⁾ diffusion codes, respectively. Since the FTR consists of hexagonal tubes, essentially all reference reactor calculations were performed in hexagonal geometry using the 2DB code.

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II. DESCRIPTION OF REFERENCE REACTOR

Geometric Model

A schematic cross section of the 2-zoned FTR core and reflector is given in Figure 2.1. The initial core consists of 91 in-core lattice positions. A breakdown of these in-core positions into their specific function is given in Table 2.I. In addition to the 3 shim rods in the core, there are 15 shim rods in the first row of the reflector.

TABLE 2.I

Initial In-Core Positions

Driver Assemblies	76
Closed Loops	6
Open Loops	3
Safety Rods	3
Shim Rods	3

Most design calculations are based on a two-dimensional, hexagonal simulation of the cross section shown in Figure 2.1. The component dimensions on which the neutronics analyses are based are given in Table 2.II. The resulting volume fractions and loadings are listed in Tables 2.III and 2.IV, respectively. Material densities are tabulated in Appendix B.

- Open Loops - 3
- Closed Loops - 6
- Control/Safety Rods - 6
- Peripheral Control Rods - 15
- Reflectors - 111
- Drivers - 76

FIGURE 2.1
Conceptual Core Arrangement

TABLE 2.II*Component Dimensions*

Parameter	English Units	CGS Units
No. Pins/Assembly	217	217
Pin Length	3 feet	91.4 cm
Pin O.D.	230 mils	0.584 cm
Clad Thickness	15 mils	0.0381 cm
Wrapper Wire Diam.	56 mils	0.142 cm
Shroud Thickness	140 mils	0.356 cm
Shroud Clearance	15 mils	0.0381 cm
Shroud Gap	100 mils	0.254 cm
Cell Flat-to-Flat	4.72 inches	11.98 cm

TABLE 2.III*Reactor Composition*

Region	Volume (liters)	Material	V/O
Inner Driver (28 Assemblies)	317.8	PuO ₂	5.7
		UO ₂	25.4
		Void	3.9
		316 SS	26.0
		Na	39.0
Outer Driver (48 Assemblies)	544.9	PuO ₂	7.6
		UO ₂	23.5
		Void	3.9
		316 SS	26.0
		Na	39.0
Radial Refl. Assembly	11.35	Void	0.0
		316 SS	34.0
		Ni	32.0
		Na	34.0
Control Assembly	11.35	Void	4.2
		B ₄ C	30.6
		316 SS	31.6
		Na	33.6
Control Follower	11.35	Void	0.0
		316 SS	37.0
		Ni	28.0
		Na	35.0

TABLE 2.III (Contd.)

Region	Volume (liters)	Material	V/O
Open Loop Assembly	11.35	PuO ₂	4.3
		UO ₂	19.1
		Void	3.6
		316 SS	40.0
		Na	33.0
Closed Loop Assembly	11.35	PuO ₂	1.1
		UO ₂	4.2
		Void	0.7
		316 SS	46.0
		Na	48.0
Axial Reflector	(T=45 cm)	Void	10.0
		316 SS	35.0
		Ni	11.0
		Na	44.0

TABLE 2.IV
Assembly Loadings (kg)

Isotope	Inner-Zone Driver	Outer-Zone Driver	Open Loop Test	Closed Loop Test	Control Rod
Pu-239	5.81	7.72	4.35	1.15	0.0
Pu-240	0.79	1.06	0.60	0.16	0.0
U-235	0.20	0.18	0.15	0.03	0.0
U-238	27.69	25.62	20.77	4.54	0.0
O	4.63	4.64	3.47	0.79	0.0
Na	3.86	3.86	3.31	4.74	3.33
Fe	15.68	15.68	24.48	28.21	19.33
Cr	3.92	3.92	6.12	7.05	4.83
Ni	2.67	2.67	4.32	4.98	3.41
Mo	0.69	0.69	1.08	1.24	0.85
B-10	0.0	0.0	0.0	0.0	2.21

Principal Neutronics Parameters

The significant neutronics parameters for the FTR conceptual design are listed in Table 2.V. Needless to say, many of the items given in this table are subject to appreciable uncertainty. An extensive critical experiment program is underway to reduce these uncertainties.

In addition to these calculational uncertainties, a significant error can probably be ascribed to inadequate definition of the core configuration and composition. For example, it is difficult to predict at this early stage the precise composition of each test loop. Thus, the parameters listed herein give a consistent, but not necessarily final, core description.

TABLE 2.V*Parameters for Initial 2-Zone Core*

<u>Parameter</u>	<u>Units</u>	<u>Value</u>
Power (Including all loops)	MWt	400
Core Height	cm	91.4
Equivalent Radius	cm	60.6
Total Core Volume	liters	1033
Total Pu ²³⁹ Mass	kg	530
Pu ²³⁹ Mass in Driver	kg	510
Delay Fraction	--	0.003
Neutron Generation Time	sec	4.0×10^{-7}
Peak Power Density	MW/l	0.75
Peak Linear Power Density	KW/cm	0.43
Radial Peak-to-Average Power	--	1.35
Axial Peak-to-Average Power	--	1.24
Neutron Flux (At core center)		
Peak	n-cm ⁻² -sec ⁻¹	7.3×10^{15}
% > 0.1 MeV	--	65
% > 1.0 MeV	--	12
Fuel Composition		
% 239 in Pu	--	88
% 240 in Pu	--	12
U ²³⁸ /Pu ²³⁹ (Zone 1)	--	4.79
U ²³⁸ /Pu ²³⁹ (Zone 2)	--	3.33
Fuel Burnup		
Breeding Ratio	--	0.53
Reactivity Cycle	days	70
Reactivity Loss/Cycle	δk	0.02

TABLE 2.V (Contd.)

<u>Parameter</u>	<u>Units</u>	<u>Value</u>
Average Burnup/Cycle	MWd/kg	11
Average Total Burnup	MWd/kg	45
Reactivity Requirements		
Safety	δk	0.04
Fuel Burnup	δk	0.02
Cold to Hot	δk	0.01
Shutdown Margin	δk	0.01
Test Compensations	δk	0.01
Miscellaneous	δk	0.01
Control Elements		
Number of Safety Rods	--	3
Number of In-Core Shim Rods	--	3
Number of Peripheral Shim Rods	--	15
Worth of Safety Rods	δk	0.064
Worth of In-Core Shim Rods	δk	0.037
Worth of Peripheral Shim Rods	δk	0.041
Worth of All Shim Rods	δk	0.071
Reactivity Coefficients		
Doppler (Reflector Shims Inserted)	$T\delta k/\delta T$	-0.004
Clad Axial Expansion	$\delta k/(\delta L/L)$	-0.30
Central Channel Pu ²³⁹ Worth	$\delta k/kg$	$+1.97 \times 10^{-3}$
Central Pu ²³⁹ Worth	$\delta k/kg$	$+2.87 \times 10^{-3}$
Core Averaged Na Worth	$\delta k/kg$	-8.5×10^{-6}
Central Channel Na Worth	$\delta k/kg$	-5.1×10^{-5}
Central Na Worth	$\delta k/kg$	-13.0×10^{-5}

TABLE 2.V (Contd.)

<u>Parameter</u>	<u>Units</u>	<u>Value</u>
Temperature and Power Coefficients (At P = 400 MWt)		
Doppler	$\delta k / \delta T$ ($\bar{T}_f \cong 1000$ °C)	-3.2×10^{-6}
Clad Axial Expansion	$\delta k / \delta T$ ($\bar{T}_{ss} \cong 425$ °C)	-5.4×10^{-6}
Na Expansion	$\delta k / \delta T$ ($\bar{T}_{Na} \cong 400$ °C)	$+0.8 \times 10^{-6}$
Doppler	$\delta k / (\delta P / P)$	-22.0×10^{-4}
Clad Axial Expansion	$\delta k / (\delta P / P)$	-6.0×10^{-4}
Na Expansion	$\delta k / (\delta P / P)$	$+0.7 \times 10^{-4}$

III. FLUX AND POWER PROFILES

Two dimensional power and flux profiles have been computed in both hexagonal and R-Z geometry. All computations were performed using the 2DB code with 13 energy groups. The axial leakage in the hexagonal calculations was simulated by a group and zone independent buckling of $6.04 \times 10^{-4} \text{ cm}^{-2}$. The core compositions in the R-Z calculations were obtained by volume weighting all test loops, control follower assemblies, and driver loops.

Figs. 3.1 and 3.2 show the average power density and flux, respectively, in selected hexagonal assemblies. The numbers in these figures are based on a core configuration in which all rods are withdrawn.

Figs. 3.3 and 3.4 give the axial and radial profiles for power, density and flux. All values are normalized to unity at the core center. In both the axial and radial direction, observe that the power profile near the core-reflector interface has a smaller gradient than the flux profile. This difference, of course, is caused by spectrum softening.

FIGURE 3.1
Power Density (MW/l) at Core Mid-Plane for a Total Power
400 MWt

FIGURE 3.2

Flux Profile ($\text{n/cm}^2\text{-sec} \times 10^{-15}$) at Core Mid-Plane for a
Total Power of 400 MWt

FIGURE 3.3
Normalized Axial Power Density and Flux Profiles at Radial Center

FIGURE 3.4
Normalized Radial Power Density and Flux Profiles at the Axial Midplane

IV. FLUX SPECTRA AND REACTION RATES

The central flux spectrum and integrated flux spectrum are shown in Figs. 4.1 and 4.2, respectively. The corresponding specific powers and spectrum averaged cross sections are given in Table 4.1. Since there is little spectrum variation in the core, these central values are essentially equal to the core averaged values.

Observe that the fraction of flux above 1 MeV is about 12%, and the fraction above 0.1 MeV is approximately 65%. These percentages are slightly higher at the interface between the 2 core zones, and slightly lower at the core periphery.

FIGURE 4.1
Central Flux Spectrum

FIGURE 4.2
Central Integrated Flux Spectrum

TABLE 4.I

*Specific Power and Spectrum Averaged
Cross Sections in Central Zone*

Isotope	Specific Power (Watts/gm) for $\phi = 5 \times 10^{15}$	σ_f (barns)	σ_a (barns)
Pu-239	770	1.86	2.26
Pu-240	150	0.36	1.05
Pu-241	990	2.42	2.76
U-235	810	1.93	2.52
U-238	20	0.047	0.337
Na	0.0	0.0	0.0016
O	0.0	0.0	0.0014
Fe	0.0	0.0	0.0103
Cr	0.0	0.0	0.0077
Ni	0.0	0.0	0.0243
B-10	0.0	0.0	2.40

V. REACTIVITY COEFFICIENTS

Reactivity coefficients profiles in R-Z geometry were computed using the PERT-IV⁽³⁾ first-order perturbation code. The thirteen-group flux and adjoint flux profiles were obtained from 2DB.

Figs. 5.1 and 5.2 show the radial and axial reactivity coefficient distributions for Pu-239, U-238, B-10, Fe, and Na. An integrated B-10 reactivity coefficient profile, which can be used to estimate rod worth as a function of axial position, is shown in Fig. 5.3. The center-point and central channel coefficients for virtually all core constituents are given in Table 5.I.

A few cautionary remarks on the interpretation of these coefficients are in order. First, the light element worths (e.g., C, O, and Na) have large error bars due to cross sections uncertainties and calculational approximations. Further analysis of the important Na coefficient is in progress.

The reader is also reminded that the integrated rod worth profile shown in Fig. 5.3 should, in reality, be slightly skewed. That is, flux depression in the rod causes the lower end of the stroke to be worth less than the upper end of the stroke. However, since the curve in Fig. 5.3 was based on perturbation theory, this effect is not seen.

FIGURE 5.1
Normalized Radial Reactivity Coefficient Profiles at Core Midplane

FIGURE 5.2
Normalized Axial Reactivity Coefficient Profiles at Radial Center

FIGURE 5.3
Integrated Axial Reactivity Coefficient Profile for B-10

TABLE 5.1*Reactivity Coefficients for Core Constituents*

Isotope	Center-Point Worth ($\delta k/kg$) $\times 10^5$	Central Channel Worth ($\delta k/kg$) $\times 10^5$
Pu-239	+ 287.3	+ 197.0
Pu-240	+ 3.5	+ 1.6
U-235	+ 221.3	+ 153.0
U-238	- 18.3	- 11.9
O	- 22.9	- 10.7
Na	- 13.0	- 5.1
C	- 38.5	- 19.7
Fe	- 6.8	- 3.4
Cr	- 6.9	- 3.3
Ni	- 10.6	- 5.6
Mo	- 26.8	- 17.2
B-10	-4291.0	-3016.0
Fission Products	- 39.3	- 26.6

VI. CONTROL REQUIREMENTS AND ROD WORTHS

A breakdown of the control requirements is shown in Table 6.I. The total requirements of 10% δk should be sufficient to meet all routine control demands. It should be noted, however, that this value does not include provisions for:

- 1) Test compensations greater than 1% δk . Such compensations, presumably, could be achieved by interchanging fuel in the inner and outer zone.
- 2) The initial reactivity loss in getting to an equilibrium cycle.
- 3) Uncertainties in calculation.

TABLE 6.I

Control Requirements

Parameter	$ \delta k $
Safety:	0.04
Shim:	
Shutdown	0.01
Cold-to-Hot	0.01
Burnup	0.02
Test Compensation	0.01
Miscellaneous	0.01
Total	0.10

The position of each control rod was shown in Figure 2.1. The reactivity worths and the associated peak-to-average power densities for these control rods are given in Table 6.II. Each rod is assumed to

TABLE 6.II*Control Worths and Peak-to-Average Power*

Configuration	- δk	Radial Peak-to-Average Power
Base Case (no rods)	0.0	1.23
3 Safety Rods	0.0646	1.30
3 In-Core Shim Rods	0.0374	1.31
5 Contiguous Reflector Rods	0.0121	1.34
All 15 Reflector Rods	0.0406	1.37
All Shim Rods	0.0706	1.48

contain enriched (35 a/o) B_4C at 34.8 v/o (= 2.21 kg B-10). For a more detailed description of the B_4C rod configuration, the reader is referred to BNWL-899. For our purposes, it is sufficient to say that each rod consists of 61 pins with an O.D. of 1.15 cm and a clad thickness of 0.10 cm. The B_4C smear density is 88%.

Perhaps the best scheme to enhance the control worth--if, indeed, a larger worth is required--is simply to add more B-10 to each rod. The control rod worth as a function of B-10 loading for in-core and peripheral rods is shown in Figure 6.1. Although the data in Figure 6.1 was generated using a single in-core rod and a smeared peripheral annulus, similar scaling factors can be used for any rod or set of rods.

Since tantalum is a possible alternate control material, a few remarks on its worth are in order. To obtain the same worth as a natural B_4C rod, much higher volume percents of Ta are required. This, in turn, leads to heavy rods. For example, an in-core Ta rod having the same worth as a B_4C rod containing 0.9 kg B-10 would contain about 80 kg's of Ta.

FIGURE 6.1
Relative Rod Worth as a Function of B-10 Mass

VII. FUEL BURNUP AND MANAGEMENT

As presently conceived, the FTR fuel management program will consist of four, 70-day cycles. At the end of each cycle, 1/4 of the driver fuel, or about 19 subassemblies, will be replaced with fresh fuel. Although the details of the reloading pattern has not been worked out, shotgun reloading with only minor shuffling appears to be satisfactory. In any event, the parameters described below are insensitive to the details of reloading.

During each cycle, the reactivity variation can be approximated by the expression

$$\delta k(t) \approx - 0.33 \times 10^{-3} \frac{Pt}{M_0} ,$$

where: t = time (days)
 P = power (MWt)
 M_0 = mass Pu-239 (kg).

Using $P = 400$ MWt and $M_0 = 540$ kg, we have

$$\delta k(t) \approx - 0.25 \times 10^{-3} t .$$

Thus, for $t = 70$ days, $\delta k = -0.017$. The corresponding mass changes for a 70-day cycle are given in Table 7.I. As a point of interest, it should be mentioned that about 20% of the above reactivity loss is caused by the buildup of fission products.

TABLE 7.1*Mass Variations During A 70-Day Cycle*

Isotope	Inner Zone (28 Drivers)		Outer Zone (45 Drivers)	
	M ₀ (kg)	ΔM(kg)	M ₀ (kg)	ΔM(kg)
Pu-239	162.5	-4.1	347.5	-9.7
Pu-240	22.3	+1.2	47.6	+2.1
Pu-241	0.0	+0.5	0.0	+0.77
U-235	5.6	-0.5	8.3	-0.5
U-238	775.2	-8.2	1153.0	-9.1
Fission Products	0.0	11.1	0.0	16.4

The average fuel burnup, expressed in terms of a/o, can be obtained from the equation

$$B(a/o) \approx 2.2 \times 10^{-2} \frac{Pt}{M_0} ,$$

or

$$B(a/o) \approx 1.6 \times 10^{-2} t .$$

Thus, at 70 days the average burnup is 1.1%, and at 4 x 70 days the average burnup is 4.4%. Note that 1 a/o corresponds to about 10 MWd/kg.

Finally, the problem of approaching the "equilibrium" cycle should be mentioned. If the initial fuel composition is used to reload, there

will be a net reactivity loss of about 2.6% during the first three cycles. This loss could be balanced by starting with a partially loaded core or by reloading with a more enriched fuel.

VIII. FLUX MAGNITUDE DURING SHUTDOWN

Since the shutdown flux profiles are similar to the power-level distributions shown in Figs. 3.3 and 3.4, it is only necessary to compute the relative flux magnitudes. This can be accomplished by examining the relative neutron source rates.

At power level (400 MWt), the neutrons source rate, S_p , is given by the expression

$$\begin{aligned} S_p &\cong 3.05 \times 10^{10} \times 4 \times 10^8 \times 3.0 \\ &\approx 3.65 \times 10^{19} \text{ neutrons/sec.} \end{aligned}$$

During shutdown, the total neutron source rate, S_s , can be computed from the equation

$$S_s = \frac{S_o}{1-k},$$

where:

S_o = Background source rate (spontaneous fission plus (α, n) reaction in oxygen).

k = Multiplication constant.

For the FTR, the background source rate is approximately

$$\begin{aligned} S_o &\approx 7.1 \times 10^7 \text{ (fission)} + 4.1 \times 10^7 \text{ (}\alpha, n\text{'s)} \\ &\approx 11.2 \times 10^7 \text{ neutrons/sec.} \end{aligned}$$

Combining the above expressions, we have

$$\phi_s = 3.1 \times 10^{-12} \frac{\phi_o}{1-k} ,$$

where ϕ_s and ϕ_o are the shutdown and power-level fluxes, respectively.

Thus, the central shutdown flux for $k = 0.9$ is

$$\phi_s \approx 3.1 \times 10^{-12} \times 10 \times 7 \times 10^{15} \approx 2.2 \times 10^5 \text{ neutrons/cm}^2\text{-sec} .$$

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2. W. W. Little, Jr., and R. W. Hardie, "2DB User's Manual -- Revision I", BNWL-831 R (February 1969).
3. R. W. Hardie and W. W. Little, Jr., "PERT-IV, A Two-Dimensional Perturbation Code in FORTRAN-IV", BNWL 409 (April 1967).

APPENDIX A

DISCUSSION OF NUCLEAR DATA

In order to assess the relative accuracy of the cross sections used in this study, numerous comparisons with critical experiments have been performed. A representative sample of these results are described below.

Table A-I gives the calculated k_{eff} and critical bucklings for selected critical assemblies. The k_{eff} calculations were performed using the 2-D diffusion code, 2DB, with 12 energy groups. As can be seen, the eigenvalue is predicted moderately well in the uranium blanket assemblies, but is underestimated by about 1-2% in the nickel reflected systems.

A tabulation of the measured and calculated central spectrum fission rate ratios is given in Table A-II. In general, the agreement is satisfactory--especially for the $\text{U}^{238}/\text{U}^{235}$ fission ratio.

TABLE A-I*Eigenvalue Calculations*

Assembly	k_{eff}	$B^2 \text{ (cm}^{-2}\text{)}$ ($k = 1.0$)
ZPR-III 48	1.002	2.56×10^{-3}
ZPR-III 49	1.003	2.31×10^{-3}
ZPR-III 50	0.995	3.07×10^{-3}
ZPR-III 51	0.982	2.90×10^{-3}
ZPR-III 52	0.987	2.90×10^{-3}
ZEBRA 6A	1.003	3.31×10^{-3}

TABLE A-II*Central Fission Rate Ratios*

Assembly	U^{238}/U^{235}		Pu^{239}/U^{235}		Pu^{240}/U^{235}	
	C	E	C	E	C	E
ZPR-III 48	0.0307	0.0307	0.955	0.976	0.214	0.243
ZPR-III 49	0.0337	0.0345	0.984	0.986	0.236	--
ZPR-III 50	0.0259	0.0251	0.861	0.903	0.168	0.183
ZPR-III 51	0.0310	0.0309	1.00	1.00	0.224	0.240
ZEBRA 6A	0.0362	0.0364	0.946	0.961	0.232	0.252

APPENDIX B

MATERIAL DENSITIES

<u>Material</u>	<u>Molecular Weight</u>	<u>Density (gm/cc)</u>	<u>Number Density (atoms/barn-cm)</u>
Be	9.01	1.84	0.12296
C	12.01	1.80	0.09026
Na	22.99	0.87	0.02284
Fe	55.85	7.87	0.08487
Ni	58.71	8.90	0.09130
Ta	180.95	16.60	0.05525
W	183.86	19.30	0.06322
B ₄ C (19.7 a/o B-10)		2.51	--
B-10	10.01	--	0.02156
B-11	11.01	--	0.08789
C	12.01	--	0.02736
316 SS		7.92	--
Cr (17 w/o)	52.01	--	0.01560
Fe (68 w/o)	55.85	--	0.05808
Ni (12 w/o)	58.71	--	0.00975
Mo (3 w/o)	95.94	--	0.00149
304 SS		7.92	--
Cr (18 w/o)	52.01	--	0.01651
Fe (74 w/o)	55.85	--	0.06321
Ni (8 w/o)	58.71	--	0.00650
BeO	25.01	2.86	--
Be	9.01	--	0.06887
O	16.00	--	0.06887

<u>Material</u>	<u>Molecular Weight</u>	<u>Density (gm/cc)</u>	<u>Number Density (atoms/barn-cm)</u>
PuO ₂ (88 a/o 239, 12 a/o 240)		11.46	--
Pu-239	239.05	--	0.02240
Pu-240	240.05	--	0.00305
O	16.00	--	0.05091
UO ₂ (0.72 a/o 235)		10.96	--
U-235	235.04	--	0.00018
U-238	238.05	--	0.02427
O	16.00	--	0.04940

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