MASTER

COMPILATION OF FAST REACTOR EXPERIMENTS VOLUME II

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SECTION 5 - REACTION RATES (INCLUDING FISSION RATIOS)

As for sample worth measurements, the utility of reaction rate measurements depends not only on the precision of the experiment itself, but also on the degree to which the experimental configuration is accounted for in the analysis. Unfortunately, some important questions such as whether a given fission counter samples the spectral distribution in a cell properly are not readily answered in the literature. An attempt has been made to include information pertinent to such questions in the entries in this section. Specific studies on the effect of heterogeneity on reaction rates are compiled in Section 10.4.

Section 5.2 is devoted to measurements of the spatial variation of reaction rates in assemblies. This does not include cell distributions which are compiled in Section 10.1.

SECTION 5.1 - CENTRAL REACTION RATES

5.1-1-2 Central Reaction Rates

Facility: ZPR-3 (1.1) Assembly: 2 (2.1-2)

Reactions Considered: Fission in U-233, U-234, U-238, and Pu-239 relative to U-235.

Experimental Method:

"Absolute" fission chambers of the Kirn Type (2) were used in this and many subsequent assemblies. The chamber is shown in Figure 1.

Figure 1

The fissile material is deposited on the base of the chamber and a circular collection plate is mounted about 0.3 in. from the base. The chamber is filled with an argon-methane mixture and sealed. The chamber is made absolute by the deposition of an accurately known quantity of fissile material on the chamber base in a very thin, adherent film spread over a circle of a precisely known diameter. This is accomplished by making the chamber base an electrode in an electrolytic cell containing as electrolyte a solution of a salt of the fissile material. The strength and volume of the solution are known accurately, and the plating is continued until nearly all the fissile material is deposited. An analysis of the strength of the residual electrolyte gives the amount of material left and

5.1-1-2 (cont'd)

hence gives the amount deposited. The second electrode of the cell consists of a rotating paddle which stirs the electrolyte, and consequently the fissile material is deposited as a uniform film. After removal from the cell, the chamber is fired to reduce the fissile material to a hard, uniform oxide film.

The chambers contain 400 to 800 μg of material deposited over an area about 1 in. in diameter, and these fissile films are sufficiently thin that the absorption of fission fragments in the film is very small. Consequently, the chambers have efficiencies close to 100%.

A chamber is inserted into the core by restacking the plates in a drawer so that a void (approximately 2 in. x 2 in. x 1 in. deep) is created at the front of one of the drawers. A fission ratio is measured by mounting one chamber in each half of the core so that when the halves are driven together the chambers are nearly in contact. One chamber of the pair is usually a U^{235} chamber, so that all the ratios include U^{235} . All the chambers contain fairly small quantities of isotopes other than the principal isotope, and corrections for their presence are made experimentally by constructing a suitable set of simultaneous equation for the reaction rates of the chambers and solving these to obtain the reaction rates of the individual isotopes.

The chamber base containing the fissile material is perpendicular to the plane of the plates of the cell and is thus considered to average over the cell spectrum.

<u>Results:</u>		Ratio of Fission Rate to that in U^{235}
<u>Material</u>	<u>Uncorrected</u>	<u>Corrected</u> (a)
U-233	1.53	1.55
U-234	0.47	0.545
U-238	0.064	0.069
Pu-239	1.16	1.17

(a) Primarily for inelastic scattering in the chamber walls See section 5.1-1-29.

References:

- (1) J. K. Long, et al., "Fast Neutron Power Reactor Studies with ZPR-III," ICPUAE, Geneva 1958, 12, 119.
- (2) W. G. Davey, "A Critical Comparison of Measured and Calculated Fission Ratios for ZPR-III Assemblies" ANL-6617 (1962).

Facility: ZPR-3 (1.1)Assembly: 2A (2.1-2a)Reactions Considered:

Fission in U-234, U-238 and Pu-239 relative to U-235

Experimental Method:

Kirk absorption fission chambers at core center (see Section 5.1-1-2).

Results:

<u>Material</u>	Ratio of Fission Rate to that in U ²³⁵	
	<u>Uncorrected</u>	<u>Corrected^a</u>
U-234	0.57	0.066
U-238	0.088	0.095
Pu-239	1.28	1.29

^a See Section 5.1-1-29Reference:

- (1) J. K. Long, et al., "Fast Neutron Power Reactor Studies with ZPR-III", ICPUAE, Geneva, 1958, 12, 119.

Facility: ZPR-3 (1.1)Assembly: 3 (2.1-3)Reactions Considered:

Fission in U-234, U-238 and Pu-239 relative to U-235

Experimental Method:Kirk absolute fission chambers placed at center of central blanket region. (5.1-1-2).Results:

Ratio of Fission Rates to that in U-235

<u>Material</u>	<u>Uncorrected</u>	<u>Corrected</u> ^a
U-234	0.34	0.394
U-238	0.031	0.033
Pu-239	1.12	1.13

^aSee Section 5.1-1-29Reference:

- (1) J. K. Long, et al., "Fast Neutron Power Reactor Studies with ZPR-III", ICPUAE, Geneva, 1958, 12, 119.

Facility: ZPR-3 (1.1)Assembly: 4 (2.1-4)Reactions Considered:

Fission in U-234, U-238 and Pu-239 relative to U-235

Experimental Method:

Kirk absolute fission chambers at center of central blanket region (5.1-1-2).

Results:

<u>Material</u>	Ratio of Fission Rates to that in U-235	
	<u>Uncorrected</u>	<u>Corrected</u> ^a
U-234	0.395	0.458
U-238	0.0375	0.405
Pu-239	1.213	1.225

^aSee Section 5.1-1-29Reference:

- (1) J. K. Long et al., "Fast Neutron Power Reactor Studies with ZPR-III", ICPUAE, Geneva, 1958, 12, 119.

Facility: ZPR-3 (1.1)Assembly: 5 (2.1-5)Reactions Considered:

Fission in U-234, U-238 and Pu-239 relative to U-235

Experimental Method:

Kirk absolute fission chambers placed at center of core (5.1-1-2).

Results:

<u>Material</u>	<u>Ratio of Fission Rate to that in U-235</u>	
	<u>Uncorrected</u>	<u>Corrected^a</u>
U-234	0.56	0.65
U-238	0.074	0.080
Pu-239	1.24	1.25

^a See Section 5.1-1-29Reference:

- (1) J. K. Long, et al., "Fast Neutron Power Reactor Studies with ZPR-III" ICPUAE, Geneva, 1958, 12, 119.

5.1-1-6F Central Reaction Rates

Facility: ZPR-3 (1.1)

Assembly: 6F (2.1-6F)

Reactions Considered:

- a) U-238 (n,f) / U-235 (n,f)
- b) U-238 (n,γ) / U-235 (n,f)

Experimental Method: (2)

a) Fission rates were determined by radiochemical analysis of depleted and enriched foils exposed at the center of the assembly. The fission product Mo⁹⁹ is separated and counted, and the fission rates are inferred using fission yields of Mo⁹⁹ of 6.3% for U-238 and 6.1% for U-235.

b) The capture rate in U-238 was also determined radiochemically; an irradiated foil is allowed to decay sufficiently to allow all the U-239 produced to decay to Np-239 which is then extracted. The chemical yield is determined using a Np-237 tracer and the Np-239 activity is counted in a calibrated β counter.

Results:

- a) $\frac{\sigma^{-28} (n,f)}{\sigma^{-25} (n,f)} = 0.069$
- b) $\frac{\sigma^{-28} (n,\gamma)}{\sigma^{-25} (n,f)} = 0.103$

(including a correction of 1.227 due to subsequent re-calibration of the Np-239 counter (3))

References:

- (1) J. K. Long, et al., "Fast Neutron Power Reactor Studies with ZPR-III," ICPUAE, Geneva, 1958, 12, 119.
- (2) R. J. Armani, et al., "Improved Techniques for Low-Flux Measurement of Prompt Neutron Lifetime, Conversion Ratio and Fast Spectra," Exponential and Critical Experiments, IAEA, Vienna, 1964.
- (3) ANL Reactor Development Program Progress Report, April 1965, ANL-7045 p. 24.

5.1-1-9

Central Reaction Rates

Facility: ZPR-3 (1.1)

Assembly: 9 (2.1-9)

Reactions Considered:

Fission in U-234, U-238 and Pu-239 relative to U-235

Experimental Method:

Kirn absolute fission chambers placed at center of core (5.1-1-2)

Results:

<u>Material</u>	<u>Ratio of Fission Rate to that in U-235</u>	
	<u>Uncorrected</u>	<u>Corrected^a</u>
U-234	0.43	0.50
U-238	0.051	0.055
Pu-239	1.30	1.31

^aSee Section 5.1-1-29

Reference:

- (1) J. K. Long, et al., "Fast Neutron Power Reactor Studies with ZPR-III", ICPUAE, Geneva, 1958, 12, 119.

5.1-1-9A

Central Reaction Rates

Facility: ZPR-3 (1.1)

Assembly: 9A (2.1-9A)

Reactions Considered:

Fission in U-234 and Pu-239 relative to U-235

Experimental Method:

Kirn chambers placed at center of core (See Section 5.1-1-2)

Results:

<u>Material</u>	Rate of Fission Rate to that in U-235	
	<u>Uncorrected</u>	<u>Corrected*</u>
U-234	0.429	0.500
Pu-239	1.30	1.31

Reference:

- (1) R. Avery et al., "Physics of Fast Reactors", ICPUAE, Geneva, 1964, 6, 112.

Facility: ZPR-3 (1.1)Assembly: 10 (2.1-10)Reactions Considered:

Fission in U-233, U-234, U-238 and Pu-239 relative to U-235.

Experimental Method:Kirk absolute fission chambers placed at center of core (5.1-1-2)Results:

<u>Material</u>	<u>Counter No.</u>	<u>Ratio of Fission Rate to that in U-235</u>	
		<u>Uncorrected</u>	<u>Corrected (a)</u>
U-233	16	1.52	1.535
U-234	8	0.332 }	0.378
	11	0.330 }	
U-238	2,3	0.0440	0.0475
Pu-239	20	1.21 }	1.24
	21	1.23 }	

(a) See Section 5.1-1-29.Reference:

- (1) W. G. Davey and P. I. Amundson, "A Re-evaluation of Fission Ratios Measured in ZPR-III Critical Assemblies", ANL-6941 (1964).

Facility: ZPR-3 (1.1)Assembly: 11 (2.1-11)Reactions Considered:

Fission in U-233, U-234, U-238 and Pu-239 relative to U-235.

Experimental Method:

Kirn absolute fission chambers placed at center of core
(5.1-1-2)

Results:

These results are the mean values for Assemblies 11 and 22 which are almost identical.

<u>Material</u>	<u>Counter No.</u>	<u>Ratio of Fission Rate to that in U-235</u>	
		<u>Uncorrected</u>	<u>Corrected (a)</u>
U-233	16	1.51	1.525
U-234	8	0.299	0.339
	11	0.293	
U-238	2,3	0.0355	0.0383
Pu-239	20	1.18	1.19
	21	1.18	

(a) See Section 5.1-1-29.

Reference:

- (1) W. G. Davey and P. I. Amundson, "A Re-evaluation of Fission Ratios Measured in ZPR-III Critical Assemblies", ANL-6941 (1964).

Facility: ZPR-3 (1.1)Assembly: 12 (2.1-12)Reactions Considered:

- A. Fission in U-233, U-234, U-238 and Pu-239 relative to U-235.
- B. U-238 (n, γ)/U-235 (n,f)

Experimental Method:

- A. Kirm absolute fission chambers placed at center of core (5.1-1-2)
- B. Np-239 counting after chemical separation from U-238 foil and Mo-99 β counting after chemical separation from U-235 foil (5.1-1-6F)

Results:

A. <u>Material</u>	<u>Counter No.</u>	<u>Rate of Fission Rate to that in U-235</u>	
		<u>Uncorrected</u>	<u>Corrected^(a)</u>
U-233	16	1.46	1.475
U-234	8	0.297	} 0.335
	11	0.288	
U-238	2,3	0.0444	0.0486
Pu-239	20	1.10	1.12

(a) See Section 5.1-1-29

B. $\bar{\sigma}^{28} (n,\gamma) / \bar{\sigma}^{25} (n,f) = 0.123$

(includes a factor of 1.227 due to a subsequent calibration of the Np-239 counter) ⁽²⁾

References:

- (1) W. G. Davey and P. I. Arundson, "A Re-evaluation of Fission Ratios Measured in ZPR-III Critical Assemblies", ANL-6941 (1964).
- (2) ANL Reactor Development Program Progress Report, April 1965, ANL-7045, p. 24.

Facility: ZPR-3 (1.1)Assembly: 14 (2.1-14)Reactions Considered:

- A. Fission in U-233, U-234, U-238 and Pu-239 relative to U-235
- B. U-238 (n, γ)/U-235 (n,f)

Experimental Method:

- A. Kirm absolute fission chambers placed at center of core (5.1-1-2).
- B. Np-239 counting after chemical separation from depleted foil and Mo⁹⁹ β counting after separation from U-235 foil. (5.1-1-6F).

Results:

A. <u>Material</u>	<u>Counter No.</u>	<u>Ratio of Fission Rate to that in U-235</u>	
		<u>Uncorrected</u>	<u>Corrected^(a)</u>
U-233	16	1.45	1.465
U-234	8	0.306	0.384
	11	0.304	
U-238	2,3	0.0550	0.0594
Pu-239	20	1.05	1.06
	21	1.05	

(a) See Section 5.1-1-29

B. $\sigma^{-28}(n,\gamma)/\sigma^{-25}(n,f) = 0.134$ (including a correction factor of 1.227 due to a subsequent calibration of the Np-239 counter⁽²⁾).

References:

- (1) W. G. Davey and P. I. Amundson, "A Re-evaluation of Fission Ratios Measured in ZPR-III Critical Assemblies", ANL-6941 (1964).
- (2) ANL Reactor Development Program Progress Report April, 1965, ANL-7045, p. 24.

Facility: ZPR-3 (1.1)Assembly: 16 (2.1-16)Reactions Considered:

Fission in U-234 and U-238 relative to U-235.

Experimental Method:

Kirn absolute fission chambers placed at center of core (5.1-1-2)

Results:

<u>Material</u>	<u>Counter No.</u>	Ratio of Fission Rate to that in U-235	
		<u>Uncorrected</u>	<u>Corrected^(a)</u>
U-234	8	0.297	0.336
U-238	2	0.0414	0.0447

(a) See Section 5.1-1-29

Reference:

- (1) W. G. Davey and P. I. Amundson, "A Re-evaluating of Fission Ratios Measured in ZPR-III Critical Assemblies", ANL-6941 (1964).

Facility: ZPR-3Assembly: 17 (2.1-17)Reactions Considered:

Fission in U-233, U-234, U-238 and Pu-239 relative to U-235.

Experimental Method:

Kern absolute fission chambers placed at center of core (5.1-1-2).

Results:

<u>Material</u>	<u>Counter No.</u>	<u>Ratio of Fission Rate to that in U-235</u>	
		<u>Uncorrected</u>	<u>Corrected (a)</u>
U-233	16	1.46	1.475
U-234	8	0.296	0.340
	11	0.299	
U-238	2, 3	0.0490	0.0529
Pu-239	20	1.07	1.10
	21	1.09	

(a) See Section 5.1-1-29

Reference:

- (1) W. G. Davey and P. I. Amundson, "A Re-evaluation of Fission Ratios Measured in ZPR-III Critical Assemblies", ANL-6941 (1964).

Facility: ZPR-3 (1.1)Assembly: 20 (2.1-20)Reactions Considered:

- A. Fission in U-233, U-234, U-238, Pu-239 and Pu-240 relative to U-235
- B. U-238 (n, γ) / U-235 (n,f)

Experimental Method:

A. Kirt absolute fission chambers placed in front of drawers 1-P-16 and 2-P-16 (5.1-1-2).

B. Np-239 counting after chemical separation (5.1-1-6F) from depleted foil and Mo-99 β counting after separation from U-235 foil.

Results:

A. Fission Ratios (2) Material	Counter No.	Ratio of Fission Rate to that in U-235	
		Uncorrected	Corrected (a)
U-233	16	1.52	2.535
U-234	8	0.304	0.342
	11	0.292	
U-238	2,3	0.0381	0.0421
Pu-239	20	1.15	1.17
	21	1.14	
Pu-240	12	0.322	0.349

(a) These corrections are described in Section 5.1-1-29 note that the values given in reference (1) are incorrect; the correction factors have been applied in the wrong sense.

B. $\bar{\sigma}^{28}(n,\gamma) / \bar{\sigma}^{25}(n,f) = 0.1227 + .005$ (including a correction factor of 1.227 due to a subsequent calibration of the Np-239 counter (2))

References:

- (1) C. E. Branyon, "Core A Critical Studies for the Enrico Fermi Atomic Power Plant on ZPR-III" ANL-6629 (1962).
- (2) W. G. Davey and P. I. Amundson, "A Re-evaluation of Fission Ratios Measured in ZPR-III Critical Assemblies" ANL-6941.
- (3) ANL Reactor Development Program Progress Report April 1965, ANL-7045, p.24.

Facility: ZPR-3 (1-1)Assembly: 22 (2.1-22)Reactions Considered:

- A. Fission in U-233, U-234, U-238 and Pu-239 relative to U-235.
- B. U-238 (n, γ)/U-235 (n,f)

Experimental Method:

- A. Kirm absolute fission chambers placed at center of core (5.1-1-2).
- B. a) Np-239 counting after chemical separation (5.1-1-6F)
 b) Mo-99 β counting after chemical separation from U-235 foil (5.1-1-6F)

Results:

A. <u>Material</u>	<u>Counter No.</u>	<u>Ratio of Fission Rate to that in U-235</u>	
		<u>Uncorrected</u>	<u>Corrected (a)</u>
U-233	16	1.51	1.525
U-234	8	0.299	0.339
	11	0.293	
U-238	2,3	0.0355	0.0383
Pu-239	20	1.18	1.19
	21	1.18	

(a) See Section 5.1-1-29

B. $\frac{\sigma^{28}(n,\gamma)}{\sigma^{25}(n,f)} = 0.112$ (including a correction factor of 1.227 due to a subsequent calibration of the Np-239 counter (2)).

References:

- (1) W. G. Davey and P. I. Amundson, "A Re-evaluation of Fission Ratios Measured in ZPR-III Critical Assemblies", ANL-6941 (1964).
- (2) ANL Reactor Development Program Progress Report April, 1965, ANL-7045, p. 24.

Facility: ZPR-3Assembly: 23 (2.1-23)Reactions Considered:

- A. Fission in U-233, U-234, U-238 and Pu-239 relative to U-235.
- B. U-238 (n, γ)/U-235 (n,f)

Experimental Method:

- A. Fission absolute fission chambers placed at center of core (5.1-1-2).
- B. a) Np-239 counting after chemical separation (5.1-1-6F)
 b) Mo-99 β counting after chemical separation from foil (5.1-1-6F)

Results:

A. <u>Material</u>	<u>Counter No.</u>	<u>Ratio of Fission Rate to that in U-235</u>	
		<u>Uncorrected</u>	<u>Corrected^(a)</u>
U-233	16	1.48	1.495
U-234	8	0.408	0.460
	11	0.396	
U-238	2,3	0.0678	0.0732
Pu-239	21	1.19	1.20

(a) See Section 5.1-1-29

B. $\sigma^{-28}(n,\gamma)/\sigma^{-25}(n,f) = 0.109$ (including a correction factor of 1.227 due to a subsequent calibration of the Np-239 counter⁽²⁾).

References:

- (1) W. G. Davey and P. I. Amundson, "A Re-evaluation of Fission Ratios Measured in ZPR-III Critical Assemblies", ANL-6941 (1964).
- (2) ANL Reactor Development Program Progress Report April, 1965, ANL-7045, p. 24.

Facility: ZPR-3 (1.1)Assembly: 24 (2.1-24)Reactions Considered:

Fission in U-233, U-238 and Pu-239 relative to U-235.

Experimental Method:

Kirk absolute fission chambers placed at center of core (5.1-1-2)

Results: (a)

<u>Material</u>	<u>Counter No.</u>	<u>Ratio of Fission Rate to that in U-235</u>	
		<u>Uncorrected</u>	<u>Corrected (b)</u>
U-233	16, 18	1.44	1.462
U-234	8	0.408	0.400
	11	0.396	
U-238	2, 3	0.0308	0.0333
Pu-239	21	1.17	1.18

(a) Mean values for Assemblies 24 and 38, which are nearly identical in composition

(b) See Section 5.1-1-29

References:

- (1) W. G. Davey and P. I. Amundson, "A Re-evaluation of Fission Ratios Measured in ZPR-III Critical Assemblies" ANL-6941 (1964).

5.1-1-25

Central Reaction Rates

Facility: ZPR-3 (1.1)

Assembly: 25 (2.1-25)

Reactions Considered:

A. Fission in U-234, U-238, and Pu-239 relative to U-235.

B. $U-235(n,\gamma)/U-235(n,f)$

Experimental Method:

A. Kirm absolute fission chambers placed at center of core (5.1-1-2).

B. a) Np-239 counting after chemical separation (5.1-1-6F)

b) Mo-99 β counting after chemical separation from U-235 foil (5.1-1-2).

Results:

A.	Material	Counter No.	Rate of Fission Rate to that in U-235 (1)	
			Uncorrected	Corrected ^(a)
	U-234	11	0.253	0.293
	U-238	3	0.0292	0.0315
	Pu-239	21	1.17	1.18

(a) See Section 5.1-1-29

B. $\frac{\sigma^{-28}(n,\gamma)}{\sigma^{-25}(n,f)} = 0.113$ (including a correction factor of 1.227 due to a subsequent calibration of the Np-239 counter⁽²⁾)

References:

- (1) W. G. Davey and P. I. Amundson, "A Re-evaluation of Fission Ratios Measured in ZPR-III Critical Assemblies" ANL-6941 (1964).
- (2) ANL Reactor Development Program Progress Report April, 1965, ANL-7045, p. 24.

5.1-1-29**Central Reaction Rates****Facility:** ZPR-3 (1.1)**Assembly:** 29 (2.1-29)**Reactions Considered:**

A. Fission in U-233, U-236, U-238, Pu-239 and Pu-240 relative to U-235

B. U-238 (n γ) / U-235 (n.f)**Experimental Method:**

A. Kirn chambers (see Section 5.1-1-2) were used; the U-235 chamber (counter no. 5) was placed in the front of drawer 1-P-16 and the second chamber, directly opposite in drawer 2-P-16. For all measurements except that for Pu-240, the fuel in the central 9 drawers in each half was "unbunched" (see Section 10.4-1-29).

B. Np-239 counting after chemical separation from U-238 foil and Mo-99 β counting after separation from U-235 foil (5.1-1-6F).

Results and Corrections:**A. Fission Rates**

TABLE I
Ratio of Fission Rate to that in U-235

<u>Material</u>	<u>Counter No.</u>	<u>Uncorrected</u>	<u>Corrected (a)</u>
U-233	16	1.47	1.485
U-234	11	0.259	0.300
U-236	24	0.082	0.087
U-238	2,3	0.0356	0.0384
Pu-239	21	1.06	1.07
Pu-240	12	0.289	0.303

(a) The corrections that have been applied subsequent to the original reporting of the data are those due to a) inelastic scattering in the counter walls (2) (3) b) the effective mass of fissionable material in the counter (4) and c) variable loss of fission counts below operating bias. These corrections can be summarized in the following table (4).

TABLE II

Corrections to Measured Fission Ratios

Material	Correction Relative to U - 235 (a)			
	Inelastic Scattering	Effective Mass	Operating Conditions	Total Correction
U-233	0	0	+ 1%	+ 1%
U-234	+ 4%	+12 %	0	+16%
U-236	+ 6%	not evaluated	0	+ 6%
U-238	+ 8%	"	0	+ 8%
Pu-239	0	0	+ 1%	+ 1%
Pu-240	+ 4%	0	+ 1%	+ 5%

a) A positive correction means that the experimental value must be increased.

The corrections for inelastic scattering were originally calculated by DSN and subsequently corroborated experimentally (5). The corrections were found to be independent of spectrum in the range of spectra characteristic of ZPR-3 assemblies (3). Thus the values of Table II have been used to correct all ZPR-3 data.

b) $\frac{\sigma^{28}(n, \gamma)}{\sigma^{25}(n, f)} = 0.141$ (including a correction factor of 1.227 due to a subsequent calibration of the Mp-239 counter (6))

References:

- (1) A. L. Hess, et al., "Critical Studies of a 450-liter Uranium Oxide Fast Reactor Core (ZPR-III Assembly 29)" ANL-6336 (1960)
- (2) W. G. Davey and R. M. Curran, "An Experimental Investigation of Some Sources of Error in the Measurement of Absolute Fission Ratios in Fast Reactors" ANL-6468 (1961).
- (3) W. G. Davey, "A Critical Comparison of Measured and Calculated Fission Ratios for ZPR-III Assemblies" ANL-6617 (1962).
- (4) W. G. Davey and P. I. Amundson, "A Re-evaluation of Fission Ratios Measured in ZPR-III Critical Assemblies" ANL-6941 (1964).
- (5) A. M. Broomfield, et al., "ZPR-3 Assembly 48: Studies of a Dilute Plutonium-fueled Assembly" p. 205 in ANL-7320 (1967).
- (6) ANL Reactor Development Program Progress Report April, 1965, ANL-7045, p. 24.

5.1-1-30

Central Reaction Rates

Facility: ZPR-3 (1.1)

Assembly: 30 (2.1-30)

Reactions Considered: Fission in U-233, U-234, U-236, U-238
and Pu-239 relative to U-235

Experimental Method:

Kirn absolute fission chambers placed at center of core (5.1-1-2)

Results:

<u>Material</u>	<u>Chamber No.</u>	<u>Ratio of Fission Rate to that in U-235</u>	
		<u>Uncorrected</u>	<u>Corrected (a)</u>
U-233	16	1.49	1.505
U-234	11	0.301	0.349
U-236	25	0.099	0.105
U-238	2	0.0427	0.0461
Pu-239	21	1.12	1.13

(a) See Section 5.1-1-29

References:

- (1) P. I. Amundson et al., "Critical Studies of a Dilute Oxide Fast Reactor Core (ZPR-III Assembly 30)" ANL-6337 (1961)
- (2) W. G. Davey and P. I. Amundson, "A Re-evaluation of Fission Ratios Measured in ZPR-III Critical Assemblies", ANL-6941 (1964).

5.1-1-31 Central Reaction Rates

Facility: ZPR-3 (1.1)

Assembly: 31 (2.1-31)

Reactions Considered: Fission in U-233, U-234, U-236, U-238, Pu-240 relative to U-235.

Experimental Method:

Kirn absolute fission chambers placed at center of core (see section 5.1-1-2)

Results:

<u>Material</u>	<u>Chamber No.</u>	<u>Ratio of Fission Rate to that in U-235</u>	
		<u>Uncorrected</u>	<u>Corrected^(a)</u>
U-233	16	1.52	1.535
U-234	11	0.334	0.387
U-236	24,25	0.106	0.112
U-238	2	0.0440	0.0475
Pu-239	19	1.17	1.21
	21	1.19	
Pu-240	6,12	0.313	0.329

(a) See section 5.1-1-29

Reference:

W. G. Davey and P. I. Amundson, "A Re-evaluation of Fission Ratios Measured in ZPR-III Critical Assemblies", ANL-6941 (1964).

5.1-1-32 Central Reaction Rates

Facility: ZPR-3 (1.1)

Assembly: 32 (2.1-32)

Reactions Considered: Fission in U-233, U-234, U-236, U-238, Pu-239, and Pu-240 relative to U-235.

Experimental Method:

Kirn absolute fission chambers placed at center of core (see section 5.1-1-2)

Results:

<u>Material</u>	<u>Chamber No.</u>	<u>Ratio of Fission Rate to that in U-235</u>	
		<u>Uncorrected</u>	<u>Corrected (a)</u>
U-233	16	1.51	1.525
U-234	11	0.367	0.426
U-236	24	0.110	0.117
U-238	2	0.0451	0.0487
Pu-239	21	1.20	1.21
Pu-240	12	0.382	0.401

(a) See Section 5.1-1-29

Reference:

- (1) W. G. Davey and P. I. Amundson, "A Re-evaluation of Fission Ratios Measured in ZPR-III Critical Assemblies", ANL-6941 (1964).

5.1-1-33**Central Reaction Rates****Facility:** ZPR-3 (1.1)**Assembly:** 33 (2.1-33)**Reactions Considered:** Fission in U-233, U-234, U-236, U-238, Pu-239, and Pu-240 relative to U-235**Experimental Method:**

Kirn absolute fission chambers placed at center of core (see section 5.1-1-2)

Results:

<u>Material</u>	<u>Chamber No.</u>	<u>Ratio of Fission Rate to that in U-235</u>	
		<u>Uncorrected</u>	<u>Corrected (a)</u>
U-233	16	1.51	1.525
U-234	11	0.370	0.429
U-236	24	0.119	0.126
U-238	2	0.0480	0.0518
Pu-239	21	1.21	1.22
Pu-240	12	0.400	0.420

(a) See section 5.1-1-29

Reference:

- (1) W. G. Davey and P. I. Amundson, "A Re-evaluation of Fission Ratios Measured in ZPR-III Critical Assemblies", ANL-6941 (1964).

5.1-1-34

Central Reaction Rates

Facility: ZPR-3 (1.1)

Assembly: 34 (2.1-34)

Reactions Considered:

A. Fission in U-233, U-234, U-236, U-238, Pu-239 and Pu-240 relative to U-235

B. U-238 (n, γ) / U-235 (n,f)

Experimental Method:

A. Kirm absolute fission chambers placed at center of core (5.1-1-2).

B. a) Mn-239 counting after chemical separation (5.1-1-6F)

b) Mo-99 β counting after chemical separation from U-235 foil (5.1-1-6F)

Results:

A.	Material	Counter No.	Ratio of Fission Rate to that in U-235	
			Uncorrected	Corrected (a)
	U-233	16	1.45	1.465
	U-234	11	0.247	0.287
	U-236	24	0.076	0.081
	U-238	2	0.0339	0.0366
	Pu-239	21	1.07	1.08
	Pu-240	6,12	0.271	0.285

(a) See Section 5.1-1-29

B. $\frac{\sigma_{-28}(n,\gamma)}{\sigma_{-25}(n,f)} = 0.150$ (including a correction factor of 1.227 due to a subsequent calibration of the Mn-239)

References:

- (1) W. G. Davey and P. I. Amundson, "A Re-evaluation of Fission Ratios Measured in ZPR-III Critical Assemblies" ANL-6941 (1964).
- (2) ANL Reactor Development Program Progress Report April, 1965, ANL-7045, p. 24.

5.1-1-36**Central Reaction Rates****Facility:** ZPR-3 (1.1)**Assembly:** 36 (2.1-36)**Reactions Considered:**

- A. Fission in U-233, U-234, U-236, U-238, Pu-239 and Pu-240 relative to U-235
- B. U-238 (n, γ) / U-235 (n,f).

Experimental Method:

- A. Kirm absolute fission chambers placed at center of core (5.1-1-2)
- B. a) Np-239 counting after chemical separation (5.1-1-6F)
b) Mo-99 β counting after chemical separation from U-235 foil (5.1-1-6F)

Results:

A.	Material	Counter No.	Ratio of Fission Rate to that in U-235	
			Uncorrected	Corrected (a)
	U-233	16	1.47	1.485
	U-234	11	0.312	0.362
	U-236	24	0.094	0.100
	U-238	2	0.0410	0.0443
	Pu-239	21	1.19	1.20
	Pu-240	6,21	0.337	0.354

(a) See Section 5.1-1-29

- B. $\sigma^{-28}(n,\gamma) / \sigma^{-25}(n,f) = 0.119$ (including a correction factor of 1.227 due to a subsequent calibration of the Np-239 counter⁽²⁾).

References:

- (1) W. G. Davey and P. I. Amundson, "A Re-evaluation of Fission Ratios Measured in ZPR-III Critical Assemblies" ANL-6941 (1964).
- (2) ANL Reactor Development Program Progress Report April, 1965. ANL-7045, p.24.

5.1-1-41

Central Reaction Rates

Facility: ZPR-3 (1.1)

Assembly: 41 (2.1-41)

Reactions Considered: Fission in U-233, U-234, U-236, U-238 and Pu-239 relative to U-235

Experimental Method:

Kirn absolute fission chambers placed at center of core
(5.1-1-2)

Results:

<u>Material</u>	<u>Chamber No.</u>	Ratio of Fission Rate to that in U-235	
		<u>Uncorrected</u>	<u>Corrected (a)</u>
U-233	18	1.454	1.483
U-234	11	0.286	0.332
U-236	24	0.085	0.090
U-238	2	0.0395	0.0427
Pu-239	21	1.16	1.17

(a) See Section 5.1-1-29

Reference:

- (1) W. G. Davey and P. I. Amundson, "A Re-evaluation of Fission Ratios Measured in ZPR-III Critical Assemblies", ANL-6941 (1964).

5.1-1-42 Central Reaction Rates

Facility: ZPR-3 (1.1) Assembly: 42 (2.1-42)

Reactions Considered: Fission in U-233, U-234, U-236, U-238, Pu-239 and Pu-240 relative to U-235

Experimental Method:

2 Kirn chambers placed in opposing central drawers (see section 5.1-1-2).

Results:

<u>Material</u>	Ratio of Fission Rate to that in U-235	
	<u>Uncorrected</u>	<u>Corrected</u> (a)
U-233	1.43	1.44
U-234	0.252	0.298
U-236	0.079	0.084
U-238	0.034	0.037
Pu-239	1.08	1.09
Pu-240	0.282	0.297

(a) See Section 5.1-1-29

Reference:

Amundson, P. I. et al., "A Two-Zone Fast Critical Experiment ,"
ANL-6733 (1964).

5.1-1-43 Central Reaction Rates

Facility: ZPR-3 (1.1) Assembly: 43 (2.1-43)

Reactions Considered:

Fission in U-233, U-236, U-238, and Pu-239 and Pu-240 relative to U-235. See Section 5.2-1-43 for spatial distributions of other reaction rates.

Experimental Method:

Both thin-wall and thick-wall counters were used. As thick-wall counters, Kirn chambers (see section 5.1-1-2) were used for the Pu measurements and gas-flow counters for the U measurements. The thick and thin-wall, gas-flow counters are shown in Fig. 1.

Fig. 1 Absolute Fission Chambers

5.1-1-43 (Cont'd).

Results:

Fission Rates Relative to U-235			
Material	Thick-Wall Counter		Thin Wall Counter
	Uncorrected	Corrected	
U-233	1.42 $\pm$ 0.02	1.43	
U-236	0.052 $\pm$ 0.001	0.055	0.0537 $\pm$ 0.0007
U-238	0.0200 $\pm$ 0.0002	0.0216	0.0217 $\pm$ 0.0002
Pu-239	0.99 $\pm$ 0.01	1.00	
Pu-240	0.183 $\pm$ 0.0002	0.192	

(a) See Section 5.1-1-23

Reference:

J. M. Gasidlo, J. K. Long and W. P. Keeney, "Doppler Effect Measurements, Sodium Void Coefficients, and Critical Studies of a 5,000-liter Fast Power Breeder Reactor by the Two-Zone Method" ANL-6838 (1964).

5.1-1-45A (Cont'd)

Results:

$$\sigma^{238} (n, \gamma) / \sigma^{235} (n, f) = 0.189 \pm 0.010$$

Reference:

W. G. Davey, "The Ratio of U^{238} Capture and U^{235} Fission Cross Sections in Fast Reactors" ANL-7034 (1965).

5.1-1-48

Central Reaction Rates

Facility: ZPR-3 (1.1)

Assembly: 48 (2.1-48)

Reactions Considered:

Fission in U-233, U-234, U-236, U-238, Pu-239 and Pu-240 relative to U-235.

Experimental Method:

Both Kirtz fission chambers (see Section 5.1-1-2) and new thin-walled, spherical back-to-back gas-flow chambers (1) were used. A sketch of the latter is shown in Figure 1.

Figure 1
Spherical, Aluminum Back-to-Back Gas-Flow Chamber
Wall Thickness = 0.030 in.

5.1-1-48 (cont'd)

The fissile material for each chamber was electroplated over a 0.7 in. diameter circle on a 0.0025 in-thick stainless steel disk. Two such disks, one always containing U-235, were placed in each double chamber. The total wall in each chamber was about 100 mg. Wall calibrations were made both by wall balance during plating and by thermal calibration (agreement generally to about 1%) (2). The reduction of the data for each foil included as fission pulses all of the counts above the trough between the alpha and fission peaks in the pulse-height distribution curves plus a linear extrapolation at the minimum trough count rate to zero pulse height.

Results:

Table I
Fission Ratios

Ratio	Kirn Chambers Uncorrected	Kirn w/Wall Correction	Spherical Chambers	Corrected Kirn/Spher.
U-233/U-235	1.454±0.023	1.454	1.480±0.015	0.98±0.02
Pu-239/U-235	0.995±0.015	0.995	0.976±0.010	1.02±0.02
Pu-240/U-235	0.231±0.003	0.241	0.243±0.003	0.99±0.02
U-234/U-235	0.193±0.003	0.201	0.204±0.003	0.99±0.02
U-236/U-235	0.0643±0.0009	0.0679	0.0670±0.0011	1.01±0.02
U-238/U-235	0.0284±0.0005	0.0307	0.0307±0.0005	1.00±0.02

The correction factors measured here for scattering in the chamber walls are compared with the estimated ones used previously (see Section 5.1-1-29) in Table II.

Table II Comparison of Wall Corrections

Fission Ratio	Measured Thick-to-Thin Wall Factor	Previously Used Correction
U-238/U-235	1.032 ± 0.005	1.08
U-236/U-235	1.055 ± 0.005	1.06
U-234/U-235	1.041 ± 0.003	1.04
Pu-240/U-235	1.042 ± 0.003	1.04

5.1-1-48 (cont'd)

References:

1. W. G. Davey and P. I. Amundson, "Inelastic Scattering Measurements in a Fast Reactor by the Spherical Shell Method", Nucl. Sci. Eng., 28, 111 (1967)
2. E. F. Bennett, et al., "Measurements of Spectrum and Spectral Indices in ZPR-3 Assemblies 48 and 48B" ZPR-TM-9 (1970)

5.1-2-1 Central Reaction Rates

Facility: ZPR-6 (1.2)

Assembly: 1 (2.2-1)

Reactions Considered:

Fission in U-233, U-235, U-238

Experimental Method:

Measurements were made about 5 cm from the center of the core by means of a thin-walled Kirn-type gas flow counter.

Measurements were also made using miniature solid-state detectors. See Section 5.1-2-2.

Results:

<u>Counter</u>	<u>U-233/U-235</u>	<u>U-238/U-235</u>
Gas Flow		0.0392 ± 0.0005
Solid State	1.625 ± 0.008	0.0395 ± 0.0015

References:

1. "Reactor Development Program Progress Report - September 1963"
ANL-6784, p. 10 (October 15, 1963)
2. "Reactor Development Program Progress Report - November 1963"
ANL-6808, p. 5-7 (December 15, 1963)

Facility: ZPR-6 (1.2)Assembly: 2 (2.2-2)Reactions Considered:

Fission in U-233, U-234, U-236, U-238, Wp-237, Pu-239, Pu-240, and Am-241, relative to fission in U-235.

Experimental Method:

Two thin-walled, Kirt-type gas-flow counters* (5 cm diameter x 2.5 cm long) were used for the measurements. They were stainless steel with 0.3-mm thick walls and contained a known quantity of fissionable material electroplated with a superficial density, of $\sim 100 \mu\text{g}/\text{cm}^2$ on a stainless steel backing plate. The two counters, one containing a U-235 coating, were placed in 2.5 cm deep by 5 cm square voids at the fronts of the two central drawers. The pulse cables and gas supply tubes were lead through axial ducts, 2.5 cm x 1.3 cm wide, located in the middle of the drawers.

Measurements were also made with two solid-state detectors of 1.3 cm diam. and $\sim 100 \mu\text{g}/\text{cm}^2$ coating thickness. These detectors were centered between the two fuel columns of the drawers at the location: row = 23, column = 21, (See Figure 1 of 2.2-2). This location is about 10 cm radially outward from the core center.

Corrections:

Corrected for fission contributions of isotopes other than the principal ones, except for Wp-237 and Am 241. See table of results.

Precision:

Experimental errors were based on counting statistics. Fissionable material deposits were known to less than 1% when prepared from solutions of gravimetrically known concentration. Errors from 1.5% to 4% were assumed when the amount was determined by alpha activity on the basis of known isotopic composition.

*See Section 5.1-1-2

Results:

Table I

Experimental Fission Rates

<u>Isotope Ratio</u>	<u>Flow Counter</u> <u>(σ_g/σ_f^{235})</u>	<u>Solid State Counter</u> <u>(σ_g/σ_f^{235})</u>
U-233/U-235	1.520 $\pm$ 0.020 1.522 $\pm$ 0.020	1.45 $\pm$ 0.04
U-234/U-235	0.225 $\pm$ 0.004	0.233 $\pm$ 0.007
U-236/U-235	0.074 $\pm$ 0.004	
U-238/U-235	0.0346 $\pm$ 0.0005 0.0354 $\pm$ 0.0005	0.0344 $\pm$ 0.0010
Np-237/U-235	0.174 ^a	0.174 $\pm$ 0.006
Pu-239/U-235	1.045 $\pm$ 0.020	1.09 $\pm$ 0.006
Pu-240/U-235	0.249 $\pm$ 0.005	
Am-241/U-235	0.101 ^a	

^a No accurate isotopic analysis is yet available; data are based on an assumed 100% isotopic purity of the Np-237 and Am-241 samples.

Reference:

G. K. Busch, et al., "Investigations of a 600-Liter Uranium Carbide Core (SRA-VI Assembly No. 2); Reactor Physics Division Annual Report - July 1, 1963 to June 30, 1964 ANL-7010, pp. 91-110 (1965)

Facility: ZPR-6 (1.2)Assembly: 3 (2.2-3)Reactions Considered:

Fission in U-233, U-234, U-236, U-238, Np-237, Pu-239, Pu-240, and Am-241, relative to U-235.

Experimental Method:

Kirn-type gas flow counters, as described in Section 5.1-2-2 were used.

Measurements were made with and without sodium in a central region of about 50 cm length and diameter.

Results:

<u>Isotopes</u>	<u>Ratio of Fission Rate to that of U-235</u>	
	<u>With Na</u>	<u>Without Na</u>
U-233	1.453	1.461
U-234	0.222	0.250
U-236	0.690	0.0716
U-238	0.0331 ^a	0.0383
Np-237	0.242	0.269
Pu-239	1.019	1.038
Pu-240	0.244	0.259

^aThe same U-238/U-235 fission ratio of 0.331 was measured when the light blanket cones were in place.

Reference:

F. H. Helm, et al., "Physics Parameters and Sodium Void Coefficients in a 980-Liter Pancake Core", Research Summary Division Annual Report - July 1, 1964 to June 30, 1965, AEC-7110, pp. 101-113, (1965)

Facility: EPR-6 (1.2)Assembly: 4Z (2.2-4Z)Reactions Considered:

Fission in U-234, U-235, U-238, and Pu-239

Experimental Method:

The detector used for these measurements is shown in Fig. 1 of 5.2-2-4Z and was placed at the center of the assembly. Measurements were done with and without sodium voided in the entire central region.

Precision:

Errors indicated are due to the uncertainty in determining the amount of fissionable material in a counter and measuring the fission rates.

Corrections:

Values given are corrected for isotopic content.

Results:Central Fission Ratios

<u>Fission Ratio</u>	<u>With Sodium</u>	<u>Without Sodium</u>
U-234/U-235	0.187 $\pm$ 0.005	0.214 $\pm$ 0.005
U-238/U-235	0.0256 $\pm$ 0.0008	0.0262 $\pm$ 0.0008
Pu-239/U-235	0.901 $\pm$ 0.024	0.987 $\pm$ 0.024
U-238/U-234	0.136 $\pm$ 0.006	

Reference:

R. A. Karam, et al., "Measured Physics Parameters in a Zoned Fast Uranium Carbide Core, Assembly 4Z", ANL-7281 (1966)

Facility: ZPR-6 (1.2)

Assembly: 5 (2.2-5)

Reactions Considered:

Fission in U-234, U-238, and Pu-239.

See Section 10.4-2-5

5.1-3-1 Central Reaction Rates

Facility: ZPR-9 (1.3)

Assembly: 1 (2.3-1)

Reactions Considered:

U-238 / U-235 fission ratio.

Experimental Method:

Enriched and depleted foils were exposed at the core center. The Mo-99 activity was counted absolutely after chemical separation. The foils were 0.005 in. thick.

Results:

$$\frac{\bar{\sigma}_f(238)}{\bar{\sigma}_f(235)} = 0.41 \pm 0.03$$

Reference:

R. C. Doerner, W. G. Knapp, K. K. Almenas, and R. A. Karam, "Physics Measurements in Tungsten-Based, Aluminum Reflected Fast Reactors", ANL-7007 (March 1967).

5.1-3-2

Central Reaction Rates

Facility: ZPR-9 (1.3)

Assembly: 2 (2.3-1)

Reactions Considered:

U-238 / U-235 fission ratio.

Experimental Method:

See 5.1-3-1

Results:

$$\frac{\bar{\sigma}_f(238)}{\bar{\sigma}_f(235)} = 0.35$$

Reference:

R. C. Doerner, W. G. Knapp, K. K. Albrecht, and R. A. Kavan,
"Physics Measurements in Tungsten-Based, Aluminum-Reflected
Fast Reactors", AML-7007 (1967)

5.1-3-3

Central Reaction Rates

Facility: WPA-9 (1.3)

Assembly: 3 (2.3-1)

Reactions Considered:

U-238 / U-235 fission ratio.

Experimental Method:

See 5.1-3-1

Results:

$$\frac{\bar{\sigma}_f(238)}{\bar{\sigma}_f(235)} = 0.33$$

Reference:

R. C. Doerner, W. G. Knapp, K. K. Almenas, and R. A. Maram,
"Physics Measurements in Tungsten-Based, Aluminum-Reflected
Fast Reactors", ANL-7007 (1967)

5.1-3-4

Central Reaction Rates

Facility: SFR-9 (1.2)

Assembly: 4 (2.1-1)

Reactions Considered:

U-238 / U-235 fission ratio.

Experimental Method:

See 5.1-3-1

Results:

$$\frac{\bar{\sigma}_f(238)}{\bar{\sigma}_f(235)} = 0.42$$

References:

R. C. Beerner, W. G. Knapp, R. K. Almonas, and R. A. Kram,
"Physics Measurements in Tungsten-Based, Aluminum-Reflected
Fast Reactors", ARL-7007 (1967)

5.1-3-4B Central Reaction Rates

Facility: XPR-9 (1.3)

Assembly: 4B (2.3-4B)

Reactions Considered:

U-238 / U-235 fission ratio.

Experimental Method:

See 5.1-3-1.

Results:

$$\frac{\bar{\sigma}_f(238)}{\bar{\sigma}_f(235)} = 0.425$$

Reference:

W. G. Klapp and R. C. Doerner, "Physics Measurements with Modified Diluent Compositions in Tungsten-Based Aluminum-Reflected Fast Reactors" ANL-7207 (Feb. 1967)

5.1-3-5

Central Reaction Rates

Facility: ZPR-9 (1.3)

Assembly: 5. (2.3-5)

Reactions Considered:

U-238 / U-235 fission ratio.

Experimental Method:

See 5.1-3-1

Results:

$$\frac{\bar{\sigma}_f(238)}{\bar{\sigma}_f(235)} = 0.43$$

Reference:

W. G. Knapp and R. C. Doerner, "Physics Measurements with Modified Diluent Compositions in Tungsten-Based, Aluminum-Reflected Fast Reactors" ANL-7207 (Feb. 1967)

S.1-3-6

Central Reaction Rates

Facility: ZPP-9 (1.3)

Assembly: 6 (2.3-6)

Reactions Considered:

U-238 / U-235 fission ratio.

Experimental Method:

See S.1-3-1

Results:

$$\frac{\bar{\sigma}_f(238)}{\bar{\sigma}_f(235)} = 0.44$$

Reference:

W. G. Klapp and R. C. Doerner, "Physics Measurements with Modified Bilvent Compositions in Tungsten-Based, Aluminum-Reflected Fast Reactors", ARL-7207 (Feb. 1967)

5.1-3-7

Central Reaction Rates

Facility: EFR-9 (1.3)

Assembly: 7 (2.3-7)

Reactions Considered:

U-238/U-235 fission ratio

Experimental Method:

See 5.1-3-1

Results:

Fission ratio, U-238/U-235 = 0.0513

References:

R. C. Deerner, W. G. Knapp, R. A. Haron, and D. K. Butler,
"EFR-9 Assemblies No. C-9 Critical Experiments", AML-7208
(March 1967)

5.1-3-8

Central Reaction Rates

Facility: EPR-9 (1.3)

Assembly: 8 (2.3-8)

Reactions Considered:

U-238 / U-235 fission ratio.

Experimental Method:

See 5.1-3-1

Results:

$$\frac{\bar{\sigma}_f(238)}{\bar{\sigma}_f(235)} = 0.48$$

Reference:

R. C. Doerner, W. G. Knapp, R. A. Karam, and D. K. Butler,
"EPR-9 Assemblies No. 6-9 Critical Experiments", AML-7208
(March 1967)

Facility: EPR-9 (1.3)Assembly: 11 (2.3-11)Reactions Considered:

Fission in U-238, U-234, and Pu-239 relative to U-235.

Experimental Method:Results:

<u>Material</u>	<u>Ratio of Fission Rate to that in U-235</u>
U-238	0.0238 $\pm$ 0.0008
U-234	0.176 $\pm$ 0.004
Pu-239	0.986 $\pm$ 0.026

Reference:

L. G. LePage, et al., "A Review of Data Pertaining to Spectral Equilibration and Spectral Matching in Sealed Cores", ANL Reactor Physics Division Annual Report, July 1, 1966 to June 30, 1967 ANL-7310, p. 199 (1968).

5.1-3-18

Central Reaction Rates

Facility: ZPR-9 (1.3)

Assembly: 18 (2.3-18)

Reactions Considered:

U-238/U-235 fission and Pu-237/U-235 fission.

Experimental Method:

See Section 5.1-3-1.

Results:

<u>Material</u>	<u>Ratio of Fission Rate to That in U-235</u>
U-238	0.0232 $\pm$ 0.0008
U-237	0.154 $\pm$ 0.005

Reference:

ANL Reactor Development Program Progress Report August 1967,
ANL-7371, p. 63 (1967)

Assemblies: EPR-9 Assemblies 24 and 24A (2.3-24, 24A) and EPR-3 Assemblies 55 and 55a (2.1-55, 55a).

Reactions Considered:

Ratio of capture to fission (α) in Pu-239 and U-235, absolute capture and fission rates.

Experimental Method:

Because of the special nature of these experiments they will not be reported in detail; the interested reader is referred to the reference.

The experiments were based on the null-reactivity zoned-assembly technique. (Parallel measurements using the reactivity-reaction rate technique are reported in Section 5.1-3-24b). The composition of each of the test zones used was adjusted so that $k_{\infty} = 1$, so that the total fission rate equalled the total absorption rate. One can then write the equation for that balance and solve for α . For the U-235 fueled assemblies, for example, one obtains:

$$\alpha^{25} = [\nu^{25} - 1] + (1/P^{25})[(\nu^{28} - 1) F^{28} - C^{28}] + (1/P^{25})[N^h - C^{SS}]$$

where C and F are the gross radiative capture and fission rates in the test region; the superscripts refer to the isotopes, SS refer to stainless steel and N^h refers to the $(n,2n)$ reaction in the heavy isotopes.

The quantities that are measured are the fission and capture rates. (The third term is calculated, but represents a small correction) A similar equation holds for the Pu-fueled assemblies.

The fission and capture rates were determined in a number of ways. These are described in the reference.

Results:

The various reaction rate ratios are compared in Table I.

Reaction-Rate Ratios (Per Atom) by Different Methods

	Assembly 24			Assembly 55		
	C^{28}/f^{25}	f^{28}/f^{25}	f^{49}/f^{25}	C^{28}/f^{49}	f^{28}/f^{49}	f^{49}/f^{25}
Thermal Calibration	0.1325 ± 0.002	---	---	0.1415 ± 0.002	---	0.902 ± 0.02
Radiochemistry	0.1281 ± 0.005	0.01814 ± 0.0006	---	0.1394 ± 0.005	0.01939 ± 0.0006	---
Fission Chambers	---	0.01914 ± 0.0006	0.930 ± 0.02	---	0.02047 ± 0.0006	0.909 ± 0.01
^{238}U Radiochemistry ^{235}U Fission Ch.	0.1310 ± 0.004	0.01831 ± 0.0006	---	---	---	---
^{238}U Radiochemistry ^{239}Pu Fission Ch.	---	---	---	0.1393 ± 0.004	0.01953 ± 0.0006	---

Table II summarizes the experimental results. The calculations were made using ENDF/B data.

Table II Experimental Results
Assembly

	24	24A	55	55A
Reaction C^{28}/f^{25}	0.1317±0.002			
Rate C^{28}/f^{49}		0.143±0.004	0.1409±0.002	0.1420±0.002
Ratios (a) f^{28}/f^{25}	0.0185±0.0004			
f^{28}/f^{49}		0.0196±0.0006	0.0197±0.0005	0.0201±0.0005
Pu-239/U-235	0.930±0.02		0.908±0.01	0.897±0.01
Fission Ratio (b)				
α (U-235) (c)	0.340±0.03			
α (Pu-239) (c)		0.42±0.05	0.426±0.03	

(a) Average, in-plate values per atom

(b) Cell average, per atom

(c) Average, in-plate value, errors are experimental only, not including errors in ν and null composition.

Reference

C. E. Till, et al., "Null Reactivity Measurements of Capture/Fission Ratio in ^{235}U and ^{239}Pu ", Nucl. Sci. Eng., **40**, 132 (1970)

Facility: ZPR-9 (1.3)Assembly: 24 (2.3-24)Reactions Considered:

Ratio of capture to fission (α) in Pu-239 and U-235.

Experimental Method:

The experiments were performed by the reactivity-reaction rate technique. Because of the special nature of these experiments and their complexity, they will not be reported in detail. A detailed description is provided in the reference. Related measurements, using the null-reactivity technique are reported in Section 5.1-3-24a.

This method consists basically in determining α by measuring the reactivity worth and fission rate in the sample in question. These quantities are interpreted on an absolute scale by similar measurements with a calibrated spontaneous fission source (Cf-252) and a "pure" absorber (Li-6)

Results:

The measured α values, corrected to infinite sample dilution were:

$$\alpha \text{ (U-235)} = 0.337 \pm 0.029$$

$$\alpha \text{ (Pu-239)} = 0.516 \pm 0.040$$

Reference:

M. M. Bretscher and W. C. Redman, "Low Flux Measurements of ^{239}Pu and ^{235}U Capture-to-Fission Ratios in a Fast Reactor Spectrum", Nucl. Sci. Eng., 39, 368 (1970)

SECTION 5.2 - SPATIAL VARIATION OF REACTION RATES

Facility: ZPR-3 (1.1)Assembly: 20 (2.1-20)

Reactions Considered: U-235 (n, fission) U-238 (n, fission).
U-238 (n, γ)

Experimental Method:

a) Radiochemical analysis of irradiated U-235 and depleted foils provided radial and axial traverses of relative U-238 capture and U-235 fission. See Fig. 1.

Fig. 1

U-238 Capture Rates in Clean-core
Assembly (Radiochemical Analysis
of Irradiated Foils)

b) U-235 foils (1 in. diameter, 0.005 in. thick) and U-238 foils (0.2% U-235, 1 in. diameter, 0.020 inch thick) wrapped in Al foil were placed inside the drawer fronts. The foil plane was thus perpendicular to that of the plates. Near the radial core-blanket interface the foils were placed between plates. The results are given in Fig. 2.

Fig. 2
Radial Fission Rates in Clean
Core at Midplane (Through P-Row
at Interface) (Foils)

Reference:

C. E. Branyan, "Core A Critical Studies for the Enrico Fermi Atomic Power Plant on ZPR-III", ANL-6629 (1962).

5.2-1-29

Spatial Variation of Reaction Rates

Facility: ZPR-3 (1.1)

Assembly: 29 (2.1-29)

Reactions Considered:

- (1) Fission in U-234, U-235 U-238 and Pu-239
- (2) B^{10} (n, α)

Details of Measurement:

Detectors. 2 in. long and 1/2 in. in diameter were moved remotely through an axial guide tube. The center drawers were modified to allow insertion of the tube. Some ambiguity in the effective center of the counters was reported.

Results:

See Table I

Reference:

A. L. Hess, et al., "Critical Studies of a 450-liter Uranium Oxide Fast Reactor Core (ZPR-III Assembly 29)" ANL-6336 (1961).

Table I
AXIAL TRAVERSE COUNTER DATA

Axial Position (in.)	Counts per 10 ⁴ Counts on U ²³⁵ Standard (Counter No. 5)				
	U ²³⁴	U ²³⁵	U ²³⁸	Pu ²³⁹	B ¹⁰
-30	53	1126	7		7416
-26	147	2611	10	602	12584
-22	650	9896	61	2340	40611
-18	2033	24998	265	5821	103359
-16	3558		661	9571	163386
-14	7022	62508	2080	14909	247061
-12	9930	79951	3553	19568	314063
-10	12069	94163	4455	23362	353931
-8	13927	106716	5158	26347	408142
-6	15024	116567	5817	28659	442517
-4	16312	124320	6064	30231	456018
-2	17151	127835	6251	31086	464599
-1	16994	128956	6251	31786	472504
0	16949	129722	6319	31757	486989
1	17257	129584	6433	31748	479940
2	17017	129381	6360	31614	473199
4	16277	124825	6128	30554	460802
6	15213	117952	5785	28166	440023
8	13975	107922	5230	26505	395272
10	12006	96564	4620	23232	358698
12	10364	81935	3674	20151	307669
14	7285	65697	2392	15628	244308
16	3993		578	10329	
18	2210	28592	311	6384	100090
21	1056	12500	92	3354	
Material	89.3% U ²³⁴ 7.7% U ²³⁵	93.2% Enriched U	Depleted U	95% Pu ²³⁹ 5% Pu ²⁴⁰	Enriched BF ₃
Mass	3.15 mg	5.1 mg		1.96 mg	

Facility: ZPR-3 (1.1)

Assembly: 32 (2.1-32)

Reactions Considered: fission in U-235, U-234, U-238 and Pu-239

Experimental Method:

Cylindrical fission chambers, 1/2 in. in diameter and 2 in. long are moved through a vertically oriented guide tube. The tube is recessed 1/4 in. into the 16th column of the movable half so the chambers are displaced 1/4 in. from the core center. A separate U-235 counter is used for monitoring power.

Results:

Fig. 1 Radial Fission Rate Distribution (U^{235})

Fig. 2 Radial Fission Rate Distribution (Pu^{239})

Fig. 3 Radial Fission Rate Distribution (U^{234})

Fig. 4 Radial Fission Rate Distribution (U^{238})

Reference:

P. I. Amundson, et al., "Critical Studies of Uranium-Steel and Uranium-Steel-Sodium Fast Reactor Cores (ZPR-III Assemblies 32 and 33)", ANL-6690 (1963).

Facility: ZPR-3 (1.1)Assembly: 33 (2.1-33)Reactions Considered: fission in U-235 and U-238Experimental Method:

See Section 5.2-1-32

Results:

Fig. 1
Radial Fission Rate
Distribution (U^{238})
Assembly 33

Fig. 2
Radial Fission Rate
Distribution (U^{235})
Assembly 33

Reference:

P. I. Amundson et al., "Critical Studies of Uranium-Steel and Uranium-Steel-Sodium Fast Reactor Cores (ZPR-III Assemblies 32 and 33)" ANL-6690 (1963).

5.2-1-34 Spatial Variation of Reaction Rates

Facility: ZPR-3 (1.1)

Assembly: 34 (2.1-34)

Reactions Considered: U-235 and U-238 fission

Experimental Method:

See section 5.2-1-32

Results:

Fig. 1 U²³⁵ Fission Traverse

Reference:

R. J. Huber et al., "Critical Studies of a Dilute Carbide Fast Reactor Core - ZPR-III Assembly 34" ANL-6401 (1961).

5.2-1-34 (cont'd)

Fig. 2 U^{238} Fission Traverse

5.2-1-36

Spatial Variation of Reaction Rates

Facility: ZPR-3 (1.1)

Assembly: 36 (2.1-36)

Reactions Considered: U-235 and U-238 fission

Experimental Method: See Section 5.2-1-32

Results:

Fission Counter Position (cm)	U ²³⁵ Traverse (a)		U ²³⁸ Traverse (L)	
	Total Counts	Normalized	Total Counts	Normalized
36.4	(Void/Blanket Interface)			
35.8	984 ± 1.9%	0.887 ± 0.001	43 ± 12.8%	0
45.7	2,753 ± 2.8%	0.882 ± 0.002	310 ± 5.7%	0.010 ± 0.001
35.6	6,922 ± 2.2%	0.885 ± 0.005	1,398 ± 2.7%	0.044 ± 0.001
25.4	17,187 ± 2.8%	0.530 ± 0.008	9,642 ± 1.5%	0.510 ± 0.003
24.8	(Blanket/Clad Interface)			
28.3	22,535 ± 1.9%	0.669 ± 0.013	16,364 ± 0.9%	0.579 ± 0.005
15.2	27,573 ± 1.9%	0.812 ± 0.005	34,281 ± 0.8%	0.766 ± 0.006
38.2	31,417 ± 1.9%	0.932 ± 0.008	35,484 ± 0.8%	0.896 ± 0.007
7.6	32,364 ± 1.9%	0.980 ± 0.008	29,873 ± 0.8%	0.942 ± 0.008
5.1	32,885 ± 1.9%	0.996 ± 0.008	30,719 ± 0.5%	0.969 ± 0.008
2.5	34,634 ± 1.9%	1.010 ± 0.009	31,366 ± 0.8%	0.980 ± 0.008
0	33,389 ± 1.9%	0.938 ± 0.008	31,982 ± 0.8%	1.006 ± 0.008
2.5	34,408 ± 1.9%	1.021 ± 0.009	31,284 ± 0.8%	0.984 ± 0.008
5.1	32,085 ± 1.9%	0.977 ± 0.008	30,673 ± 0.8%	0.967 ± 0.008
7.6	31,889 ± 1.9%	0.955 ± 0.008	29,343 ± 0.8%	0.925 ± 0.007
38.2	29,842 ± 1.9%	0.885 ± 0.007	28,669 ± 0.8%	0.904 ± 0.007
15.2	26,994 ± 1.9%	0.789 ± 0.005	23,341 ± 0.8%	0.739 ± 0.006

(a) 93.2 % U-235, 6 % U-234, 1 % U-238

(b) + 80 ppm U-235

Reference:

J. M. Gasidlo, J. K. Long, and R. L. McVean, "Critical Studies of a Fast Reactor Core Containing Depleted Uranium and Sodium as Diluents (ZPR-III Assembly 36)" ANL-6494 (1962).

5.2-1-39

Spatial Variation of Reaction Rates

Facility: ZPR-3 (1.1)

Assembly: 39 (2.1-39)

Reactions Considered: U-235 and U-238 fission

Experimental Method:

Scans were performed in two radial directions, one perpendicular, the other parallel to the fuel plates. For the parallel scan the plates of drawer P-16 in both halves were rearranged as shown in Fig. 1.

Fig. 1 Drawer Loading for Fission Rate Scan Parallel to Fuel-plate Planes

For the perpendicular scan the core material in the row P drawers of Half No. 2 of the reactor was moved back 1 in. and the front of the drawers loaded as shown in Fig. 2.

Fig. 2 Drawer Loading for Fission Rate Scan Perpendicular to the Fuel Plates

5.2-1-39 (cont'd)

Results:

Table I

Radial U-235 (a) Fission Distributions			
Scan Parallel to Fuel		Scan Perpendicular to Fuel	
Distance from Core Center, (in.)	Relative Fission Rate	Distance from Core Center, (in.)	Relative Fission Rate
0.09	$1.000 \pm 0.004^{(b)}$	0	$1.000 \pm 0.004^{(b)}$
-	----	5	0.947 ± 0.004
6.16	0.928 ± 0.004	-	----
-	----	7	0.890 ± 0.004
9.19	0.828 ± 0.003	9	0.833 ± 0.003
12.16	0.725 ± 0.003	12	0.716 ± 0.003
-	----	14	0.619 ± 0.002
17.16	0.495 ± 0.002	17	0.482 ± 0.002
		18	0.429 ± 0.004
		18.5	0.387 ± 0.004
		19.5	0.310 ± 0.003

(a) 93.2% U²³⁵, 6% U²³⁸, and 1% U²³⁴.

(b) The errors are based purely on statistics and represent 66% confidence limits.

Table II

Radial U-238 Fission Distributions

Distance from Core Center, (in.)	Scan Parallel to Fuel Relative Fission Rate	Scan Perpendicular to Fuel Relative Fission Rate
0	1.000 ± 0.015	1.000 ± 0.015
2	1.000 ± 0.015	0.918 ± 0.009
5	0.940 ± 0.014	0.896 ± 0.009
12	0.687 ± 0.010	0.648 ± 0.006
14	0.574 ± 0.009	0.551 ± 0.006
17	0.352 ± 0.005	0.348 ± 0.003
18	0.218 ± 0.003	0.254 ± 0.003

It was concluded that, with a realistic estimate of the experimental errors, the two distributions are in agreement.

Reference:

J. C. Bates, et al. "Two Spherical Fast Critical Experiments (ZPR-III Assemblies 38 and 39)" ANL -6582 (1962).

Facility: ZPR-3 (1.1)Assembly: 42 (2.1-42)Reactions Considered:

- (1) U^{235} (n,f), U^{234} (n,f), U^{238} (n,f), Pu^{239} (n,f),
 B^{10} (n, γ)
- (2) U^{238} (n, γ)
- (3) Na (n, γ)

Method of Measurement and Results:

(1) Fission ratios obtained from radial and axial scans made with cylindrical chambers in guide tubes. Enriched BF_3 counters used for boron results. See figure 1-10.

(2) Natural uranium foils distributed about movable half of assembly. Enriched foils at symmetric positions in fixed half. All foils sandwiched in Al and exposed for fission and capture ratios. See Table I.

(3) Sodium nitrate powder encapsulated in Al activated, and activity due to 2.85 kev resonance found using scintillation counter. U^{235} (n,f) also found at symmetrical points to yield $\sigma_a(Na)/\sigma_f(U-235)$. See figure 1-11 and Table II.

Reference:

Amundson, P. I. et al., "A Two Zone Fast Critical Experiment", ANL-6733 (1964).

5.2-1-42 (cont'd)

Fig. 1 Radial U²³⁵ Fission Distribution

Fig. 2 Radial U²³⁸ Fission Distribution

Fig. 3 Radial Pu²³⁹ Fission Distribution

Fig. 4 Radial U²³⁴ Fission Distribution

5.2-1-42 (cont'd)

Fig. 5 Axial U^{235} Fission Distribution

Fig. 6 Axial U^{238} Fission Distribution

Fig. 7 Axial Pu^{239} Fission Distribution

Fig. 8 Axial U^{234} Fission Distribution

S.2-1-42 (cont'd)

Fig. 9 Radial BF_3 Reaction Rate Distribution

Fig. 10 Axial BF_3 Reaction Rate Distribution

Fig. 11 Radial $\frac{\bar{\sigma}_a(\text{Na})}{\sigma_f(\text{U}^{235})}$ Distribution

Table I

ENRICHED AND NATURAL URANIUM FOIL IRRADIATIONS

Distance from Core Center (cm)		Fissions/g U^{238}	Captures/g U^{238}	Fissions/g U^{235}	$\frac{\bar{\sigma}_f U^{238}}{\bar{\sigma}_f U^{235}}$	$\frac{\bar{\sigma}_c U^{238}}{\bar{\sigma}_f U^{235}}$
Axial	Radial					
0	2.5	6.39×10^8	1.91×10^9	1.85×10^{10}	0.035	0.103
17.8	2.5	5.95×10^8	1.76×10^9	1.56×10^{10}	0.038	0.113
35.6	2.5	3.79×10^8	1.21×10^9	1.12×10^{10}	0.034	0.108
0.6	8.3	6.13×10^8	1.95×10^9	1.80×10^{10}	0.034	0.108
0.6	18.8	5.43×10^8	1.81×10^9	1.61×10^{10}	0.034	0.112
0.6	29.8	5.18×10^8	1.47×10^9	1.31×10^{10}	0.039	0.112

Table II

SODIUM ACTIVATION MEASUREMENTS

Sample	Location	Radius, cm	$\frac{\text{Counts/m}}{\text{g NaNO}_3}$	$\frac{\bar{\sigma}_a(\text{Na})}{\bar{\sigma}_f(U^{235})}$ (normalized to unity at core center)
1D	1 O 16	0	4313	1.0
2D	1 M 16	11.2	4311	1.057
3D	1 K 16	22.1	3934	1.044
5D	1 I 16	33.3	3285	1.10
8D	1 G 16	44.2	1018	1.06
7D	1 G 17	44.7	748	0.804

Facility: ZPR-3 (1.1)Assembly: 43 (2.1-43)Reactions Considered:

U-235 (n,f), U-238 (n,f), U-238 (n, γ), Al-27 (n, α) and Na (n, γ).

Experimental Method:

Voids (of the following sizes:) were created at various axial and radial positions:

- a) Central zone and axial blanket: 5.08 cm x 5.08 cm x 0.952 cm thick.
- b) Buffer and outer blanket: 5.08 cm wide x 1.27 cm high x 0.952 cm thick.
- c) Driver: 5.08 cm wide x 2.54 cm high x 0.952 cm thick.

The following foils and counting methods were used.

U-235 (n,f) and U-238 (n,f): Analyzed radiochemically, counting Mo-99 activity.

U-238 (n, γ): count Np-239 activity after separation. The U foils were placed in the cell to integrate over the cell fine structure.

Na-23 (n, γ): NaNO₃ powder placed in 0.952 cm diameter x 5.08 cm long Al capsules. Dimensions of NaNO₃ = 0.793 cm diameter x 4.61 cm long. Counting as for Al-27 (n, α): Al foils, 0.307 cm thick x 1.27 cm wide x 5.08 cm long, counted in a calibrated scintillation well counter.

Results:

Activation ratios at center of assembly:

$$\frac{\text{U-238 (n,f)}}{\text{U-235 (n,f)}} = 0.0224 \pm 0.0011$$

$$\frac{\text{U-238 (n,\gamma)}}{\text{U-235 (n,f)}} = 0.114 \pm 0.005$$

$$\frac{\text{Al-27 (n,\alpha)}}{\text{U-235 (n,f)}} = 3.1 \pm 0.3 \times 10^{-5}$$

$$\frac{\text{Na-24 (n,\gamma)}}{\text{U-235 (n,f)}} = 5.8 \pm 0.6 \times 10^{-4}$$

Table I

RELATIVE SODIUM, ALUMINUM, AND URANIUM ACTIVATIONS IN ASSEMBLY 43

Position (cm)	$U^{235} (\bar{\sigma}_f)$	$U^{238} (\bar{\sigma}_f)$	$U^{238} (\bar{\sigma}_{n,\gamma})$	Na ($\bar{\sigma}_{n,\gamma}$)	Al ($\bar{\sigma}_{n,\alpha}$)
Axial Measurements					
0.48	$1.00 \pm 3\%$	$1.00 \pm 4\%$	$1.00 \pm 3\%$	$1.00 \pm 3\%$	$1.00 \pm 3\%$
16.66	$0.91 \pm 4\%$	$0.88 \pm 5\%$	$0.91 \pm 4\%$	$0.94 \pm 4\%$	$0.90 \pm 4\%$
35.41	$0.63 \pm 4\%$	$0.55 \pm 5\%$	$0.66 \pm 4\%$	$0.67 \pm 4\%$	$0.59 \pm 4\%$
43.26	Core-blanket Interface				
43.97	$0.51 \pm 4\%$	$0.289 \pm 5\%$	$0.53 \pm 4\%$	$0.57 \pm 4\%$	$0.354 \pm 4\%$
53.34	$0.382 \pm 4\%$	$0.107 \pm 7\%$	$0.43 \pm 4\%$	$0.47 \pm 4\%$	$0.115 \pm 4\%$
69.70	$0.168 \pm 8\%$	$0.018 \pm 10\%$	$0.173 \pm 4\%$	$0.189 \pm 5\%$	$0.034 \pm 5\%$
Radial Measurements					
2.29	$1.00 \pm 3\%$	$1.00 \pm 4\%$	$1.00 \pm 3\%$	$1.00 \pm 3\%$	$1.00 \pm 3\%$
8.97	$1.01 \pm 4\%$	$0.82 \pm 5\%$	$1.03 \pm 4\%$	$1.09 \pm 4\%$	$0.95 \pm 4\%$
14.50	$0.99 \pm 5\%$	$0.85 \pm 5\%$	$1.09 \pm 4\%$	$1.10 \pm 4\%$	$0.98 \pm 4\%$
20.04	$0.98 \pm 5\%$	$0.88 \pm 5\%$	$1.07 \pm 4\%$	$1.22 \pm 4\%$	$1.00 \pm 4\%$
24.87	Core-buffer Interface				
28.14	$0.97 \pm 5\%$	$1.06 \pm 5\%$	$1.02 \pm 4\%$	$1.75 \pm 4\%$	$1.13 \pm 4\%$
30.40	Buffer-driver Interface				
33.66	$0.95 \pm 4\%$	$1.61 \pm 5\%$	$0.96 \pm 4\%$	$1.72 \pm 4\%$	$1.75 \pm 4\%$
39.19	$0.78 \pm 4\%$	$1.27 \pm 5\%$	$0.81 \pm 4\%$	$1.30 \pm 4\%$	$1.48 \pm 4\%$
44.25	Driver-blanket Interface				
46.00	$0.50 \pm 4\%$	$0.64 \pm 5\%$	$0.51 \pm 4\%$	$0.65 \pm 4\%$	$0.66 \pm 4\%$

Fission chamber traverses are reported in Section 5.2-1-43a.

Reference:

J. M. Gasidlo, J. K. Long and W. P. Keeney, "Doppler Effect Measurements Sodium Void Coefficients, and Critical Studies of a 5,000-liter Fast Power Breeder Reactor by the Two-Zone Method", ANL-6838 (1964)

Facility: ZPR-3 (1.1)Assembly: 43A (2.1-43)Reactions Considered:Fission in U-235, U-234, U-238, and Pu-239 and B^{10} (n, α).Experimental Method:

Fission chambers and a BF_3 proportional counter, 1.1 cm in diameter and 5 cm long were traversed axially and radially through the core.

For the radial traverses, the blanket and core drawers in half No. 2, Column 16 were modified to accept a 1.27 cm-OD, vertical thimble, approximately 0.8 cm back of the midplane. The affected columns of materials in the high-density blanket, driver, buffer and central zone drawers were simply pushed back to provide room for the thimble. The loading master for the central zone is shown in Fig. 1. Thus, as the detectors were driven down through the thimble, they passed between large parallel sheets of core material formed by the vertically loaded plates in each drawer lining up throughout the core.

Fig. 1 Central Zone Loading Master for
Radial Reaction-rate Traverses

The axial traverses were performed by driving the detectors horizontally through a thimble loaded in the 1-P-16 and 2-P-16 matrix locations (see Fig. 1.1-3). In this case, the fuel column was moved to the side of the drawer. The loading of the central zone drawer is shown in Fig. 2. The low density blanket was loaded in a similar manner.

5.2-1-43a (Cont'd)

Fig. 2 Front View of Central Zone Loading Master for Axial Traverses

In the following tables, the reaction rates are integrated (by the length of the counter) over 2.5 cm on either side of the indicated position. Results have not been corrected for the isotopic contents of the chambers which are:

Principal Isotope	Composition
U^{235}	93.2% U^{235} , 5.8% U^{234} , 1.0% U^{238}
U^{234}	89.34% U^{234} , 7.72% U^{235} , 2.89% U^{238}
U^{238}	99.99% U^{238} , 80 ppm U^{235}
Pu^{239}	94.92% Pu^{239} , 4.75% Pu^{240} , 0.32% Pu^{241} , 0.01% Pu^{242}

Table I

Normalized Radial Reaction-Rate Traverses in Assembly 43

Pos (cm)	U234	U235	U238	Pu239	BF3	Pos (cm)	U234	U235	U238	Pu239	BF3		
0	0.996 ± 1.3%	1.000 ± 0.6%	1.000 ± 0.8%	1.006 ± 0.8%	1.007 ± 0.7%	31.75	1.261 ± 1.4%	-	-	-	-		
2.54	0.997 ± 1.3%	1.000 ± 0.6%	1.000 ± 0.8%	0.992 ± 0.8%	0.996 ± 0.7%	33.02	1.336 ± 1.3%	0.995 ± 0.9%	1.025 ± 0.9%	1.006 ± 1.2%	0.911 ± 1.0%		
5.08	0.996 ± 1.3%	1.014 ± 0.9%	0.990 ± 1.0%	1.001 ± 1.2%	1.002 ± 1.0%	34.29	1.351 ± 1.3%	-	1.030 ± 0.9%	-	-		
10.16	0.991 ± 1.5%	1.000 ± 0.6%	0.976 ± 1.1%	1.001 ± 1.2%	1.006 ± 1.0%	35.56	1.336 ± 1.3%	0.991 ± 0.9%	1.007 ± 0.8%	1.000 ± 1.2%	0.861 ± 1.0%		
15.24	1.000 ± 1.5%	1.018 ± 0.9%	0.977 ± 1.1%	1.012 ± 1.2%	1.025 ± 1.0%	36.83	-	0.988 ± 0.9%	1.033 ± 0.9%	-	0.804 ± 1.0%		
17.73	-	1.031 ± 0.9%	-	-	1.040 ± 1.0%	39.12	-	0.993 ± 0.9%	1.015 ± 0.9%	0.945 ± 1.2%	0.712 ± 1.0%		
20.32	1.006 ± 1.5%	1.024 ± 0.9%	0.966 ± 1.1%	1.013 ± 1.2%	1.029 ± 1.0%	41.66	1.221 ± 1.4%	0.933 ± 0.9%	1.015 ± 0.9%	0.945 ± 1.2%	0.620 ± 1.0%		
22.30	0.991 ± 1.5%	1.040 ± 0.9%	0.997 ± 1.1%	1.034 ± 1.2%	1.034 ± 1.0%	44.20	1.074 ± 1.4%	0.727 ± 0.9%	1.018 ± 1.0%	0.806 ± 1.2%	0.527 ± 1.0%		
24.84	0.996 ± 1.5%	1.034 ± 0.9%	0.952 ± 1.1%	1.032 ± 1.2%	1.036 ± 1.0%	46.74	0.836 ± 1.6%	0.627 ± 0.9%	1.032 ± 1.0%	0.780 ± 1.3%	0.414 ± 1.0%		
27.38	1.052 ± 1.5%	1.031 ± 0.9%	1.047 ± 1.1%	1.059 ± 1.2%	1.040 ± 1.0%	50.80	0.573 ± 1.7%	0.487 ± 0.9%	0.785 ± 1.4%	0.350 ± 1.3%	0.287 ± 1.0%		
29.21	-	-	-	-	1.008 ± 1.0%	55.88	0.349 ± 2.1%	0.336 ± 1.0%	0.341 ± 2.0%	0.379 ± 1.7%	0.184 ± 1.0%		
30.43	1.227 ± 1.4%	1.017 ± 0.9%	1.305 ± 1.0%	1.062 ± 1.2%	0.980 ± 1.0%	60.96	-	0.230 ± 1.0%	0.093 ± 4.5%	-	0.116 ± 1.1%		
								0.103 ± 3.7%	0.131 ± 1.1%	0.047 ± 6.2%	0.142 ± 2.9%		
Core-buffer Interface						Buffer-driver Interface						Driver-blanket Interface	

*Core-buffer Interface

•Buffer-driver Interface

•Driver-blanket Interface

Table II

Normalized Axial Reaction-Rate Traverses in Assembly 43

Position (cm)	ρ_{234}	ρ_{235}	ρ_{236}	ρ_{237}	β_{23}	Position (cm)	ρ_{234}	ρ_{235}	ρ_{236}	ρ_{237}	β_{23}
0	1.000 ± 0.00	1.000 ± 1.75	1.000 ± 1.10	1.000 ± 1.25	1.000 ± 1.05	30.10	0.927 ± 1.95	0.905 ± 1.75	0.870 ± 1.95	0.976 ± 1.75	0.990 ± 1.05
2.34	0.978 ± 1.25	1.007 ± 1.25	0.990 ± 1.15	0.994 ± 1.25	1.000 ± 1.05	40.64	0.846 ± 1.95	0.946 ± 1.75	0.827 ± 1.95	0.937 ± 1.75	0.993 ± 1.05
5.60	0.900 ± 1.25	0.982 ± 1.75	0.990 ± 1.15	0.990 ± 1.25	1.000 ± 1.05	42.10	0.800 ± 1.75	0.915 ± 1.25	0.900 ± 1.95	0.990 ± 1.75	0.990 ± 1.05
7.40	0.873 ± 1.25	0.980 ± 1.75	0.980 ± 1.15	0.980 ± 1.25	0.972 ± 1.05	45.72	0.947 ± 1.75	0.900 ± 1.25	0.927 ± 1.05	0.943 ± 1.75	0.922 ± 1.05
17.70	0.871 ± 1.95	0.880 ± 1.75	0.880 ± 1.15	0.894 ± 1.25	0.880 ± 1.05	46.10	0.894 ± 1.75	0.854 ± 1.25	0.871 ± 1.75	0.834 ± 1.05	0.900 ± 1.05
27.04	0.732 ± 1.05	0.740 ± 1.75	0.727 ± 1.05	0.732 ± 1.25	0.736 ± 1.05	51.34	-	-	-	-	0.870 ± 1.05
33.02	0.600 ± 1.05	0.604 ± 1.75	-	0.600 ± 1.25	0.679 ± 1.05	50.42	0.166 ± 2.45	0.305 ± 1.75	0.063 ± 4.05	0.205 ± 1.05	0.353 ± 1.05
35.70	-	-	-	-	0.610 ± 1.05	60.50	0.090 ± 30.05	-	-	0.160 ± 1.75	0.202 ± 1.05

*Core-rod Blanket Interface

Reference:

J. M. Gasidlo, J. K. Long and W. P. Keeney, "Doppler Effect Measurements, Sodium Void Coefficients, and Critical Studies of a 5,000-liter Fast Power Breeder Reactor by the Two-Zone Method", ANL-6838 (1964).

5.2-1-45A

Spatial Variation of Reaction Rates

Facility: ZPR-3 (1.1)

Assembly: 45A (2.1-45A)

Reactions Considered:

Fission in U-238 and Pu-239

Experimental Method:

See Section 5.2-1-43a

Results:

The data represent count rates averaged over the 2 in. length of the counter.

Distance from Reactor Midplane, cm	Pu ²³⁹	U ²³⁸	U ²³⁸ /Pu ²³⁹
+62.2	0.174 ± 1.2%	0.044 ± 3.6%	0.251 ± 3.8%
+54.6	0.275 ± 1.0%	0.106 ± 2.4%	0.385 ± 2.6%
+49.5	0.358 ± 1.0%	0.180 ± 1.8%	0.502 ± 2.0%
+47.0	0.398 ± 1.0%	0.299 ± 1.6%	0.575 ± 1.9%
+45.8	Axial blanket/core interface		
+44.4	0.452 ± 0.9%	0.326 ± 1.4%	0.722 ± 1.6%
+41.9	0.488 ± 0.9%	0.409 ± 1.5%	0.838 ± 1.7%
+39.4	0.542 ± 0.9%	0.486 ± 1.4%	0.898 ± 1.6%
+34.3	0.640 ± 0.8%	0.610 ± 1.3%	0.953 ± 1.5%
+29.2	0.735 ± 0.8%	0.727 ± 1.2%	0.988 ± 1.4%
+24.1	0.828 ± 0.8%	0.825 ± 1.1%	0.996 ± 1.4%
+19.0	0.903 ± 0.8%	0.893 ± 1.1%	0.989 ± 1.3%
+14.0	0.942 ± 0.8%	0.942 ± 1.0%	1.000 ± 1.3%
+8.9	0.981 ± 0.8%	1.006 ± 1.0%	1.025 ± 1.3%
+3.8	1.000 ± 0.8%	1.009 ± 1.0%	1.008 ± 1.3%
+1.3	0.994 ± 0.8%	0.990 ± 1.0%	0.995 ± 1.3%
0	Reactor midplane		
-1.3	1.008 ± 0.8%	0.996 ± 1.0%	0.989 ± 1.3%
-3.8	0.987 ± 0.8%	1.000 ± 1.0%	1.014 ± 1.3%
-6.3	0.981 ± 0.8%	0.991 ± 1.0%	1.010 ± 1.3%
-11.4	0.984 ± 0.8%	0.941 ± 1.0%	0.956 ± 1.3%
-16.5	0.924 ± 0.8%	0.921 ± 1.0%	0.997 ± 1.3%
-21.6	0.856 ± 0.8%	0.844 ± 1.1%	0.987 ± 1.3%
-26.7	0.774 ± 0.8%	0.765 ± 1.1%	0.989 ± 1.4%

Reference:

Reactor Development Program Progress Report January 1965,
ANL-7003.

5.2-1-48

Spatial Variation of Reaction Rates

Facility: ZPR-3 (1.1)

Assembly: 48 (2.1-48)

Reactions Considered:

Fission in U-235, U-238 and Pu-239 and B-10 (n,α)

Experimental Method:

Fission chambers and a BF₃ proportional counter, all 0.375 in. in diameter and 2 in. long, were traversed through the assembly in a radial direction at 1.25 in. from the core midplane. The counters are described in Table I.

Table I

Traverse Counter Specifications

Counter Type	Material Analysis	Material Loading
Enriched U Fission	93.2 w/o ²³⁵ U 5.5 w/o ²³⁸ U 1.3 w/o ²³⁴ U + ²³⁶ U	0.94 mg ^a
Depleted U Fission	80 ppm ²³⁵ U	2.8 mg ^a
Plutonium Fission	94.9 w/o ²³⁹ Pu 4.8 w/o ²⁴⁰ Pu + ²⁴² Pu 0.3 w/o ²⁴¹ Pu	2.04 mg ^a
¹⁰ B (n,α)	Approx. 90 atom % ¹⁰ B in B of BF ₃	40 cm Hg filling of BF ₃

a. Annular plating on inside of cylinder, annulus is 3.21 cm long x 0.84 cm diam.

b. Cylindrical chamber, inside dimensions of 0.95 cm diam. x 5.0 cm length.

5.2-1-48 (cont'd)

A horizontal channel was created at the point of the P-row matrix tubes as shown in Figure 1.

Figure 1
Side View of Front 3 in. of
Drawers Containing Traverse Tube

Results:

The results are given in Table II. The slight shift of the peak in the negative direction (the direction of smaller drawer numbers) is explained by the effect of the asymmetric fuel loading. See also Section 5.2-1-48.

Reference:

A. L. Hess and Q. L. Baird, "Reaction Rate Distributions in ZPR-3 Assemblies 48, 48A, and 48B", ZPR-TM-6 (1969).

Table II. Radial Traverses of Neutron Detectors Through Assembly 48

Radial Distance from Center, cm	Relative Reaction Rate, Normalized to Core Center			
	Plutonium Fission Counter	Enriched Uranium Fission Counter	Depleted Uranium Fission Counter	$^{10}\text{B}^1\text{F}_3$ Proportional Counter
-60.96	4.67 $\pm$ 0.06		0.68 $\pm$ 0.02	4.32 $\pm$ 0.03
-55.88	8.16 $\pm$ 0.08		1.66 $\pm$ 0.03	7.32 $\pm$ 0.04
-50.80	14.0 $\pm$ 0.1		3.98 $\pm$ 0.05	12.4 $\pm$ 0.1
-45.72	23.3 $\pm$ 0.2	21.6 $\pm$ 0.2	10.6 $\pm$ 0.1	21.5 $\pm$ 0.2
-43.18	30.8 $\pm$ 0.2		18.3 $\pm$ 0.1	28.2 $\pm$ 0.2
-41.91	35.0 $\pm$ 0.2		23.7 $\pm$ 0.2	32.1 $\pm$ 0.2
-40.64	38.6 $\pm$ 0.2	36.8 $\pm$ 0.2	28.8 $\pm$ 0.2	35.7 $\pm$ 0.3
-38.10	44.9 $\pm$ 0.2	43.0 $\pm$ 0.2	38.3 $\pm$ 0.3	43.1 $\pm$ 0.3
-35.56	51.2 $\pm$ 0.2	49.5 $\pm$ 0.2	44.4 $\pm$ 0.3	49.9 $\pm$ 0.4
-30.48	64.1 $\pm$ 0.3	62.4 $\pm$ 0.3	59.3 $\pm$ 0.4	62.2 $\pm$ 0.4
-25.40	74.7 $\pm$ 0.3	73.0 $\pm$ 0.3	71.2 $\pm$ 0.4	74.1 $\pm$ 0.5
-20.32	83.2 $\pm$ 0.4	81.6 $\pm$ 0.4	82.0 $\pm$ 0.4	82.2 $\pm$ 0.6
-15.24	92.2 $\pm$ 0.4	89.7 $\pm$ 0.4	89.7 $\pm$ 0.4	90.1 $\pm$ 0.6
-10.16	96.6 $\pm$ 0.4	94.8 $\pm$ 0.4		95.9 $\pm$ 0.7
- 5.08	100.4 $\pm$ 0.4	98.1 $\pm$ 0.4	98.4 $\pm$ 0.5	98.1 $\pm$ 0.7
- 2.54	101.3 $\pm$ 0.4	98.9 $\pm$ 0.4	96.9 $\pm$ 0.5	100.2 $\pm$ 0.7
0	100.0 $\pm$ 0.4	100.0 $\pm$ 0.4	100.0 $\pm$ 0.5	100.0 $\pm$ 0.7
2.54	100.5 $\pm$ 0.4	99.3 $\pm$ 0.4	97.5 $\pm$ 0.5	99.5 $\pm$ 0.7
5.08	99.6 $\pm$ 0.4	98.4 $\pm$ 0.4	98.1 $\pm$ 0.5	97.6 $\pm$ 0.7
10.16	95.8 $\pm$ 0.4	94.3 $\pm$ 0.4	94.4 $\pm$ 0.5	95.3 $\pm$ 0.7
15.24	90.0 $\pm$ 0.4	89.6 $\pm$ 0.4	88.9 $\pm$ 0.4	89.3 $\pm$ 0.6
20.32	82.4 $\pm$ 0.4	82.4 $\pm$ 0.4	80.4 $\pm$ 0.4	81.9 $\pm$ 0.6
25.40	73.1 $\pm$ 0.3	73.8 $\pm$ 0.3	69.1 $\pm$ 0.4	72.2 $\pm$ 0.5
30.48	62.4 $\pm$ 0.3	62.7 $\pm$ 0.3	58.0 $\pm$ 0.4	61.9 $\pm$ 0.4
35.56	49.7 $\pm$ 0.2	49.9 $\pm$ 0.3	43.2 $\pm$ 0.3	49.4 $\pm$ 0.4
38.10	43.7 $\pm$ 0.2	43.8 $\pm$ 0.2	36.6 $\pm$ 0.3	42.8 $\pm$ 0.3
40.64	36.8 $\pm$ 0.2	37.0 $\pm$ 0.2	27.6 $\pm$ 0.2	36.2 $\pm$ 0.3
41.91	33.7 $\pm$ 0.2	33.9 $\pm$ 0.2	22.2 $\pm$ 0.2	32.2 $\pm$ 0.2
43.18	29.6 $\pm$ 0.2	29.3 $\pm$ 0.2	17.0 $\pm$ 0.1	27.0 $\pm$ 0.2
45.72	22.4 $\pm$ 0.2	22.7 $\pm$ 0.2	9.9 $\pm$ 0.1	20.9 $\pm$ 0.2
50.80	13.4 $\pm$ 0.1	13.4 $\pm$ 0.1	3.78 $\pm$ 0.05	12.5 $\pm$ 0.1
55.88	7.9 $\pm$ 0.1	7.8 $\pm$ 0.1	1.56 $\pm$ 0.02	7.5 $\pm$ 0.1
60.96	4.60 $\pm$ 0.06	4.64 $\pm$ 0.06	0.64 $\pm$ 0.02	4.46 $\pm$ 0.03
66.04	2.63 $\pm$ 0.04	2.72 $\pm$ 0.04	0.285 $\pm$ 0.008	2.64 $\pm$ 0.02
68.58	2.00 $\pm$ 0.03			2.04 $\pm$ 0.02
71.12	1.55 $\pm$ 0.03	1.42 $\pm$ 0.03	0.112 $\pm$ 0.004	1.67 $\pm$ 0.02
73.36	1.26 $\pm$ 0.03			1.70 $\pm$ 0.02

5.2-2-3

Spatial Variation of Reaction Rates

Facility: ZPR-6 (1.2)

Assembly: 3 (2.2-3)

Reactions Considered:

Fission in U-235 and U-238

Experimental Method:

See Section 5.2-2-4Z

Results:

Fig. 1 Radial Traverse

Fig. 2 Axial Traverse

Related Experiments:

1. Axial and radial fission rate traverses were done with low density axial blanket zones in place. See Ref. 1 pp. 108-113.
2. A diametric fission rate traverse was taken through the center of a sodium voided sector. See Ref. 1, pp. 111

Reference:

F. H. Helm, et al., "Physics Parameters and Sodium Void Coefficients in a 950-Liter Pancake Core", Reactor Physics Division Annual Report- July 1, 1964 to June 30, 1965, ANL-7110 (1965)

Facility: ZPR-6 (1.2) Assembly: 42 (2.2-42)

Reactions Considered:

Fission in U-234, U-235, U-236, and U-238.

Experimental Method:

A. Axial and radial traverses were performed with the detector shown in Figure 1. The position was regulated with a stepping motor attached to a lead screw.

The axial traverse was performed in the central drawer as shown in Figure 2. The radial traverse was performed 1-3/4 in. from the midplane in the central column of drawers. See Figure 3 for drawer arrangement in central region and Figure 4 for driver region.

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Fig. 1 Small Surface-Barrier Solid-state Counter Used for Fission-ratio Traverses

Fig. 2 Axial Traverse Drawer

IN VOIDED CENTRAL REGION BROOM CAMS WERE REPLACED WITH 1/4 IN. PHONES

C DEPLETED U 1/4 IN. NO CAM PUL.

5.2-2-4Z (cont'd)

B. Axial and radial traverses with U-235 and U-238 foils were also made. All foils were 0.0125 cm thick with diameters of 1.27 cm and 0.95 cm for U-238, and U-235 respectively. Foils were placed vertically between adjacent plates in the core region and horizontally on top of U-238 blocks in the reflector.

Axial traverses were made at the core center, the boundary of the central region, and the middle of the driver. Radial traverses were made 7.62 cm from the interface and in vertical, horizontal, and diagonal (45°) directions.

A-ALUMINUM TRAVERSE TUBE 3/16 O.D. $\pm$ 0.020 WALL

IN VOIDED CORE ALL NO CANS WERE REPLACED WITH 1/4 IN. FRAMES

Fig. 3 Radial (vertical) Traverse Drawer

A-ALUMINUM TRAVERSE TUBE 3/16 O.D. $\pm$ 0.020 WALL

IN VOIDED CORE ALL NO CANS WERE REPLACED WITH 1/4 IN. FRAMES

Fig. 4 Arrangement of Driver Drawer to Allow Space for Fission Counter

Precision:

In part A, positions were found to be reproducible to within 0.002 cm.

Results:

Table 1
Fission Ratios as Function of Axial Distance

Axial Position, cm from Center $\pm$ 0	Measured Fission Ratios			
	U^{234}/U^{235}	U^{236}/U^{235}	Pu^{239}/U^{235}	U^{238}/U^{235}
0.0	0.107 ± 0.005	0.0256 ± 0.0008	0.901 ± 0.004	0.136 ± 0.006
10.0	0.179 ± 0.005	0.0061 ± 0.0008	0.876 ± 0.023	0.146 ± 0.006
20.0	0.178 ± 0.005	0.0063 ± 0.0008	0.854 ± 0.023	0.147 ± 0.006
30.0	0.173 ± 0.005	0.0066 ± 0.0009	0.875 ± 0.025	0.154 ± 0.006
40.0	0.168 ± 0.005	0.0062 ± 0.0009	0.863 ± 0.026	0.155 ± 0.006
50.0	0.165 ± 0.005	0.0238 ± 0.0009	0.858 ± 0.026	0.142 ± 0.006
56.3	0.160 ± 0.005	0.0251 ± 0.0008	0.857 ± 0.027	0.144 ± 0.005

5.2-2-4Z (cont'd)

Table 2
Fission Ratios as a Function of Radial Distance

Radial Position, cm from Center (Z = 4.345 cm)	Measured Fission Ratios				Calculated Fission Ratios ^a			
	U^{234}/U^{235}	U^{238}/U^{235}	Pu^{239}/U^{235}	U^{238}/U^{234}	U^{234}/U^{235}	U^{238}/U^{235}	Pu^{239}/U^{235}	U^{238}/U^{234}
0.0	0.172 ± 0.004	0.0238 ± 0.0007	0.967 ± 0.025	0.138 ± 0.004	0.222 (0.221)	0.0267 (0.0266) ^b	0.943 (0.941)	0.120 (0.120)
15.0	0.168 ± 0.004	0.0230 ± 0.0008	0.964 ± 0.025	0.137 ± 0.004	0.222 (0.021)	0.0266 (0.0265)	0.942 (0.941)	0.120 (0.120)
30.0	0.163 ± 0.004	0.0232 ± 0.0008	0.956 ± 0.025	0.142 ± 0.005	0.220 (0.219)	0.0263 (0.0261)	0.940 (0.939)	0.119 (0.119)
40.5	0.174 ± 0.004	0.0228 ± 0.0008	0.934 ± 0.025	0.131 ± 0.005	0.223 (0.213)	0.0262 (0.0249)	0.930 (0.933)	0.108 (0.116)
50.0	0.241 ± 0.007	0.0377 ± 0.0012	1.020 ± 0.027	0.156 ± 0.008	0.324 (0.314)	0.0491 (0.0439)	1.052 (1.040)	0.151 (0.139)
55.0	-	0.0395 ± 0.0015	1.047 ± 0.030	-	-	0.0418 (0.0397)	1.062 (1.050)	-

^aValues in parentheses were obtained by diffusion theory with Set 226; the other values were obtained with the same cross-section set by transport S_4 calculations.

^bThis ratio was also obtained by diffusion theory with Set 801.

See Figures 5-8

Figure 5
Axial Traverse of U-235 Fission Rates in ZPR-6 Assembly 42

5.2-2-4Z (cont'd)

Figure 6
Axial Traverse of U-238 Fission Rates in ZPR-6 Assembly 42

Figure 7
Radial Traverse of U-235 Fission Rates in ZPR-6 Assembly 42

5.2-2-4Z (cont'd)

Figure 8
Radial Traverse of U-238 Fission Rates in ZPR-6 Assembly 4Z

Reference:

R. A. Karam, et al., "Measured Physics Parameters in a Zoned Fast Uranium Carbide Core, Assembly 4Z", ANL-7291 (1966)

Facility: ZPR-6 (1.2)Assembly: 5 (2.2-5)Reactions Considered:

Fission in U-235 and U-238 and capture in U-238

Experimental Method:

Three experimental techniques were used:

1) Foils: Enriched (93%) and depleted (0.2%) uranium foils were irradiated in the central drawer: once with sodium "in" and once with sodium "out" from a central region of 25 cm radius and extending the axial length of the core. Relative fission product activities (see section 5.2-2-6) were determined by scintillation counting, and coincidence counting was used to determine the relative captures in U-238 foils. Absolute determination of fission and captures was by radiochemical means described in 10.1-2-5.

2) Solid State Track Recorders (SSTR) (2) (3): Enriched (93% U-235 in the form of a thin deposit of U_3O_8 on platinum) and depleted (0.2% U-235, 0.125 mm metal) foils were placed in contact with mica. The mica was cleaved to ~0.1 mm and foil-etched. The foil sandwich is shown in Fig. 1.

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Fig. 1. Geometrical configuration of the SSTR and fission sources.

For the axial traverses the foil sandwiches were placed between sodium and depleted uranium plates in the core drawers and on top of the depleted uranium blocks of the reflector. In the radial case, the foils were placed 10 cm from the interface gap.

After exposure the mica was etched in HF and the track density was determined by visual scanning (2).

3) Fission Chambers: Small fission chambers were traversed axially through the core. Some sodium plates were removed to accommodate the chamber.

Results:

The results of the foil measurements are shown in Table 1.

Table 1
Distribution of Fission Rates and U-238
Capture - effect of Na

Z, cm	Foil No.	Absolute Fissions/g U-235		Absolute Captures/g U-238		Absolute Fissions/g U-235	
		with Na, 10 ⁹	w/o Na, 10 ⁹	with Na, 10 ⁹	w/o Na, 10 ⁹	with Na, 10 ¹¹	w/o Na, 10 ¹¹
0.00	1	3.06	3.09	1.92	1.63	1.480	1.208
5.08	2	3.04	3.07	1.92	1.61	1.474	1.207
10.16	3	3.04	3.09	1.89	1.59	1.477	1.201
15.24	4	3.57	3.61	1.84	1.57	1.448	1.271
20.32	5	3.46	3.41	1.89	1.51	1.399	1.228
25.40	6	3.37	3.31	1.74	1.45	1.337	1.190
30.48	7	3.09	3.13	1.66	1.40	1.290	1.130
35.56	8	3.06	2.99	1.61	1.31	1.298	1.072
40.64	9	2.72	2.76	1.47	1.28	1.122	0.994
45.72	10	2.79	2.51	1.38	1.14	1.045	0.912
50.80	11	2.22	2.26	1.21	1.08	0.931	0.838
55.88	12	1.92	1.94	1.07	0.91	0.909	0.783
60.96	13	1.65	1.62	0.98	0.81	0.710	—
66.04	14	1.29	—	0.76	—	—	—

The results of the SSFR measurements are compared with fission counters (for the axial case) or foils (for the radial case) in Figures 2 and 3.

Fig. 2 A comparison of the SSTR and fission counter methods for an axial traverse of ZPR-6, Assembly 5, with (a) ^{235}U and (b) ^{238}U .

Fig. 3 A comparison of the SSTR and radiochemical methods for a radial traverse of ZPR-6, Assembly 5, with (a) ^{235}U and (b) ^{238}U .

5.2-2-5 (Cont'd)

The disagreement between the two methods for the axial traverse has been attributed to the removal of Na in the fission chamber case (3).

References:

1. R. A. Karon, et al., "Measured Physics Parameters in a Large Uranium Carbide Fast Core, Assembly 5 ZPR-6" Reactor Physics Division Annual Report - July 1, 1966 to July 30, 1967, ANL-7310 p. 167-189 (1968)
2. R. Gold, R. J. Armani and J. H. Roberts, "Absolute Fission Rate Measurements with Solid-State Track Recorders", Nucl. Sci. Eng. 34, 13 (1968).
3. J. H. Roberts, et al., "Fission Rate Measurements in Low-Power Fast Critical Assemblies with Solid-State Track Recorders" Nucl. Appl. 5 (4), 247-252 (1968).

Facility: ZPR-6 (1.2)

Assembly: 6 (2.2-6)

Reactions Considered:

Fission in U-235 and U-238, capture in U-238.

Experimental Methods:

Axial and radial reaction rates were measured with foils. U-235 foils were 0.0045 in. thick by 0.375 in. in diameter and were 93% enriched. U-238 foils were 0.005 in. thick by 0.625 in. in diameter and were 0.2% enriched. U-235 foils were in the movable half while U-238 foils were in the stationary half at corresponding locations. For radial traverses the foils were placed in every drawer, between the 1/16 in. U-235 plate and the 1/8 in. U-238 plate at a distance of 2 in. from the mid-section of the core. Along the axial direction the foils were placed between the same plates in the central drawers of the two halves. In the blanket the foils were placed on top of the 2 x 2 x 5 in. depleted uranium blocks.

Five hours after irradiation, the activities of the U-238 foils were counted for fission fragments (counting gammas with energies above 0.41 MeV) and for capture by counting the 106 keV gammas from excited Pu-239 in coincidence with x-rays produced by internal K conversion electrons. The five-hour waiting period was necessary to allow sufficient time for the buildup of Pu-239 from U-238. The U-235 foils were counted for fission fragments in a manner similar to that for counting the U-238 foils. A few foils were radiochemically analyzed for the absolute number of fissions in U-235 and absolute number of fissions and captures in U-238. The radiochemical results were used to obtain efficiency factors to convert the rest of the data to absolute numbers.

Results:

Fig. 1 Radial U-235 Fission Distribution

Fig. 2 Axial U-235 Fission Distribution

5.2-2-6 (cont'd)

Fig. 3 Radial U-238 Fission Distribution

Fig. 4 Axial U-238 Fission Distribution

Fig. 5 Radial U-238 Capture Distribution

Fig. 6 Axial U-238 Capture Distribution

References:

1. R. A. Karam, et al., "A 4000-Liter Uranium Oxide Fast Core, Assembly 6 of ZPR-6", Reactor Physics Division Annual Report - July 1, 1967 to June 30, 1968, AML-7410 (1969)
2. R. A. Karam, et al., "Heterogeneity Effects in a Large UO_2 Core Assembly 6 of ZPR-6", Reactor Physics Division Annual Report - July 1, 1967 to June 30, 1968, AML-7410 (1969)

5.2-3-1

Spatial Variation of Reaction Rates

Facility: ZPR-9 (1.3) **Assembly:** 1 (2.3-1)

Reactions Considered:

Fission in U-235 and U-238

Experimental Method:

Measure fission product activation of enriched and depleted foils.

Results:

see Figure 1

Reference:

R. C. Doerner, W. G. Knapp, K.K. Almenas, and R. A. Karam,
"Physics Measurements in Tungsten-Based, Aluminum-Reflected
Fast Reactors," ANL-7007 (1967)

5.2-3-1 (cont'd)

Figure 1
U-238 and U-235 Space-dependent Fission Rates and
Their Ratios in Assembly No. 1

5.2-3-2

Spatial Variation of Reaction Rates

Facility: ZPR-9 (1.3)

Assembly: 2 (2.3-2)

Reactions Considered:

Fission in U-235 and U-238

Experimental Method:

Measure fission product activation of enriched and depleted foils.

Results:

See Figure 1.

Reference:

R. C. Doerner, W. G. Knapp, K. K. Almenas, and R. A. Karam, "Physics Measurements in Tungsten-Based Aluminum-Reflected Fast Reactors," AML-7007 (1967)

5.2-3-2 (cont'd)

Figure 1
U-238 and U-235 Space-dependent Fission Rates and
Their Ratios in Assembly No. 2

5.2-3-3

Spatial Variation of Reaction Rates

Facility: ZPR-9 (1.3)

Assembly: 3 (2.3-3)

Reactions Considered:

Fission in U-235, U-238, Pu-237 and Np-237.
Capture in B-10.

Experimental Method:

Fission foils of U-235 and U-238 were measured.
Also, miniature solid-state detectors were used for fission distributions of U-235, U-238, Pu-239 and Np-237. A miniature BF₃ counter was used for capture rates in boron.

Results:

See Figures 1-4.

Reference:

R. C. Doerner, W. G. Knapp, K. K. Almenas, and R. A. Karam, "Physics Measurements in Tungsten-Based, Aluminum-Reflected Fast Reactors", ANL-7007 (March 1967)

5.2-3-3 (cont'd)

Figure 1

U-238 and U-235 Space-dependent Fission Rates and Their Ratios in Assembly No. 3

5.2-3-3 (cont'd)

Figure 2
Pu-239 Space-dependent Fission Rate in Assembly No. 3

Figure 3
Pu-237 Space-dependent Fission Rate in Assembly No. 3

Figure 4
B-10 (n, α) Space-dependent Fission Rate in Assembly No. 3

5.2-3-4

Spatial Variation of Reaction Rates

Facility: ZPR-9 (1.3)

Assembly: 4 (2.3-4)

Reactions Considered:

Fission in U-235, U-238, U-234

Capture in B-10

Experimental Method:

Fission foils of U-235 and U-238 were measured. Also, fission rate of U-234 was measured with miniature solid-state detector. A miniature BF₃ counter was used for capture rate in boron.

Results:

See Figures 1-3

Reference:

R. C. Doerner, W. G. Knapp, K. K. Almenas, and R. A. Karam, "Physics Measurements in Tungsten-Based Aluminum-Reflected Fast Reactors", AML-7(07) (March 1967)

5.2-3-4 (cont'd)

Figure 1

U-238 and U-235 Space-dependent Fission Rates and Their Ratios in Assembly No. 4

5.2-3-4 (cont'd)

Figure 2

U-234 Space-dependent Fission Rate in Assembly No. 4

Figure 3

B-10 (n,α) Space-dependent Fission Rate in Assembly No. 4

5.2-3-4B

Spatial Variation of Reaction Rates

Facility: ZPR-9 (1.3) Assembly: 4B (2.3-4B)

Reactions Considered:

Fission in U-235 and U-238.

Capture in B-10

Experimental Method :

Same as for (5.2-3-6), but in addition a small Lionel Anton BF_3 counter was traversed to measure the boron reaction rate.

Results :

Figure 1
Radial U-235 Fission Rate in ZPR-9 Assembly No. 4B

Reference:

W. G. Knapp and R. C. Doerner, "Physics Measurements with Modified Diluent Compositions in Tungsten-Based Aluminum Reflected Fast Reactors", ANL-7207 (Feb. 1967)

5.2-3-4B (cont'd)

Figure 2
Radial U-238 Fission Rate in ZPR-9 Assembly No. 4B

Figure 3
Radial U-238/U-235 Fission Ratios
in ZPR-9 Assembly No. 4B

5.2-3-4B (cont'd)

Figure 4
Radial B-10 Capture Rate in ZPR-9 Assembly No. 4B

Figure 5
Axial U-235 and U-238 Fission Rates and
U-238/U-235 Fission Ratios in ZPR-9 Assembly No. 4B

5.2-3-4B (cont'd)

Reference:

W. G. Knapp and R. C. Doerner, "Physics Measurements with Modified Diluent Compositions in Tungsten-Based Aluminum Reflected Fast Reactors" ANL-7207 (1967)

5.2-3-5

Spatial Variation of Reaction Rates

Facility: ZPR-9 (1.3)

Assembly: 5 (2.3-5)

Reactions Considered:

Fission in U-235 and U-238

Capture in B-10 (n, α)

Experimental Method:

Same as for 5.2-3-6, but in addition a small Liconel-Anton BF₃ counter was traversed to measure the boron reaction rate.

Results:

See Figures 1-5.

Figure 1

Radial U-235 Fission Rate in ZPR-9 Assembly No. 5

5.2-3-5 (cont'd)

Figure 2

Radial U-238 Fission Rate in ZPR-9 Assembly No. 5

Figure 3

Radial U-238/U-235 Fission Ratios in ZPR-9 Assembly No. 5

5.2-3-5 (cont'd)

Figure 4
Radial B-10 Capture Rate in ZPR-9 Assembly No. 5

Figure 5
Axial U-235 and U-238 Fission Rates and
U-238/U-235 Fission Ratios in ZPR-9 Assembly No. 5

5.2-3-5 (cont'd)

Reference:

W. G. Knapp and R. C. Doerner, "Physics Measurements with Modified Diluent Compositions in Tungsten-Based, Aluminum-Reflected Fast Reactors", ANL-7207 (Feb. 1967).

5.2-3-6

Spatial Variation of Reaction Rates

Facility: EPR-9 (1.3)

Assembly: 6 (2.1-6)

Reactions Considered:

Fission in U-235 and U-238.

Experimental Method:

U-235 and U-238 foils and a solid-state detector-converter foil combination were used to measure the reaction rate of fissionable nuclides at the reactor center plane. In addition foils were used to measure the axial fission rate for U-235 and U-238.

Results:

See Figure 1-4.

Figure 1

Radial U-235 Fission Rate in EPR-9 Assembly No. 6

5.2-3-6 (cont'd)

Figure 2
Radial U-238 Fission Rate in EPR-9 Assembly No. 5

Figure 3
Radial U-238/U-235 Fission Ratios in EPR-9 Assembly No. 6

5.2-3-6 (cont'd)

Figure 4
Axial U-235 and U-238 Fission Rates and
U-238/U-235 Fission Ratios in SFR-9 Assembly No. 6

Reference:

W. G. Knapp and R. C. Doerner, "Physics Measurements with Modified Diluent Composition in Tungsten-Based Aluminum-Reflected Fast Reactor" ANL-7207 (Feb. 1967).

5.2-3-7

Spatial Variation of Reaction Rates

Facility: ZPR-9 (1.3)

Assembly: 7 (2.3-7)

Reactions Considered:

Fission in U-238, U-235, U-233, U-234, U-236, Np-237 and Pu-239.

Capture (n, α) in B-10.

Experimental Method:

Foils of U-235 and U-238 were 5 mils thick with cross-sectional areas of from $\frac{1}{4}$ to 10 cm². The foils were arranged on a horizontal line in the core midplane which passed through the core center.

For all other fission rate measurements, small solid-state counters were traversed through a vertical, 0.290 cm diameter, voided Al tube at the core mid-plane.

The B-10 capture distribution was measured with a miniature BF₃ counter.

Results:

See Figures 1-9.

Figure 1

U-238 Reaction-rate Traverse in Assembly No. 7

5.2-3-7 (cont'd)

Figure 2

U-235 Reaction-rate T versus in Assembly No. 7

Figure 3

U-238/U-235 Fission Ratio in Assembly No. 7

5.2-3-7 (cont'd)

Figure 4

U-233 Reaction-rate Traverse in Assembly No. 7

Figure 5

U-234 Reaction-rate Traverse in Assembly No. 7

5.2-3-7 (cont'd)

Figure 6
U-236 Reaction-rate Traverse in Assembly No. 7

Figure 7
Pu-237 Reaction-rate Traverse in Assembly No. 7

5.2-3-7 (cont'd)

Figure 8

Pu-239 Reaction-rate Traverse in Assembly No. 7

Figure 9

B-10 Reaction-rate Traverse in Assembly No. 7

Reference:

R. C. Doerner, W. G. Knapp, R. A. Karam, and D. K. Butler,
 "EPX-9 Assemblies No. 6-9 Critical Experiments" ANL-7208
 (March 1967)

Facility: ZPR-9 (1.3)Assembly: 8 (2.3-8)Reactions Considered:

Fission in U-235 and U-238

Capture (n, α) in B-10Experimental Method:

Same as for ZPR-9, Assembly 7 (See Section 5.2-3-7).

An additional measurement made was a microscopic U-235 fission rate traverse through a fuel plate at the core edge. The fission rate was measured through the outer fuel plate of the core by replacing a $\frac{1}{2}$ in-sq. section of fuel plate by a $\frac{1}{2}$ in-sq. stack of twelve 0.005-in-thick U-235 foils wrapped in aluminum foil. Two such measurements were made. In the first, shown in Fig. 5, there was a $\frac{1}{8}$ -in-thick column of 45% density aluminum plus a $\frac{1}{8}$ -in-thick column of Al_2O_3 between the last fuel plate and the BeO. In the second measurement, shown in Fig. 6, tungsten replaced the Al_2O_3 column.

Results:

Figure 1

U-238 Reaction-rate Traverse in Assembly No. 8

5.2-3-8 (cont'd)

Figure 2
U-235 Reaction-rate Traverse in Assembly No. 8

Figure 3
B-10 Capture Rate in Assembly No. 8

5.2-3-8 (cont'd)

Figure 4
U-238/U-235 Fission Ratio in Assembly No. 8

Figure 5
Microscopic U-235 Reaction-rate Traverse without Tungsten

5.2-3-8 (cont'd)

Figure 6
Microscopic U-235 Reaction-rate Traverse with Tungsten

Reference:

R. C. Doerner, W. G. Knapp, R. A. Karam, and D. K. Butler,
"ZPR-9 Assemblies No. 6-9 Critical Experiments", ANL-7208
(March 1967)

Facility: ZPR-9 (1.3)

Assembly: 9 (2.3-9)

Reactions Considered:

Fission in U-238, U-235, U-233, U-234, U-236,
Np-237 and Pu-239.

Capture in B-10.

Experimental Method:

Same as for ZPR-9 Assembly 7. See Section 5.2-3-7.

Results:

See Figures 1-8.

Figure 1

U-238 Reaction-rate Traverse in Assembly No. 9

5.2-3-9 (cont'd)

Figure 2
U-235 Reaction-rate Traverse in Assembly No. 9

Figure 3
U-233 Reaction-rate Traverse in Assembly No. 9

5.2-3-9 (cont'd)

Figure 4
 U-234 Reaction-rate Traverse in Assembly No. 9

Figure 5
 U-236 Reaction-rate Traverse in Assembly No. 9

5.2-3-9 (cont'd)

Figure 6
Np-237 Reaction-rate Traverse in Assembly No. 9

Figure 7
Pu-239 Reaction-rate Traverse in Assembly No. 9

5.2-3-9 (cont'd)

Figure 8
B-10 Reaction-rate Traverse in Assembly No. 9

Reference:

R. C. Doerner, W. G. Knapp, R. A. Karam, and D. K. Butler,
"ZPR-9 Assemblies No. 6-9 Critical Experiments", ANL-7208

5.2-3-11

Spatial Variation of Reaction Rates

Facility: ZPR-9 (1.3)

Assembly: 11 (2.3-11)

Reactions Considered:

Fission in U-238

Experimental Method:

Depleted uranium foils, 1.27 cm diameter, 0.0125 cm thick were placed vertically between plates.

Results:

Fig. 1. Measured and calculated radial fission-rate distributions in ^{238}U foils.

Reference:

A. Ancarani, "Fission Rates in a Zoned-Core Fast Critical Assembly", Trans Am. Nucl. Soc. 10, 1, 275, (1967)

Facility: ZPR-9 (1.3)Assembly: 12 (2.3-12)Reactions Considered:

U-235 and U-238 fission, B capture

Experimental Method:

Holes 1.125 in. in diameter were drilled through the plates of the core and matrix material on the core radius (on the vertical center-line). A double-walled stainless steel tube, with total wall thickness of 0.020 in. was inserted into the hole and acted as the sample tube. The same tube was used for the reactivity traverses (4.2-3-19).

The fission and BF_3 chambers were positioned in the tube with a 1/8 in. diameter stainless steel rod. The fission chambers were $\frac{1}{2}$ in. in diameter and $\frac{3}{8}$ in. long (active volume). The fissionable material was deposited in an area of $\frac{1}{4}$ in. in diameter on a $\frac{1}{2}$ in. diameter stainless steel disk. The BF_3 counter was $\frac{1}{4}$ in. in diameter and $\frac{7}{8}$ in. long and contained enriched B^{10}F_3 gas at a pressure of 40 cm Hg.

Results:

Figure 1 Measured and Calculated Ratios of U-238 Fission Counter to BF_3 Counter Versus Radial Position for ZPR-9 Assembly 12 (Normalized to 1.0 at $r = 0$)

5.2-3-12 (cont'd)

Figure 2
Measured and Calculated Ratios of U-238 Fission
Counter to U-235 Fission Counters Versus Radial
Position for ZPR-9 Assembly 12
(Normalized to 1.0 at $r = 0$)

Figure 3
Measured and Calculated Ratios of U-235 Fission
Counter to BF_3 Counter Versus Radial Position for
ZPR-9, Assembly 12 (Normalized to 1.0 at $r = 0$)

Reference:

L. G. LeSage et al., "A Review of Data Pertaining to Spectral
Equilibration and Spectral Matching in Zoned Cores" ANL-
Reactor Physics Division Annual Report, July 1, 1966 to June 30
1967, ANL-7310 p. 199 (1968)

Facility: ZPR-9 (1.3)Assembly: 19 (2.3-19)Reactions Considered:

U-235 and U-238 fission and capture in B-10.

Experimental Method:

The counters and techniques are described in Section 5.2-3-12.

Traverses were also made using enriched (93%) foils 0.0044 in. thick and 0.322 in. in diameter. The foils were placed in a horizontal position on top of the drawers that form the horizontal diameter of the movable half about 1/2 in. from the interface.

Table I shows the data from the chambers ⁽¹⁾ and in Figure 1 the U-235 foils and chamber data are compared ⁽²⁾.

Table I
Normalized Count Rates for U-235 and U-238
Fission Counters and a BF₃ Counter^a

Radial Position, cm	Assembly 19			Assembly 21			Assembly 22		
	U-235	U-238	BF ₃	U-235	U-238	BF ₃	U-235	U-238	BF ₃
0	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000
2.76	0.993	0.993	—	0.993	1.006	0.987	1.008	1.030	1.038
5.52	0.977	1.004	1.015	0.980	0.988	1.005	0.996	0.972	0.977
8.28	—	—	—	1.005	1.001	0.980	1.005	1.005	0.987
11.06	1.007	1.601	1.005	0.995	1.009	1.009	0.971	0.975	0.973
13.80	—	—	—	0.999	0.990	1.006	0.982	1.003	1.001
16.56	0.994	1.000	1.017	0.996	1.043	0.982	0.978	0.959	1.033
19.32	—	—	—	0.971	—	0.938	0.970	0.962	1.063
22.10	0.992	0.995	1.012	0.932	1.415	0.886	0.978	1.006	1.059
24.84	1.021	0.984	1.029	0.899	—	0.892	0.968	0.995	1.006
27.60	0.994	1.052	1.007	0.880	1.153	0.940	0.980	1.085	1.006
30.36	0.966	1.237	—	0.912	—	1.172	0.969	1.238	0.973
33.12	0.932	1.380	0.915	0.965	0.616	1.333	0.920	1.378	0.922
35.88	0.882	—	—	—	—	—	0.850	1.322	0.895
38.63	0.858	1.137	0.883	0.979	0.301	1.670	0.853	1.147	0.893
41.39	0.824	—	—	—	—	—	0.815	—	0.938
44.16	0.794	0.578	0.943	0.805	0.158	1.396	0.768	0.589	0.961
46.97	0.617	0.257	0.763	0.545	—	0.865	0.606	0.233	0.780
55.19	0.339	0.121	0.373	0.253	—	0.347	0.317	—	0.411
60.71	0.178	0.052	—	0.122	—	0.149	0.173	—	0.256
66.23	0.101	—	—	—	—	—	—	—	—

^a All counts were normalized to 1.0 at assembly center. Statistical uncertainties were less than 1% for the U-235 and BF₃ counters and about 1.5% for the U-238 counter.

Figure 1
Radial U-235 Fission Rate Measurement ZPR-9, Assembly 19

References:

1. L. G. LeSage, E. F. Groh, and W. R. Robinson, "An Investigation of the Properties of Zoned Systems", ANL Reactor Physics Division Annual Report July 1, 1967 to June 30, 1968, ANL-7410 p. 145 (1969)
2. L. G. LeSage and W. R. Robinson, "ZPR-9 Assemblies 19, 21, 22 and 24", ANL Reactor Physics Division Annual Report July 1, 1967 to June 30, 1968, ANL-7410 p. 80 (1969)

5.2-3-21

Spatial Variation of Reaction Rates

Facility: ZPR-9 (1.3)

Assembly: 21 (2.3-21)

Reactions Considered:

U-235 and U-238 fission, B (n, α)

Experimental Method:

See Section 5.2-3-12 and 5.2-3-19.

Results:

The U-235 chamber and foil data are compared in Figure 1. Other results are given in Table 1 of section 5.2-3-19.

Fig. 1 Radial U-235 Fission Rate Measurements
ZPR-9 Assembly 21

Reference:

L. G. LeSage and W. R. Robinson, "ZPR-9 Assemblies 19, 21, 22 and 24" ANL Reactor Physics Division Annual Report July 1, 1967 to June 30, 1968, ANL-7410 p. 80 (1969).

5.2-3-22 Spatial Variation of Reaction Rates

Facility: ZPR-9

Assembly: 22 (2.3-22)

Reactions Considered:

U-235 and U-238 fission, B-10 (n, α)

Experimental Method:

see Section 5.2-3-19

Results:

The U-235 chamber and foil data are compared in Fig. 1. Other results are given in Table 1 of section 5.2-3-19.

Figure 1

Radial U-235 Fission Rate Measurement ZPR-9, Assembly 22

Reference:

L. G. Le sage and W. R. Robinson, "ZPR-9 Assemblies 19, 21, 22, and 24," ANL Reactor Physics Division Annual Report July 1, 1967 June 30, 1968, ANL-7410 p. 80 (1969).

SECTION 6 - SODIUM VOID COEFFICIENTS

Sodium void coefficients are of major importance with regard to the safety of LMFBR's. The coefficients depend both on the position of the Na within the cell and the position of the cell in the assembly. Most measurements reported in this section were made as a function of position in the assembly. Although some experiments related to dependence on position within the cell are included here, specific experiments dealing with the effects of heterogeneity on Na void coefficients are compiled in Section 10.3.

Facility: ZPR-3 (1.1)Assembly: 43 (2.1-43)Experimental Method:

A) Central worth:

Each of the five 6-in. long sodium cans in the front of the central zone drawer (see Section 2.1-43) was replaced with one 4-in- and one 2-in-long can for the reference. Then the 4-in sodium-filled cans were replaced with identical empty cans on both sides of the reactor midplane for the central measurements. The experiment was performed for the two opposing central drawers 10 central drawers (N16, 015, 016, 017, and P16 on each half) and 18 central drawers (the 3 x 2 central array on each half).

B) Axial worth:

Empty cans were substituted for Na-filled cans for the full length of the core and axial blanket. Substitutions were performed for two sodium-void fractions. The loading master for the 60% void measurement is shown in Fig. 1. The 100% void fraction measurements were similar except that all of the sodium-filled cans were replaced with empty cans.

Fig. 1

Loading Master for Full-length Axial
Sodium Void Measurements with 60% Void Fraction

Results:

	Sodium Mass Removed (kg)	Reactivity Change (Ih $\pm$ 0.5) ^a	Worth of Sodium (Ih/kg)
<u>A. Central Measurements</u>			
<u>100% of Sodium Removed</u>			
2 Central Drawers	0.244	+0.3	-1.3 $\pm$ 2
10 Central Drawers	1.22	+1.6	-1.3 $\pm$ 0.4
18 Central Drawers	2.20	+3.3	-1.5 $\pm$ 0.3
<u>B. Axial Measurements</u>			
<u>60% of Sodium Removed</u>			
2 Central Drawers	1.203	-1.1	+0.9 $\pm$ 0.4
10 Central Drawers	6.014	-6.46	+1.09 $\pm$ 0.08
18 Central Drawers	10.825	-11.43	+1.06 $\pm$ 0.05
<u>100% of Sodium Removed</u>			
2 Central Drawers	2.005	-2.15	+1.1 $\pm$ 0.3
10 Central Drawers	10.02	-10.6	+1.06 $\pm$ 0.05

^aA variation of ± 0.5 Ih is assigned to all reactivity measurements due to uncertainty in the closure of the halves.

Reference:

J. M. Gasidlo, J. K. Long and W. P. Keeney, "Doppler Effect Measurements, Sodium Void Coefficients, and Critical Studies of a 5,000-liter Fast Power Breeder Reactor by the Two-Zone Method" ANL-6838 (1964).

6-1-48

Sodium Void Coefficient

Facility: ZPR-3 (1.1)Assembly: 48 (2.1-48)Experimental Method:

Sodium cans were removed from normal, unbunched, and bunched drawers at the center and edges of the core: they were replaced by an amount of steel equivalent to the can. The reactivity was determined with a calibrated control rod. The drawer loadings and positions of measurement are shown in section 10.2-1-48. In all cases the measurements were made in both halves of the reactor.

Results:

No. of Drawers	Fuel Configuration	Core Location	Drawer Designation	Sodium Removed (kg)	Reactivity Change (1h/kg)
<u>Sodium-void Measurements^(a)</u>					
18	Normal	Center	A	1.32	7.9 ± 0.8
24	Normal	Center	B	1.76	3.2 ± 0.6
42	Normal	Center	A and B	3.08	5.2 ± 0.3
42	Bunched	Center	A and B	3.08	3.1 ± 0.3
42	Unbunched	Center	A and B	3.08	5.7 ± 0.3
42	Standard	Center	A and B	3.08	5.6 ± 0.3
42	Normal	Axial Edge	A and B	3.08	-20.7 ± 0.3
42	Bunched	Axial Edge	A and B	3.08	-23.6 ± 0.3
42	Standard	Axial Edge	A and B	3.08	-20.5 ± 0.3
42	Bunched (15 in.)	Axial Edge	A and B	3.08	-23.5 ± 0.3
12	Standard	Radial Edge	C	0.88	-17.6 ± 1.1
10	Standard	Radial Edge	D	0.73	-19.6 ± 1.4
22	Standard	Radial Edge	C and D	1.61	-18.5 ± 0.6

(a) 1% $\Delta k/k = 1002.5$ 1hReference:

J. M. Gasidlo, et al., "Doppler-Effect, Heterogeneity, and Sodium-Voiding Experiments in ZPR-3 Assemblies 48 and 48B" ZPR-TM-8 (1969)

Facility: ZPR-6 (1.2)Assembly: 2 (2.2-2)Experimental Method:

1. Establish critical rod position for normal core configuration.
2. Replace cans containing Na by stainless steel frames and find critical rod position.
3. Determine worth of stainless to correct for slight difference (~10%) between cans and frames.

Precision:

Results found to be reproducible to between 0.5 and 1.0 Ih.

Results:

Note: $1\% \Delta k/k = 451 \text{ Ih.}$

I Coefficients of Central Regions

<u>Region*</u>	<u>Coefficient</u> <u>(Ih/kg of Na)</u>
A	-1.6
B	-2.07
C	-2.99
E**	-4.89

* See Figure 1

**Same as Region E in Figure 2 except that the Na-voided region is 91.4 cm long and centrally located on the axis.

Figure 1
Voided Regions

II Coefficients of Axial Cylindrical Regions

<u>Region*</u>	<u>Coefficient</u> <u>(I_h/kg of Na)</u>
D	-6.34
E	-4.89
F	-4.5
G I	-1.9
G II	-4.0
G III	-7.5

* See Figure 2

6-2-2 (cont'd)

☒ CONTROL ROD
□ SAFETY ROD

HEAVILY OUTLINED AREAS ARE No-VOID
REPLACEMENT SECTORS
DIMENSIONS ARE IN INCHES (CENTIMETERS)

Figure 2

III Coefficients of Rings

Region*	Coefficients (Ih/kg of Na)	Total Coefficient (Ih)
H I	-4.4	-18.29
H II		-17.54
H III		-22.54
I I	-5.96	-35.77
I II		-33.13
I III		-30.97
J I	-7.26	-62.64
J II		-45.82
J III		-29.25

* See Figure 3

Figure 3

Other Experiments Performed:

1. Sodium void coefficients were measured for a central region diluted with stainless steel, niobium, or sodium. See reference 2, page 105.

2. Sodium void coefficients were measured for modified drawer loadings. Alternate drawers in a central region were filled with sodium while the intermediary drawers contained fuel, depleted uranium and graphite. See Reference 2, page 108.

References:

1. W. Y. Kato, G. K. Rusch, and F. H. Helm, "Sodium-void Coefficients", Proc. Conf. on Safety, Fuels, and Core Design in Large Fast Power Reactors, ANL-7120 pp. 598-602 (1966)
2. G. K. Rusch, et al., "Investigations of a 600-Liter Uranium Carbide Core (ZPR-VI Assembly No. 2)", Reactor Physics Division Annual Report - July 1, 1963 - June 30, 1964, ANL-7010, pp. 91-110 (Jan. 1965)

6-2-3 Sodium Void Coefficients

Facility: ZPR-6 (1.2) Assembly: 3 (2.2-3)

Experimental Method:

As for Assembly 2 (2.2-2)

Precision:

Results found to be reproducible within about 0.5 Δk with isolated discrepancies up to 2.5 Δk .

Results:

Note: 1% $\Delta k/k$ = 451 Δk .

I. Coefficients of Central Regions

<u>Region*</u>	<u>Coefficient ($\Delta k/k$ of Na)</u>
A	0
B	-1.31
C	-2.29
D	-3.23
E	-3.54
F	-3.48

*See Figure 1

II Coefficients of Axial Cylindrical Regions

<u>Region*</u>	<u>Coefficient ($\Delta k/k$ of Na)</u>
G	-0.18
H	-2.33
I	-5.0
K	-5.8

*See Figure 1

III Coefficients of Rings

<u>Region*</u>	<u>Coefficient (Ih/kg of Na)</u>
L	-1.06
M	-2.16
N	-2.25
O	-2.13

* See Figure 1

IV Coefficient for a Voided Sector

See Figure 1, Region P

Coefficient (Ih/kg of Na) = -2.83

Other Experiments:

Coefficients were measured for several axial, cylindrical regions with zones of the axial blanket replaced by a low density blanket. The low density blanket composition was the same as for the core except that U-235 was replaced by U-238. See Reference, page 108-109.

Reference:

F. H. Helm, et al., "Physics Parameters and Sodium Void Coefficients in a 950-Liter Pancake Core", Reactor Physics Division Annual Report, July 1, 1964 to June 30, 1965, ANL-7110, pp. 101-113 (1965)

6-2-3 (cont'd)

Figure 1 Regions for Sodium Void Measurements

6-2-4Z

Sodium Void Coefficients

Facility: ZPR-6 (1.2)

Assembly: 4Z (2.2-4Z)

Experimental Method:

Axial regions were voided, each region consisting of two equal zones located symmetrically with respect to the midplane of the assembly. Voiding was accomplished by removal of sodium from the nine central drawers in both halves of the assembly. The regions are defined as:

<u>Region*</u>	<u>Inner Boundary of Voided Region, cm</u>	<u>Outer Boundary of Voided Region, cm</u>
A	0	5
B	0	10
C	10	20
D	20	30.5
E	30.5	40.6
F	40.6	50.8
G	50.8	61.0

*The boundaries designated for these regions are representative for both halves of the assembly.

The drawer configuration was changed from normal to that shown in Figure 1, and measurements were made using the same regions indicated above. The worth of the sodium was determined by a calibrated control rod.

Figure 1
Material Arrangement in the Three Basic Drawers
for the Sodium-void Coefficient

6-2-4Z (cont'd)

The sodium void coefficient was also measured as a function of radius

Precision:

Rod position was reproducible to within 0.002 cm. The uncertainty based on reproducibility of the measurements was 0.1-0.2 Ih.

Results:

Table I

Sodium-void Coefficient
in Axial Direction,
Sodium Next to U-238
(Normal Loading)

<u>Region</u>	<u>Ih/kg</u>
A	+1.07
B	+0.96
C	+0.87
D	0.00
E	-0.75
F	-1.69
G	-2.02
BCDEFG	-

Table II

Sodium-Void Coefficient
in Axial Direction,
Sodium Next to Graphite

<u>Region</u>	<u>Ih/kg</u>
A	+0.79
B	+0.62
C	+0.40
D	-0.32
E	-0.97
F	-1.63
G	-2.25
BCDEFG	-

6-2-4Z (cont'd)

Table III
Radial Sodium-void Coefficient

Matrix Position* of Sample Removed	Weight of Sample, g	Effective Radius, cm	Measurements, I_h^b	
			Worth of Sample	Worth/kg
22-22, 23-22, 24-22 22-24, 23-24, 24-24	1281.40	8.30	+1.16 $\pm$ 0.20	+0.905
22-19, 23-19, 24-19 22-27, 23-27, 24-27	1248.80	24.20	+1.710 $\pm$ 0.20	+0.568
22-16, 23-16, 24-16 22-30, 23-30, 24-30	1259.0	40.76	+1.66 $\pm$ 0.30	+1.320
22-14, 23-14, 24-14 22-32, 23-32, 24-32	1227.0	48.82	+1.60 $\pm$ 0.30	+1.30
22-13, 23-13, 24-13 22-33, 23-33, 24-33	1227.0	54.34	-5.20 $\pm$ 0.30	-4.24

* See Fig. 1 of 2.2-4Z

Reference:

R. A. Karam, et al., "Measured Physics Parameters in a Zoned Fast Uranium Carbide Core, Assembly 4Z", ANL-7281 (Dec. 1966)

Facility: ZPR-6 (1.2)Assembly: 5 (2.2-5)Experimental Method:

The sodium void coefficient was measured as a function of axial and radial position. For the axial measurements, nine central drawers per half were used. See Figure 1 for the axial locations of the regions voided.

Figure 1

Sections Where Sodium was Removed at Center
of Assembly 5 of ZPR-6

The radial measurements were made by voiding sodium from the front 8 in (20.32 cm) of these drawers in each half of the assembly at five different radial positions. The centers of the drawers were located along a horizontal radius.

In addition to the above experiment, done in a normal core loading, sodium void measurements were done in three other core loadings. (See Figures 1 and 2 of 10.1-2-5). In each case the front 4 in., 8 in., and entire axial length of each half of the assembly was voided for the central 18 drawers.

Additional experiments were done for the loadings shown in Figures 3, 4 and 5. Sodium was voided from the front 8 in. and from the entire axial length of the central 18 drawers (9 in each half of the assembly.)

6-2-5 (cont'd)

Fig. 2 Position Effects of Bunched U-238 Loading Patterns on Na-Void Coefficient

Fig. 3 Uniform Loading with $\frac{1}{4}$ in. Sodium Cans in Three-Drawer Cell

6-2-5 (cont'd)

Fig. 4 Bunching of Na in Three-Drawer Cell

Fig. 5 Loading Pattern to Study Environmental Effects on the Na-Void Coefficient

Results:**Table I Sodium-Void Coefficient As a Function of the Axial Direction**

Section Voided ^a	Measured Worth, lh	Na Weight, kg	Specific Worth ^b , lh/kg
A	1.02	0.886	1.15
B	1.98	1.772	1.12
C	+1.90	1.955	0.97
D	+0.85	1.955	0.44
E	-0.40	1.955	-0.20
F	-2.00	1.955	-1.02
G	-2.64	1.955	-1.35
H	-3.00	1.955	-1.54
B + C	+4.69	3.727	+1.26
B + C + D + E + F + G + H	-3.95	13.502	-0.293

Table II Sodium-Void Coefficient As a Function of the Radial Direction

Section Voided (drawers from reactor center)	Na Weight, kg	Specific Worth ^a , lh/kg
0	1.242	+1.10
0	1.242	+1.08
4	2.585	+0.65
4	2.585	+0.64
7	2.585	-0.15
10	2.585	-0.79
10	2.585	-0.83
10	2.585	-0.77
14	2.585	-0.75
14	2.585	-0.77

^a 1% $\Delta k/k$ = 476 lh.^a See Fig. III-6-19 for explanation of sections.^b 1% $\Delta k/k$ = 476 lh.**Table III Sodium-Void Coefficient Dependence on Loading Pattern**

Section Voided, kg	Loading Pattern	Reactivity Effect ^a of Sample, lh
Front 4 in. (1.772)	Normal Loading ^b	+1.98 ± 0.20
(1.955)	SuCEdCDSDCD CEDSDSDCEdD CDSdCDCEdSD	
(1.955)	Ref. Case (Fig. 1) ^d	+1.51 ± 0.10
(1.955)	DCSEdDCDCD DCESDCEdCD DCSEdDCDCD	
(1.955)	Bunched Case (Fig. 2) ^d	-0.017 ± 0.30
(1.955)	CCSEdDDDDC CCESSDDDED CCEdDDDDC	
	Position effect of bunching (Fig. 2)	+0.10 ± 0.30
	CCSEdDDDD CCESSDDDED CCEdDDDD	
Front 8 in. (3.727)	Normal Loading	+4.11 ± 0.20 ^a
(3.91)	Ref. Case (Fig. 1)	+2.70 ± 0.24
(3.91)	Bunched Case (Fig. 2)	+0.13 ± 0.20
28 in. (13.50)	Normal Loading	-3.95 ± 0.20
(13.62)	Ref. Case (Fig. 1)	-10.17 ± 0.02
(13.62)	Bunched Case (Fig. 2)	-16.97 ± 0.03
(13.62)	Position effect of bunching (Fig. 2)	-17.11 ± 0.55

^a 1% $\Delta k/k$ = 476 lh.^b Key: C = $\frac{1}{8}$ in. thick graphite.S = $\frac{1}{2}$ in. thick Na can.d = $\frac{1}{16}$ in. thick depleted uranium (0.2% enrichment).D = $\frac{1}{8}$ in. thick depleted uranium (0.2% enrichment).E = $\frac{1}{16}$ in. thick enriched uranium (93% enrichment).^a This number is the sum of two adjacent 4 in. sections.

d. Figure refers to Section 10.1-2-5.

6-2-5 (cont'd)

Table IV
Sodium-Void Coefficient in Different Environments

Configuration	Comments	Front 8 in. void		Entire 56 in. void	
		lb/kg	lb	lb/kg	lb
I Normal Loading	Sodium contiguous to depleted uranium	+1.08	+4.01 ±0.2	-0.29	-3.95 ±0.3
II Uniform loading with 1/4 in. Na cans (Fig. 3)	Sodium contiguous to depleted uranium and to carbon	+0.97	3.35 ±0.3	—	—
III 1 in. column of sodium loaded in center of drawer (Fig. 4)	Void full 1 in. width	+0.70	2.33 ±0.3	-0.70	-9.52 ±0.03
	Void two 1/4 in. outside columns	+0.65	+1.10 ±0.3	-0.49	-2.90 ±0.30
IV Drawer loadings arranged to bunch carbon on one side of drawer and depleted uranium on the other side of the drawer (Fig. 5)	All sodium voided in region indicated	+0.65	+2.18 ±0.3	—	—
	Sodium bounded by carbon voided	-0.23	-0.48 ±0.3	-1.02	-6.01 ±0.3
	Sodium bounded by depleted uranium voided	+1.60	+2.67 ±0.3	+0.14	+0.82 ±0.3

Related Experiments:

See Sections 5.2-2-5, 10.1-2-5, and 10.5-2-5.

Reference:

R. A. Karam, et al., "Measured Physics Parameters in a Large Uranium Carbide Fast Core, Assembly 5 ZPR-6", Reactor Physics Division Annual Report July 1, 1966 to June 30, 1967, ANL-7310, pp. 167-189 (1968)

6-2-6 Sodium Void Coefficient

Facility: ZPR-6 (1.2) Assembly: 6 (2.2-6)

Experimental Method:

The sodium void coefficient was mapped axially and radially for the normal loading pattern (See Figure 1). Axial worths were determined for 4-in. long sections of sodium in each half of the assembly. (See Fig. 2). At each position the nine central drawers were voided.

ALL DIMENSIONS IN INCHES

FIG. -2. Sections Where Sodium Was Removed at Center of Assembly 6.

6-2-6 (cont'd)

Radial measurements were made by voiding the entire amount of sodium from the specific drawers in the radial direction. The sodium was removed from four drawers, two to the left and two to the right of the center of the core, in each measurement. At the core center only two drawers were voided. In each case the opposing drawers in each half were voided.

Precision:

The uncertainties of the coefficients are generally ± 0.1 Ih.

Results:

Note: $1\% \Delta k/k = 458$ Ih.

TABLE I. SODIUM-VOID COEFFICIENT AS A FUNCTION OF AXIAL HEIGHT IN NORMAL LOADING OF ASSEMBLY 6 OF ZPR-6

Sections Voided	ΔZ , ^(a) cm	Sodium Weight, kg	Sodium-Void Coefficient ^b	
			Normal Loading	
			Ih/section	Ih/kg Na
A	0.0-5.08	0.8906	-0.082	-0.093
B	5.08-15.2	1.955	-0.088	-0.045
C	15.2-25.4	1.955	-0.415	-0.212
D	25.4-35.6	1.955	-0.825	-0.422
E	35.6-45.7	1.955	-1.643	-0.840
F	45.7-55.9	1.955	-2.477	-1.267
G	55.9-66.0	1.955	-2.941	-1.504
H	66.0-76.2	1.955	-3.203	-1.639
All sections (measured)	0.0-76.2	14.401	-13.464	-0.861
All sections (summed)	0.0-76.2	14.401	-11.271	-0.783

^a ΔZ is the length of each section in each half.

^b $1\% \Delta k/k = 458$ Ih.

TABLE II. SODIUM-VOID COEFFICIENT AS A FUNCTION OF RADIUS IN NORMAL LOADING PATTERN OF ASSEMBLY 6 OF ZPR-6

Drawers Voided	Equivalent Radius, cm	Sodium Removed, kg	Sodium-Void Coefficient ^b	
			Ih/section	Ih/kg Na
S/M 2323 ^a	0	1.604	-1.610	-1.004
S/M 2321 + 2325	11.0	3.248	-2.980	-0.918
S/M 2319 + 2327	22.1	3.248	-3.078	-0.948
S/M 2316 + 2330	38.7	3.248	-2.681	-0.825
S/M 2313 + 2333	55.3	3.267	-2.700	-0.826
S/M 2310 + 2336	71.9	3.254	-2.449	-0.753
S/M 2307 + 2339	88.5	2.767	-1.532	-0.554

^a The letters S/M represent the stationary/movable halves of ZPR-6 and the numbers 2323, for example, designate the drawers occupying matrix position 2323, the location of the center of the core.

^b $1\% \Delta k/k = 458$ Ih.

Related Experiments:

Refer to Section 10.3-2-6 for central sodium void coefficient measurements for plate, rodde and homogeneous samples.

6-2-6 (cont'd)

Reference:

R. A. Karam, J. E. Marshall, L. R. Dates, and G. K. Rusch,
"Sodium Void Effects in a Large UO₂ Core, Assembly 6 of
ZPR-6", Reactor Physics Division Annual Report - July 1, 1967
to June 30, 1968, ANL-7410 (1969)

Facility: ZPR-6 (1.2)Assembly: 6 (2.2-6)Reactivity Coefficient: Sodium void versus UO_2 enrichment.Experimental Method:

The sodium void coefficient was also measured by the oscillation technique with nominally 2 in³ rodde samples as a function of the enrichments of UO_2 . (see Section 10.3-2-6a). The following enrichments of 1/4 in. diam. by 1/2 in. long pellets were used: 0.22, 11.94, 15.86, and 19.95%.

Results:

Table I

Sodium-Void Coefficients in Rodded Samples
As a Function of UO_2 Enrichment

Sample Description	Sodium Weight g	Coefficient, Th^a	Specific Coefficient, Th/kg Na
0.22% enriched pellet type	40.748	+0.0182±0.0024	+0.448±0.059
11.94% enriched pellet type	40.803	+0.0052±0.0021	+0.128±0.052
15.86% enriched pellet type	40.712	0.0000±0.0020	0.000±0.050
19.95% enriched pellet type	40.909	-0.0027±0.0026	-0.066±0.064

^a1% $\Delta k/k = 450 \text{ Th}$.

Reference:

R. A. Karam, et al, "Sodium Void Effects in a Large UO_2 Core Assembly 6 of ZPR-6", Reactor Physics Division Annual Report, July 1, 1967 to June 30, 1968, ANL 7410, 125 (1969).

SECTION 7 - DOPPLER COEFFICIENTS

Measurements of the Doppler Coefficient are among the most complex made in fast critical assemblies. While the descriptions provided in this section are adequate for a survey of the measurements, the references should be consulted for more complete descriptions.

7-1-43

Doppler Coefficients

Material Studied: U-238 in UO_2 Pellets

Facility: ZPR-3 (1.1)

Assembly: 43 (2.1-43)

Experimental Method:

Samples were inserted into two heating elements specially designed to constrain axial expansion of the pellets and to minimize heat loss. See Figure 1. Each element was fastened to a drive mechanism, and placed in empty matrix channels above and below the reactor center line. See Figure 2.

The positions of one hot and one cold sample were periodically interchanged from core center (or flux symmetric positions, designated 1 and 2, about core center) to outside the assembly. The cold sample was out when the hot was in, and vice-versa. The reactivity worth of each exchange was determined by measuring the difference in the critical position of a fine control rod which had an uncertainty in reproducibility of less than ± 0.002 Ih; a number of exchanges were made at each temperature.

To investigate the effects of flux depression, measurements were made with minimized flux depression near the Doppler element by moving all sodium to the side of drawers nearest the element, as shown in Figures 3 and 4. For maximized flux depression, all sodium was moved away, and fuel was placed nearest the element, as shown in Figures 3 and 5.

To investigate all non-Doppler effects caused by heating the Doppler element, the element was surrounded by boron (see Figures 6 and 7) to perturb the flux and importance of neutrons in the energy region of a few kilovolts, which should reduce the Doppler effect. Non-Doppler effects should be largely unperturbed.

Thermocouples placed along the sample axis monitored temperatures, the average of these being used to determine the temperature of the heated sample.

A precision of approximately 5% was quoted for the results.

7-1-43 (cont'd)

Figure 1 Design of Doppler Element

7-1-43 (cont'd)

Figure 2
Location of Doppler Elements during an Experiment

Figure 3
Front View of Modified Loading Masters for Central Zone

7-1-43 (cont'd)

Figure 4
Schematic Representation of Minimized Resonance-depression
Effect Experiment

Figure 5
Schematic Representation of Maximized Resonance-depression
Effect Experiment

Figure 6
Front View of Loading Masters for Boron Test Experiment

Figure 7
Schematic Representation of Boron Test Experiment

7-1-43 (cont'd)

Results

Table 1

<u>Configuration</u>	<u>Measured Reactivity Change (Inhours)</u>
1. <u>Minimized Resonance Flux Depression*</u>	
1 Hot vs 2 Cold	-0.251
1 Cold vs 2 Cold	<u>+0.151</u>
Difference	-0.402
2 Hot vs 1 Cold	-0.549
2 Cold vs 1 Cold	<u>-0.151</u>
Difference	-0.398
$-0.400 \text{ Ih} = 0.90 \times 10^{-5} \Delta k/k = 0.462 \times 10^{-5} \Delta k/k/kg \text{ U-238}$	
2. <u>Maximized Resonance Flux Depression**</u>	
2 Hot vs 1 Cold (net)	-0.382
3. <u>Boron Test ***</u>	
1 Hot vs 2 Cold (net)	-0.077

* See Figures 3 and 4

** See Figures 3 and 5

*** See Figures 6 and 7

7-1-43 (Cont'd)

Reference:

J. M. Gasidlo, J. K. Long and W. P. Keeney, "Doppler Effect Measurements, Sodium Void Coefficients, and Critical Studies of a 5000-Liter Fast Power Breeder Reactor by the Two-Zone Method" p. 32, ANL-6838 (1964).

7-1-43A

Doppler Coefficients

Material Studied: U-238 in UO_2 Pellets

Facility: ZPR-3 (1.1)

Assembly: 43A (2.1-43A)

Experimental Method:

Identical to that described in Section 7-1-43.

After the measurements on Assembly 43, the central zone and buffer regions were modified to provide a softer spectrum (See Section 2.1-43A). These experiments were performed with minimized resonance depression effects as shown in Figure 4 of Section 7-1-43 and Figure 1.

Figure 1
Front View of Loading Masters for Measurements in Assembly 43A

Additional experiments were performed to evaluate the size and importance of possible non-Doppler reactivity effects.

Results:

Table 1

<u>Configuration</u>		<u>Measured Reactivity Change (Inhours)</u>
1. <u>High-temperature Measurements</u>		
1 Hot vs 2 Cold ($T = 800^{\circ}\text{K}$)		-0.593
1 Cold vs 2 Cold		+0.090
	Difference	-0.683
2 Hot vs 1 Cold		-0.765
2 Cold vs 1 Cold		-0.088
	Difference	-0.677
2. <u>Reduced-temperature Measurements</u>		
2 Hot vs 1 Cold ($T = 553^{\circ}\text{K}$)		-0.437
2 Cold vs 1 Cold		-0.089
	Difference	-0.348
3. <u>Measurements of Possible Non-Doppler Effects</u>		
Axial Expansion of U^{238}		+0.0086 lh/mm
Element at Core Center		+0.003 lh/mm
Element at Out Position		-0.005 lh/mm
Dummy Element at Core Center		+0.002 lh/mm

* Positional errors

Reference:

J. M. Gasidlo, J. K. Long, and W. P. Keeney, "Doppler Effect Measurements, Sodium Void Coefficients, and Critical Studies of a 5000-Liter Fast Power Breeder Reactor by the Two-Zone Method", p. 32, ANL-6838 (1964)

7-1-45

Doppler Coefficients

Materials Studied: U-238 in UO_2 pellets
Pu-239 in PuO_2 pellets

Facility: ZPR-3 (1.1)

Assembly: 45 and 45A
(2.1-45, 2.1-45A)

Experimental Method:

Samples were inserted into two heating elements mounted in line, then fastened to a drive mechanism, and placed in an empty matrix channel. The two elements were positioned symmetrically about the axial center of the facility. See Figure 1.

Figure 1

Schematic Diagram Showing Arrangement of Doppler Element Drive System in ZPR-3 (Proportions have been distorted).

One end of each sample was fixed so both expanded in the same direction upon heating, resulting in a first-order cancellation of flux gradient expansion effects. See Figure 2.

7-1-45 (cont'd)

Notes

1. The inner surface of the outer can is silvered to prevent radiation losses. The space between outer can and sample can is evacuated. Cooling air is circulated outside the outer can.
2. The heater is an insulated resistance wire spirally wrapped around the sample can. The pitch of the spiral is varied to compensate for heat losses from the end.
3. There are three thermocouples located on the center line, and one on the sample can of each sample.

Figure 2 Doppler Elements

7-1-45 (cont'd)

The elements were periodically interchanged with a reference material from the central core position to a position outside the assembly. The reactivity worth of each exchange was determined by measuring the difference in the critical position of a fine control rod. A number of exchanges were made per run at each temperature.

Temperatures were determined by thermocouples.

Sources of error were reactor drift (mostly because of the decay heat of plutonium fuel) and uncertainties in expansion effects. An experimental uncertainty of $\pm 0.40 \times 10^{-6} \Delta k/k$ was assigned to all results.

Results:

Table I
Doppler Effect Reactivity Changes for U-238 and U-238 Mixed
with Pu-239 in 7:1 Ratio
($\Delta T = 500$ to $1100^\circ K$)

Measurements ^a	Assembly	Experiment			Theory ($\Delta k/k \times 10^{-6}$)		
		$\Delta\rho, \text{lh}$ ($\pm \sim 0.020$)	$\Delta k/k \text{ per } \text{hg}$ $\times 10^{-6}$	$\Delta k/k \times 10^{-6}$	Columbia Data	Virk, et al. data	Exact calculation
U-238 1 in. D.E.	45	-0.327	-27.7	-6.30	-6.19	—	—
U-238 1.0 in. D.E.	45	-1.36	-23.3	-26.8	-26.11	-23.8	-26.4
Dummy U-238 1 in. D.E.	45	-0.006	—	-0.112	—	—	—
U-238, Pu-239 7:1 Mixture 1 in. D.E.	45A	-0.164	-16.0	-3.68	-2.08	—	—
U-238, Pu-239 7:1 Mixture 1 in. D.E.	45A	-0.136	-13.6	-3.12	-1.98	—	—

^a D.E. = Doppler element.

Table 2
Doppler Effect Reactivity Changes for Pu-239 and U-235
($\Delta T = 500$ to 1100°K)

Measurement	Assembly	Experiment					Theory
		$\Delta\rho$, Inhours ($\pm \sim 0.020$)	$\Delta k/k \times 10^{-4}$	$\Delta k/k$ per kg $\times 10^{-4}$	Expansion* $\Delta k/k \times 10^{-4}$	Corrected $\Delta k/k \times 10^{-4}$	$\Delta k/k \times 10^{-4}$
Pu-239	45A	-0.000	-1.21	-5.26	—	—	—
Pu-239—3 weeks later	45A	-0.049	-0.96	-4.39	—	—	—
Pu-239 Average	45A	-0.055	-1.10	-4.88	+1.35	+0.25	+5.06
Pu-239 Homogeneous	45A	-0.034	-0.68	-3.02	+1.35	+0.67	—
Pu-239 in B ₂ C	45A	+0.042	+0.83	+3.70	+0.90	+1.73	+2.33
Pu-239 1 Element	45A	-0.044	—	—	—	—	—
Pu-239	45	-0.024	-0.56	-2.49	+1.63	+1.07	+12.95
(U-235; JG, PA)	45A	+0.054	+1.1	+4.64	+1.05	+2.15	+2.90

* Calculated correction to experiment.

References:

1. G. J. Fischer, et al., "Doppler Effect Measurements in Plutonium-Fueled Fast Power Breeder Reactor Spectra" p. 120 ANL Reactor Physics Division Annual Report, July 1, 1964 to June 30, 1965, ANL-7110 (1965)
2. ANL Reactor Development Program Progress Report September 1964, p. 13, ANL-6944 (1964)

7-1-47

Doppler Coefficients

Materials Studied: Natural U in UO_2 pellets
 U-235 in UO_2 pellets
 Pu-239 in PuO_2 pellets
 U-238:Pu-239 = 8:1 in $\text{UO}_2 + \text{PuO}_2$ pellets

Facility: ZPR (1.1)

Assembly: 47 (2.1-47) SEFOR Mockup

Experimental Method:

Identical to that described in 7-1-45.

Sample compositions are given in Table 3.

Results:

Errors quoted are those obtained from the control (balancing) rod data.

Table 1
 Measured Reactivity Change Due to Heating in the SEFOR Mockup.

Sample	Date	Environment	lb/kg of Metal Component at			
			300°K	500°K	800°K	1100°K
Natural U	4-21	Normal Core	0 ± 0.060	-0.806 ± 0.074	-1.831 ± 0.068	—
"	4-22	"	0 ± 0.025	-0.947 ± 0.040	-1.906 ± 0.066	-2.567 ± 0.060
"	5-9	"	0 ± 0.033	-0.947 ± 0.060	-1.993 ± 0.066	—
"	Average	"		-3.926	-1.903	-2.567
"	4-25	Sodium Void	0 ± 0.017	-0.736 ± 0.060	-1.575 ± 0.060	-2.133 ± 0.033
"	4-28	Boron Sleeve	0 ± 0.033	-0.120 ± 0.041	-0.227 ± 0.060	-0.331 ± 0.068
U/Pu	4-10	Normal Core	0 ± 0.062	-0.797 ± 0.096	-1.645 ± 0.062	-2.204 ± 0.061

* Not Corrected for expansion effects.

** Radius of Sodium voided region was 19.4 cm.

*** Thickness of natural boron carbide sleeve was 1.27 cm.

7-1-47 (cont'd)

Table 2
Results for Fissile Samples in the SEFOR Mockup*

T(°K)	lb/kg of Metal Component			
	Pu-239 in Boron Sleeve	Pu-239 in Normal Core (First Run) ^(a)	Pu-239 in Normal Core (Second Run)	U-235 in Normal Core
300	—	—	—	0 ± 0.025
350	0 ± 0.026	-0.160 ± 0.071	—	—
370	—	—	0 ± 0.026	—
500	+0.270 ± 0.039	-0.213 ± 0.077	-0.270 ± 0.062	+0.378 ± 0.026
650	—	—	-0.391 ± 0.045	+0.579 ± 0.043
800	+0.665 ± 0.045	-0.546 ± 0.084	-0.472 ± 0.032	+0.833 ± 0.056
900	—	—	-0.429 ± 0.045	+1.039 ± 0.087
1050	—	—	-0.086 ± 0.052	+1.537 ± 0.049
1100	+1.498 ± 0.039	-0.044 ± 0.077	-0.112 ± 0.045	+1.901 ± 0.043

^(a) The reference run at 350°K was performed in a manner inconsistent with all other experiments. The results were normalized to place the other three data on the curve of the second run.

*Not corrected for expansion effects. If the data beyond 800° are ignored, preliminary calculations for expansion effects for plutonium in the normal core indicate a Doppler effect near zero.

** Thickness of natural boron carbide sleeve was 1.27 cm.

*** Data beyond 800°K suspect because of unexplained reactivity variations.

Table 3
Masses and Compositions of the ZPR-3 Doppler-effect Samples
For Assemblies 47 and 48

Isotope	Atoms/cm ³ × 10 ⁻²⁴			
	Pu-239	8:1 U-Pu	U-238	U-235
U-238	—	0.01327	0.0156	0.0004
U-235	—	0.00010	0.0001	0.0154
Pu-239	0.0147	0.00163	—	—
Pu-241	0.0004	0.00012	—	—
O	0.0302	0.03012	0.0314	0.0316
Total Mass (g)	263.8	261.8	274.4	275.8

Reference:

J. M. Gasidlo, "Results of Recent Doppler Experiments in ZPR-3 " p. 345 Proc. Int. Conf. on Fast Critical Experiments and Their Analysis, ANL-7320 (1966)

7-1-48

Doppler Coefficients

Materials Studied:

U-238 in UO₂ pellets
Pu-239 in PuO₂ pellets

Facility: ZPR-3 (1.1)

Assembly: 48 (2.1-48)

Experimental Method:

Identical to that described in 7-1-45.

Sample compositions are given in 7-1-47

Results:

Errors quoted are those obtained from control (balancing) rod data.

Table I
Experimental Results for the Doppler-Effect
Sample of Natural Uranium Oxide in Assembly 48*

<u>$\delta k \times 10^6$</u>					
T(°K)	Experimental Results		Fitted to 1/T Curve **		Average
	<u>1st Run</u>	<u>2nd Run</u>	<u>1st Run</u>	<u>2nd Run</u>	
300	0±0.07		-0.08	0	0
322		0±0.04		-0.20	-0.20
500	-1.67±0.09	-1.19±0.06(a)	-1.75		-1.75
800	-3.65±0.10	-3.59±0.05	-3.73	-3.79	-3.76
1100	-4.75±0.13	-4.76±0.10	-4.83	-4.96	-4.89

*Not corrected for expansion effects

**Data from each run were fitted to $\frac{dk}{dT} = \frac{1}{T}$ curves;

the curves were forced to zero at 300°K, and the data points re-specified from the resultant curves.

^a This datum is inconsistent with the remainder and was not included in the analysis or the average.

7-1-48 (cont'd)

Table II
Experimental Results for the Doppler-Effect
Sample of Plutonium Oxide in Assembly 48*

$\delta k \times 10^6$

<u>T (°K)</u>	<u>Plutonium Sample</u>	<u>Empty Cans</u>
300		0 ± 0.046
355	0 ± 0.04	
500	-0.29 ± 0.05	$+0.055 \pm 0.052$
650	-0.49 ± 0.06	
800	-0.87 ± 0.06	-0.23 ± 0.056
900	-1.01 ± 0.06	
1052	-1.25 ± 0.06	
1100	-1.16 ± 0.08	-0.015 ± 0.061

*Not corrected for expansion effects. Preliminary calculations for expansion effects indicate a Doppler effect near zero.

Reference:

J. M. Gasidlo, "Results of Recent Doppler Experiments in ZPR-3" Proc. Int. Confer. on Fast Critical Experiments and Their Analysis, p. 345 ANL-7320 (1966)

Materials Studied:

Natural U in UO_2 pellets.
Depleted U in UC
Depleted U in U metal
Enriched U (12.5%, 33.3%, 50%, 66.7%) in UO_2 pellets
U-235 in UO_2 pellets
U-235 in U metal

Facility: ZPR-6 (1.2)**Assembly:** 4Z (2.2-4Z)**Experimental Method:**

The Doppler sample was oscillated repetitively in and out of the assembly while a servo-controlled, calibrated boron rod held the reactor critical. The difference in the average position of the calibrated rod, sample-in minus sample-out, gave the reactivity worth of the sample relative to the reference. Repeating the procedure with the sample at various temperatures gave the reactivity change of the sample with temperature.

Figure 1 shows the oscillator and assembly.

FIG. 1. Doppler Equipment in ZPR-6.

Four general sample types were used. The FE type was freely-expanding, reduced density UO_2 , the NE type allowed radial but no axial expansion of reduced density UO_2 , the NNE type allowed no axial expansion and constrained radial expansion of reduced density UO_2 , and the last type was U metal pellets.

In the main, samples were about 1 in. in diameter by 12 inches long. They were contained in elements evacuated between the inside heater tube and the outer shell, an example of which is given in Figure 2 for the NE case.

FIG. 2. Type 2 Element (Expansion Constrained).

In the FE sample elements the heater tube was Inconel and the sample itself about 70% theoretical density UO_2 pellets resting on the tube; the sample length increased with the expansion characteristics of the Inconel.

In the NE sample elements the heater tube was Inconel, and the pellet stack was held by a piston at each end sealed through bellows which allowed the heater tube to expand over the pistons, thus axially constraining sample expansion.

In the NNE sample elements the heater tube was Invar and the samples were inserted by a vibratory compaction process to maintain no clearance between tube and UO_2 , the ends being held by pistons. From room temperature to about 200°C , the samples were constrained axially and radially; above 200°C the Invar expansion is very similar to Inconel, and the element acted much like the NE sample.

In the metal sample elements the heater tube was Inconel, and the samples made of ten 1 in. diameter by 1 in. long pellets with 1 in. long quartz dummy elements at each end to keep the axial temperatures flat so that the alpha-beta phase change occurred throughout the sample at once.

The "normal" sample environment was the regular Assembly 42 core except the four drawers nearest the sample were loaded more homogeneously. In the "no sodium" environment, the sodium cans were replaced throughout the core by steel frames to preserve the core steel concentration. In the "1/2 in. B_4C " environment, the first 0.5 in. of the environment outside the sample was replaced by natural B_4C . The "isolated" environment had the U-238 plates in the eight drawers adjacent to the sample replaced by an amount of Mo calculated to retain the fluxes and adjoints at the sample position. The "reference void" environment was a variation in the standard loading so that a 1 in. increase in the void radius around the sample, called the "1-in. void increase" environment, could be made easily.

TABLE 1. SUMMARY OF DOPPLER MEASUREMENTS IN ZPR-6 ASSEMBLY 4Z

Date of Measurement	Sample No.	Diameter and Length, in.	Material	Enrichment, %	Total U Mass, kg	Sample Type ^(a)	Sample Environment	Calculated Expansion Correction 1000 $\frac{\Delta T}{T}$ (°K/°K)	σ , lb/kg U	Measured Reactivity Changes vs Temperature (60 lb = 1% $\Delta k/k$)	
										Temp., °K (Data normalized to 0.00 lb/kg U at 293°K and uncorrected for expansion)	lb/kg U
6/7/65	N1	1 x 12	UO ₂	Nat. U	1.115	FE	Normal	0.000	±0.002	346 531 1091	
6/16/65	N1	1 x 12	UO ₂	Nat. U	1.116	FE	Normal	0.000	±0.002	-121 -212 -272	
3/18/66	N1	1 x 12	UO ₂	Nat. U	1.116	FE	Normal	0.000	±0.002	359 533 416 525 537 736 952 1647	
7/11/65	N1	1 x 12	UO ₂	Nat. U	1.116	FE	1/2-in. H ₂ O	0.000	±0.002	-631 -667 -683 -115 -157 -301 -332 -371	
3/23/66	N3	1 x 12	UO ₂	Nat. U	1.116	FE	No Na	0.000	±0.002	493 1227	
6/22/65	N5	3/8 x 12	UO ₂	Nat. U	0.214	FE	Normal	0.000	±0.011	-137 -252	
7/5/65	N6	3/8 x 12	UO ₂	Dep. U 22% U ₂₃₅	0.400	FE	Normal	0.000	±0.011	511 778 1056	
6/25/65	N9	1 x 12	UO ₂	Nat. U	1.673	NE	Normal	0.000	±0.002	-631 -667 -682	
9/20/65	N9	1 x 12	UO ₂	Nat. U	1.673	NE	Normal	0.000	±0.002	945 790 1056	
7/12/65	N9	3/8 x 12	UO ₂	Nat. U	0.633	FE	Normal	0.000	±0.002	-676 -123 -170	
3/23/66	N11	1 x 10	U Metal	Dep. U 47% U ₂₃₅	2.300	Metal	Normal	0.000	±0.002	471 458 646 1056	
6/9/65	LE1	1 x 12	UO ₂	12.5	0.901	FE	Normal	+0.002	±0.004	-676 -123 -170	
6/11/65	LE1	1 x 12	UO ₂	12.5	0.901	FE	Isolated	+0.002	±0.004	-676 -123 -170	
6/17/65	LE1	1 x 12	UO ₂	12.5	0.901	FE	Reference Void	+0.002	±0.002	511 778 1056	
6/18/65	LE1	1 x 12	UO ₂	12.5	0.901	FE	1-in. Void Insurance	+0.002	±0.004	-676 -123 -170	
3/21/66	LE1	1 x 12	UO ₂	12.5	0.901	FE	No Na	+0.002	±0.002	471 458 646 1056	
9/2/66	LE2A	1 x 12	UO ₂	12.5	0.994	NNE	Normal	<+0.002	±0.002	-676 -123 -170	
7/6/65	LE2	1 x 12	UO ₂	12.5	0.995	NE	Normal	<+0.002	±0.004	511 778 1056	
7/26/65	HE1	1 x 12	UO ₂	97.7	0.995	FE	Normal	+0.002	±0.004	-676 -123 -170	
7/29/65	HE2	1 x 12	UO ₂	97.7	1.000	NE	Normal	+0.002	±0.004	511 778 1056	
6/1/66	HE2	1 x 12	UO ₂	97.7	1.000	NE	No Na	+0.002	±0.002	-676 -123 -170	
6/12/65	HE2	1 x 12	U Metal	92.1	2.300	Metal	Normal	+0.235	±0.002	471 458 646 1056	
6/24/65	HE4	3/8 x 12	UO ₂	97.6	0.710	FE	Normal	+0.002	±0.002	-676 -123 -170	
3/22/66	HE5	1 x 12	UO ₂	97.7	0.712	NNE	Normal	+0.011	±0.002	511 778 1056	
8/5/65	E23	1 x 12	UO ₂	32.2	0.990	FE	Normal	+0.002	±0.004	-676 -123 -170	
6/10/65	E26	1 x 12	UO ₂	60.0	1.907	FE	Normal	+0.002	±0.004	511 778 1056	
6/10/65	E26	1 x 12	UO ₂	60.7	1.915	FE	Normal	+0.011	±0.002	-676 -123 -170	
9/13/65	Empty Sample Can					FE	Normal		±0.002	511 778 1056	

^(a) FE = freely expanding element type; NE = axially constrained element type; NNE = axially and radially constrained element type.^(b) Expansion correction is proportionately less at lower temperatures above the room-temperature reference (see Sec. 2.1).

7-2-42 (Cont'd)

Results:

(See 7-2-5 for comparative U-235 expansion effects in Assemblies 4Z and 5 of ZPR-6, and Assemblies 11 and 12 of ZPR-9).

The data, uncorrected for expansion, are listed in Table I.

The U-235 results, corrected for U-238 content, but not expansion are given in Figure 3. The effect of expansion was determined by measuring the reactivity jump associated with the alpha-beta phase transition and, on a similar sample, the expansion associated with the transition. The value obtained was -0.0165 Ih/mil of expansion per kg. of U-235. This was used to correct the metal results directly, and to correct the oxide results by use of a calculated oxide-to-metal expansion ratio. The U-235 Doppler effects corrected for expansion are given in Table II.

The Doppler effect for several enrichments of U-235 is given in Figure 4.

Fig. 2 U-235 Experimental Results

Table II Doppler Effects Corrected for Expansion

Temperature		Measured Reactivity, $\beta/\text{kg U}^{235}$				Expansion Correction, $\beta/\text{kg U}^{235}$				Doppler Effect, $\beta/\text{kg U}^{235}$			
°C	°K	Metal	FE	NE	NKE	Metal	FE	NE	NKE	Metal	FE	NE	NKE
100	373	+0.008	+0.014	+0.016	+0.027	+0.017	+0.008	+0.008	=	+0.008	+0.013	+0.015	+0.027
200	473	+0.004	+0.008	+0.008	+0.014	+0.008	+0.008	+0.004	=	+0.008	+0.007	+0.007	+0.014
300	573	-0.002	+0.002	+0.002	+0.004	+0.002	+0.002	+0.002	+0.001	+0.001	+0.001	+0.001	+0.007
400	673	-0.014	+0.006	+0.002	+0.002	+0.004	+0.002	+0.006	+0.002	+0.002	+0.001	+0.002	+0.002
500	773	-0.026	+0.006	+0.002	+0.002	+0.012	+0.002	+0.012	+0.002	+0.002	+0.001	+0.001	+0.004
600	873	-0.037	+0.006	+0.004	+0.002	+0.017	+0.002	+0.015	+0.002	+0.002	+0.001	+0.001	+0.002
700	973	-0.066	+0.002	+0.002	+0.002	+0.021	+0.002	+0.015	+0.002	+0.002	+0.001	+0.001	+0.002
800	1073	-0.126	+0.002	+0.002	+0.002	+0.031	+0.002	+0.015	+0.001	+0.002	+0.001	+0.001	+0.002

Fig. 4 1-in. UO_2 Corrected for Expansion

7-2-42 (Cont'd)

Reference:

C. E. Till, R. A. Lewis, and W. M. Hwang, "ZPR-6 Doppler Measurements and Comparison with Theory," p. 319, Proc. Int. Conf. on Fast Critical Experiments and their Analysis, October 10-13, 1966 ANL-7320 (1966).

7-2-5

Doppler Coefficients

Materials Studied: Natural U in UO_2 pellets
 U-235 in UO_2 pellets
 Pu-239 in PuO_2 pellets

Facility: ZPR-6 (1.2)Assembly: 5 (2.2-5)Experimental Method:

Identical to that described in Section 7-2-4Z.

Results:

Assembly 5 data are listed in Table I and shown in Figure 1. Note: $1\% \Delta k/k = 476$ Ih.

Comparative U-235 expansion effects in Assemblies 5 and 4Z of ZPR-6 and Assemblies 11 and 12 of ZPR-9 are shown in Table II and Figures 1, 2, and 3. The Assembly 5 data were most carefully measured on the full scale version of the carbide mock-up; these data are the most precise of all data listed here and should be weighted most heavily. The Assembly 4Z data show more scatter, and the results from Assemblies 11 and 12 illustrate the difficulty in obtaining precise measurements in small zone assemblies in which it was known that spectral equilibrium was not achieved.

Table I
 Results of Doppler Measurements in Assembly 5 of ZPR-6

Sample No. ^a	Material	Enrichment	Total Mass, kg	Sample Type ^b	Measured Reactivity Change versus Temperature ^{c,d} , $\Delta k/k$ or $\Delta \rho/\rho$
N1	UO_2	Natural	1.116	FE	$0.000(200)$, $-0.100(400)$, $-0.104(407)$, $+0.001(200)$, $-0.300(370)$, $-0.301(1001)$
HE3	UO_2	97.7	0.9123	FE	$0.000(200)$, $+0.001(400)$, $+0.000(412)$, $+0.000(200)$, $+0.000(100)$
HE10	UO_2	97.7	0.9071	NNE	$0.000(200)$, $+0.000(200)$, $+0.004(270)$, $+0.000(200)$, $+0.000(400)$, $+0.000(500)$
Pu1NV2	PuO_2	89.6 w/o ^{240}Pu	0.9047	NNE	$+0.007(200)$, $+0.000(200)$, $+0.000(774)$, $+0.000(200)$, $0.000(211)$, $-0.004(200)$, $-0.001(200)$, $-0.000(200)$, $-0.000(157)$, $-0.004(400)$, $-0.000(200)$, $-0.000(200)$, $-0.000(717)$, $-0.000(500)$

^a All samples were 1 in. diam and 12 in. long.^b FE—freely expanding; NNE—constrained axially and radially.^c Experimental precision was 0.001 Ih/kg in all samples.^d Data normalized to 0.000 Ih/kg at nominal room temperature and uncorrected for expansion.

Table II
Experimental Results

Assembly and Sample Type	Sample Temperature, °K	Measured lb/kg of U-235	Std. Dev.	Correction for U-235, lb/kg of U-235	Correction for Expansion, lb/kg of U-235	Doppler Effect, lb/kg of U-235
Assembly 6 ZPR-6; Sample HE-8 (FE, 97.7%, 892.3 g U-235)	426	0.021	0.001	+0.001	+0.004	0.020
	612	0.040		+0.002	+0.008	0.061
	839	0.049		+0.002	+0.017	0.086
	1068	0.060		+0.003	+0.023	0.076
Assembly 6 ZPR-6; Sample HE-10 (NNE, 97.7%, 786.5 g U-235)	339	0.010	0.001	0	0	0.010
	370	0.014		+0.001	0	0.014
	395	0.020		+0.001	0	0.023
	428	0.025		+0.001	0	0.026
	469	0.030		+0.001	0	0.031
	486	0.034		+0.001	0	0.035
	579	0.041		+0.002	+0.001	0.044
	774	0.064		+0.002	+0.006	0.062
Assembly 42 ZPR-4; Sample HE-1 (FE, 97.7%, 942.2 g U-235)	628	0.061	0.001	+0.003	+0.010	0.074
	480	0.032		+0.001	+0.006	0.039
	609	0.046		+0.002	+0.012	0.060
	821	0.066		+0.002	+0.019	0.076
	1061	0.091		+0.003	+0.025	0.099
	319	0.006		0	+0.001	0.007
Assembly 42 ZPR-4; Sample HE-2 (NE, 97.7%, 860.9 g U-235)	326	0.016	0.003	0	+0.002	0.018
	362	0.017		0	+0.002	0.013
	410	0.022		+0.001	+0.006	0.026
	466	0.028		+0.001	+0.004	0.033
	589	0.041		+0.002	+0.007	0.053
	783	0.068		+0.002	+0.012	0.079
	1011	0.085		+0.003	+0.017	0.083
	329	0.013		0	0	0.013
Assembly 42 ZPR-4; Sample HE-5 (NNE, 97.7%, 696.0 g U-235)	364	0.026	0.003	+0.001	+0.002	0.026
	417	0.033		+0.001	0	0.031
	447	0.037		+0.001	0	0.038
	479	0.042		+0.001	0	0.044
Assembly 42 ZPR-4; Sample HE-4 (FE, 97.8%, 226.1 g U-235)	721	0.062	0.003	+0.002	+0.004	0.061
	935	0.089		+0.003	+0.009	0.081
	1048	0.016		+0.001	+0.002	0.019
	1068	0.035		+0.003	+0.008	0.046
Assembly 42 ZPR-4; Sample HE-3 (Metal, 53.1%, 2206.3 g U-235)	474	-0.021	0.002	+0.006	+0.040	+0.041
	706	-0.026		+0.010	+0.008	+0.040
	899	-0.066		+0.013	+0.137	+0.104
	1030	-0.111		+0.014	+0.209	+0.112
	1030	-0.123		+0.015	+0.226	+0.118
	331	0.011		0	+0.001	0.012
Assembly 11 ZPR-9; Sample HE-2 (NE, 97.7%, 860.5 g U-235)	377	0.016	0.004	+0.001	+0.004	0.020
	424	0.040		+0.001	+0.004	0.045
	472	0.046		+0.001	+0.005	0.046
	616	0.076		+0.002	+0.012	0.059
	866	0.092		+0.003	+0.020	0.116
	1064	0.112		+0.004	+0.030	0.136
Assembly 11 ZPR-9; Sample HE-5 (NNE, 97.7%, 636.0 g U-235)	342	0.007	0.004	0	0	0.007
	378	0.017		+0.001	0	0.018
	443	0.036		+0.001	0	0.037
	464	0.046		+0.001	+0.001	0.047
	626	0.046		+0.002	+0.003	0.045
	820	0.061		+0.003	+0.009	0.068
Assembly 12 ZPR-9; Sample HE-5 (NNE, 97.7%, 696.0 g U-235)	1032	0.113	0.004	+0.004	+0.016	0.132
	225	+0.032		0	0	0.032
	364	+0.060		+0.001	0	0.060
	426	+0.064		+0.001	+0.001	0.065
	479	+0.062		+0.001	+0.001	0.064
	604	+0.079		+0.002	+0.004	0.085
	847	+0.076		+0.003	+0.013	0.095

7-2-5 (cont'd)

Figure 1
Assembly 5, ZPR-6, 1 in. dia. U-235 O₂ Samples

References:

1. C.E. Till, R. A. Lewis, and E. F. Groh, "U-235 Doppler Effect Measurement Data and Techniques", ANL Reactor Physics Division Annual Report, July 1, 1966 to June 30, 1967, 151, ANL-7310 (1968)
2. R. A. Karam, et al., "Measured Physics Parameters in a Large Uranium Carbide Fast Core, Assembly 5 ZPR-6" ANL-7310, 167 (1968)

7-2-6

Doppler Coefficient

Materials Measured: Natural U in UO_2 pellets
U-235 in UO_2 pellets

Facility: ZPR-6 (1.2) Assembly: 6 (2.2-6)

Experimental Method:

Identical to that described in 7-2-42.

The standard environmental configuration included a 0.5 in. stainless steel filter between the Doppler sample and the reactor core. See 7-3-18 for the original investigation of various filter arrangements. In one case, a 0.75 in. filter was used, as indicated in the tables of results.

Results:

Sample Temperature (°K) ^a Reactivity Change [1h/(kg-U)] ^b				Comments
<u>Natural Uranium</u>				
505	811	1081		
-0.132	-0.266	-0.353		
.				
<u>Enriched Uranium^c</u>				
759	764	1023	1041	
+0.050	+0.050	+0.059	+0.055	
509	768	1042		3/4-in. Stainless Steel Filter
+0.041	+0.061	+0.075		
406	642	826	1030	Expansion Constrained
+0.019	+0.041	+0.047	+0.058	

^aData normalized to 0.000 1h/(kg-U) at 293 $^{\circ}K$ and uncorrected for expansion.

^bRoot-mean-square error $\sigma \approx 0.002$ 1h/(kg-U).

^c97.7% ^{235}U .

Reference:

J. W. Daughtry and R. B. Pond, "Doppler Experiments,"
ANL Reactor Development Program Progress
Report August 1969, 19, ANL-7606 (1969)

7-3-11

Doppler Coefficients

Materials Studied: Natural U in UO_2 pellets
U-235 in UO_2 pellets and in UO_2 powder

Facility: ZPR-9 (1.3) Assembly: 11 (2,3-11)

Experimental Method:

Similar to that described in Section 7-2-4Z.

Four sample types were used. The FE type was freely expanding, the NE type allowed radial but no axial expansion, and the NNE type allowed no axial expansion and constrained radial expansion over the first 240 to 300°C above room temperature. The HT or high temperature sample holder was an alumina inner can with a bare tantalum heater wire used to investigate techniques for possible Doppler measurements to 2000°K.

Results:

(See also Section 7-2-5 for comparative U-235 expansion effects in Assemblies 4Z and 5 of ZPR-6, and Assemblies 11 and 12 of ZPR-9).

Table I
Assembly 11 Doppler Measurements

Date of Meas.	Sample No.	Nominal Diam. and Length, in.	Material	% Enrichment	Total U Mass, kg	Sample Type	Sample Environment	Experimental Precision, lh/kg-U	Measured Reactivity Change vs. Temp. ^a Temp., °K; (Reactivity, lh/kg-U)			
8/1/66	N13	$\frac{3}{8} \times 12$	UO_2	natural	0.344	FE	Regular Core	0.009	1061			
									(-0.307)			
8/1/66	N16	$\frac{3}{8} \times 12$	UO_2	natural	0.344	FE	Regular Core	0.009	497	773	1064	
									(-0.184)	(-0.746)	(-0.300)	
8/24/66	N1	1×12	UO_2	natural	1.116	FE	Regular Core	0.008	495	865	878	1064
									(-0.145)	(-0.381)	(-0.335)	(-0.300)
10/6/66	N16	1×12	UO_2	natural	0.716	HT	Regular Core	0.007	899	890	747	1063
									(-0.200)	(-0.338)	(-0.300)	(-0.403)
									1907	1513		
									(-0.500)	(-0.610)		
10/7/66	empty can	1×12	—	—	—	HT	Regular Core	0.008	840	723	1030	
									(-0.045)	(-0.088)	(-0.100) ^b	
8/10/66	HE5	$\frac{3}{8} \times 12$	UO_2	97.6	6.340	FE	Regular Core	0.010	805	804 ^c	875	1060 ^d
									(+0.048)	(+0.077)	(+0.005)	(+0.072)
9/3/66	HE2	1×12	UO_2	97.7	1.004	NE	Regular Core	0.008	331	377	424	475
									(+0.011)	(+0.012)	(+0.000)	(+0.007)
									618	808	1004	
									(+0.073)	(+0.000)	(+0.100)	
9/13/66	HE5	1×12	UO_2	97.7	0.712	NNE	Regular Core	0.008	343	375	445	494
									(+0.007)	(+0.017)	(+0.005)	(+0.044)
									628	820	1003	
									(+0.000)	(+0.000)	(+0.110)	

^a 420 lh = 1% $\Delta k/k$. Data are normalized to 0.000 lh/kg-U at 295°K and uncorrected for expansion.

^b Per kg uranium in N19.

^c $\sigma = 0.015 \text{ lh/kg-U}$.

^d $\sigma = 0.005 \text{ lh/kg-U}$.

7-3-11 (cont'd)

Reference:

R. A. Lewis, L. G. LeSage, and R. A. Karam, "ZPR-9 Assemblies 11 and 12 - Two Small Zoned Versions of the 2600-Liter Assembly 5 ZPR-6", p. 190, ANL Reactor Physics Annual Report - July 1, 1966 to June 30, 1967, ANL-7310 (1968)

7-3-12

Doppler Coefficients

Materials Studied: Natural U in UO_2 pellets
U-235 in UO_2 powder

Facility: ZPR-9 (1.3)Assembly: 12 (2.3-12)Experimental Methods:

Similar to that described in Section 7-2-4Z.

Results:

See also Section 7-2-5 for comparative U-235 expansion effects in Assemblies 4Z and 5 of ZPR-6, and Assemblies 11 and 12 of ZPR-9).

Table I
Assembly 12 Doppler Measurements

Date of Meas.	Sample No.	Nominal Diam. and Length, in.	Material	% Enrichment	Total U Mass, kg	Sample Type	Sample Environment	Experimental Precision σ , lb/kg-U	Measured Reactivity Change vs. Temp. ^a Temp., °K; (Reactivity, lb/kg-U)			
12/16/66	N1	1 x 12	UO	natural	1.116	FE	Regular Core	0.004	335 (-0.086)	514 (-0.147)	678 (-0.216)	867 (-0.284)
12/20/66	N1	1 x 12	UO	natural	1.116	FE	Min. Het.	0.002	520 (-0.153)	682 (-0.231)	867 ^b (-0.291)	1082 (-0.361)
1/9/67	N1	1 x 12	UO	natural	1.116	FE	Min. Het.	0.003	354 (-0.306)			
1/3/67	HE5	1 x 12	UO	97.7	0.712	NNE	Min. Het.	0.005	325 (+0.032) 604 (+0.077)	364 (+0.068) 347 (+0.077)	426 (+0.063)	479 (+0.080)

^a 451 lb = 1% $\Delta k/k$. Data are normalized to 0.000 lb/kg-U at 293°K and uncorrected for expansion.

^b σ = 0.011 lb/kg uranium.

* The eight drawers immediately adjacent to the sample location were rearranged from the regular core loading to minimize the plate heterogeneity around the sample.

Reference:

R. A. Lewis, L. G. LeSage, and R. A. Karam, "ZPR-9 Assemblies 11 and 12 - Two Small Zoned Versions of the 2600-Liter Assembly 5 ZPR-6", p. 190, ANL Reactor Physics Annual Report - July 1, 1966 to June 30, 1967, ANL-7310 (1968)

7-3-13

Doppler Coefficients

Materials Studied: U-235 in UO_2 pellets and in UO_2 powder
U-238 in UO_2 pellets

Facility: ZPR-9 (1.3)

Assemblies: 13, 14, 16 and 17
(2.3-13, 14, 16, 17)

Experimental Method:

Similar to that described in 7-2-4Z, but in this work, the calibrated auto rod used to measure reactivity was designed for worth linearity over its full travel.

In the U-235 measurements, one sample was constrained and the other allowed to freely expand. The U-238 sample was freely expanding.

Results:

For comparison and convenience, a brief summary of approximate compositions of Assemblies 13, 14, 16, 17 and 17Z is included in Table I. Detailed compositions are given in the appropriate parts of Section 2. Table II lists results, and Figures 1 and 2 show trends of the Doppler effect from assembly to assembly.

Table I
Material Compositions for ZPR-9 Assemblies Nos. 13, 14, 16, 17 and 17Z

Type of Zone Material	Atom Densities, (atoms/cm ³) $\times 10^{21}$							
	Fe	Ni	Cr	Na	C	U-235	U-238	H
Core of Assy. 12	0.000086	0.001148	0.002593	0.000024	0.012968	0.001547	0.010623	0.009754
Core of Assy. 13					0.014920	0.001541	0.027380	
Core of Assy. 14					0.016363	0.001539	0.006880	
Core of Assy. 15					0.017305	0.001537	0.005771	
Core of Assy. 16					0.019249	0.001536	0.004962	
Core of Assy. 17	0.008629	0.001116	0.002520	0.003017	0.027966	0.001535	0.000110	—
Buffer					0.024630	0.000020	0.009704	
Driver					0.007833	0.008784	0.006335	
Depleted reflector					—	—	—	
Steel reflector					—	—	—	

- * Assembly 17Z had the composition of Assembly 17 Core only in a 15.57 cm. radius zone at the center. This zone in turn was surrounded by Assembly 16 Core zone composition material out to a radius of 25.87 cm., and the buffer, driver, and reflector compositions were identical to Assembly 17.

Table 11
Experimental Doppler Results

Assembly ^a	Element ^b	(Type) ^c	Average Temperature, Average Reactivity Worth, Standard Deviation of the Average Reactivity Worth									
			%	lb/kgU ^d	±lb/kgU ^d	%	lb/kgU ^d	±lb/kgU ^d	%	lb/kgU ^d	±lb/kgU ^d	%
13	N1	(FE)	243 877	0.0 -1.818	0.010 0.006	283 1076	-0.276 -1.916	0.009 0.040	498	-0.880	0.010	715
13	11E-8	(FE)	291 887	0.0 +0.461	0.004 0.002	423 1074	+0.131 +0.716	0.005 0.003	817	+0.217	0.003	628
13	11E-9	(NNE)	283 880	0.0 +0.283	0.004 0.003	289 882	-0.484 +0.283	0.003 0.004	413 883	+0.111 +0.832	0.003 0.001	489 1070
14	N1	(FE)	288 886	0.0 -1.710	0.005 0.003	280 1072	-0.283 -2.032	0.002 0.002	495	-0.715	0.001	681
14	11E-8	(FE)	231 889	0.0 +0.882	0.004 0.002	428 1075	+0.130 +0.672	0.002 0.003	815 878	+0.228 +0.106	0.001 0.006	528
14	11E-9	(NNE)	282 888	0.0 +0.282	0.005 0.003	287 883	+0.116 +0.282	0.004 0.007	485 911	+0.160 +0.813	0.004 0.001	807 1067
16	N1	(FE)	282	0.0	0.002	280	-0.282	0.006	480	-0.720	0.002	680
16	N1	(FE)	281 879	0.0 -2.314	0.005 0.004	485	-0.640	0.003	823	-1.972	0.002	882
16	11E-8	(FE)	281 719	0.0 +0.281	0.004 0.003	489 844	+0.082 +0.281	0.002 0.002	821 1076	+0.183 +0.314	0.004 0.002	884
172	11E-9	(FE)	282 888	0.0 -0.782	0.005 0.003	429	-0.282	0.004	882	-0.381	0.003	813
17	11E-8	(FE)	283 788	0.0 -0.483	0.003 0.006	486 886	-0.178 -0.612	0.003 0.003	496 1081	-0.283 -0.722	0.003 0.006	883
17	11E-10	(NNE)	283 889	0.0 -0.487	0.007 0.007	482 717	-0.280 -0.482	0.007 0.006	473 880	-0.310 -0.707	0.007 0.006	821 1018
17	N1	(FE)	284 785	0.0 -2.088	0.003 0.006	285 1082	-0.624 -2.281	0.002 0.006	487 1081	-1.202 -3.585	0.001 0.006	810

^a Assembly 13: 1 X 12 in. = 427.88 lb.
 Assembly 14: 1 X 12 in. = 425.88 lb.
 Assembly 16: 1 X 12 in. = 425.88 lb.
 Assembly 172: 1 X 12 in. = 418.18 lb.
 Assembly 17: 1 X 12 in. = 281.78 lb.

^b N1: 1.2880 kg U²³⁵; 1.1882 kg U²³⁸; 0.8877 kg U²³⁵.
 11E-8: 1.2880 kg U²³⁵; 0.8877 kg U²³⁸; 0.8877 kg U²³⁵.
 11E-9: 0.8877 kg U²³⁵; 0.8877 kg U²³⁸; 0.8877 kg U²³⁵.
 11E-10: 0.8877 kg U²³⁵; 0.8877 kg U²³⁸; 0.8877 kg U²³⁵.
 11E-10: 0.8877 kg U²³⁵; 0.8877 kg U²³⁸; 0.8877 kg U²³⁵.
 N1 element w/o U: 1.2880 kg U²³⁵; 1.1882 kg U²³⁸; 0.8877 kg U²³⁵.
 N1 element w/o U: 1.2880 kg U²³⁵; 1.1882 kg U²³⁸; 0.8877 kg U²³⁵.
 U²³⁵ element w/o U: 1.2880 kg U²³⁵; 1.1882 kg U²³⁸; 0.8877 kg U²³⁵.
 U²³⁸ element w/o U: 1.2880 kg U²³⁵; 1.1882 kg U²³⁸; 0.8877 kg U²³⁵.
 U²³⁵ element w/o U: 1.2880 kg U²³⁵; 1.1882 kg U²³⁸; 0.8877 kg U²³⁵.
 U²³⁸ element w/o U: 1.2880 kg U²³⁵; 1.1882 kg U²³⁸; 0.8877 kg U²³⁵.

^c FE: Fully expanding U²³⁵ element 1 X 12 in. outer cylinder.
 NNE: Non expanding U²³⁵ element 1 X 12 in. powder cylinder (total axial constraints and limited radial constraints).

^d U²³⁵ element w/o U: 1.2880 kg U²³⁵; 1.1882 kg U²³⁸; 0.8877 kg U²³⁵.
 U²³⁸ element w/o U: 1.2880 kg U²³⁵; 1.1882 kg U²³⁸; 0.8877 kg U²³⁵.

7.3-13 (cont'd)

Figure 1
U-238 Doppler Effect Measurements on Assemblies 13,14,16 & 17 ZPR-9

Figure 2
U-235 Doppler Effect Measurements on Assemblies 13,14,16 & 17 ZPR-9
Uncorrected for expansion

Reference:

R. Pond and C. E. Till, "Doppler Effect Measurements in CH_2 -Softened Spectra; ZPR-9 Assemblies 13-17" ANL Reactor Physics Division Annual Report - July 1, 1966 to June 30, 1967, ANL-7310 159, (1968)

7-3-18

Doppler Coefficients

Material Studied: Natural U in UO_2 pellets

Facility: ZPR-9 (1.3)

Assembly: 18 (2.3-18)

Experimental Method:

Similar to that described in 7-2-4Z.

The environment immediately surrounding the Doppler sample was varied in the following ways to study the effect on the measurement:

- 1) The Standard Configuration shown in Figure 1.
- 2) The Revised Standard Configuration, which consisted of turning plates so that their flat sides rather than edges faced the sample, then subdividing plates so that each adjacent drawer contained four identical 0.5 inch thick sections, as shown in Figures 2 and 3.
- 3) The 90° Rotated Revised Standard Configuration, which consisted of rotating through 90° the adjacent four drawers in the Revised Standard Configuration so that the plate edges faced the sample.
- 4) The Revised Standard Configuration with filters, which had filter zones adjacent to the sample consisting of 0.25 in., 0.5 in., or 1.0 in. thick stainless steel, or 0.25 in., 0.5 in., or 1.0 in. thick depleted uranium. The 0.5 in. stainless steel case is shown in Figure 4.
- 5) The cases of the Standard, Revised Standard, 90° Rotated Revised Standard, 0.5 in. Stainless Steel Filter, and 1.0 in. stainless steel filter, all with sodium removed from the core.

Results:

The complete data are listed in Table I, and some normalized data for a specific temperature change summarize environmental affects in Table 2.

Assembly 18 is specified as having the same composition as Assembly 12. For Assembly 12:

$$1\% \frac{\Delta k}{k} = 451.2 \text{ Ih}$$

7-3-18 (cont'd)

Table I
1 in. Natural UO₂ Sample, ZPR-9 Assembly 18

Sample Environment	Sodium in Core	Date (1967)	Standard Deviation Each Point, lb/kg-U	Sample Temperature, °K ^(a) Reactivity Change, lb/kg-Uranium			
Standard	Yes	6/23	0.002	293 0.000	510 -0.155	773 -0.289	1058 -0.394
		7/13	0.002	293 0.000	762 -0.289		
		8/25	0.002	293 0.000	756 -0.278	1044 -0.388	
		8/28	0.003	293 0.000	776 -0.284		
Revised Standard	Yes	6/28	0.002	293 0.000	499 -0.157	782 -0.363	1059 -0.418
		8/8	0.002	293 0.000	768 -0.301		
		8/21	0.002	293 0.000	749 -0.291	1044 -0.408	
		8/23	0.001	293 0.000	748 -0.291	1043 -0.404	
		8/23	0.002	293 0.000	748 -0.292	1043 -0.406	
¼-in. SS Filter	Yes	8/17	0.001	293 0.000	479 -0.151	758 -0.299	1051 -0.408
½-in. SS Filter	Yes	7/11	0.001	293 0.000	771 -0.316		
		7/11	0.001	293 0.000	771 -0.318		
		8/11	0.001	293 0.000	505 -0.160	775 -0.320	
		8/16	0.001	293 0.000	760 -0.320	1064 -0.444	
1-in. SS Filter	Yes	7/5	0.002	293 0.000	505 -0.181	786 -0.353	1075 -0.477
90 deg-Rotated Rev. Stan.	Yes	8/22	0.001	293 0.000	751 -0.289	1045 -0.393	
		8/22	0.001	293 0.000	748 -0.281		
		8/24	0.002	293 0.000	752 -0.283	1046 -0.394	
¼-in. Depl. U Filter	Yes	8/18	0.002	293 0.000	749 -0.263	1043 -0.356	

7-3-18 (cont'd)

Table I (continued)

Sample Environment	Sodium in Core	Date (1967)	Standard Deviation Each Point, lb/kg-U	Sample Temperature, °K ^(a) Reactivity Change, lb/kg-Uranium			
1/2-in. Depl. U Filter	Yes	7-10	0.002	293 0.000	511 -0.128	796 -0.239	1047 -0.318
1 in. Depl. U Filter	Yes	7-6	0.002	293 0.000	512 -0.089	796 -0.173	1057 -0.233
Standard	No	9-5	0.002	293 0.000	772 -0.199	1043 -0.270	
Rev. Stan.	No	9-6	0.001	293 0.000	766 -0.205	1043 -0.279	
90 deg-Rot. Rev. Stan.	No	9-6	0.001	293 0.000	1043 -0.269		
1/2 in. SS Filter	No	9-8	0.001	293 0.000	768 -0.215	1043 -0.296	
1 in. SS Filter	No	9-11	0.002	293 0.000	770 -0.242	1043 -0.320	

^a Data normalized to 0.00 at 293°K.

Table II
Summary of Experimental Results for
1-in. Natural UO₂ Sample, 293-1100°K

Configuration	Na in Core	Normalized to Rev. Stan. with Na	Normalized to Rev. Stan. without Na
Revised Standard	Yes	1.000	
90 deg Rotated Rev. Stan.	Yes	0.958 ± 0.010	
Standard	Yes	0.948 ± 0.010	
1/4-in. Extra SS Filter	Yes	0.995 ± 0.010	
1/2-in. Extra SS Filter	Yes	1.068 ± 0.010	
1-in. Extra SS Filter	Yes	1.138 ± 0.010	
1/4-in. Depl. U Filter	Yes	0.869 ± 0.010	
1/2-in. Depl. U Filter	Yes	0.779 ± 0.011	
1-in. Depl. U Filter	Yes	0.566 ± 0.010	
Revised Standard	No	0.683 ± 0.009	1.000
90 deg Rotated Rev. Stan.	No		0.962 ± 0.010
Standard	No		0.969 ± 0.010
1/4-in. Extra SS Filter	No		1.062 ± 0.015
1-in. Extra SS Filter	No		1.144 ± 0.018

7-3-18 (cont'd)

Fig. 1 Standard Doppler Sample Environment

Fig. 2 Revised Standard Configuration

Fig. 3 Detail of Repeated $\frac{1}{2}$ in Thick Section of Revised Standard Configuration

Fig. 4 $\frac{1}{2}$ -in. Stainless Steel Filter Configuration

7-3-18 (cont'd)

Reference:

R. A. Lewis and T. W. Johnson, "Sensitivity of Small-Sample Doppler-Effect Measurements to Environment", ANL Reactor Physics Division Annual Report, July 1, 1967 to June 30, 1968
ANL-7410, 96, (1969)

7-3-19

Doppler Coefficients

Material Measured: Natural U in UO_2 pellets

Facility: ZPR-9 (1.3)

Assembly: 19 (2.3-19)

Experimental Method:

Identical to that described in 7-2-4Z.

The standard environmental configuration included a 0.5 in. stainless steel filter between the Doppler sample and the reactor core. See Section 7-3-18 for the original investigation of various filter arrangements.

Results:

Table I
Doppler Measurement in ZPR-9 Assembly 19*

Natural Uranium	
<u>Sample Temperature ($^{\circ}\text{K}$)^a</u>	<u>Reactivity Change ($\text{Ih}/(\text{kg-U})$)^b</u>
800	- 0.497
1055	- 0.628

^aData normalized to $0.000 \text{ Ih}/(\text{kg-U})$ at 293°K and uncorrected for expansion.

^bRoot-mean-square error $\sigma \approx 0.002 \text{ Ih}/(\text{kg-U})$.

*435.5 $\text{Ih} = 1\% \Delta k/k$.

Reference:

J. W. Daughtry & R. A. Lewis, "Uranium Doppler Measurements in ZPR-6 and -9 Oxide Cores", Reactor Physics Division Annual Report - July 1, 1968 to June 30, 1969, ANL-7610, 193 (1969)

7-3-21

Doppler Coefficients

Materials Measured: Natural U in UO_2 pellets
U-235 in UO_2 pellets

Facility: ZPR-9 (1.3)

Assembly: 21 (2.3-21)

Experimental Method:

Identical to that described in Section 7-2-4Z.

The standard environmental configuration included a 0.5 in. stainless steel filter between the Doppler sample and the reactor core. See Section 7-3-18 for the original investigation of various filter arrangements.

Results:

<u>Sample Temperature ($^{\circ}\text{K}$)^a</u>	<u>Reactivity Change ($\text{Ih}/(\text{kg-U})$)^b</u>
Natural Uranium	
776	- 0.594
1041	- 0.795
Enriched Uranium ^c	
765	+ 0.111
1048	+ 0.114

^aData normalized to $0.000 \text{ Ih}/(\text{kg-U})$ at 293°K and uncorrected for expansion.

^bRoot-mean-square error $\sigma \approx 0.002 \text{ Ih}/(\text{kg-U})$.

^c97.7% U-235.

Reference:

J. W. Daughtry and R. B. Pond, "Doppler Experiments", ANL Reactor Development Program Progress Report August 1969, ANL-7606, 19 (1969)

7-3-23

Doppler Coefficients

Materials Studied: Natural U in UO_2 pellets
Molybdenum metal
Rhenium metal
Tantalum metal
Tungsten metal

Facility: ZPR-9 (1.3)

Assembly: 23 (2.3-23)

Experimental Method:

Similar to that described in Section 7-2-4Z, but new sample holders were made of graphite in place of nickel alloy to allow measurement at high temperatures. See Figures 1 and 2.

Fig. 1 Graphite Sample Holder and UO_2 Pellets

Fig. 2 Assembled Doppler Element with Graphite Heater

7-3-23 (cont'd)

The environment adjacent to the sample is shown in Figure 3.

Fig. 3 Environment of Doppler Sample

Results:

The results are given in graphical form below.

Note: $1\% \Delta k/k = 457 \text{ Ih.}$

Fig. 4 Measured Doppler Effect; No Expansion Correction

7-3-23 (cont'd)

Reference:

J. W. Daughtry, et al., "Doppler Effect Measurements to 2000°K in ZPR-9", Reactor Physics Division Annual Report July 1, 1967 to June 30, 1968, ANL-7410 p. 103

SECTION 8 - TEMPERATURE COEFFICIENTS

8-1-20

Temperature Coefficient

Facility: ZPR-3 (1.1)

Assembly: 20 (2.1-20)

Method of Measurement:

Vary equilibrium temperature of the assembly over a range of 8.2 °C, measure reactivity by calibrated rod position.

Results:

Net temperature coefficient = $-1.5 \pm 0.2 \text{ Ih}/^\circ\text{C} = -3.3 \pm 0.4 \times 10^{-5} \Delta k/k^\circ\text{C}$.

Reference:

E. E. Branyan, "Core A Critical Studies for the Enrico Fermi Atomic Power Plant on ZPR-III" ANL-6629 (1962).

SECTION 9 - OTHER REACTIVITY COEFFICIENTS

9-1-30

Reactivity Coefficients

Type of Measurement: Reactivity worth of substituting materials throughout a section of core.

Facility: ZPR-3 (1.1)

Assembly: 30 (2.1-30)

Experimental Method:

The reference drawer contains the following materials:

position: 1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16
material: 5 5 6 A 1 2 A 6 5 5 5 5 5 5 6 A where 1 = U-235
2 = U-238, 5 = 45% Al (perforated), 6 = SS (304) and A = Al_2O_3
all plates are 0.318 cm thick (see Section 1.1).

Three substitutions were made:

- 1) 45% Al for the Al_2O_3 in positions 4, 6, and 14.
- 2) 100% Al " " " " " " " " " " " " " " " "
- 3) 45% Al for the SS in position 14.

These substitutions were made in a section of the core indicated in figure 1, and the resultant reactivity changes were measured by changes in rod positions. The results can be interpreted as distributed material worths.

Figure 1
Core Face Showing Section
Utilized in Wedge Sub-
stitution Experiments

Results:

<u>Material</u>	<u>Mass</u> <u>(kg)</u>	<u>Total Worth</u> <u>(Ih)</u>	<u>Material Worth</u> <u>(Ih/kg)</u>
Aluminum	5.302	+36.60	+6.90
Al ₂ O ₃	9.183	+114.40	+12.46
O (by inference)	4.321	+80.85	+18.71
SS	9.277	+21.76	+2.35

Related experiments indicated that these worths were distinctly dependent on position in the cell (1).

Reference:

P. I. Amundson, et al., "Critical Studies of a Dilute Oxide Fast Reactor Core (ZPR-III Assembly 30)" ANL-6337 (1961).

9-1-34

Reactivity Coefficients

Type of Measurement: Reactivity worth of materials distributed throughout the core.

Facility: ZPR-3 (1.1)

Assembly: 34 (2.1-34)

Experimental Method:

The normal cell structure was altered in a wedge-shaped section of one half of the core, and the reactivity measured by calibrated control rods.

Results:

The changes made, and the reactivity worths, extrapolated to the full core are given in Table I.

Table I
Distributed Reactivity Worths

<u>Material</u>	<u>Cell Alteration</u>	<u>Worth, $\Delta k/k$*</u>
U-235	Add 0.010 in thick foils of enriched U to cell (69.7 g U-235 per drawer)	44.1
U-238	Replace one column of depleted U by a column of 45% density Al	-0.846
Al	Replace 2 columns of 45% Al by 2 columns of solid Al	4.65
SS	Replace one column of SS by graphite	1.66
Graphite	Replace one column of graphite by Al	16.3

* $1\% \Delta k/k = 444 \Delta k/k$

Reference:

R. J. Huber et al., "Critical Studies of a Dilute Carbide Fast Reactor Core - ZPR III Assembly 34" ANL-6401 (1961)

9-2-2

Other Reactivity Coefficients

Type of Measurement: Reactivity coefficient of expansion.

Facility: ZPR-6 (1.2)

Assembly: 2 (2.2-2)

Experimental Method:

28 drawers comprising a sector having a volume approximately 1/8 that of the stationary half were modified such that the sodium plates extended fully into the blanket region at the back of each drawer. A reference run was then made using the metal spring, normally contained in a drawer, to hold the drawer constituents firmly against the front of the drawer. See Figure 1. The holding springs were removed from each of the 28 drawers and the constituents were arranged as shown in Figure 2 to simulate expansion. The 15.3 cm of blanket material within each drawer was also moved, as a whole, toward the back of the drawer.

Results:

Measured reactivity change = - 14.3 $\Delta k/k$

Whole core change = - 228. $\Delta k/k$ = - 0.51% $\Delta k/k$

Reference:

G. K. Rusch, et al., "Investigations of a 600-Liter Uranium Carbide Core (ZPR-VI Assembly No. 2)," Reactor Physics Division Annual Report - July 1, 1963 to June 30, 1964, ANL-7010, pp. 91-110 (Jan. 1965).

01 02 03 04 05 06 07 08 09 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29 30 31 32 33 34 35 36 37 38 39 40 41 42 43 44 45

Figure 1

9-2-2 (cont'd)

Figure 2

Type of Measurement: Reflector replacement worths

Experimental Method:

Horizontal slabs of aluminum (45% density), carbon or rhenium were installed above and below the radial surface of the core and extending its full length. Reactivity changes relative to the normal aluminum blanket were measured with calibrated control rods. Various slab sizes were used for each material.

Figure 1 illustrates the geometry where the

Figure 1
Geometry for Material Worths at Core Edge

effective mean radius was determined for each slab from the following expression:

$$\bar{r} = \frac{\int_{\text{slab}} r(x,y) \, dx \, dy}{\int_{\text{slab}} dx \, dy} = \frac{1}{WT} \int_{-W/2}^{+W/2} dx \int_0^T \sqrt{(d+y)^2 + x^2} \, dy$$

9-3-4 (cont'd)

where x and y are in the directions of W and T, respectively.
The effective solid angle is then

$$\bar{\Omega} = 2 \tan^{-1} \frac{W/2}{\bar{r}} \text{ radians}$$

Results:

Table I
Reflector Replacement Worths*

Aluminum Removed, kg	Material Added				$\bar{\Omega}$, radians	$\bar{r}$, cm	$\Delta\rho$, Relative to Aluminum, lh	$\Delta\rho/\text{weight}$, Relative to Void, lh/kg
	d, cm	W, cm	T, cm	Weight, kg				
Aluminum (45% Density)								
11.430	24.89	16.59	2.54	4.984	1.26	26.59	-66.74	+10.35
11.430	27.65	16.59	2.54	4.984	1.19	29.31	-44.61	+6.92
38.101	35.95	27.65	5.08	16.613	1.79	31.89	-40.24	+1.87
53.341	41.48	38.71	5.08	23.258	1.76	45.41	-29.66	+0.99
Carbon								
11.430	24.89	16.59	2.54	7.032	1.26	26.59	+45.55	+13.63
11.430	27.61	16.59	2.54	7.032	1.14	29.27	+37.87	+11.73
22.860	30.42	16.59	5.08	14.064	1.00	33.31	+65.21	+8.95
38.101	35.95	27.65	5.08	23.440	1.79	31.89	+93.14	+5.69
53.341	41.48	38.71	5.08	32.816	1.76	45.41	+98.03	+3.89
Rhenium								
5.715	24.88	16.59	1.28	43.100	1.30	25.96	-133.48	-3.68
5.715	27.03	16.59	1.28	43.100	1.20	28.08	-72.12	-2.11
11.430	30.41	16.59	2.54	86.200	1.04	32.04	-71.72	-1.02
9.525	40.09	27.65	1.28	71.834	1.36	41.50	-23.21	-0.40
13.335	45.62	30.71	1.28	100.567	1.33	47.10	-7.88	-0.12

*1% $\Delta k/k = 430$ lh

Reference:

R. C. Doerner, W. G. Knapp, K. K. Almenas, and R. A. Karem,
"Physics Measurements in Tungsten-Based, Aluminum-Reflected
Fast Reactors", ANL-7007 (March 1967)

Type of Measurement:

Reflector poisoning by boron.

Facility: ZPR-9 (1.3)

Assembly: 6 (2.3-6)

Experimental Method:

The aluminum radial reflector was poisoned by the addition of enriched boron. The boron was contained in 2 x 2 x 1/8 in. aluminum cans. The average welded can weight was determined to be 6.295 ± 0.052 g. Each can was filled with 4.873 ± 0.139 g. of enriched boron or 4.20 ± 0.12 g. of B-10.

The cans were loaded into the reflector in positions shown as a through f (for one octant) in Fig. 1 of Section 2.3-6, and whose effective radii are given in Table I.

Table I
Geometric Properties of Boron Can Positions

<u>Position Designation</u>	<u>Subtended Angle</u>	<u>Mean Radius, cm</u>
a	6.84°	41.7
b	6.72°	42.0
c	6.39°	43.1
d	6.20°	42.3
e	5.73°	42.4
f	5.06°	43.4

a row of 10 cans (20 in) was loaded at each position.

9.3-6 (cont'd)

Results:

Table I
Boron Worths in Reflector**

Run No.	Locations of Boron Addition*	Excess Reactivity, 1h	Reactivity Increment, 1h	Total Worth of Boron, 1h	Subtended Angle Added	Total Subtended Angle	Reactivity Increment/Angle Increment, 1h/deg
22-1	Reference core-- 459.69206 kg of U ²³⁵	112.5	-	0	-	0	-
22-2	31-23 (a) 31-22, 31-24 (b)	87.5	-25.0	-25.0	20.28°	20.28°	-1.23
22-3	30-21, 30-25 (c) 30-20, 30-26 (d)	57.8	-29.7	-54.7	25.18°	45.46°	-1.17
22-4	29-19, 29-27 (e) 29-18, 29-28 (f)	31.3	-26.5	-81.2	21.58°	67.04°	-1.22
22-5	28-17, 28-29 (f) 27-17, 27-29 (e)	5.0	-26.3	-107.5	21.58°	88.62°	-1.21
23-1		180.5	Add two rows of fuel to drawer 22-23, both halves. Total addition: 1.647 kg of U ²³⁵ .				
23-2	26-16, 26-30 (d) 25-16, 25-30 (c)	154.7	-25.8	-133.3	25.18°	113.80°	-1.02
23-3	24-15, 24-31 (b) 22-15, 22-31 (b) 23-15, 23-31 (a)	115.2	-39.5	-172.8	40.56°	154.36°	-0.97
23-4	21-16, 21-30 (c) 20-16, 20-30 (d)	91.2	-24.0	-196.8	25.18°	179.54°	-0.95
23-5	19-17, 19-29 (e) 18-17, 18-29 (f)	68.5	-22.7	-219.5	21.58°	201.12°	-1.05
23-6	17-18, 17-28 (f) 17-19, 17-27 (e)	44.3	-24.2	-243.7	21.58°	222.70°	-1.12
23-7	16-20, 16-26 (d) 16-21, 16-25 (c)	18.1	-26.2	-269.9	25.18°	247.88°	-1.04
24-1		64.5	Add one row of fuel to drawer 22-23, stationary half. Total addition: 0.41172 kg of U ²³⁵ .				
24-2	15-22, 15-24 (b) 15-23 (a)	45.6	-18.9	-288.8	20.28°	268.16°	-0.93
25-1		43.0	Remove one row of fuel from drawers 22-23, both halves, and add one row of fuel to drawers 24-27, both halves. No decrease in U ²³⁵ mass.				
25-2	Remove boron added for Run 24-2	63.5	20.5		-20.28°		-1.01
25-3	Add extra boron can to 11 drawers on each half	54.7	-8.8				
25-4	Restore boron removed for Run 25-2	35.5	-19.2		20.28°		-0.94

*Letters in parentheses refer to position designations in Table VII.

**1% $\Delta k/k = 430.6$ 1h

Reference:

W. G. Knapp and R. C. Doerner, "Physics Measurements with Modified Diluent Compositions in Tungsten-Based Aluminum-Reflected Fast Reactors", ANL-7207 (Feb. 1967)

SECTION 10 - EFFECTS OF HETEROGENEITY

If fast breeder reactors and the critical experiments on which their design is based were homogeneous, fast reactor physics would be at least an order of magnitude simpler than it is. As it is, the experiments and the reactors are just sufficiently heterogeneous for these effects to be significant. Furthermore, they are not heterogeneous in the same way, i.e. the fuel in fast breeders is generally in the form of thin pins surrounded by the cooling medium, whereas the fuel and diluents in critical experiments are generally in the form of plates. In the past this gap has been bridged by correcting the experimental results to an equivalent homogeneous situation which forms the locus of comparison with theoretical results. As analysis becomes more explicit in dealing with heterogeneous situations, the locus of comparison will move to the real experimental situation.

The two most common methods of investigating the effect of a heterogeneous cell structure on criticality have been the measurement of foil activation distributions in cells and the measurement of the reactivity effect of altering the arrangement (e.g. "bunching") of the cell constituents. Such measurements are reported in Sections 10.1 and 10.2 respectively. Sections 10.3 through 10.8 are devoted to experiments dealing with the effect of heterogeneous cell structures on a variety of parameters as follows:

Effect of Heterogeneity on

Section 10.3 - Reactivity Coefficients

10.4 - Reaction Rates

10.5 - Sample Reactivity Worths

10.6 - Neutron Lifetime

10.7 - Leakage (Streaming Effects)

10.8 - Neutron Spectra

SECTION 10.1 - CELL ACTIVATION DISTRIBUTIONS

10.1-1-20

Cell Activation Distributions

Type of Cell: Plate (U-Al-Zr-SS-Mo)

Facility: ZPR-3 (1.1)

Assembly: 20 (2.1-20).

Environment:

A 21-drawer test region, 7 in. deep in each half, was constructed at the core center for each of the configurations shown below.

Experimental Method:

U-235 and U-238 foils were located 1 in. back from the assembly interface across the width of the central drawers.

Results:

The distributions are shown in Figures 1 - 5. The data have been normalized at the left-most point.

Reference:

C. E. Branyan, "Core A Critical Studies for the Enrico Fermi Atomic Power Plant on ZPR-III", ANL-6629, (1962).

Fig. 1 Single-drawer Fission Rate Distribution, Configuration A

Fig. 2 Single-drawer Fission Rate Distribution, Configuration B

Fig. 3 Single-drawer Fission Rate Distribution, Configuration C

10.1-1-20 (Cont'd)

Fig. 4 Single-drawer Fission Rate Distribution, Configuration D

Fig. 5 Single-drawer Fission Rate Distribution, Configuration E

10.1-1-48

Cell Activation and Fission Distributions

Type of Cell: Plate (Pu-U-Na-SS)

Reactions Considered: U-235 (n,f), U-238 (n,f) U-238 (n,γ)

Facility: ZPR-3 (1.1)

Assembly: 48 (2.1-48)

Environment: "normal" cell configuration

Experimental Method:

The measurements were made in the central drawer, 1 in. from the core midplane. Depleted (0.2% U-235) and enriched (93.1%) foils were placed between the plates of the cell. The fission rates were determined by integral gamma counting of the foils (see section 5.1-1-45A).

Results:

See Figures 1, 2, and 3.

Fig. 1 Relative Reaction Rates through Normal Central Drawer

Fig. 2 Relative Reaction Rates through Bunched Central Drawer

10.1-1-48 (cont'd)

Fig. 3 Relative Reaction Rates
through Unbunched Central Drawer

Reference:

J. M. Gasidlo, et al., "Doppler Effect, Heterogeneity, and Sodium-Voiding Experiments on ZPR-3 Assemblies 48 and 48B" ZPR-TM-8 (1969)

10.1-2-2

Cell Activation Distributions

Type of Cell: Plate (U-C-Na)

Facility: ZPR-6 (1.2)

Assembly: 2 (2.2-2)

Reactions Considered:

U-235 and U-238 fission.

Environment:

Same as core.

Experimental Method:

U-235 and U-238 foils (9.27 mm diam. 0.13 mm. thick) were placed as pairs between constituent plates of the drawers located at row 23 and column 22 or 25 on the stationary half. The foils were 10 cm from the interface of the two halves. Subsequent to irradiation, gamma activities above 412 keV were measured. The foils were calibrated by means of fission chambers exposed, along with foils, in the same run.

Foil traverses were measured both with and without sodium-filled stainless steel cans.

Results:

See Figures 1 and 2.

Reference:

G. K. Rusch, et al., "Investigations of a 600-Liter Uranium Carbide Core (ZPR-VI Assembly No. 2)" Reactor Physics Division Annual Report - July 1, 1963 to June 30, 1964, ANL-7010, (1965)

Figure 1
Relative Distribution of Fissions within Core Drawer filled
with Na Filled SS Cans (Central Core Region)

10.1-2-2 (cent'd)

Figure 2
Relative Distribution of Fissions within Core Drawer with
Empty SS (void) Cans. (Central Core Region)

10.1-2-3

Cell Activation Distributions

Type of Cell: Plate (U-C-Na)

Facility: ZPR-6 (1.2)

Assembly: 3 (2.2-3)

Reactions Considered:

U-238/U-235 fission.

Environment:

Same as core with the exception that distributions were also measured with sodium voided from a central region 50 cm long and 50 cm diameter.

Experimental Method:

See Section 10.1-2-2.

Results:

Figure 1
Measured Fine Structure of U-238/U-235 Fission Ratio Across Drawers With and Without Sodium

10.1-2-3 (cont'd)

Reference:

F. H. Helm, et al, "Physics Parameters and Sodium Void Coefficients in a 950-Liter Pancake Core", Reactor Physics Division Annual Report - July 1, 1964 to June 30, 1965, ANL-7110, 101, (1965)

10.1-2-4Z

Cell Activation Distributions

Type of Cell: Plate

Facility: ZPR-6 (1.2)

Assembly: 4Z (2.2-4Z)

Reactions Considered:

U-235 and U-238 fission.

Environment:

Normal drawer configurations in central, buffer, and driver regions.

Experimental Method:

The fine structure of the U-238 fission rate was measured at the core center and across a region consisting of central, buffer, and driver zones.

The fine structure of the U-235 fission rate was measured adjacent to the core center.

Results:

Fig. 1 Fine Structure of U-238 Fission Rate in Central Region

Fig. 2 Fission Rate of U-238 Foils, placed between Plates in ZPR-6 Assembly 4Z

10.1-2-4Z

Cell Activation Distributions

Figure 3
Fine Structure of U-235
Foils, Placed between Plates
in ZPR-6 Assembly 4Z

Reference:

R. A. Karam, et al., "Measured Physics Parameters in a Zoned Fast Uranium Carbide Core, Assembly 4Z", ANL-7281 (1966)

Type of Cell: Plate

Facility: ZPR-6 (1.2)

Assembly: 5 (2.2-5)

Reactions Considered: U-235 and U-238 fission

Environment:

The central 9 by 9 drawers (49.72cm) in each half of the reactor were loaded according to the patterns shown in Figure 1 and measurements were taken. A second set of measurements were done with the central 9 by 9 drawers in the pattern shown in Figure 2.

Fig. 1 Reference Loading Patterns of Three-Drawer Cell for Bunching U-238 in Assembly 5 of ZPR-6

Fig. 2 Bunched U-238 Loading Patterns in Three-Drawer Cell of Assembly 5 of ZPR-6

10.1-2-5 (cont'd)

Experimental Method:

U-235 (.005 in. x .375 in.) and U-238 (.005 in. x .625 in.) foils were placed between every plate in the 3-drawer cell. U-235 foils were located at an axial distance of 2 in. from the core center in one half of the reactor. The U-238 foils were located similarly in the other half.

The counting procedures are described in Section 5.2-2-6.

Results:

Fig. 3 Absolute Fission Rates Per Gram of U-235 in Bunched U-238 Case of Assembly 5 of ZPR-6

Fig. 4. Absolute Fission Rates Per Gram of U-238 in Bunched U-238 Case of Assembly 5 of ZPR-6

10.1-2-5 (cont'd)

Fig. 5 Absolute Capture Rates Per Gram of U-238 in Bunched U-238 Case of Assembly 5 of ZPR-6

Fig. 6 Absolute Fission Rates Per Gram of U-235 in Unbunched U-238 Case of Assembly 5 of ZPR-6

Fig. 7 Absolute U-238 Captures Per U-235 Fission in Bunched U-238 Case of Assembly 5 of ZPR-6

Fig. 8 Absolute Fission Rates Per Gram of U-238 in Unbunched U-238 Case of Assembly 5 of ZPR-6

Fig. 9 Absolute U-238 Fission Per U-235 Fission in Bunched U-238 Case of Assembly 5 of ZPR-6

Fig. 10 Absolute Capture Rates Per Gram of U-238 in Unbunched U-238 Case of Assembly 5 of ZPR-6

10.1-2-5 (cont'd)

Reference:

R. A. Karam, et al., "Measured Physics Parameters in a Large Uranium Carbide Fast Core, Assembly 5 ZPR-6", Reactor Physics Division Annual Report - July 1, 1966 to June 30, 1967, ANL-7310, 167 (1968)

10.1-2-6

Cell Activation Distributions

Type of Cell: Plate

Facility: ZPR-6 (1.2)

Assembly: 6 (2.2-6)

Reactions Considered:

U-235 and U-238 fission. Capture in U-238.

Environment:

Normal cell structure.

Experimental Method:

Depleted uranium foils (0.2% enrichment, 5 mil thick by 5/8 in. diam.) and U-235 foils (93% enrichment, 4.5 mil thick by 3/8 in. diam.) were placed at corresponding locations about the center of the core in the two halves of the ZPR-6 machine. The foils were placed among the adjoining surfaces of all the plates in the unit cell. Foils were counted by the technique described in 5.2-2-6.

In another experiment, a 2 x 2 x 1/16 in. plate of 93% enriched uranium and a 2 x 2 x 1/8 in. plate of depleted uranium (0.2 enriched) that were normally in the core were replaced by packets of foils equivalent in weight and enrichment to the two plates. The packets were irradiated in the manner described earlier. For each of the 2 x 2 x 0.005 in. foils in the packets, a 5/8 in. diameter foil was punched out from the center and counted. This procedure obviated edge effect corrections.

Results:

Refer to Figures 1-4.

10.1-2-6 (cont'd)

Fig. 1 Measured and Calculated Activation Distributions in "Normal" Loading of Assembly 6 of ZPR-6

Fig. 2 Measured and Calculated Activation Distributions in "Bunched" Loading Patterns of Assembly 6 of ZPR-6

10.1-2-6 (cont'd)

Fig. 3 Measured and Calculated Values of Activation Distributions in an "Isolated Stainless Steel" Loading Pattern of Assembly 6 of ZPR-6

Fig. 4 Relative Fissions in U-235 and U-238 Packets and Relative Capture in U-238 Packet Simulating the U-235 and U-238 Plates in Assembly 6 of ZPR-6

Reference:

R. A. Karam, et al., "Heterogeneity Effects in a Large UO_2 Core, Assembly 6 of ZPR-6", Reactor Physics Division Annual Report - July 1, 1967 to June 30, 1968, ANL-7410, 133 (1969)

10.1-3-11

Cell Activation and Fission Distributions

Type of Cell: Plate (U, Na, C)

Facility: ZPR-9 (1.3)

Assembly: 11 (2.3-11)

Reactions Considered: U-238 fission

Environment:

Standard (i.e. same) cell configuration

Experimental Method:

U-238 foils placed between plates of cell.

Results:

Figure 1
Measured Radial Fine Structure of the U-238 Fission Rates

Reference:

A. Ancarani, "Fission Rates in a Zoned-Core Fast Critical Assembly" Trans. Am. Nucl. Soc. 10, 1, 275 (1967).

SECTION 10.2 - REACTIVITY EFFECT OF HETEROGENEITY

Type of Cell: Plate (U-SS)

Facility: ZPR-3 (1.1)

Assembly: 32 (2.1-32)

Type of Measurement: Fuel plate bunching

Method of Measurement:

In a representative wedge of the core the following cell structures were created:

A: 61'6'6666'1'66661'6'6666'1'6

B: (reference)6661666666661666

C: 6666666116656666

where 1 = U-235, 6 = SS (304), the primed numbers indicate a plate thickness of 0.1587 cm, and the unprimed, 0.3175 cm (see section 1.1). Reactivity was measured with a calibrated control rod.

Results:

The reactivity worths of changing the entire core from the reference structure to A and C were:

B $\rightarrow$ A = -227 Ih

B $\rightarrow$ C = +553 Ih

Extrapolation to the full core was made merely by dividing by the fraction of drawers used in the experiment.

Reference:

P. I. Amundson, et al., "Critical Studies of Uranium-Steel and Uranium-Steel-Sodium Fast Reactor Cores (ZPR-III Assemblies 32 and 33)" ANL-6690 (1963).

Type of Cell: Plate

Facility: ZPR-3 (1.1) Assembly: 42 (2.1-42)

Type of Measurement: Plate orientation effects

Method of Measurement:

- A. The plates in drawers at locations L-14, L-18, R-14, and R-18 in both halves were rotated 90° . The effective radius at these positions is 6.17 in.
- B. The plates in drawers at locations I-15, I-16, I-17, U-15, U-16 and U-17 were rotated 90° . The effective radius is 13.22 in.

Results:

- A. -1.26 Ih (for the total of 9.2 kg of enriched U rotated)
- B. +2.36 Ih (for the total of 27.6 kg enriched U rotated)

Reference:

Amundson, P. I., et al., "A Two-Zone Fast Critical Experiment," ANL 6733 (1964).

10.2-1-42b Reactivity Effects of Heterogeneity

Type of Cell: Plate

Facility: ZPR-3 (3.1) Assembly: 42 (2.1-42)

Environment: Similar cells

Type of Measurement: Fuel plate bunching effects.

Method of Measurement:

A. Core: The cell structure in a wedge of the central zone of the core containing 10 elements was altered in two steps ("bunched" and "unbunched"). In addition to the reference structure (See Section 2.1-42) the following cells were considered:

bunched: 6554327117345566

unbunched: 6551'724334271'5566

where 1 = U-235, 2 = U-238, 3 = Al, 4 = 63% (perforated Al, 5 = 45% Al, 6 = Stainless Steel (304) and 7 = Graphite. All plates are 0.318 cm thick except 1' which are 0.159 cm thick (see 1.1).

B. Driver: A representative 6 drawer section was bunched and unbunched in the following way (see Section 2.1-42 for the reference cell structure):

bunched: 7672767117677677

unbunched: 71'67761'7271'67761'77

where the numbering is defined above.

Results:

The reactivity worths of bunching and unbunching relative to the reference condition and extrapolated linearly to the total size of the regions are:

	<u>Unbunched</u>	<u>Bunched</u>
Core	-121.8 Ih	+193 Ih
Driver	- 44.6 Ih	+158
Total	-166.4 Ih	+351 Ih

10.2-1-42b (cont'd)

Extrapolation to the homogeneous (i.e. zero plate thickness) situation indicates a correction of -335 lb or -13.4 kg U-235 from the reference condition, this change in amount of fuel occurring at the outer boundary of the driver region.

Reference:

P. I. Amundson, et al., "A Two-Zone Fast Critical Experiment"
ANL-6733 (1964)

Type of Cell: Plate

Assembly: 48 (2.1-48)

Facility: ZPR-3 (1.1)

Experimental Method:

The fuel plates were bunched and unbunched at the center and edges of the core and the reactivity measured with a calibrated control rod. Figures 1 and 2 show the location of the drawers used in the experiments.

Figure 2 Drawers Used in Heterogeneity Experiments in Half No. 2 of Assembly 48 of ZPR-3

Figure 1 Drawers Used in Heterogeneity Experiments in Half No. 1 of Assembly 48 of ZPR-3

The central core measurements were made in the front 4 in. of the drawers marked A and B. Measurements at the axial core edge were made in the outer 4 in. of core material in the drawers marked A and B. Measurements at the radial core edge were made in the front 4 in. of the drawers marked C and D. Figure 3 is an end view of the arrangement of material in the 4 in. section of the core drawers used in the experiment.

Figure 3 Material Arrangements in Core Drawers for Heterogeneity Experiments with ZPR-3 Assembly 48

Results:

<u>No. of Drawers</u>	<u>Experiment</u>	<u>Core Location</u>	<u>Drawer (a) Designation</u>	<u>Change in (b) Reactivity (Ih)</u>
9	Bunching	Center	A (Half #1)	16.1 ± 1.0
9	Bunching	Center	A (Half #2)	17.1 ± 1.0
24	Bunching	Center	B	41.7 ± 1.0
42	Bunching	Center	A and B	75.2 ± 1.0
42	Bunching	Axial Edge	A and B	16.8 ± 1.0
18	Unbunching	Center	A	-8.3 ± 1.0
24	Unbunching	Center	B	-11.5 ± 1.0
42	Unbunching	Center	A and B	-20.0 ± 1.0

(a) All measurements made in both halves of reactor, except where stated otherwise.

(b) $1\% \Delta k/k = 1002.5 \text{ Ih}$.

Related Measurements:

Sodium void coefficient (6-1-48).

Reference:

J. M. Gasidlo et al, "Doppler Effect, Heterogeneity, and Sodium-Voiding Experiments in ZPR-3 Assemblies 48 and 48B", ZPR-TM-8 (1969)

10.2-2-1

Reactivity Effect of Heterogeneity

Type of Cell: Plate

Facility: ZPR-6 (1.2)

Assembly: 1 (2.2-1)

Type of Measurement:

Fuel plate bunching and unbunching

Method of Measurement:

Bunching was accomplished by combining two 1/8-inch fuel columns in a drawer into one 1/4-inch fuel column in the center of the drawer. Unbunching was accomplished by splitting the two 1/8-inch fuel columns per drawer into four 1/16-inch fuel columns and evenly distributing them in the drawer. Several different experiments were performed:

A. Bunching experiments

- (1) All drawers in one quadrant of the core symmetrical to its horizontal midplane were bunched (Section 1 of Figure 1)
- (2) All drawers in one quadrant of the core symmetrical to the vertical midplane were bunched (Section 2 of Figure 1).
- (3) All drawers in the horizontal midplane of the core were bunched (Section 3 of Figure 1).
- (4) All drawers in the vertical midplane of the core were bunched (Section 4 of Figure 1).

B. Unbunching experiments

- (1) All drawers in the core excepting those in the quadrant of A-(1) above were unbunched.
- (2) All drawers in the core excepting those in the quadrant of A-(2) above were unbunched.

C. Other experimentation

- (1) The fuel columns in the six central drawers were unbunched by a separation of the 1/8-inch fuel columns into 1/32-inch fuel columns.

10.2-2-1 (cont'd)

Figure 1 Cross Section through the Cylindrical Core of ZPR-6 Assembly 1. Bunched fuel zones are cross hatched.

Results:

<u>Experiment</u>	<u>Reactivity Effect (Δk)</u> <u>(480 Δk = 1% $\Delta k/k$)</u>
A- (1)	+ 97 $\pm$ 5
A- (2)	+112 $\pm$ 5
A- (3)	+57.5 $\pm$ 3
A- (4)	+78.8 $\pm$ 3
B- (1)	-55.8 $\pm$ 5
B- (2)	-58.5 $\pm$ 5
C- (1)	-14.1 $\pm$ 3

Reference:

"Reactor Development Program Progress Report - January 1964"
ANL-6840, pp. 17-19 (1964)

10.2-2-5

Reactivity Effect of Heterogeneity

Type of Cell: Plate

Facility: ZPR-6 (1.2)

Assembly: 5 (2.2-5)

Type of Measurement:

Fuel plate bunching

Method of Measurement:

The normal core cell pattern was changed to the reference pattern shown in Figure 1 of 10.1-2-5 and the reactivity change was measured. The reference pattern was then changed in steps to that shown in Figure 2 of 10.1-2-5. Control rod loadings were unchanged.

Results:

The reactivity change in going from the normal cell pattern to the reference pattern was -13.4 ± 0.4 lh.

The reactivity changes in going from the reference pattern to the bunched core is shown in Table I.

Table I
Reactivity Change Due to Bunching
U-238 in Assembly 5 of ZPR-6

$R_i - R_o^{(a)}$, cm	No. of Drawers	Reactivity Change ^b , lh
0.0-9.35	18	$+17.5 \pm 0.4$
9.35-15.85	32	$+10.6 \pm 0.4$
15.85-21.95	41	$+21.6 \pm 0.4$
21.95-28.45	63	$+40.9 \pm 0.4$

^a R_i and R_o are the equivalent inner and outer radii of cylindrical rings.

^b The bunching in a ring was performed around a bunched region. $1\% \Delta k/k = 476$ lh.

Reference:

R. A. Karam, "Measured Physics Parameters in a Large Uranium Carbide Fast Core, Assembly 5 ZPR-6" Reactor Physics Division Annual Report - July 1, 1966 to June 30, 1967, ANL-7310 167, (1968)

10.2-2-6a

Reactivity Effect of Heterogeneity

Type of Cell: Plate

Facility: ZPR-6 (1.2) Assembly: 6 (2.2-6)

Type of Measurement: Bunching

Experimental Method:

The reactivity change associated with the change in loading pattern from normal (see Fig. 1 of 6-2-6) to that shown in Fig. 1 of 10.3-2-6b was measured. The changes were made in a central region of the core comprising 76 drawers in each half (see Fig. 1)

12 D.P. CONTROL ROD (CONTROL ROD DRAWERS WERE NOT BUNCHED)

13 INSERTION SAFETY ROD (DARK AREA REPRESENTS)

Figure 1 Regions in Assembly 6
of ZPR-6 D.P. Control Rod Drawers Were Not Bunched

The change was also made in the central 9 drawers per assembly half.

The change in reactivity corresponding to the change in loading pattern from normal to that shown in Fig. 2 of 10.3-2-6b was measured. The change was limited to the central 25 drawers per half in a square region.

Results:

See Table I.

10.2-2-6a (Cont'd)

Table I

<u>Cell Pattern</u>	<u>No. of Drawers per Half</u>	<u>Worth (Ih) *</u>
Normal (Fig. 1 of 6-2-6)		0
Bunched (Fig. 1 of 10.3-2-6b)	76	93.06
	9	11.53
Isolated SS (Fig. 2 of 10.3-2-6b)	25	4.47

* $1\% \Delta k/k = 458 \text{ Ih}$

Reference:

R. A. Karam, K. D. Dance, L. R. Dates, W. Y. Kato, J. E. Marshall, T. Nakamura, and G. K. Rusch, "Heterogeneity Effects in a Large UO_2 Core, Assembly 6 of ZPR-6," Reactor Physics Division Annual Report - July 1, 1967 to June 30, 1968, ANL-7410, pp. 133-140 (Jan. 1969).

10.2-2-6b Reactivity Effect of Heterogeneity

Type of Cell: Plate, rod, and homogeneous

Facility: ZPR-6 (1.2) Assembly: 6 (2.2-6)

Type of Measurement: Reactivity effect of cell geometry and environment changes.

Experimental Method:

The relative reactivity worths of nominal 2 in. x 2 in. $\approx$ 2 in. plate, rod, and homogeneous samples were measured. The compositions of all the samples were essentially the same (see Table I). The plate-type sample had the normal loading pattern shown in Fig. 1 of 6-2-6.

Table I
Samples Used In Heterogeneity Effects Study

Sample		Composition, g						Enrichment of ^{235}U , w/o U-238
No.	Type	Stainless Steel		Fe	Total SS + Fe	Na	UO_2	
		In Sample	Can					
1	Homogeneous	36.00	46.10	45.71	189.81	—	392.87	16.42
2	"	37.97	46.18	45.72	189.87	32.80	392.89	16.42
3	"	—	46.30	—	46.30	—	392.89	16.42
4	"	—	46.30	—	46.30	—	392.87	16.42
5	"	—	46.05	—	46.05	42.17	392.87	16.42
6	Plate	40.39	46.00	—	186.39	41.23	392.63	16.42
11a	"	61.17	46.00	—	107.17	—	392.40	16.42
11b	"	61.17	46.00	32.31	139.48	—	392.40	16.42
11c	Plate	45.18	46.08	46.28	137.51	42.91	399.29	16.41
11d	"	45.18	46.04	46.28	137.51	—	399.29	16.41
21a	"	45.33	31.25	46.88	123.20	—	393.30	16.40
21b	"	45.33	31.25	46.88	123.20	42.91	393.30	16.40

The samples, method of measurement, and types of environments are described in Section 10.3-2-6a.

Precision:

The errors listed reflect the relative errors associated with the oscillation method and not the period method.

10.2-2-6b (cont'd)

Results:

Table II
Measured Reactivity Worths at Center of Assembly 6 of ZPR-6

Sample No.	Sample Type	Sample Worth Relative to Void, β	
		Normal Plate Environment	Pellet Environment
1	Homogeneous with 20, no Na	1.591 ± 0.003	1.370 ± 0.002
11b	Pellet, no Na	1.331 ± 0.004	1.345 ± 0.003
11b	Plate, no Na	1.309 ± 0.005	—
20a	Plate, no Na	—	1.307 ± 0.002
2	Homogeneous with Na and 20	1.301 ± 0.002	1.273 ± 0.003
20a	Plate, with Na	1.330 ± 0.003	1.307 ± 0.003
2	Pellet, with Na	1.317 ± 0.003	1.300 ± 0.003

Reference:

R. A. Karam, et al., "Heterogeneity Effects in a Large UO_2 Core, Assembly 6 of ZPR-6" Reactor Physics Division Annual Report - July 1, 1967 to June 30, 1968, ANL-7410, 133 (1969)

10.2-3-4**Reactivity Effect of Heterogeneity****Type of Cell :** Plate**Facility :** ZPR-9 (1.3)**Assembly :** 4 (2.3-4)**Type of Measurement:** Fuel plate "bunching"**Experimental Method:**

The 0.0625 in. thick fuel columns in two octants of the core were bunched into 0.125-in and 0.1875- in thick columns, with the resulting change in reactivity determined by a calibrated control rod. The full core reactivity change was obtained by assuming a linear scaling factor between the octant measurement and full core condition

Results :**Table I.**

<u>Fuel Thickness (in.)</u>	<u>No. of fuel Columns per Drawer</u>	<u>Center-to-Center Spacing between Fuel Columns (in.)</u>	<u>Reactivity Added for Fuel Core</u>
0.0625	6	0.922	0
0.125	3	1.843	+ 0.66
0.1875	2	2.765	+ 2.06

* The width of an average matrix tube divided by the number of fuel columns contained within it.

Reference :

R.C. Doerner, W.G., Knapp, K.K. Almenas, and R.A. Karam, "Physics Measurements in Tungsten-Based, Aluminum- Reflected Fast Reactors," ANL-7007 (March, 1967)

10.2-1-6

Reactivity Effect of Heterogeneity

Type of Cell: Plate

Facility: ZPR-9 (1.1)

Assembly: 6 (2.2-6)

Type of Measurement: Fuel plate "bunching and unbunching"

Experimental Method:

Fuel in the upper east octant of the stationary half, and the lower west octant of the movable half, were loaded with 1/16 in. fuel columns consisting of pairs of 1/32 in. fuel plates. The remaining octants of both halves were loaded with 1/16 in. plates. In subsequent measurements, the fuel in these zones was either distributed into more homogeneous geometry or bunched into thicker fuel columns. See Figure 1.

The fuel columns made from the 1/32 in. fuel plates, when placed together to make an equivalent 1/16 in. column of fuel, were some 8 1/4% lighter in fuel content than a normal 1/16 in. fuel plate. The reactivity loss associated with the smaller fuel density in the two test-zone regions accounted for a 20-kg increase in the critical mass over that with full density fuel.

The reactivity was determined by the difference in critical rod positions. The rod worths were essentially the same for the "bunched" and the "unbunched" situations.

Figure 1

ZPR-9 Assembly No. 6 Core Geometry for Heterogeneity Measurements

10.2-3-6 (cont'd)

Results:

Note: $1\% \Delta k/k - 430.6$ Ih

Table I
Fuel-bunching Measurements in ZPR-9 Assembly No.6

Remarks	1/32-in. Fuel Plates in Test Zones						0.446 kg of ^{235}U Removed				1/32-in. Fuel Plates in Test Zones				1/16-in. Fuel Plates			
	32	33	34	35	36	37 ^a	38 ^b	39	40	41 ^c	42 ^d	43 ^e	44 ^f	45 ^g	46 ^h	47	48	49
Fuel Element Upper East Stationary	1/16	1/32	1/32	1/8	1/8	1/8	1/8	1/8	1/8	$\frac{1}{8} + \frac{1}{16}$	1/8	1/16	1/16	1/16	1/16	1/16	1/8	1/16
Brigance Lower East Stationary	-	-	-	-	-	-	-	-	-	-	-	-	-	1/16	-	-	-	-
In Test Zone Lower West Stationary	1/16	1/16	1/32	1/32	1/8	1/8	1/8	1/8	1/16	$\frac{1}{8} + \frac{1}{16}$	1/16	1/16	1/32	-	1/16	1/8	1/8	1/16
Total Excess Reactivity	99.5	54.4	59.2	100.7	156.6	156.0	40.1	102.0	156.5	62.5	170.7	60.2	153.0	175.0	76.5	134.5	152.4	79.7

^aVertical column 1231 interchanged with horizontal row 1231 in lower west movable column.

^bSame geometry as 38, eight half drawers removed from core edge 1-0.446 kg of ^{235}U .

^cCenter-to-center.

^dSame geometry as 40, eight half drawers added 1-0.446 kg of ^{235}U .

^eSame as 37.

^fReverse of 38.

^gLower east quadrant of stationary fuel used as second zone.

^hReplaced both zones with 1/16-in. fuel plates.

Reference:

W. G. Knapp and R. C. Doerner, "Physics Measurements with Modified Diluent Compositions in Tungsten-Based Aluminum-Reflected Fast Reactors", ANL-7207 (Feb. 1967)

SECTION 10.3 - EFFECT OF HETEROGENEITY ON REACTIVITY COEFFICIENTS

10.3-2-3

Effect of Heterogeneity on Reactivity Coefficients

Type of Cell: Plate

Facility: ZPR-6 (1.2)

Assembly: 3 (2.2-3)

Reactivity Coefficient: Sodium void

Experimental Method:

Heterogeneity effects were studied at the core center in a test region which contained 21 drawers in each half of the assembly. See Figure 1. The axial length of the region was either 30.5 cm or 50.8 cm (through the whole core).

Procedure was as follows:

- (1) Establish critical rod position for drawer configuration being studied.
- (2) Replace cans containing Na by stainless steel frames, find critical rod position.
- (3) Determine worth of stainless to correct for slight difference (10%) between cans and frames.

Figure 1

Precision:

Results were reproducible within 0.5 lh with isolated discrepancies up to 2.5 lh.

10.3-2-3 (cont'd)

Results:

Refer to Table I and Figure 2.

Table I			
<u>Material Arrangement^a</u>	<u>Axial Length of Void (cm)</u>	<u>Fig. 2</u>	<u>Th/kg</u>
Different materials next to sodium:			
Normal drawer loading	30.5	a	-1.31
Sodium predominantly next to U-235 and C	30.5	b	-1.48
Sodium predominantly next to U-238	30.5	c	-1.25
Modified drawer loading where two sodium columns are surrounded by U-238 and two mostly by C:			
Sodium voided next to U-238	30.5	d	-1.20
	50.8		-3.10
Sodium voided next to C	30.5	e	-1.97
	50.8		-4.06
All sodium voided	30.5	f	-1.34
	50.8		-3.59

^aIf not otherwise noted the drawers in the movable and in the stationary halves were loaded symmetrically, so that all material columns were aligned at the interface.

Figure 2 Cell Configuration

10.3-2-3 (cont'd)

Reference:

F. H. Helm, et al., "Physics Parameters and Sodium Void Coefficients In a 950-Liter Pancake Core", Reactor Physics Division Annual Report - July 1, 1964 to June 30, 1965, ANL-7110, 101, (1965)

10.3-2-5

Effect of Heterogeneity on Reactivity Coefficients

Type of Cell: Plate

Facility: ZPR-6 (1.2)

Assembly: 5 (2.2-5)

Reactivity Coefficient: Sodium void

Experimental Method:

The worth of sodium in a homogeneous sample was compared to that in a heterogeneous sample. Samples were the same as those in Table I of 10.5-2-5. In addition, two depleted uranium carbide (0.306% enrichment) samples, one with and one without sodium, were used. The compositions of the samples are given in Table I.

Table I
Description of Samples Used in Heterogeneity Effects Study

Sample No.*	Physical Description	Composition					
		U-235	U-238	C	Na	Stainless Steel	
						Powder	Can
9	Homogeneous	2.09	680.24	36.25	0.00	0.00	71.58
10	Homogeneous	2.08	679.69	36.22	47.74	0.00	69.40
11	Heterogeneous	1.40	700.02	25.00	48.00	—	73.25
12	Heterogeneous	1.40	700.02	25.00	0.000	—	73.25

* These samples were 1.875 x 1.875 x 1.875 in.

Figure 1 shows the arrangement of the heterogeneous samples. The measurements were carried out in both the normal core environment and the bunched environment. (Fig. 2 of 10.1-2-5)

10.3-2-5 (cont'd)

Figure 1
Depleted Uranium Heterogeneous Loading Pattern

Results:

Table II
Heterogeneity Effects on the Sodium-Void Coefficient

Samples Used in Measurement	Description of Sample ^a	Worth of Na ^b , lb/kg	
		Loading of Surrounding	
		Normal ^c	Enriched ^d
1 & 2	Enriched heterogeneous	1.108 ± 0.046	1.148 ± 0.062
3 & 4	Enriched homogeneous	1.717 ± 0.070	1.089 ± 0.079
5 & 6	Enriched homogeneous with SS Powder	1.997 ± 0.038	1.198 ± 0.079
7 & 8	Pellets in SS assembly	1.026 ± 0.043	—
9 & 10	Depleted homogeneous	2.514 ± 0.073	—
11 & 12	Depleted heterogeneous	1.283 ± 0.060	—

^aSee Table 1 of 10.5-2-5 for further description of samples

^b1% Δk/k = 476 lh

^cLoading of surrounding is normal core loading.

^dLoading of surrounding is shown in Fig 2 of 10.1-2-5.

Reference:

R. A. Karam, et al., "Measured Physics Parameters in a Large Uranium Carbide Fast Core, Assembly 5 ZPR-6," Reactor Physics Division Annual Report - July 1, 1966 to June 30, 1967
ANL-7310, 189 (1968)

10.3-2-6a

Effect of Heterogeneity on Reactivity Coefficients

Type of Cell: Plate, rod and homogeneous

Facility: ZPR-6 (1.2)

Assembly: 6 (2.2-6)

Reactivity Coefficient: Sodium void versus cell type

Experimental Method:

The sodium void coefficient was measured in homogeneous rod, and plate samples at the center of the core. The measurements consisted of oscillating "matched" samples, one with and one without sodium, relative to each other. The samples were nominally 2 in. x 2 in. x 2 in.

The compositions of the samples are given in Table I.

Table I
Composition of Matched Samples

Sample		Composition, g					Enrichment of ^{235}U , w/o ^{238}U	Remarks
No.	Type	SS In-Sample	Can	Fe	Total SS + Fe	Na	UO_2	
1	Homogeneous	54.00	40.30	45.74	139.54	—	392.57	1 & 2 matched
2	Homogeneous	57.57	40.18	45.72	143.47	42.50	392.40	
4	Homogeneous	—	40.30	—	40.30	—	392.57	1 & 3 matched
9	Homogeneous	—	40.05	—	40.05	40.17	392.57	
8	Pellet	60.39	40.00	—	104.39	41.33	392.45	5 & 11a matched
11a	Pellet	61.67	40.00	—	107.17	—	392.40	
11b	Pellet	61.17	40.00	32.31	133.48	—	392.40	11a & 11b matched
11a	Plate	45.18	40.08	40.26	125.54	42.91	390.29	
11b	Plate	45.18	40.08	40.26	125.54	—	390.29	21a & 21b matched
21a	Plate	45.25	34.25	40.88	120.38	—	403.34	
21b	Plate	45.45	34.25	40.88	120.58	42.91	403.30	21a & 21b matched
21b	Plate	45.45	34.25	40.88	120.58	—	403.30	

All samples were encased in 1.90 x 1.90 x 2.00 in. SS (304) cans with wall thickness of 0.015 in. The homogeneous samples were mixtures of UO_2 (16.4% enrichment) powder and sodium or UO_2 powder, sodium, and stainless steel powder--the latter to

10.3-2-6a (cont'd)

mock-up the stainless steel tubes in the rodded samples. The stainless steel and UO_2 powders were blended together and sintered; the particle size of the blend was such that only about 53% of the volume would be used when packed under vibration. The remaining volume fraction was filled with sodium. Tests conducted with x-rays and an actual opening of one can, revealed that the mixture was quite homogeneous. The rodded samples consisted of 0.25 in. diameter by 1/2-5/8 inch long pellets inserted in 36-tube calandria cans. The pellets consisted of two enrichments, 19.9% and 15.9%, pro-rated to give an average enrichment of 16.43%. A test of the reactivity worth of UO_2 with each enrichment and the two combined showed that the assumption of linear dependence was adequate. See Figure 1 for rodded sample diagrams.

SPACES BETWEEN TUBES FILLED WITH SODIUM

PELLET COLUMN IDENTIFICATION

A. 6-15.881% ENR. + 1-15.947% ENR.

B. 6-15.831% ENR.

MT. NO PELLETS

Figure 1
Square Calandria Can for Rodded Zone

10.3-2-6a (cont'd)

The measurements were conducted as follows: the reactivity of a sample was measured relative to void by the period method (at least 5 times); the reactivity relative to other samples was measured using an oscillator in conjunction with an autorod.

For some measurements a rodged central zone was constructed of 26 (2.00 x 2.00 x 3.17 in.) calandria cans (Fig. 1) around the sample being oscillated. This rodged zone comprised the first 3 in. of 13 drawers in each half of the assembly. The relative reactivity worth of the plate, rodged, and homogeneous samples were measured in the rodged zone (or "pellet") environment as well as the plate environment.

Fig. 2 Normal Drawer Loading Pattern

Fig. 3 Bunched Drawer Loading Pattern

Results:

Table II
Heterogeneity Effects on Na Void Coefficient
in Assembly 6 of ZPR-6 (16.42% Enrichment)

Sample Type	Na Void Coefficient, 1h/kg	
	Plate Environment	Pellet Environment
Plate heterogeneous (normal loading pattern, Fig. III-13-1a)	+0.147 ± 0.079	-0.014 ± 0.070
Bunched plate heterogeneous (Fig. III-13-1b)	+0.047 ± 0.040	—
Pellet heterogeneous	+0.189 ± 0.027	+0.302 ± 0.039
Homogeneous without SS powder	+0.821 ± 0.020	+0.888 ± 0.020
Homogeneous with SS powder ^a	+0.199 ± 0.025	+0.221 ± 0.017

^a Stainless steel powder (304L) was added to mock up the SS in the tubes housing the pellets.

10.3-2-6a (cont'd)

Reference:

R. A. Karam, et al., "Sodium Void Effects in a Large UO_2 Core Assembly 6 of ZPR-6" Reactor Physics Division Annual Report - July 1, 1967 to June 30, 1968, ANL-7410, 125 (1969)

10.3-2-6b

Effect of Heterogeneity on Reactivity Coefficients

Type of Cell: Plate

Facility: ZPR-6 (1.2)

Assembly: 6 (2.2-6)

Reactivity Coefficient: Sodium void versus loading pattern

Experimental Method:

The sodium void coefficient was measured in the 18 central drawers with the loading patterns shown in Figure 1 and Figure 2, as well as the normal loading pattern (see Fig. 1 of 6-2-5). All loading patterns extended from the center of the core to a radius of 27.4 cm, some 19 cm beyond the region where the actual voiding took place.

Fig. 1 Bunched Drawer Loading Pattern

Fig. 2 Isolated SS Loading Pattern

In each case the measurement comprised removal of the sodium from the front 6 in. of the entire axial length (30 in.) in each half of the reactor.

10.3-2-6b (cont'd)

Results:

Table I
Sodium-Void Coefficient Dependence
on Loading Patterns

Section Voided	Amount of Sodium, kg	Loading Pattern	Na-Void Coefficient, 1h/kg ^(a)
Front 6 in.	2.835	Normal loading pattern	-0.060
Front 6 in.	2.835	Bunched case	-0.736
Front 6 in.	2.648	Isolated SS case	-0.116
30 in.	14.401	Normal case	-0.861
30 in.	14.401	Bunched case	-1.473
30 in.	13.424	Isolated SS case	-1.101

^a 1% $\Delta k/k = 45 \pm 1$ h.

Reference:

R. A. Karam, et al., "Sodium Void Effects in a Large UO₂ Core Assembly 6 of ZPR-6" Reactor Physics Division Annual Report - July 1, 1967 to June 30, 1968, ANL-7410, 125 (1969)

SECTION 10.4 - EFFECT OF HETEROGENEITY ON REACTION RATES

10.4-1-29 Effect of Heterogeneity on Fission Ratios

Type of Cell: Plate

Facility: ZPR-3 (1.1)

Assembly: 29 (2.1-29)

Experimental Method:

The U-236 / U-235 fission ratio was measured at the core center with fuel in the central core region in three configurations.

<u>Configuration</u>	<u>Cell Structure</u>
I (normal)	6A2AA1AA2A665555
II (unbunched)	A2AlaA6556A2AlaA655
III (unbunched and seeded)	As II with A2AlaA6556A2AlaA655 replaced with 1/2 plate U-238 and 1/2 plate U-235

where 1 and 1a = U-235, 2 = U-238, 5 = 45% (perforated) Al,
6 = SS (304) and A = Al_2O_3 . All plates are 0.318 cm thick
except 1a which are 0.159 cm thick. (see section 1.1).

Results:

<u>Configuration</u>	<u>U-236 / U-235 count ratio</u> ^(a)
I	0.110 $\pm$ 0.002
II	0.111 $\pm$ 0.002
III	0.111 $\pm$ 0.002

(a) raw data - uncorrected for isotopic content of chambers.

Reference:

A. L. Hess, et al., "Critical Studies of a 450-Liter Uranium Oxide Fast Reactor Core (ZPR-III Assembly 29)" ANL-6336 (1961).

Facility: ZPR-3 (1.1)Assembly: 38 (2.1-38)Type of Measurement:

Effect of local plate environment on U-238/U-235 fission ratios as measured by Kirn absolute fission chambers (See Section 5.1-1-2).

Results:

<u>Case</u>	<u>Fuel Arrangement^a</u>	<u>Uncorrected U-238/U-235 Fission Ratio</u>
1.	2 x 1/16 in. plates symmetrical in counter drawers. (Normal arrangement for counter measurements)	0.0308
2.	2 x 1/16 in. plates both on one side of counter drawers.	0.0318
3.	4 x 1/16 in. plates in counter drawers; 9 central drawers homogenized	0.0309
4.	2 x 1/8 in. plates in counter drawers; 9 central drawers homogenized.	0.0307

^aSee Section 2.1-38.

Thus homogenization of surrounding drawers, and the number of fuel plates in the counter drawer have little effect on the measured ratio whereas an asymmetrical arrangement (case 2) can have a 3% effect.

Reference:

J. C. Bates, et al., "Two Spherical Fast Critical Experiments (ZPR-III Assemblies 38 and 39)", ANL-6582 (1962)

SECTION 10.5 - EFFECT OF HETEROGENEITY ON SAMPLE REACTIVITY WORTHS

Type of Cell: Plate

Facility: ZPR-6 (1.2)

Assembly 5 (2.2-5)

Experimental Method:

The worths of plate, rod and homogeneous samples were compared. Sample reactivities were measured using the period and oscillator methods. Figure 1 shows plate-type sample and Figure 2 shows pellet-type sample.

Fig. 1 Heterogeneous Plate Sample Loading Pattern

Fig. 2 Pellet Sample Loading Pattern

Compositions of the various samples are given in Table I.

10.5-2-5 (cont'd)

Table I
Description of Samples Used in Heterogeneity Effects Study

Sample No.	Physical Description	Composition g					
		U-235	U-238	C	Na	Stainless Steel	
						Powder ^a	Can ^b
1	Heterogeneous ^c	82.95	635.78	25.00	0.00	49.51 ^d	23.74
2	Heterogeneous	82.95	635.78	25.00	48.00	49.65 ^d	23.74
3	Homogeneous	107.31	739.68	45.05	0.00	0.00	24.47
4	Homogeneous	108.96	751.01	45.74	34.36	0.00	43.37
5	Homogeneous	86.63	597.08	36.36	0.00	60.14	30.08
6	Homogeneous	87.86	605.59	36.88	44.44	61.01	29.08
7	Pellets	86.73	597.83	33.47	0.00	59.96 ^e	54.52
8	Pellets	87.38	602.30	33.71	42.29	59.90 ^e	54.56

^a Some samples had SS powder (type 304) mixed with the UC and sodium.

^b All samples were canned in SS (type 304).

^c These samples were 1.875 x 1.875 x 1.875 in.

^d This quantity of SS was in the form of cans (2 x 2 x 1/2 in.) some of which were filled with sodium.

^e The SS tubes weigh about 60 g; it is not powder.

The samples were measured in two environments: (1) the normal core loading and (2) the bunched loading, Fig. 2 of 10.1-2-5 in the 8 drawers per half surrounding the oscillator drawer.

Results:

Table II
Measured Heterogeneity Effects At the
Center of the Assembly 5 of ZPR-6

Sample No. ^a	Description of Samples	Worth of UC ^b , lh/kg		Worth of (UC + Na + SS) ^c Mix, lh/kg	
		Loading of Surrounding		Loading of Surrounding	
		Normal ^d	Bunched ^e	Normal	Bunched
1	Heterogeneous without Na	3.436 ± 0.070	3.301 ± 0.009	—	—
2	Heterogeneous with Na	—	—	2.698 ± 0.010	2.644 ± 0.008
3	Homogeneous without Na	3.027 ± 0.007	2.982 ± 0.007	—	—
5	Homogeneous with SS powder and Na	—	—	2.234 ± 0.006	2.221 ± 0.008
6	Homogeneous with SS powder and no Na	2.991 ± 0.011	2.932 ± 0.012	—	—
7	UC pellets in SS assembly without Na	2.970 ± 0.009	—	—	—
8	UC pellets in SS assembly with Na	—	—	2.227 ± 0.007	—

^a See Table III-6-IV for further description of the composition of the samples.

^b Reactivity worth adjusted relative to sample 6.

^c Reactivity worth adjusted relative to sample 5.

^d Loading of surroundings shown in Fig. III-6-3.

^e Loading of surroundings shown in Fig. III-6-5.

10.5-2-5 (cont'd)

Reference:

R. A. Karam, et al., "Measured Physics Parameters in a Large Uranium Carbide Fast Core, Assembly 5 ZPR-6" Reactor Physics Division Annual Report - July 1, 1966 to June 30, 1967, ANL-7310, 167 (1968)

SECTION 10.6 - EFFECT OF HETEROGENEITY ON LIFETIME

**SECTION 10.7 - EFFECT OF HETEROGENEITY ON LEAKAGE (STREAMING
EFFECTS)**

10.7-2-5

Effect of Heterogeneity on Leakage (Streaming Effect)

Type of Cell: Plate

Facility: ZPR-6 (1.2)

Assembly: 5 (2.2-5)

Experimental Method:

The streaming effect on the sodium-void coefficient was investigated in the following manner:

1. The drawers in each half of the matrix, loaded with the unbunched loading pattern of the central drawer of the three-drawer cell (middle drawer shown in Fig. 1 of Section 10.1-2-5) were used. These drawers were at the center of the core and their positions in the matrix were 24-23, 23-23 and 22-23 in both halves of the reactor. Matrix position 23-23 is the center of the core.

2. The reactivity effect on the system due to the rotation of 180° of the materials in these drawers was measured. It is pointed out that the relative positions of the U-235 plates is not affected by the rotation of the materials.

3. The sodium-void coefficients of the front 8 in. section and of the entire columns (28 in.) were measured in the three drawers per half with and without the rotation of 180° .

4. The sodium-void coefficients of step No. 3 were measured with the second, fourth, and sixth 4 in. sections of the materials in each drawer rotated 180° . The objective of this loading pattern shown in Fig. 1 was to reduce by approximately one half the width of the sodium channels which, because of their low density, may be seen as "windows" by the neutrons.

Figure 1

Rotation of Second, Fourth and Sixth 4 in. Sections of 180° .

10.7-2-5 (cont'd)

Results:

Table I
Streaming Effect on Sodium-Void Coefficients

<u>Section Voided,</u> <u>(kg of Na)</u>	<u>Remarks on</u> <u>Loading Pattern</u>	<u>Reactivity Effect</u> <u>of Sodium Sample,</u> <u>Ih</u>
Front 8 in., (0.326)	Not rotated	$+0.507 \pm 0.136$
" (0.326)	Rotated 180 deg.	$+0.415 \pm 0.312$
" (0.326)	throughout drawer	
	Partial rotation of	$+0.511 \pm 0.116$
	180° (Fig. 5 of 6-2-5)	
Entire Column, (2.270)	Not rotated	-4.385 ± 0.136
(28 in.)		
"	Rotated 180 deg,	-4.232 ± 0.100
	throughout drawer	
"	Partial rotation of	-3.749 ± 0.137
	180° (Fig. 5 of 6-2-5)	

Reference:

R. A. Karam, et al., "Measured Physics Parameters in a Large Uranium Carbide Fast Core, Assembly 5 ZPR-6" Reactor Physics Division Annual Report - July 1, 1966 to June 30, 1967, ANL-7310 167, (1968)

SECTION 10.8 - EFFECT OF HETEROGENEITY ON SPECTRA

SECTION 11 - NEUTRON SPECTRA

Because of their specialized nature, and because of a separate compilation in this area⁽¹⁾ it was originally intended not to include measurements of neutron spectra in this Compilation. It was decided, however, that a summary of such measurements would be useful, and that, as such measurements become used in routine data testing, the section should be expanded. For example, in future editions the spectra should be tabulated in some common format rather than presented as graphs.

Reference:

"Neutron Spectra" EACRP-L-62 (Sept. 1966) and supplements of Dec. 1966, Feb. 1967, and Oct. 1968.

11-1-48

Neutron Spectra

Type of Measurement:

- A) Hydrogen and methane proportional counters
- B) Li^6 semiconductor sandwich spectrometer

Facility: ZPR-3 (1.1)

Assembly: 48 (2.1-48)

Method of Measurement:

The counters, the experimental techniques and the analysis of the experiments have been well described in the literature (1), (2), (3) and have been summarized, along with corrected results, in an ANL technical memorandum (4). Only a very brief recapitulation will be given here.

A) Two proton recoil counters were used to cover the range 1 keV - 2 MeV.

1) (1-150 keV) contained 5 atm of hydrogen (plus some CH_4 and some N_2 for calibration); effective length (as defined by needles having an OD of 0.3 mm) = 3.2 cm, diameter = 1.59 cm.

2) (100 keV - 2 MeV) contained 5 atm methane; effective length = 3.2 cm, diameter = 2.5 cm.

Both counters were made with stainless steel walls, 0.8 mm thick. A sleeve of lead, 6 mm thick, surrounded the sensitive volumes. The active volumes of the counters were situated at 10 cm from the reactor midplane in a near-central drawer (4).

The measurements were made with the core approximately 5.5% subcritical (with 196 kg Pu-239 in the core compared to the critical 272 kg). At this level of subcriticality the spectrum at the core center is expected to be virtually the same as at critical (5).

11-1-48 (cont'd)

The data were corrected for wall- and end-effects (6). This correction introduces large uncertainties above 1 MeV (up to 20% (FWHM) except below 5 keV where statistical effects in ionization enter.

B) The Li^6 semiconductor detector⁽³⁾ was made up of a $5 \mu\text{g}/\text{cm}^2$ Li^6 layer on a thin ($<10 \mu\text{g}/\text{cm}^3$) organic film between two surface barrier diodes, (sensitivity about 10^{-5} counts per thermal neutron/ cm^2). Those pulses in coincidence were summed to get the pulse spectrum.

Results:

Results of the proton recoil measurements are given in Table I and Figure 1. The Li^6 pulse spectrum was compared to that derived from the proton recoil results over the energy range from 1-720 keV and the results were found to be consistent.

References:

- (1) E. F. Bennett, "Fast Neutron Spectroscopy by Proton-Recoil Proportional Counting", Nucl. Sci. Eng. 27 16 (1967)
- (2) E. F. Bennett, "Neutron Spectrum Measurements in a Fast Critical Assembly", Nucl. Sci. Eng., 27, 28 (1967)
- (3) R. J. Huber, "In-core Experiments with a Lithium-6 'Sandwich' Fast Neutron Spectrometer", Trans Am. Nucl. Soc. 7, (2), 368 (1964)
- (4) E. F. Bennett, et al., "Measurements of Spectrum and Spectral Indices in ZPR-3 Assemblies 48 and 48B", ZPR-TM-9 (1970)
- (5) A. Travelli, "A Comment on the Comparison of Theoretical and Experimental Neutron Spectra in Fast Critical Assemblies" Proc. Int. Conf. on Fast Critical Experiments and Their Analysis, ANL-7320 p. 481 (1967)
- (6) E. F. Bennett, R. Gold and I.K. Olson, "Analysis and Reduction of Proton Recoil Data", ANL-7394 (1968)

TABLE I

ASSEMBLY-48 NEUTRON SPECTRUM MEASURED WITH PROTON-RECOIL SPECTROMETER

ENERGY & POINT, KEV	RELATIVE FLUX	PROBABLE ERROR	ENERGY & POINT, KEV	RELATIVE FLUX	PROBABLE ERROR
2311.5000	870.890	49.496	257.2600	2829.600	62.561
2201.4000	1021.600	48.584	245.0100	2743.300	68.530
2096.6000	965.720	44.329	233.3400	2793.400	63.704
1996.7000	1164.800	44.465	222.2300	2757.100	59.300
1901.7000	1248.200	44.138	211.6500	2670.500	54.001
1811.1000	1258.800	40.733	201.5700	2565.900	60.893
1724.9000	1361.600	41.609	191.9700	2728.500	57.235
1642.7000	1479.400	41.835	182.8300	2693.200	52.270
1564.5000	1549.400	42.733	174.1200	2645.200	47.819
1490.0000	1453.700	39.882	165.8300	2741.700	56.119
1419.0000	1475.600	40.653	157.9300	2577.700	51.460
1351.5000	1402.000	42.332	150.4100	2528.600	47.420
1287.1000	1569.300	39.168	143.2500	2544.200	43.707
1225.8000	1535.100	40.683	136.4300	2664.700	54.110
1167.5000	1644.300	38.214	129.9300	2586.700	49.821
1111.9000	1790.000	35.741	123.7500	2416.000	46.098
1058.9000	1859.600	37.261	117.8500	2199.000	31.648
1008.5000	1816.100	40.060	112.2400	2072.900	29.786
960.4700	1856.200	32.573	106.9000	2157.100	52.547
914.7300	1936.100	34.787	101.8100	2182.900	48.948
871.1700	1992.600	32.238	96.9580	2189.000	44.834
829.6900	2187.800	35.609	92.3410	2215.400	50.621
790.1800	2296.000	33.117	87.9440	2289.500	38.368
752.5500	2355.500	36.353	83.7560	2281.000	43.550
716.7100	2402.300	33.989	79.7680	2242.600	39.999
682.5800	2454.700	32.072	75.9690	2201.800	46.975
650.0800	2461.000	36.230	72.3520	2100.500	43.277
619.1200	2416.300	27.481	68.9060	2053.500	39.910
589.6400	2520.700	31.239	65.6250	1996.500	36.798
561.5600	2663.600	36.829	62.5000	1837.100	33.845
534.8200	3320.700	28.481	59.5240	1677.400	27.118
509.3600	3224.600	26.220	56.6890	1530.700	30.052
485.1000	3169.600	31.138	53.9900	1553.400	46.698
462.0000	3068.300	28.780	51.4190	1545.200	49.119
440.0000	2994.200	26.587	48.9700	1528.600	44.907
419.0500	2981.500	24.518	46.6390	1650.400	48.096
399.0900	3088.400	30.572	44.4180	1667.700	44.436
380.0900	2886.000	26.340	42.3030	1663.100	40.920
361.9900	2831.300	68.041	40.2880	1542.600	43.913
344.7500	3082.200	71.703	38.3700	1470.500	40.842
328.3300	3143.400	77.697	36.5430	1427.300	44.840
312.7000	3117.400	71.566	34.8020	1339.900	34.380
297.8100	3050.600	78.510	33.1450	1130.300	37.714
283.6300	2725.700	72.693	31.5670	1113.300	35.064
270.1200	2819.700	67.536	30.0640	1122.100	32.152

TABLE I ...CONTINUED

ENERGY ^a POINT, KEV	RELATIVE FLUX	PROBABLE ERROR	ENERGY ^a POINT, KEV	RELATIVE FLUX	PROBABLE ERROR
28.6320	1304.200	36.344	6.3093	624.960	24.617
27.2690	1367.800	33.422	6.6099	661.420	25.008
25.9700	1396.500	30.608	5.7228	576.570	23.048
24.7330	1496.500	36.122	5.4502	561.550	21.105
23.5560	1372.300	33.291	5.1907	515.920	20.587
22.4340	1255.800	30.727	4.9435	494.350	22.578
21.3660	1064.500	23.306	4.7081	484.940	20.775
20.3480	950.070	21.476	4.4839	428.180	21.703
19.3790	913.990	34.739	4.2704	418.490	19.993
18.4560	850.600	36.044	4.0671	399.840	18.273
17.5780	853.800	33.165	3.8734	347.490	16.788
16.7410	978.890	34.900	3.6889	345.290	17.761
15.9430	1002.400	32.240	3.5133	292.940	16.244
15.1040	971.520	34.068	3.3460	268.170	17.432
14.4610	1019.800	31.472	3.1867		
13.7720	1065.100	29.226	3.0349		
13.1170	1003.700	31.566	2.8904	141.580	17.571
12.4920	985.170	25.945	2.7527	144.260	16.069
11.8970	939.880	26.789	2.6217	173.300	16.142
11.3510	861.480	24.612	2.4968	218.860	16.111
10.7910	759.710	27.107	2.3779	286.140	14.761
10.2770	655.880	24.833	2.2647	343.400	14.927
9.7879	650.520	23.126	2.1568	353.700	13.761
9.3218	619.970	21.153	2.0541	392.230	12.716
8.8779	609.280	19.459	1.9563	375.150	12.980
8.4551			1.8632	409.300	12.030
8.0526			1.7744	425.020	12.411
7.6691			1.6899	432.720	11.555
7.3038	574.800	26.157	1.6095	432.890	10.614
6.9560	622.070	26.197	1.5328	427.910	11.066
6.6248	640.880	26.577			

a. Midpoint of .048-width lethargy interval

Fig. 1: Assembly 48 Spectrum Measured With Proton-Recoil Spectrometer

10.4-2-5 Effect of Heterogeneity on Reaction Rates

Type of Cell: Plate

Facility: ZPR-6 (1.2)

Assembly: 5 (2.2-5)

Experimental Method:

To obtain axial and radial fission ratio traverses through this assembly, different cell structures were required. See Figs. 2, 3, and 4 of 5.2-2-4Z. Thus there are central fission ratios for the two arrangements. The detectors used are shown in Fig. 1 of 5.2-2-4Z.

Results:

Table I Central Fission Ratios

<u>Fission Ratio</u>	Radial (No Spacer) ^a	<u>Axial</u>	Radial (Spacer) ^a
U-238/U-235	0.0238 $\pm$ 0.0008	0.0223 $\pm$ 0.0009	0.0234 $\pm$ 0.0007
U-234/U-235	0.166 $\pm$ 0.004	0.159 $\pm$ 0.005	0.163 $\pm$ 0.004
Pu-239/U-235	0.968 $\pm$ 0.025	0.949 $\pm$ 0.025	0.963 $\pm$ 0.024

^a Spacer consists of 1/2 in. thick U-238 plate located in front drawers only when radial traverse measurements were made.

References:

1. R. A. Karam, et al., "Measured Physics Parameters in a Large Uranium Carbide Fast Core, Assembly 5 ZPR-6" Reactor Physics Division Annual Report - July 1, 1966 to June 30, 1967, ANL-7310 pp. 167-189, (1968)
2. R. A. Karam, et al., "Experimental Verification of Reactor Physics Parameters of Zones Cores", Proceedings of Int. Conf. on Fast Critical Experiments and Their Analysis, October 10-13, 1966, ANL-7320, pp. 403-420 (1966).

11-2-4Z

Neutron Spectra

Type of Measurement:

Hydrogen and methane proton-recoil proportional counters

Facility: ZPR-6 (1.2)

Assembly: 4Z (2.3-4Z)

Method of Measurement:

Two counters were used:

- a) (1-100 keV) contained 3 atm of hydrogen (plus 20 cm Hg of methane and 10 cm Hg of N for calibration); effective length = 6.5 cm, diameter = 2.5 cm.
- b) (100 keV - 1MeV) contained 3 atm of methane (plus 10 cm Hg of N); effective length = 7.5 cm, diameter = 3.8 cm.
See also Section 11-1-48.

Both counters were made with stainless steel walls, 1.2 mm thick. Lead sleeves, 0.6 cm thick, were placed around the sensitive volumes of the counter for gamma shielding. One counter at a time was placed in the central drawer with the mid-point of the counter displaced axially about 11 cm from the core center. The part of the drawer not needed for the detector or preamplifier was filled with representative core material except for U-235.

The measurements were made with the reactor subcritical (within 1% of critical).

Pulse rise-time analysis was used to discriminate against pulses due to gammas⁽¹⁾ but only limited end and wall effect corrections were made. This may introduce a maximum error of 20% in the results near 1 MeV. The preamplifier used is described in reference (3).

Results:⁽²⁾

The resolution is about 20% (FWHM) except below 3 keV where it worsened to about 60% at 1 keV. Statistical uncertainty was about $\pm 2\%$ of the maximum flux level.

11-2-4Z

Figure 1
Spectrum at the Center of Assembly No. 42

References:

1. E. F. Bennett, "Fast Neutron Spectroscopy by Proton-Recoil Proportional Counting", Nucl. Sci. Eng. 27, 16-27 (1967)
2. E. F. Bennett, "Neutron Spectrum Measurement in a Fast Critical Assembly", Nucl. Sci. Eng. 27, 28-33 (1967)
3. E. F. Bennett, "A Modified Cascade Preamplifier for Proportional Counters", Nucl Instr. Methods 48, 170 (1967)

11-2-5

Neutron Spectra

Type of Measurement:

Hydrogen and methane proton recoil proportional counters.

Facility: ZPR-6 (1.2)

Assembly: 5 (2.2-5)

Method of Measurement:

See Sections 11-2-4Z and 11-1-48. An improved energy calibration has been used and improved resolution (10% at higher energies) has been attained.

Results:

Figure 1
Central Spectrum of Assembly 5 Compared with an Energy Group
Calculation

Reference:

E. F. Bennett, "Experimental Neutron Spectrum Comparison of a Zoned and a Homogeneous Fast Critical Assembly" ANL Reactor Physics Division Annual Report, July 1, 1966 to June 30, 1967 p. 198 ANL-7310 (1968)

11-2-6

Neutron Spectra

Type of Measurement:

Hydrogen and methane proton recoil proportional counters

Facility: ZPR-6 (1.2)

Assembly: 6 (2.2-6)

Method of Measurement:

Spectra were measured at various positions in the assembly and under conditions of sodium voiding. The counters are described in Section 11-2-4Z. End and wall effects have been corrected for using measured response functions according to the technique described in reference (2).

The detectors were placed in the core in the place of a drawer at various radii. The center of the detectors were displaced axially from the reactor midplane by about 12 cm.

Results:

Figure 1
Central Neutron Spectrum in ZPR-6, Assembly 6
Together with a 22-group Diffusion Theory Calculation

11-2-6 (cont'd)

Figure 2
Edge Spectrum ($r = 88.0$ cm) in ZPR-6, Assembly 6
Together with a 22-group Diffusion Theory Calculation

Figure 3
Blanket Spectrum ($r = 108.0$ cm) in ZPR-6, Assembly 6
Together with a 22-group Diffusion Theory Calculation

Figure 4 shows the central spectrum taken with a region of radius equal to 18 cm voided in sodium.

Figure 4
Central Neutron Spectrum in a Sodium-voided Region of ZPR-6
Assembly 6 Together with an MC² Calculation for an Infinite
Sodium-voided Region

References:

1. E. F. Bennett and R. A. Karam, "Neutron Spectroscopy in ZPR-6 Assembly 6 Using Proton-Recoil Counters" ANL Reactor Physics Division Annual Report July 1, 1967 to June 30, 1968, ANL-7410 p. 151 (1969).
2. E. F. Bennett, R. Gold and I. K. Olson, "Analysis and Reduction of Proton-Recoil Data," ANL-7394 (1968).

11-3-11

Neutron Spectra

Type of Measurement:

Hydrogen and methane proton-recoil proportional counters.

Facility: ZPR-9 (1.3)

Assembly: 11 (2.3-11A)

Method of Measurement:

See Section 11-1-48.

Two counters were used:

a) (1.5-300 kev) contained 5 atm of hydrogen; effective length = 3.1 cm, diameter = 1.6 cm.

b) (300 kev - 1 Mev) contained 5 atm of methane; effective length = 5.0 cm, diameter = 2.5 cm.

Both counters were made with stainless steel walls 0.8 cm thick. No correction was made for end and wall effects (could contribute about a 20% error to the data at 1 MeV). Energy resolutions $\approx$ 20% (FWHM).

The measurements were performed in the central drawer at a counter position 4 in. back from the reactor midplane at about 7% and 0.1% $\Delta k/k$ subcritical. The results show no significant differences in the spectra as a function of subcriticality in this range.

11-3-11 (cont'd)

Results:

Figure 1

Spectrum as Measured in the Central Drawer at the Midplane in ZPR-9 Assembly 11A at 0.1% $\Delta k/k$ Subcritical

The results are in very close agreement with the spectrum from ZPR-6 Assembly 42.

Reference:

ANL Reactor Development Program Progress Report May, 1966
ANL-7219, p. 11 (1966)

11-3-12

Neutron Spectra

Type of Measurement:

Hydrogen and methane proportional counters.

Facility: ZPR-9 (1.3)

Assembly: 12 (2.3-12)

Method of Measurement:

See Section 11-2-4Z.

Results:

Figure 1 shows a comparison between the measured spectrum and that measured in ZPR-6 Assembly 5. (See Section 11-2-5).

Fig. 1 Percent Deviation of Spectrum in ZPR-9 Assembly 12 from that in ZPR-6 Assembly 5

Reference:

E. F. Bennett, "Experimental Neutron Spectrum Comparison of a Zoned and a Homogeneous Fast Critical Assembly" ANL Reactor Physics Division Annual Report, July 1, 1966 to June 30, 1967 p. 198 ANL-7310 (1968)

SECTION 12 - NEUTRON LIFETIMES

This section is divided into two parts: measurements of prompt neutron lifetimes (12.1) and measurements of various effects on lifetime measurements (such as effects of counter position) (12.2). Experiments related to the effects of heterogeneity on lifetimes are compiled in Section 10.6.

SECTION 12.1 NEUTRON LIFETIMES

12.1-1-6F

Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1)

Assembly: 6F (2.1-6F)

Experimental Details:

Two BF_3 proportional counters, with sensitive volumes of 1 x 24 in. were placed near the center of the core but not in adjacent drawer positions. To assure a constant flux level (on a macroscopic time scale) the measurements are made with the assembly slightly subcritical.

Several runs were made at different subcriticalities so that the α at delayed critical could be obtained by extrapolation.

The data were fit to a single exponential plus background.

Results:

$$\alpha \text{ (delayed critical)} = 9.4 \pm .3 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (calculated)} = 0.0073$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 7.7 \times 10^{-8} \text{ sec.}$$

Reference:

G. S. Brunson, et al., "A Survey of Prompt - Neutron Lifetimes in Fast Critical Systems" ANL-6681 (1963).

12.1-1- 9

Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1)

Assembly: 9 (2.1-9)

Experimental Details:

Two BF_3 counters were placed near the core center. The data were fit to a single exponential. See 12.1-1-6F.

Results:

$$\alpha \text{ (delayed critical)} = 8.8 \pm 0.3 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (calculated)} = 0.0073$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 8.3 \times 10^{-8} \text{ sec.}$$

Reference:

G. S. Brunson, et al., "A Survey of Prompt Neutron Lifetimes in Fast Critical Systems", ANL-6681 (1963).

12.1-1-10 Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1) Assembly: 10 (2.1-10)

Experimental Details:

Two BF_3 counters were placed near the core center. The data were fit to a single exponential. See 12.1-1-6F.

Results:

$$\alpha \text{ (delayed critical)} = 10.7 \pm .2 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (calculated)} = 0.0073$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 6.82 \times 10^{-8} \text{ sec}$$

Reference:

G. S. Brunson, et al., "A Survey of Prompt Neutron Lifetimes in Fast Critical Systems", ANL-6681 (1963).

12.1-1-11

Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1)

Assembly: 11 (2.1-11)

Experimental Details:

Two BF_3 counters were placed near the core center. The data were fit to a single exponential. See 12.1-1-6F.

Results:

$$\alpha \text{ (delayed critical)} = 10.3 \pm .3 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (calculated)} = 0.0073$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 7.08 \times 10^{-8} \text{ sec}$$

Reference:

G. S. Brunson, et al., "A Survey of Prompt Neutron Lifetimes in Fast Critical Systems", ANL-6681 (1963).

12.1-1-12

Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1)

Assembly: 12 (2.1-12)

Experimental Details:

Two BF_3 counters were placed near the core center. The data were fit to a single exponential. See 12.1-1-6F.

Results:

$$\alpha \text{ (delayed critical)} = 7.02 \pm .14 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (calculated)} = 0.0074$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 10.5 \times 10^{-8} \text{ sec}$$

Reference:

G. S. Brunson, et al., "A Survey of Prompt Neutron Lifetimes in Fast Critical Systems", ANL-6681 (1963).

12.1-1-13

Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1)

Assembly: 13 (2.1-13)

Experimental Details:

Two BF_3 counters were placed near the core center. The data were fit to a single exponential. See 12.1-1-6F.

Results:

$$\alpha \text{ (delayed critical)} = 9.37 \pm .19 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (calculated)} = 0.0073$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 7.79 \times 10^{-8} \text{ sec}$$

Reference:

G. S. Brunson, et al., "A Survey of Prompt Neutron Lifetimes in Fast Critical Systems", ANL-6681 (1963).

12.1-1- 14 Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1) Assembly: 14 (2.1-14)

Experimental Details:

Two BF_3 counters were placed near the core center. The data were fit to a single exponential. See 12.1-1-6F.

Results:

$$\alpha \text{ (delayed critical)} = 3.98 \pm .12 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (calculated)} = 0.0075$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 18.8 \times 10^{-8} \text{ sec}$$

Reference:

G. S. Brunson, et al., "A Survey of Prompt Neutron Lifetimes in Fast Critical Systems", ANL-6681 (1963).

12.1-1-17

Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1)

Assembly: 17 (2.1-17)

Experimental Details:

Two BF_3 counters were placed near the core center. The data were fit to a single exponential. See 12.1-1-6F.

Results:

$$\alpha \text{ (delayed critical)} = 5.2 \pm 0.2 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (calculated)} = 0.0075$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 14.4 \times 10^{-8} \text{ sec}$$

Reference:

G. S. Brunson, et al., "A Survey of Prompt Neutron Lifetimes in Fast Critical Systems", ANL-6681 (1963).

12.1-1-20

Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1)

Assembly: 20 (2.1-20)

Experimental Details:

Two BF_3 counters were placed near the core center. The data were fit to a single exponential. See 12.1-1-6F.

Results:

$$\alpha \text{ measured} \approx 4.55 \times 10^4 \text{ sec}^{-1}$$

$$\alpha \text{ (delayed critical)} = 4.47 \pm 0.09 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (calculated)} = 0.0069 \pm 0.0003$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 15.4 \times 10^{-8} \text{ sec.}$$

The value of α was found not to depend on the position of the initiating and terminating counters (See Section 12.2-1-20).

Reference:

G. S. Brunson, et al., "A Survey of Prompt Neutron Lifetimes in Fast Critical Systems", ANL-6681 (1963).

12.1-1-22

Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1)

Assembly: 22 (2.1-22)

Experimental Details:

Two BF_3 counters were placed near the core center. The data were fit to a single exponential. See 12.1-1-6F.

Results:

$$\alpha \text{ (delayed critical)} = 9.9 \pm 0.3 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (calculated)} = 0.00073$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 7.37 \times 10^{-8} \text{ sec}$$

Reference:

G. S. Brunson, et al., "A Survey of Prompt Neutron Lifetimes in Fast Critical Systems", ANL-6681 (1963).

12.1-1- 23 Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1) Assembly: 23 (2.1-23)

Experimental Details:

Two BF_3 counters were placed near the core center. The data were fit to a single exponential. See 12.1-1-6F.

Results:

$$\alpha \text{ (delayed critical)} = 5.63 \pm .11 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (calculated)} = 0.0068$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 12.07 \times 10^{-8} \text{ sec}$$

Reference:

G. S. Brunson, et al., "A Survey of Prompt Neutron Lifetimes in Fast Critical Systems", ANL-6681 (1963).

12.1-1-24

Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1)

Assembly: 24 (2.1-24)

Experimental Details:

Two BF_3 counters were placed near the core center. The data were fit to a single exponential. See 12.1-1-6F.

Results:

$$\alpha \text{ (delayed critical)} = 8.6 \pm 0.3 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (calculated)} = 0.0073$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 8.49 \times 10^{-6} \text{ sec}$$

Reference:

G. S. Brunson, et al., "A Survey of Prompt Neutron Lifetimes in Fast Critical Systems", ANL-6681 (1963).

12.1-1-25

Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1)

Assembly: 25 (2.1-25)

Experimental Details:

Two BF₃ counters were placed near the core center. The data were fit to a single exponential. See 12.1-1-6F.

Results:

$$\alpha \text{ (delayed critical)} = 9.0 \pm .2 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (calculated)} = 0.0073$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 8.11 \times 10^{-8} \text{ sec.}$$

Reference:

G. S. Brunson, et al., "A Survey of Prompt Neutron Lifetimes in Fast Critical Systems", ANL-6681 (1963).

12.1-1-29

Neutron Lifetime

Type of Measurement: Rossi- α (2 counters)

Facility: ZPR-3 (1.1) Assembly: 29 (2.1-29)

Experimental Details:

Two BF_3 counters placed near the core center. Data fit to a single exponential. See 12.1-1-6F.

Results:

$$\alpha \text{ (delayed critical)} = 3.17 \pm 0.10 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (estimated)} = 0.0073$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 23 \times 10^{-8} \text{ sec.}$$

Reference:

A. L. Hess, et al., "Critical Studies of a 450-liter Uranium Oxide Fast Reactor Core (ZPR-III Assembly 29)" ANL-6336 (1961)

12.1-1-30

Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1)

Assembly: 30 (2.1-30)

Experimental Details:

Two BF_3 counters placed near core center. The data were fit to a single exponential. See 12.1-6F.

Results:

$$\alpha \text{ (delayed critical)} = 4.10 \pm 0.10 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (estimated)} = 0.0073$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 17.8 \times 10^{-8} \text{ sec.}$$

Reference:

P. I. Amundson, et al., "Critical Studies of a Dilute Oxide Fast Reactor Core (ZPR-III Assembly 30)" ANL-6337 (1961)

12.1-1-31

Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1)

Assembly: 31 (2.1-31)

Experimental Details:

Two BF_3 counters placed near core center. The data were fit to a single exponential.

Results:

$$\alpha \text{ (delayed critical)} = 4.45 \pm 0.10 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (estimated)} = 0.0073$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 16.4 \times 10^{-8} \text{ sec}$$

Reference:

J. M. Gasidlo, J. K. Long, and R. L. McVean, "Critical Studies of A Dilute Fast Reactor Core (ZPR-III Assembly 31)" ANL-6338 (1961).

12.1-1-32

Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1)

Assembly: 32 (2.1-32)

Experimental Details:

Two BF_3 counters were placed near the core center. The data were fit to a single exponential. See 12.1-1-6F.

Results:

$$\alpha \text{ (delayed critical)} = 5.37 \pm 0.10 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (estimated)} = 0.0068$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 12.7 \times 10^{-8} \text{ sec.}$$

Reference:

P. I. Amundson, et al., "Critical Studies of Uranium-Steel and Uranium-Steel-Sodium Fast Reactor Cores (ZPR-III Assemblies 32 and 33)" ANL-6690 (1962)

12.1-1-33

Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1)

Assembly: 33 (2.1-33)

Experimental Details:

Two BF_3 counters were placed near the core center. The data were fit to a single exponential. See 12.1-1-6F.

Results:

$$\alpha \text{ (delayed critical)} = 5.46 \pm 0.1 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (estimated)} = 0.0068$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 12.5 \times 10^{-8} \text{ sec.}$$

Reference:

P. I. Amundson, et al., "Critical Studies of Uranium-Steel and Uranium-Steel-Sodium Fast Reactor Cores (ZPR-III Assemblies 32 and 33)" ANL-6690 (1963)

12.1-1-34

Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1)

Assembly: 34 (2.1-34)

Experimental Details:

Two BF_3 counters were placed near the core center. The data were fit to a single exponential. See 12.1-1-6F.

Results:

$$\alpha \text{ (delayed critical)} = 2.92 \pm 0.08 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (calculated)} = 0.0073$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 25 \times 10^{-8} \text{ sec.}$$

Reference:

G. S. Brunson, et al., "A Survey of Prompt Neutron Lifetimes in Fast Critical Systems", ANL-6681 (1963).

12.1-1-36

Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1)

Assembly: 36 (2.1-36)

Experimental Details:

Two BF_3 counters placed near the core center. The data were fit to a single exponential. See 12.1-1-6F.

Results:

$$\alpha \text{ (delayed critical)} = 9.4 \pm 0.1 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (estimated)} = 0.0074 \pm 0.0003$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 7.9 \times 10^{-8} \text{ sec.}$$

Reference:

J. M. Gasidlo, J. K. Long, and R. L. McVean, "Critical Studies of a Fast Reactor Core Containing Depleted Uranium and Sodium as Diluents (ZPR-III Assembly 36)" ANL-6494 (1962).

12.1-1-41

Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1)

Assembly: 41 (2.1-41)

Experimental Details:

Two BF_3 counters were placed at the core center. The data were fit to a single exponential. In this case two methods were used to obtain subcriticality; in one, control rods were withdrawn, and in the other, the average core fuel density was decreased. Higher than normal degrees of subcriticality were also used. See Section 12.1-1-6F.

Results:

$$\alpha \text{ (delayed critical)} = 5.55 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (estimated)} = 0.0073$$

$$\text{Lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 13.2 \times 10^{-8} \text{ sec.}$$

Reference:

ANL Reactor Development Program Progress Report, December 1962, ANL-6672

12.1-1-42

Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1)

Assembly: 42 (2.1-42)

Experimental Details:

Two BF₃ counters were located in various arrangements in the assembly³ - See section 12.2-1-42. The data were fit to single exponentials.

Results:

$$\text{Average } \alpha \text{ (delayed critical)} = 2.71 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (estimated)} = 0.0072$$

$$\text{lifetime} = \beta_{\text{eff}}/\alpha = 19.4 \times 10^{-8} \text{ sec}$$

Reference:

P. I. Amundson, et al., "A Two-Zone Fast Critical Experiment (ZPR-III Assembly 42)" ANL-6733 (1964).

12.1-1-43

Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-3 (1.1) Assembly: 43 (2.1-43)

Experimental Details:

Two He^3 counters were used in various positions in the core although no significant differences in the resulting α 's were found (see Section 12.2-1-43).

Results:

$$\alpha \text{ (delayed critical)} = 2.42 \pm 0.1 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (estimated)} = 0.00705$$

$$\text{lifetime} = \beta_{\text{eff}}/\alpha = 2.9 \times 10^{-7} \text{ sec}$$

Reference:

J. M. Gasidlo, J. K. Long and W. P. Keeney, "Doppler Effect Measurements, Sodium Void Coefficients, and Critical Studies of a 5,000 liter Fast Power Breeder Reactor By the Two-Zoned Method", ANL-6838 (1964).

12.1-2-1

Neutron Lifetime

Type of Measurement: Rossi- α

Facility: ZPR-6 (1.2)

Assembly: 1 (2.2-1)

Experimental Details:

Results:

$$\alpha \text{ (delayed critical)} = (10.8 \pm 0.7) \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (estimated)} = 0.0074$$

$$\text{lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = (6.85 \pm 0.7) \times 10^{-8} \text{ sec}$$

Reference:

"ANL Reactor Development Program Progress Report - September 1963", ANL-6784, p. 10 (1963)

12.1-2-2

Neutron Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-6 (1.2) Assembly: 2 (2.2-2)

Experimental Details:

A fission chamber located in the central drawer S 2323 was used to start the timing cycle, while a fission chamber located at drawer S 2314 was used to detect neutrons during the time cycle. Pulses were recorded in a time analyzer with a channel width of 1.6×10^{-6} sec. The measurement was also done using a single fission counter.

Results:

At delayed critical:

$$2 \text{ counter: } \alpha = (3.44 \pm 0.12) \times 10^4 \text{ sec}^{-1}$$

$$1 \text{ counter: } \alpha = (3.52 \pm 0.15) \times 10^4 \text{ sec}^{-1}$$

$$\text{Average } \alpha = (3.48 \pm 0.10) \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (calculated)} = 0.0073$$

$$\text{lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = (21.4 \pm 0.6) \times 10^{-8} \text{ sec.}$$

Reference:

G. K. Rusch, et al., "Investigations of a 600-Liter Uranium Carbide Core (ZPR-VI Assembly No. 2)," Reactor Physics Division Annual Report-July 1, 1963 to June 30, 1964, ANL-7010, 91 (1965)

12.1-2-3

Neutron Lifetime

Type of Measurement: Rossi- α

Facility: ZPR-6 (1.2)

Assembly: 3 (2.2-3)

Experimental Details:

Results:

$$\alpha \text{ (delayed critical)} = 3.14 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (calculated)} = 0.0072$$

$$\text{lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 0.23 \mu \text{ sec}$$

Reference:

F. H. Helm, et. al., "Physics Parameters and Sodium Void Coefficients In a 950-Liter Pancake Core," Reactor Physics Division Annual Report-July 1, 1964 to June 30, 1965, ARL-7110, 110, (1965)

12.1-2-4Z Neutron Lifetime

Type of Measurement: Rossi- α (2 counters)

Facility: ZPR-6 (1.2)

Assembly: 4Z (2.2-4Z)

Experimental Details:

Data from two fission chambers feeding a 400-channel analyzer were fit to the function.

$$f(t) = B + A_1 e^{a_1 t} + A_2 e^{a_2 t}$$

Results:

$$a_1 = (2.9 \pm 1.4) \times 10^5 \text{ sec}^{-1}$$

$$a_2 = (2.75 \pm 0.13) \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}}(\text{calculated}) = 0.00688$$

$$\text{lifetime}_1 = \frac{\beta_{\text{eff}}}{a_1} = 23.63 \times 10^9 \text{ sec}$$

$$\text{lifetime}_2 = \frac{\beta_{\text{eff}}}{a_2} = 249.9 \times 10^9 \text{ sec}$$

References:

1. R. A. Karam, et al., "Measured Physics Parameters in a Zoned Fast Uranium Carbide Core, Assembly 4Z", ANL-7281 (Dec. 1966)
2. R. A. Karam, "Spatial Dependence of Prompt-Neutron Chains", Reactor Physics Division Annual Report - July 1, 1964 to June 30, 1965 ANL-7110, 330 (1965)

12.1-2-5

Neutron Lifetime

Type of Measurement: Pulsed

Facility: ZPR-6 (1.2)

Assembly: 5 (2.2-5)

Experimental Method:

A Kaman Nuclear neutron generator produced essentially isotropic 14 MeV neutrons from the $T(D,n) He^4$ reaction. The target was located at the center of the core. The decay was monitored by a 1024 channel analyzer. U-235 and U-238 fission chambers were used (2 in. diam. x 10 in. long) at various radial positions. They contained 1.7 g enriched uranium and 1.68 g depleted uranium respectively. Measurements were made at varying degrees of subcriticality.

Results:

See Table I (next page) and Figure 1.

Figure 1
Prompt Decay Constant for Different Reactivity Conditions at
Different Core Locations

Reference:

P. H. Heim, "Pulsed Neutron Measurements on ZPR-6, Assembly No. 5", Reactor Physics Annual Report - July 1, 1966 to June 30, 1967, ANL-7310, 217 (1968)

Table I Pulsed Neutron Measurements on Assembly 5 of ZPR-6

Run No.	Detector		Table Separation cm	DP Rods Fully In ^a	Subcriticality ^b %	Pulse Frequency kc	Pulse Width μ sec	Channel Width μ sec	No. of Channels	Channel 4 Indication ^c	Count Time min	Measured σ^d sec ⁻¹	Remarks
	Location	Type											
1	2923	U ₃₂₃	180	none		3	1.5	0.125	1024		10		
2	2923	U ₃₂₃	7.65	3, 8	4.1	3	1.5	0.25	1024	0.10×10^{-9}	5	1.062×10^4	
3	2923	U ₃₂₃	2.0	3, 8	1.73	3	1.5	0.25	1024	0.32×10^{-9}	5	5.295×10^4	
4	2923	U ₃₂₃	2.0	3, 8	1.73	3	1.5	0.25	1024	0.85×10^{-10}	5	5.5×10^4	
5	2923	U ₃₂₃	0	3, 8	0.982	3	1.5	0.25	1024	0.15×10^{-9}	5	3.81×10^4	
6	2923	U ₃₂₃	0	2, 3, 4, 6, 8, 10	0.552	3	1.5	0.25	1024	0.3×10^{-9}	5	2.07×10^4	
7	2923	U ₃₂₃	0	2, 3, 4, 6, 8, 10	0.552	1	1.5	0.5	1024	0.68×10^{-10}	10	2.87×10^4	
8	2923	U ₃₂₃	0	all except 1	0.224	1	1.5	0.5	1024	0.142×10^{-9}	10	2.364×10^4	
9	2923	U ₃₂₃	0	all except 1	0.224	1	1.5	1.0	512	0.43×10^{-10}	10	2.42×10^4	
10	2923	U ₃₂₃	0	all	0.116	1	1.5	1.0	512	0.63×10^{-10}	10	2.12×10^4	
11	Runs 11-13 are time-dependent spectra measurements												
14	3723	U ₃₂₃	0	all except 6	0.224	1	1.5	1.0	512	0.12×10^{-9}	10	2.33×10^4	Rod 6 on 15.1
15	3723	U ₃₂₃	0	2, 3, 4, 8, 10	0.661	1	1.5	1.0	512	0.425×10^{-10}	15	3.10×10^4	same
16	3723	U ₃₂₃	0	2, 3, 4, 8, 10	0.661	3	1.5	0.25	1024	0.235×10^{-9}	10	3.20×10^4	same
17	3723	U ₃₂₃	0	3, 8	1.00	3	1.5	0.25	1024	0.15×10^{-9}	10	3.91×10^4	Drawer 1023 out
18	3723	U ₃₂₃	0	3, 8	3.18	3	1.5	0.25	1024	0.75×10^{-10}	10	5.87×10^4	B-10 rods No. 1, 2, 5, 9, 10, 12 in
19	4023	U ₃₂₃	0	all except 6	0.224	1	1.5	1.0	512	0.544×10^{-9}	15	2.39×10^4	Rod 6 on 20.0
20	4023	U ₃₂₃	0	2, 3, 4, 8, 10	0.661	1	1.5	1.0	512	0.175×10^{-9}	15	3.23×10^4	same
21	4023	U ₃₂₃	0	3, 8	1.05	3	1.5	0.25	1024	0.66×10^{-9}	5	3.694×10^4	
22	4023	U ₃₂₃	2.0	3, 8	1.79	3	1.5	0.25	1024	0.378×10^{-9}	10	5.12×10^4	
23	4023	U ₃₂₃	7.5	3, 8	4.16	3	1.5	0.25	1024	0.188×10^{-9}	10	1.024×10^4	
24	2923	U ₃₂₃	0	all except 6	0.224	1	1.5	1.0	512	0.71×10^{-10}	10	2.40×10^4	Rod 6 on 40.6
25	2923	U ₃₂₃	7.6	3, 8	4.12	3	1.5	0.25	1024	0.31×10^{-10}	10	1.05×10^4	
26	2923	U ₃₂₃	2.0	3, 8	1.75	3	1.5	0.25	1024	0.595×10^{-9}	5	5.32×10^4	
27	2923	U ₃₂₃	0	3, 8	1.01	3	1.5	0.25	1024	0.302×10^{-9}	5	3.70×10^4	
28	2923	U ₃₂₃	0	all except 6	0.224	1	1.5	0.5	1024	0.145×10^{-9}	20		Rod 6 on 40.8
29	2923	U ₃₂₃	0	all except 6	0.224	1	1.5	0.125	1024	0.145×10^{-9}	10		same
30	2923	U ₃₂₃	0	2, 3, 4, 8, 10	0.661	1	1.5	0.5	1024	0.498×10^{-9}	20		same
31	2923	U ₃₂₃	0	3, 8	0.982	3	1.5	0.25	1024	0.96×10^{-9}	10	3.80×10^4	same
32	2923	U ₃₂₃	2.0	3, 8	1.73	3	1.5	0.25	1024	0.855×10^{-9}	10		same
33	2923	U ₃₂₃	7.5	3, 8	4.1	3	1.5	0.25	1024	0.266×10^{-9}	20		same
34	4023	U ₃₂₃	0	3, 8	0.982	10	1.5	0.5	256	0.383×10^{-9}	30		Rod 6 on 19) Drawer 1023 out
35	4023	U ₃₂₃	0	all except 6	0.224	3	1.5	1.0	256	0.490×10^{-9}	20		same
36	Runs under No. 20 were intensity measurements with a beamhole in the assembly.												
37	2923	U ₃₂₃	180	none		3	0.15	0.125	1024	0.51×10^{-11}	50		
38	2923	U ₃₂₃	7.5	3, 8	4.1	3	0.15	0.125	1024	0.115×10^{-10}	15	1.16×10^4	Rod 6 on 40.8
39	2923	U ₃₂₃	0	3, 8	0.982	3	0.15	0.125	1024	0.32×10^{-10}	15	3.92×10^4	same
40	2923	U ₃₂₃	0	all except 6	0.224	3	0.15	0.125	1024	0.115×10^{-9}	15	2.37×10^4	same
41	3723	U ₃₂₃	7.5	3, 8	4.1	2	0.15	0.125	1024	0.106×10^{-10}	30	1.15×10^4	Rod 6 on 15)
42	3723	U ₃₂₃	0	3, 8	0.982	3	0.15	0.125	1024	0.32×10^{-10}	60		same
43	3723	U ₃₂₃	0	all except 6	0.224	3	0.15	0.125	1024	0.13×10^{-9}	45	2.32×10^4	same

^a All other dual purpose rods are fully out except as shown under Remarks. The adjustments indicated in the Remarks column were generally made in order to maintain comparable reactivities for different counter locations.

^b Subcriticality values were calculated from rod and fuel drawer worths, referencing to a critical loading, for all cases with the halves together. For the cases with the halves separated the values were estimated—very roughly—from the observed source multiplication.

^c These data are given as an indication of the relative average neutron flux during the measurement.

^d This column gives the observed fundamental decay constant.

^e These values were found by graphical extrapolation. All other σ values given were found by a least squares fit.

12.1-2-5 (cont'd)

12.1-3-1 Measurements of Lifetime

Type of Measurement: Rossi- α

Facility: ZPR-9 (1.3)

Assembly: 1 (2.3-1)

Experimental Method:

Two U-235 fission chambers were inserted into chamber drawers on the stationary half extending from the midplane to approximately 10 in. into the assembly. Data were collected with a 100 channel analyzer and were fit to a single exponential.

Results:

$$\alpha = 8.67 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} (\text{calc}) = 0.00728$$

$$\text{lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 84.0 \times 10^{-9} \text{ sec}$$

Reference:

E. C. Doerner, W. G. Knapp, K. K. Almenas, and R. A. Karam, "Physics Measurements in Tungsten-Based, Aluminum-Reflected Fast Reactors", ANL-7007 (March 1967)

12.1-3-3 Measurements of Lifetime

Type of Measurement: Rossi- α

Facility: ZPR-9 (1.3) Assembly: 3 (2.3-3)

Experimental Method:

Two U-235 fission chambers were placed near the core center. The data were collected in a 100 channel analyzer and fit to two exponentials.

Results:

$$\alpha = 6.57 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} (\text{calc}) = 0.00708$$

$$\text{lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 108 \times 10^{-9} \text{ sec}$$

Reference:

R. C. Doerner, W. G. Knapp, K. K. Almenas, and R. A. Karam, "Physics Measurements in Tungsten-Eased, Aluminum-Reflected Fast Reactors", ANL-7007 (March 1967)

12.1-3-4 Measurements of Lifetimes

Type of Measurement: Rossi-a

Facility: ZPR-9 (1.3)

Assembly: 4 (2.3-4)

Experimental Method:

Two U-235 fission chambers were placed near the core center. The data were collected in a 100 channel analyzer and fit to two exponentials.

Results:

$$\alpha = 5.19 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} (\text{calc}) = 0.00664$$

$$\text{lifetime} = \frac{\beta_{\text{eff}}}{\alpha} = 128 \times 10^{-9} \text{ sec}$$

Reference:

R. C. Doerner, W. G. Knapp, K. K. Almenas, and R. A. Karam, "Physics Measurements in Tungsten-Based, Aluminum-Reflected Fast Reactors", ANL-7007 (March 1967)

12.1-3-5 Measurement of Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-9 (1.3)

Assembly: 5 (2.3-5)

Experimental Method:

See Section 12.1-3-3.

Results:

Table 1
Least-squares Fitting of Rossi-alpha Data

Run	Location of A Detector ^b	Location of B Detector ^b	Channel Width μ sec	Bkgt/ 10^4	$A1/10^{10}$ ^c	$a1/10^{-6}$	$A2/10^{10}$ ^d	$a2/10^{-6}$
1	23-23	23-24	0.2	3.02 ± 0.02	3.67 ± 2.10	2.23 ± 0.63	2.38 ± 0.03	3.92 ± 0.14
2	23-23	23-25	0.2	2.93 ± 0.02	2.12 ± 2.30	2.00 ± 1.00	2.40 ± 0.01	3.98 ± 0.14
3	23-23	23-25	0.4	6.92 ± 0.02	0.51 ± 6.0	1.25 ± 0.13	4.91 ± 0.07	3.88 ± 0.10

^bThe letters A and B refer to detectors A and B; the first number refers to a row, and the second number refers to a column. Position 23-23 is the center of the core.

^cA1 is the coefficient of the exponential associated with $a1$.

^dA2 is the coefficient of the exponential associated with $a2$.

References:

1. W. G. Knapp and R. C. Doerner, "Physics Measurements with Modified Diluent Composition in Tungsten-Based, Aluminum-Reflected Fast Reactors", ARL-7207 (1967)
2. R. A. Karam, "Measurement of Rossi- α in Reflected Reactors" Trans. Am. Nucl. Soc. 7, 283 (1964)

12.1-3-6 Measurement of Lifetime

Type of Measurement: Rossi- α (2 counter)

Facility: ZPR-9 (1.3)

Assembly: 6 (2.3-6)

Experimental Method:

See Section 12.1-3-3.

Results:

Table 1
Least-squares Fitting of Rossi-alpha Data

Run	Location of A Detector ^b	Location of B Detector ^b	Channel Width μ sec	Bkgd/ 10^6	$A1/10^{10}$ ^c	$\alpha 1/10^{-4}$	$A2/10^{10}$ ^d	$\alpha 2/10^{-4}$
6-1	23-23	23-24	0.2	3.02 ± 0.01	1.70 ± 0.10	2.36 ± 1.10	1.60 ± 0.02	4.67 ± 0.31
6-2	23-23	23-30	0.2	1.33 ± 0.07	3.33 ± 0.11	2.60 ± 0.41	0.57 ± 0.01	4.10 ± 0.30

^bThe letters A and B refer to detectors A and B; the first number refers to a row, and the second number refers to a column. Position 23-23 is the center of the core.

^c $A1$ is the coefficient of the exponential associated with $\alpha 1$.

^d $A2$ is the coefficient of the exponential associated with $\alpha 2$.

References:

1. W. G. Knapp and R. C. Doerner, "Physics Measurements with Modified Diluent Composition in Tungsten-Based, Aluminum-Reflected Fast Reactors", ANL-7207, (Feb. 1967)
2. R. A. Karam, "Measurement of Rossi- α in Reflected Reactors" Trans. Am. Nucl. Soc. 7, 283 (Nov. 1964)

12.1-3-7 Measurements of Lifetime

Type of Measurement: Rossi- α

Facility: ZPR-9 (1.3) Assembly: 7 (2.3-7)

Experimental Method:

Two U-235 fission chambers were placed near the core center. The data were fit to three exponentials.

Results:

$$\begin{aligned}\alpha_1 &= 8.58 \pm 0.92 \times 10^5 \text{ sec}^{-1} \\ \alpha_2 &= 6.36 \pm 0.58 \times 10^4 \text{ sec}^{-1} \\ \alpha_3 &= 1.51 \pm 0.10 \times 10^3 \text{ sec}^{-1} \\ \beta_{eff}(\text{calc}) &= 0.00660 \\ \text{lifetime} &= \frac{\beta_{eff}}{\alpha_3} = 4370 \text{ nsec.}\end{aligned}$$

Reference:

R. C. Doerner, W. G. Knapp, R. A. Karam, and D. K. Butler,
"ZPR-9 Assemblies No. 6-9 Critical Experiments", ANL-7208
(March 1967)

12.1-3-8

Measurements of Lifetime

Type of Measurement: Rossi- α

Facility: ZPR-9 (1.3)

Assembly: 8 (2.3-8)

Experimental Method:

See Section 12.1-3-7.

Results:

$$\alpha_1 = 5.8 \pm 1.7 \times 10^4$$

$$\alpha_2 = 3.5 \pm 0.5 \times 10^3$$

$$\alpha_3 = 1.27 \pm 0.27 \times 10^3$$

$$\beta_{\text{eff}} (\text{calc}) = 0.00664$$

$$\text{lifetime} = \frac{\beta_{\text{eff}}}{\alpha_3} = 5230 \text{ nsec.}$$

Reference:

R. C. Doerner, W. G. Knapp, R. A. Karam, and D. K. Butler,
"ZPR-9 Assemblies No. 6-9 Critical Experiments", ANL-7208
(March 1967)

12.1-3-9

Measurements of Lifetime

Type of Measurement: Rossi- α

Facility: ZPR-9 (1.3)

Assembly: 9 (2.3-9)

Experimental Method:

See Section 12.1-3-7.

Results:

$$\alpha_1 = > 10^6 \text{ sec}^{-1} \text{ (est.)}$$

$$\alpha_2 = 1.67 \pm 0.28 \times 10^5 \text{ sec}^{-1}$$

$$\alpha_3 = 1.55 \pm 0.02 \times 10^3 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (calc)} = 0.00659$$

$$\text{lifetime} = \frac{\beta_{\text{eff}}}{\alpha_3} = 4250 \text{ nsec.}$$

Reference:

R. C. Doerner, W. G. Knapp, R. A. Karam, and D. K. Butler,
"ZPR-9 Assemblies No. 6-9 Critical Experiments", ANL-7208
(March 1967).

SECTION 12.2 PERTURBATIONS ON LIFETIME MEASUREMENTS

12.2-1-20

Perturbations on Lifetime Measurements

Type of Measurement: Effect of counter position on Rossi- α measurements

Facility: ZPR-3 (1.1)

Assembly: 20 (2.1-20)

Experimental Details:

Thirteen runs were made with different locations of the initiating and terminating counters. BF_3 proportional counters were used, one for initiation and a bundle of four (in parallel) for termination. The various counter positions used are shown in Figure 1. The data were fit to a single exponential (plus background).

Graphical summary of counter runs in Assembly 20. Each arrow represents a run; the tail of the arrow at the location of the initiating counter and the point at the position of the terminating counter.

Figure 1

Results:

No significant difference in alpha was found for any of the counter arrangements. However, the total probability of the terminating counter detecting a chain related neutron was found to depend on position. The results are shown in Figure 2 where the points are time integrated probability of coincidence, the crosses are the fluxes and the triangles are the ratios of probability to flux. Thus the probability is shown to be proportional to the flux.

Figure 2

Reference:

G. S. Brunson, et al., "A Survey of Prompt - Neutron Lifetimes in Fast Critical Systems", ANL-6681 (1963)

12.2-1-42

Perturbations on Lifetime Measurements

Type of Measurement:

Effect of counter position on Rossi- α measurements in a zoned core.

Facility: ZPR-3 (1.1)

Assembly: 42 (2.1-42)

Experimental Details:

Two BF_3 counters were located in three different arrangements. For each case the data were fit to a single exponential.

Results:

a) Both initiating and terminating counters at center of core	$\alpha \times 10^{-4} \text{ sec}^{-1}$
3.68
b) Initiating counter at center of core, terminating counter in driver region.	3.79
c) Reverse of (b)	3.67

Average lifetime of a), b), and c) taking β_{eff} as 0.0072:

$$= \frac{\beta_{\text{eff}}}{\alpha} = 19.4 \times 10^{-8} \text{ sec}$$

Reference:

P. I. Amundson, et al., "A Two-Zone Fast Critical Experiment (ZPR-III Assembly 42)," ANL-6733 (1964).

Type of Measurement:

Effect of counter position on Rossi- α measurements in a zoned assembly.

Facility: ZPR-3 (1.1)

Assembly: 43 (2.1-43)

Experimental Details:

Two He^3 counters were located in four arrangements: a) both in the central zone, b) both in the driver, c) the initiating counter in the central zone and the terminating zone in the driver and d) the reverse of c). The measured decays were fit to single exponentials.

Results:

The α 's for the four cases were the same within experimental errors, the decay being fit to a single exponential.

$$\text{average } \alpha \text{ (delayed critical)} = 2.42 \pm 0.1 \times 10^4 \text{ sec}^{-1}$$

$$\beta_{\text{eff}} \text{ (estimated)} = 0.00705$$

$$\text{lifetime} = \beta_{\text{eff}}/\alpha = 2.9 \times 10^{-7} \text{ sec}$$

Reference:

J. M. Gasidlo, J. K. Long and W. P. Keeney, "Doppler Effect Measurements, Sodium Void Coefficients, and Critical Studies of a 5,000-liter Fast Power Breeder Reactor by the Two-Zone Method," ANL-6838 (1964).

Type of Measurement:

Effect of counter position and length of time channel on Rossi- α measurement.

Facility: ZPR-9 (1.3)

Assembly: 2 (2.3-2)

Experimental Details:

Two U-235 fission chambers were located at radial distances d_1 and d_2 from the core center near the core mid-plane. The data were analyzed with single exponentials.

Results:

d_1 , cm	d_2 , cm	Time Channel	$\alpha \times 10^4 \text{ sec}^{-1}$
		μs	
0	16.6	0.4	9.2 ± 0.7
0	16.6	0.8	7.8 ± 0.4
0	16.6	2.0	7.8 ± 0.4
0	0	0.4	10.9 ± 1
0	0	0.8	7.3 ± 0.7
16.5	0	0.4	10.8 ± 1.2

Reference:

R. C. Doerner, W. G. Knapp, K. K. Almenas, and R. A. Karam, "Physics Measurements in Tungsten-Based, Aluminum-Reflected Fast Reactors", ANL-7007 (March 1967)

Type of Measurement:

Effect of counter position on Rossi- α measurement.

Facility: ZPR-9 (1.3)

Assembly: 4B (2.3-4B)

Experimental Details:

Two U-235 fission chambers were placed at variable radial positions near the core midplane. The data were fit to two exponentials.

Results:

Table 1
Least-squares Fitting of Rossi-alpha Data

Run*	Location of A Detector ^b	Location of B Detector ^b	Channel Width μ sec	Bkgd. 10^4	$A_1/10^4$ ^c	$\lambda_1/10^{-6}$ ^b	$A_2/10^4$ ^d	$\lambda_2/10^{-6}$ ^e
4B-1	23-23	23-24	0.2	2.08 $\pm$ 0.01	1.26 $\pm$ 0.34	0.94 $\pm$ 0.34	4.10 $\pm$ 0.08	7.00 $\pm$ 0.16
4B-2	23-23	23-25	0.2	1.83 $\pm$ 0.01	8.56 $\pm$ 8.40	3.98 $\pm$ 3.0	3.90 $\pm$ 0.04	7.31 $\pm$ 0.24
4B-3	23-23	23-27	0.2	1.12 $\pm$ 0.00	1.74 $\pm$ 1.40	1.79 $\pm$ 1.02	3.36 $\pm$ 0.10	7.17 $\pm$ 0.14
4B-4	23-23	23-28	0.2	6.92 $\pm$ 0.01	1.60 $\pm$ 0.70	1.64 $\pm$ 0.51	1.62 $\pm$ 0.03	6.96 $\pm$ 0.16
4B-5	23-23	23-28	0.2	1.19 $\pm$ 0.01	0.45 $\pm$ 0.20	0.98 $\pm$ 0.50	1.58 $\pm$ 0.03	7.10 $\pm$ 0.20
4B-6	23-27	23-28	0.2	3.45 $\pm$ 0.01	0.42 $\pm$ 0.20	0.71 $\pm$ 0.50	1.56 $\pm$ 0.07	6.75 $\pm$ 0.42

*The letters A and B refer to detectors A and B; the first number refers to a row, and the second number refers to a column. Position 23-23 is the center of the core.

^c A_1 is the coefficient of the exponential associated with λ_1 .

^e A_2 is the coefficient of the exponential associated with λ_2 .

References:

1. W. G. Knapp and R. C. Doerner, "Physics Measurements with Modified Diluent Composition in Tungsten-Based, Aluminum-Reflected Fast Reactors", ANL-7207 (Feb. 1967)
2. R. A. Karam, "Measurements of Rossi- α in Reflected Reactors", Trans. Am. Nucl. Soc. 7, 283 (Nov. 1964)

SECTION 13 - CONTROL ROD EFFECTS

This section compiles experiments on control rod effects that could be used in developing analytical techniques. Routine control rod worth measurements made for operational purposes are not included.

13-1-48A

Control Rod Effects

Type of Measurement:

Reaction rate distributions with and without insertion of B_4C control rods in reflector.

Facility: ZPR-3 (1.1)

Assembly: 48A (2.1-48A)

Experimental Method:

See Section 5.2-1-48. Figure 1 shows the core configuration during these experiments. Details of the drawer structures are given in the reference. Pu, U-238 and B-10 chambers were traversed radially through the core and reflector with and without the presence of B_4C rods at the inner edge of the Inconel reflector.

Results:

The results are shown in Figures 2-4.

Reference:

A. L. Hess and Q. L. Baird, "Reaction-Rate Distributions in ZPR-3 Assemblies 48, 48A, and 48E" ZPR-TM-6 (1969)

Fig. 1 : Assembly 48A Drawer Arrangement in Half No. 1 for Radial Traverses Through B₄C Mockup Rods in the Inconel Reflector

FIG. 2 : Radial Profiles of ^{239}Pu Fission Rate at Midplane of Assembly 48A With and Without B_4C Rods in the Reflector

Fig. 3 Radial Profiles of ^{235}U Fission Rate at Midplane of Assembly 48A With and Without B_4C Rods in the Reflector

Fig. 4

Radial Profiles of $^{10}\text{B}(n, \alpha)$ Reaction Rate at Midplane of Assembly 48A
With and Without B_4C Rods in the Reflector.

SECTION 14 - NEUTRON IMPORTANCE

14-2-4Z

Neutron Importance

Facility: ZPR-6 (1.2)

Assembly: 4Z (2.2-4Z)

Experimental Method:

A 1-microgram quantity of Cf-252, housed in a container and attached to a drive mechanism was used for the source of fission neutrons. See Figure 1. The core materials were arranged as shown in Figures 2, 3 and 4 of 5.2-2-4Z to accommodate the container and drive screw.

Figure 1
Californium Sample Holder

The procedure was:

1. Criticality was established with one control rod, with source out.
2. The source was inserted to a specified location.
3. The linear rise in power level was monitored by a fission chamber located at the driver-reflector interface.

The procedure above was carried out both with and without the Cf-252 source. The Cf-252 sample holder was found to have no measurable effect on reactivity anywhere in the assembly.

14-2-4Z (cont'd)

Precision:

The uncertainties in the measured points are within the circles on Figures 2 and 3.

Results:

Fig. 2 Radial Traverse of Cf-252 Neutron Importance

Fig. 3 Axial Traverse of Cf-252 Neutron Importance

Reference:

R. A. Karam, et al., "Measured Physics Parameters in a Zoned Fast Uranium Carbide Core, Assembly 4Z" ANL-7281 (1966)

14-2-5

Neutron Importance

Facility: ZPR-6 (1.2)

Assembly: 5 (2.2-5)

Experimental Method:

See Section 14-2-4Z.

Results:

Figure 1
Importance of Fission Neutrons as a Function of
Radius in Assembly 5 of ZPR-6

Reference:

R. A. Karam, et al., "Measured Physics Parameters in a Large Uranium Carbide Fast Core, Assembly 5 ZPR-6", Reactor Physics Division Annual Report - July 1, 1966 to June 30, 1967, ANL-7310, 167 (1968)

14-3-11

Neutron Importance

Facility: ZPR-9 (1.3)

Assembly: 11 (2.3-11)

Experimental Method:

Traverse Cf-252 Source. See Section 14-2-4Z.

Results:

Fig. 1 Distribution of Importance of Fission Neutrons in ZPR-9 Assembly 11

Normalization integral = $4.22 \pm 0.2 \times 10^5 \text{ sec}^{-1}$ (by normalizing $\sum_i (\nu \Sigma_f)_i \phi_i = 1$ and $\sum_i \chi_i \phi_i^* = 1$ at the core center).

Reference:

R. A. Lewis, L. G. LeSage, and R. A. Karam, "ZPR-9 Assemblies 11 and 12 - Two Small Zoned Versions of the 2600-Liter Assembly 5 ZPR-6", ANL Reactor Physics Division Annual Report, July 1, 1966 to June 30, 1967, 190 (1968)

14-3-12

Neutron Importance

Facility: ZPR-9 (1.3)

Assembly: 12 (2.3-12)

Experimental Method:

Traverse Cf-252 source. See Section 14-2-4Z.

Results:

Fig. 1 Distribution of Importance of Fission Neutrons in Assembly 12 of ZPR-9

Normalization integral = $4.47 \pm 0.2 \times 10^5 \text{ sec}^{-1}$ (obtained by normalizing $\sum_i (v \Sigma_f)_i \phi_i = 1$ and $\sum_i \chi_i \phi_i^* = 1$ at the core center).

Reference:

R. A. Lewis, L. G. LeSage, and R. A. Karam, "ZPR-9 Assemblies 11 and 12 - Two Small Zoned Versions of the 2600-Liter Assembly 5 ZPR-6", ANL Reactor Physics Division Annual Report, July 1, 1966 to June 30, 1967, ANL-7310, 190 (1968)