REACTOR PHYSICS DEPARTMENT
TECHNICAL ACTIVITIES QUARTERLY REPORT
JANUARY, FEBRUARY, MARCH, 1967

OCTOBER 1967

**AEC RESEARCH &
DEVELOPMENT REPORT**

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3000 STEVENS DRIVE, P. O. BOX 999, RICHLAND, WASHINGTON 99352

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REACTOR PHYSICS DEPARTMENT
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JANUARY, FEBRUARY, MARCH, 1967

By
The Reactor Physics Department Staff
F. G. Dawson, Manager

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FOREWARD

The Technical Activities Quarterly Report replaces the Physics Research Quarterly Report previously published by Battelle-Northwest's Reactor Physics Department. The objective of the report is to inform the scientific community in a timely manner of the technical progress made on the many phases of work within the department. The report contains brief technical discussions of accomplishments in all areas where significant progress has been made during the quarter. The results presented should be considered preliminary, and do not constitute final publication of the work. A list of publications and papers by the department staff is given in the report. Anyone wishing to obtain additional information on the work presented is encouraged to contact the author directly.

PREVIOUS REPORTS IN THIS SERIES

HW-42181	October, November, December	1955
HW-43441	January, February, March	1956
HW-44525	April, May, June	1956
HW-47012	July, August, September	1956
HW-48893	October, November, December	1956
HW-50598	January, February, March	1957
HW-51983	April, May, June	1957
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HW-54591	October, November, December	1957
HW-55879	January, February, March	1958
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HW-57861	July, August, September	1958
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HW-61181	April, May, June	1959
HW-62727	July, August, September	1959
HW-63576	October, November, December	1959
HW-64866	January, February, March	1960
HW-66215	April, May, June	1960
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HW-68389	October, November, December	1960
HW-69475	January, February, March	1961
HW-70716	April, May, June	1961
HW-71747	July, August, September	1961
HW-72586	October, November, December	1961
HW-73116	January, February, March	1962
HW-74190	April, May, June	1962
HW-75228	July, August, September	1962
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HW-77871	April, May, June	1963
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HW-80020	October, November, December	1963
HW-81659	January, February, March	1964
HW-83187	April, May, June	1964
HW-84369	July, August, September	1964
HW-84608	October, November, December	1964
BNWL-95	January, February, March	1965
BNWL-149	April, May, June	1965
BNWL-193	July, August, September	1965
BNWL-222	October, November, December	1965
BNWL-284	January, February, March	1966
BNWL-315	April, May, June	1966
BNWL-340	July, August, September	1966
BNWL-400	October, November, December	1966

TABLE OF CONTENTS

1.0	SUMMARY	1.1
2.0	REACTOR THEORY AND CODE DEVELOPMENT	2.1
	BNW Master Cross Section Library-Status Report -	
	K. B. Stewart	2.1
	An Angular Synthesis Approximation Applied to Thin Critical	
	Slabs - H. S. Zwibel.	2.1
	Computer Code Abstracts	2.3
3.0	THERMAL REACTORS.	3.1
	Comparison of PCTR Measurements Using Unpoisoned and Poisoned	
	Techniques on PuO ₂ -UO ₂ Graphite Systems	
	Comparison of PCTR Measurements Using Unpoisoned and Poisoned	
	Techniques on PuO ₂ -UO ₂ Graphite Systems - R. I. Smith,	
	L. D. Williams and E. C. Davis, Jr.	3.1
	The Application of Multigroup Neutron Transport Theory to	
	Phoenix Fuel Cores - C. M. Heeb.	3.9
	Pre-Experimental Calculations of the PRCF - Phoenix Core -	
	G. J. Busselman and D. N. Adrian.	3.11
	Investigation of Power Peak Suppressors in the MTR-Phoenix	
	Core - W. W. Porath	3.12
	PRCF-Phoenix Fuel Experiment - R. I. Smith, L. D. Williams,	
	and E. C. Davis	3.14
	Burnup Data from Low Exposure Plutonium-Aluminum Fuel -	
	D. E. Christensen, E. B. Reppond and W. A. Reardon.	3.18
	Burnup Data from High Exposure Plutonium-Aluminum Fuel -	
	D. E. Christensen, E. B. Reppond and W. A. Reardon.	3.18
	Analysis of Plutonium-Aluminum Fuel Burnup Data -	
	R. P. Matsen.	3.18
	A Fitting Formula For Critical Mass Data: H ₂ O Moderated,	
	Cylindrical Arrays of Rods - W. P. Stinson.	3.19
	FERTF Zero-Power Tests in the PRTR - J. W. Kutcher and	
	J. H. Lauby	3.23
	Correlations of Effective Cross Section Ratios of Plutonium	
	Isotopes - R. C. Liikala and W. L. Purcell.	3.25
	Photoneutron Calculations for D ₂ O Systems - U. P. Jenquin . .	3.26
	The Dancoff Correction and its Effect on Resonance Integrals	
	and Reactivity - G. L. Simmons.	3.27
	Power Peaking in UO ₂ -2 wt% PuO ₂ -H ₂ O Cores - V. O. Uotinen	
	and S. Kobayashi.	3.30
	Calculations for H ₂ O Moderated Lattices of UO ₂ -4 wt% PuO ₂	
	Fuel Rods - S. Kobayashi and V. O. Uotinen	3.31
4.0	FAST REACTORS	4.1
	Effect of BeO on FTR Neutronics - R. B. Kidman and	
	W. W. Little.	4.1
	FTR Control Rods - R. A. Harris	4.2
	Analysis of Reactor Transients Induced by Fuel Rearrangement-	
	W. W. Little.	4.6
	Considerations on the Fuel-Rod Plug Concept - A. E. Waltar. .	4.9
	Analysis of SEFOR Central Sodium Void Experiment - L. L. Maas	4.13
	Doppler Coefficient Studies - R. E. Schenter	4.14

Development of PERT, A 2-D Perturbation Code -	
R. Wayne Hardie	4.20
FTR Phase-A Critical Experiments and Preliminary Analyses -	
R. A. Bennett, S. L. Engstrom and Q. L. Baird	4.20
FTR Shielding Studies - D. R. Marr, C. A. Mansius and	
W. L. Bunch	4.36
Gamma Intensity from FTR Sodium - R. E. Shaver, C. A. Mansius	
and W. L. Bunch	4.42
5.0 CRITICAL MASS PHYSICS	5.1
Basic Criticality Experiments with Plutonium Nitrate Solutions	
in Slab Geometry - R. C. Lloyd, E. D. Clayton, and	
L. E. Hansen.	5.1
Criticality Experiments with Casks of Plutonium Metal -	
R. C. Lloyd and E. D. Clayton	5.4
Critical Mass and Buckling Measurements for 2.1 wt% Enriched	
Uranium Fuel Tubes in Light Water - L. E. Hansen.	5.6
Criticality of Poisoned PuO ₂ Fueled Assemblies - L. E. Hansen,	
R. C. Lloyd, and S. R. Bierman.	5.8
Minimum Critical Enrichment of Hydrogenous Uranyl Nitrate	
Systems - S. R. Bierman and G. M. Hess.	5.9
Rossi-Alpha Measurement - G. M. Hess.	5.12
P ₀ Method of Noise Analysis - G. M. Hess.	5.14
Plutonium Criticality Studies - Undermoderated Region -	
S. R. Bierman, L. E. Hansen, R. C. Lloyd and E. D. Clayton. .	5.15
6.0 NEUTRON PHYSICS	6.1
Completion of Fast-Neutron Total-Cross-Section Measurements -	
D. G. Foster, Jr. and D. W. Glasgow	6.1
7.0 PUBLICATIONS AND PRESENTATIONS.	7.1
Publications.	7.1
Papers Accepted for Publication	7.1
Papers Presented.	7.2
Papers Accepted for Presentation.	7.2
8.0 DISTRIBUTION.	8.1

REACTOR PHYSICS DEPARTMENT
TECHNICAL ACTIVITIES QUARTERLY REPORT
JANUARY, FEBRUARY, MARCH, 1967

1.0 SUMMARY

REACTOR THEORY AND CODE DEVELOPMENT

Eighteen isotopes in the BNW Master Cross Section Library have been updated with additional and more accurate data.

Angular synthesis and variational methods applied to analyzing a thin critical slab, produce more accurate results than diffusion theory.

THERMAL REACTORS

Measurements to obtain adjoint weighted excess neutron production cross sections, $\langle \Sigma_p \rangle$, are complete for four lattices of $\text{PuO}_2\text{-UO}_2$ fuel in graphite moderator. The measurements were made in the Physical Constants Test Reactor (PCTR), and a value of the infinite medium neutron multiplication factor (k_∞) was obtained for one lattice using the traditional PCTR null reactivity technique. This value of k_∞ is compared with the corresponding value obtained from the measured $\langle \Sigma_p \rangle$ for the same lattice. The agreement of the values for k_∞ is encouraging, and further evaluation of the comparison will be made when similar values are available for other lattices.

Analysis of the CAF-Phoenix water reflected core with 17 energy groups yields a calculated reactivity within

1% of experiment using diffusion theory and 3% using transport theory.

Pre-experiment calculations on the PRCF-Phoenix core revealed that k dropped as more core-reflector interface detail was added. Final critical loading estimates were less than 9% low.

Flux suppressors for the MTR-Phoenix core must be carefully placed to avoid having the far-out flux increases outweigh the close-in flux decreases.

A full-core banked-shim critical loading was achieved in the PRCF-Phoenix fuel experiment, with the shims 65% withdrawn. Power distribution measurements in the shimmed core have shown the expected power peaking in the fuel followers just below the bottom of the core.

Compilations have been issued of the burnup data obtained by destructive analysis of Al-1.8 wt% Pu and Al-2.6 wt% Pu fuels which were irradiated in PRTR.

Analysis of the burnup data from the Aluminum-Plutonium fuels irradiated in PRTR has been completed. The isotopic concentration data were processed using multivariable regression analysis to obtain a unique set of cross section ratios.

An empirical formula has been derived for use in fitting critical mass data

from cylindrical arrays of rods, moderated with H_2O . The formula provides a method of accurate interpolation to obtain the fuel-to-moderator ratio for minimum critical mass.

Relative rod power measurements have been made in PRTR to determine the power match between fuel rods in elements in the Batch Core and fuel rods in several possible test configurations in the Fuel Element Rupture Test Facility (FERTF). The relative rod powers were determined by gamma scanning. The reactivity changes associated with loss of coolant from the PRCF have also been measured for several fuel compositions and test configurations.

A theory-experiment correlation study of ratios of effective cross sections has been performed. A result of the study is that the calculated ratios agree best with the experimental ratios using the Leonard normalization for the ^{239}Pu and ^{241}Pu thermal cross sections.

Calculations of the photoneutron production in D_2O moderated reactor systems fueled with 19-rod clusters of Al-Pu, UO_2 - PuO_2 , and UC have been made.

A calculational study has been made comparing values of the Dancoff correction obtained via various methods. The corresponding effect of resonance integrals and reactivity have also been determined.

Calculations have been made of power peaking factors in H_2O moderated loadings of UO_2 -2 wt% PuO_2 fuel. The calculated values agree with measured values to within $\pm 5\%$.

Calculations have been performed to help predict critical masses and various lattice parameters for H_2O moderated lattices containing UO_2 -4 wt% PuO_2 fuel rods. The study was made assuming two types of fuel differing in ^{240}Pu content.

FAST REACTORS

A considerable nuclear design effort on FTR has been mounted during the past quarter.

A survey has been performed to investigate the effect of BeO on FTR neutronics.

Items considered included the sodium coefficient, Doppler coefficient, critical mass, and flux spectrum dependence on BeO volume fractions.

A detailed analysis of FTR control rod configurations has been carried out. A variety of representative rod patterns were examined and their reactivity worth determined.

The effects of fuel failure and fuel relocation (slumping) in the FTR are being re-examined. A series of kinetics calculations have been performed for a range of reactivity ramps and shutdown coefficients. The possible utilization of fuel-rod plugs to minimize the effects of fuel slumping has also been considered.

To check calculational techniques used in FTR design, analysis of the sodium void experiments performed on the SEFOR mockup in ZPR-III has been undertaken. A fairly detailed analysis of the Doppler coefficient measurements in the same assembly is also being

carried out. Incorporation of crystal-line binding effects in the analysis caused the Doppler coefficient to diverge from the $1/T$ dependence in qualitative agreement with the experimental results.

The two-dimensional perturbations code, 2-D PERT, is now operational. This code will be used for driver fuel and test management studies.

The Phase-A critical experiments in ZPR-III have been completed, and analysis of the experiments has started.

The attenuation characteristics of a number of alternate shield arrangements have been compared analytically to provide a basis for the conceptual design of the FTR. Estimates of gamma intensity at selected locations within two different closed loop cell concepts have also been made.

CRITICAL MASS PHYSICS

Criticality experiments, bare and water reflected, were continued with the 42 x 42 in. slab assembly having adjustable thickness. The plutonium concentration was 58 g/l and acid molarities were 2.3 and 5. The plutonium contained 4.6 wt% ^{240}Pu . Evaluation of the effect of vessel walls, lattice reinforcement, and hood walls was made. Correlation of slab experimental data and initial calculation theory were in poor agreement. The clean critical bare and water reflected infinite slab was experimentally estimated to be 15.7 and 10.1 cm respectively for 58 g Pu/l at an acid molarity of 2.3.

Criticality experiments were performed to provide data for nuclear

safety guidance on handling, storing, and shipping of United Kingdom nut type casks containing 6.6 kg plutonium metal each. Experiments were carried out with the nine casks bare and reflected with Lucite. Effect of Lucite moderator and cadmium plating was studied qualitatively. The bare array indicated criticality at about 19 casks; the reflected array gave 10-1/2 casks for criticality.

Critical bucklings and masses were measured for a range of lattice spacings of 2.1 wt% enriched U fuel tubes in light water. The data provide additional guidance for handling and processing 2.1 wt% enriched fuel tubes.

Criticality experiments were performed in support of the Gas Cooled Fast Breeder Reactor (GCFR) program. The experiments were designed to simulate water entry into the GCFR core and to check basic neutronic data and computational techniques. The analysis of the experiments is being performed at General Atomic.

An experimental program to provide data for determining the minimum critical ^{235}U enrichment of hydrogenous uranyl nitrate systems was completed in the PCTR. Data reduction and analysis is currently in progress. Some experimental results are presented as raw data.

The dead time problem connected with Rossi-alpha measurement using multi-channel equipment has been circumvented by using, essentially, a multiple single channel approach. A system has been assembled employing fast solid state equipment in hope that it will be useful for epithermal and fast neutron systems.

A system for experimentally measuring the P_0 probability of reactor noise for various time intervals has been assembled in hopes of providing an independent measurement of the prompt neutron decay constant.

A new series of critical experiments was begun at the Critical Mass Laboratory with PuO_2 -polystyrene compacts and the Remote Split-Table Machine. These experiments are a continuation of the basic research program to provide data for evaluating the effects of moderation and ^{240}Pu on intermediate neutron spectra plutonium systems.

The current series of experiments are concerned with fuel having an atomic H/Pu of 5 and a ^{240}Pu isotopic content of 11.5 wt%.

NEUTRON PHYSICS

The experimental program to measure the total neutron cross section of 78 naturally-occurring elements and 14 separated isotopes for $2.25 \leq E_n \leq 15.0$ MeV has been completed. Data reduction has been completed and the results are ready for analysis, publication, and distribution.

2.0 REACTOR THEORY AND CODE DEVELOPMENT

BNW MASTER CROSS SECTION LIBRARY- STATUS REPORT K. B. Stewart

A supplement to BNWL-CC-325 containing all isotopes updated between September 1, 1965 and April 1, 1967 will be published in early April. Prior to publication the updated isotopes will be plotted by the automated plotting routines of the Master Library for a final consistency check.

Since October 1, 1966 additional and more accurate data, obtained primarily from BNWL-325, Supplement 2, and KFK 352, have been added to the Master Library for the following isotopes:

1. Boron-10

2. Copper-63 and 65
3. Silver-107 and 109
4. Indium-115
5. Iodine-127
6. Praseodymium-141
7. Samarium-147, 149, and 151
8. Europium-151 and 153
9. Tungsten-183
10. Gold-197
11. Thorium-228, 232, and 234
12. Protactinium-231 and 233
13. Uranium-233, 235, and 238
14. Plutonium-238, 239, 240 and 243
15. Americium-241 and 243
16. Curium-243, and 244
17. Lutetium-175 and 176
18. Tantalum-181

AN ANGULAR SYNTHESIS APPROXIMATION APPLIED TO THIN CRITICAL SLABS^(a) H. S. Zwibel

It is well known that diffusion theory is inadequate for small (on the order of 2 or 3 mean-free-paths) systems. The main reason for this inadequacy is due to the highly anisotropic nature of the flux for these small systems. By expanding the angular dependence into forward and backward directions (rather than as a

two term Legendre polynomial expansion), a considerable improvement can be achieved.

This is demonstrated quite clearly by solving the criticality problem for a homogeneous infinite slab (in the one-speed approximation) with isotropic scattering. The Boltzmann equation for this system is

$$\frac{\mu}{2} \frac{\partial}{\partial z} [f(z, \mu)] + f(z, \mu) = \frac{c}{2} \int_{-1}^1 f(z, \mu) d\mu.$$

-1

(a) Abstract of paper to be presented at the 13th Annual Meeting at the American Nuclear Society, June 11-15, 1967 at San Diego, California

μ is the particle's direction cosine with respect to the slab normal, and

c is the eigenvalue to be determined.
 $c = (\Sigma_s + v\Sigma_f)/\Sigma_t$; z is the distance
 from the center of the slab measured
 in mean-free-paths, i.e. $z = \Sigma_t x$, where
 x is the physical distance.

It is now assumed that $f(z, \mu)$ can
 be adequately represented by

$$f(z, \mu) = g(\mu, b) [R(z) H(\mu) + L(z) H(-\mu)]$$

where $H(\mu)$ is the heavyside step
 function, and $g(\mu, b)$ is a known trial
 function, which contains an adjustable
 parameter, b . The approximation, when
 inserted into a variational expression
 for c , yields equations for $R(z)$ and
 $L(z)$. The solution of these equations
 for c gives

$$a = \frac{\bar{\mu}(b)/\sqrt{\gamma}}{\sqrt{c - \gamma(b)}} \tan^{-1} \left([c/\gamma(b)] - 1 \right)^{-1/2}$$

where a is the half-width of the slab
 and $\bar{\mu}(b)$ and $\gamma(b)$ are

$$\bar{\mu}(b) = \int_0^1 \mu g^2(\mu, b) d\mu$$

$$\gamma(b) = \int_0^1 g^2(\mu, b) d\mu.$$

It is apparent that c will depend on
 the value of b . The optimum value for
 b is obtained by solving the equation

$\frac{dc(b)}{db} = 0$, i.e., that value of b for
 which c is stationary.

This has been done for two different
 choices of $g(\mu, b)$. These are

$$g_1(\mu, b) = \frac{1 - b\mu}{1 - b/2}$$

$$g_2(\mu, b) = \frac{bc^{-b\mu}}{1 - e^{-b}}$$

For small b these functions are
 equal, but they are quite different for
 larger b . The eigenvalues are presented
 below in Table 2.I for several different
 thicknesses (c_1 results from g_1 , and c_2
 results from g_2) and are compared with
 the exact results due to Mitsis⁽¹⁾ (c_M)
 and ordinary P_1 theory (c_D). The
 total fluxes are compared in Table 2.II.

References

1. G. J. Mitsis. *Transport Solutions to the Monoenergetic Critical Problems*, Argonne National Laboratory, ANL-6787, 44 1963.

TABLE 2.I. Eigenvalue Comparison

$2a$	c_1	c_2	c_D	c_M
1.780	1.29	1.29	1.38	1.30
1.022	1.56	1.56	1.77	1.60
.621	1.98	1.95	2.40	2.00

TABLE 2.II. Flux Comparison

z/a	$2a = 1.780$			$2a = 1.022$			$2a = 0.621$		
	ρ_2	ρ_M	ρ_D	ρ_2	ρ_M	ρ_D	ρ_2	ρ_M	ρ_D
0	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
0.25	0.96	0.97	0.97	0.97	0.97	0.98	0.97	0.98	1.00
0.50	0.84	0.87	0.89	0.88	0.89	0.93	0.90	0.92	0.98
0.75	0.66	0.71	0.76	0.73	0.77	0.84	0.77	0.80	0.96
1.00	0.43	0.45	0.59	0.54	0.55	0.71	0.60	0.62	0.94

COMPUTER CODE ABSTRACTSComputer Code - Dancoff

1. NAME OR DESIGNATION OF CODE - DANCOP
2. COMPUTER FOR WHICH CODE IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC 1107
3. NATURE OF PHYSICAL PROBLEM SOLVED - Computes the Dancoff correction factor to the resonance escape probability.
4. METHOD OF SOLUTION - Dancoff-Ginsburg's and Sauer's Approximations.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - Valid for Square and Hexagonal Lattices and for the nearest 10 sets of rods.
6. TYPICAL RUNNING TIME - Approximately 5 sec.
7. UNUSUAL FEATURES OF THE PROGRAM - None
8. RELATED AUXILIARY PROGRAMS - None
9. STATUS - In production use.
10. REFERENCES - G. L. Simmons and W. L. Purcell, BNWL-CC-1099, Comparison of Methods for Obtaining Dancoff Corrections.
11. MACHINE REQUIREMENTS - UNIVAC 1107, approximately 5 K total memory requirements.
12. PROGRAMMING LANGUAGE(S) USED - 100% FORTRAN IV
13. OPERATING SYSTEM - CSC EXEC II, Package B
14. OTHER INFORMATION - None.
15. NAME AND ESTABLISHMENT OF AUTHOR -
G. L. Simmons
W. L. Purcell
Battelle-Northwest; P.O.
Box 999; Richland,
Washington 99352
16. MATERIAL AVAILABLE - Authority has been granted to release the following material for use subject to further testing (source program approximately 300 cards). Material can be obtained directly from the Analytical Physics Unit, Battelle-Northwest.

3.0 THERMAL REACTORSCOMPARISON OF PCTR MEASUREMENTS USING
UNPOISONED AND POISONED TECHNIQUES ON
PuO₂-UO₂ GRAPHITE SYSTEMSR. I. Smith, L. D. Williams,
E. C. Davis, Jr.Introduction and Summary

The infinite medium neutron multiplication factors (k_{∞}) for many reactor lattices have been measured in the Physical Constants Test Reactor (PCTR).^(1,2) The basic technique used to determine k_{∞} is to accurately measure the amount of neutron absorber required to absorb the excess neutrons of a test sample. This technique is called the poisoned technique, and the required amount of absorber is determined by a null reactivity condition on the test sample.

An unpoisoned technique has been developed recently⁽³⁾ which gives information about the neutron multiplication properties of a medium without having to add absorber to the medium. More specifically, an adjoint weighted excess neutron production cross section, $\langle \Sigma_p \rangle$, can be measured.

The experimentally obtained quantity, $\langle \Sigma_p \rangle$, is directly comparable to multi-group calculations. A value of $k_{\infty} - 1$ may then be obtained from $\langle \Sigma_p \rangle$ by including a calculated leakage dependent term.

This report presents the experimental results obtained from a series of PCTR experiments on graphite moderated lattices with square cells containing a single 0.5 in. diam rod of PuO₂-UO₂ fuel. Values of $\langle \Sigma_p \rangle$ are reported for all lattices studied. Values of k_{∞} evaluated by the poisoned technique and the unpoisoned technique have been obtained for one lattice and are presented. Corresponding values for the additional lattices will be reported as they are obtained. The experimental results are summarized in Table 3.I.

Discussion

The method for obtaining k_{∞} from experimental data by using the poisoned technique has been previously described

TABLE 3.I. *Summary of Results*

Lattice Pitch	$\frac{\text{Pu}}{\text{PuO}_2\text{-UO}_2}$	$\frac{^{240}\text{Pu}}{\text{Pu}}$	$\frac{^{235}\text{U}}{\text{U}}$	$\langle \Sigma_p \rangle_0$	k_{∞}	k_{∞}
in.	wt%	at. %	at. %	cm ⁻¹	Unpoisoned	Poisoned
2.0	0.76	7.24	0.234	$3.1 \pm 0.1 \times 10^{-4}$		
2.0	2.29	29.34	0.182	$-2.8 \pm 0.2 \times 10^{-4}$		
4.0	0.76	7.24	0.234	$1.21 \pm 0.02 \times 10^{-3}$	1.339	1.339
4.0	2.29	29.34	0.182	$2.47 \pm 0.05 \times 10^{-3}$		

in detail.^(1,2) The presence of plutonium in the system requires a modified development for k_∞ . The following expression for k_∞ is developed in the Appendix and is used to evaluate k_∞ via the poisoned technique. Quantities used in the calculation are given in Table 3.II.

$$k_\infty = \frac{TP}{A_2} \frac{A'_2 + A'_{2a} + pA'_{1a}}{TP'} \quad (3.1)$$

To obtain expressions for $\langle \sum_p \rangle$ and k_∞ by using the unpoisoned technique, we start with a general perturbation formula,⁽³⁾

$$\chi \Delta \rho^{\text{cell}} = \sum_{i=1}^m \phi_i^+ \int_S J_i^S dS \quad (3.2)$$

The change in reactivity, $\Delta \rho^{\text{cell}}$, is due to the insertion of a cell with volume V and surface area S into a void in the center of the reactor. The neutron importance, ϕ_i^+ , is evaluated at the surface of the void, and the current, J_i^S , is evaluated at the surface of the cell for the energy group i . If the surface current J_i^S is averaged over the surface S ,

$$\chi \Delta \rho^{\text{cell}} = \sum_{i=1}^m \phi_i^+ J_i^S \quad (3.3)$$

The quantity χ is a normalization involving an integration over the whole reactor. The necessity for calculating χ is removed if the reactivity worth, $\Delta \rho^a$, of a standard neutron absorber is also measured. An equation similar to (3.2) can be written for $\Delta \rho^a$. In the case of the absorber, however, the integral of the neutron current over the surface of the absorber equals

the absorption rate in the absorber. The result can be expressed as follows:

$$\chi \Delta \rho^a = \frac{1}{\phi_{th}^+} \sum_{i=1}^m \phi_i^+ (\sum_i \phi_i V)_a \quad (3.4a)$$

If the absorber is primarily a thermal neutron absorber with some epithermal absorptions, Equation (3.4a) can be rewritten in two-group theory as

$$\chi \Delta \rho^a = -\phi_{th}^+ [A_{2a} + pA_{1a}] \quad (3.4b)$$

The quantities A_{2a} and A_{1a} are thermal and epithermal absorption rates in the absorber, and the quantity p is approximately the ratio of epithermal to thermal neutron importances, ϕ_1^+ / ϕ_{th}^+ . The division between thermal and epithermal energies is the cadmium cutoff energy for 0.040 in. cadmium 0.64eV.

If Equation (3.3) is divided by Equation (3.4b), the following relationship can be obtained:

$$\left(\frac{-\Delta \rho^{\text{cell}}}{\Delta \rho^a} \right) \frac{(A_{2a} + pA_{1a})}{S} = \frac{1}{\phi_{th}^+} \sum_{i=1}^m \phi_i^+ J_i^S \quad (3.5)$$

The goal of this development is to obtain an expression for the adjoint weighted excess neutron production cross section, $\langle \Sigma_p \rangle$. The quantity $\langle \Sigma_p \rangle$ is normalized to the thermal flux at the surface of the cell. It is convenient, experimentally, to use copper as the absorber and to define the thermal flux at the surface to be that flux measured by the thermal activation of $1/v$ detector at the surface.

TABLE 3.II. Quantities Used in k_{∞} Calculations

A. Total Production - Unpoisoned Case

Material	ν	σ_f, b	g	Concentration, Atoms/ 10^{-24} cm^3	Total Activation	Relative Production
^{235}U	2.442	577.1	0.9739	4.968×10^{-5}	11.115	0.758
^{239}Pu	2.898	740.6	1.0545	1.6964×10^{-4}	13.125	5.039
^{241}Pu	3.01	950.0	1.0494	1.09×10^{-6}	13.125	0.043

TP = 5.840

B. Total Thermal Absorption - Unpoisoned Case

Material	$(1+a)g\sigma_f, b$	σ_a, b	Concentration, Atoms/ 10^{-24} cm^3	Volume, Vol. Fuel	Thermal Activation	Relative Absorption
^{235}U	659.2	---	4.968×10^{-5}	1.0	10.711	0.351
^{239}Pu	1146.5	---	1.6964×10^{-4}	1.0	12.777	2.485
^{241}Pu	1304.0	---	1.09×10^{-6}	1.0	12.777	0.018
^{240}Pu	---	290	1.322×10^{-5}	1.0	10.396	0.040
^{234}U	---	95	1.09×10^{-6}	1.0	10.396	0.001
^{238}U	---	2.73	2.118×10^{-2}	1.0	10.396	0.601
Zr	---	0.188	4.1418×10^{-2}	0.2545	11.173	0.022
Graphite	---	0.0038	8.2121×10^{-2}	78.988	12.596	0.310

 $A_2 = 3.828$

C. Total Production - Poisoned Case

Material	ν	σ_f, b	g	Concentration, Atoms/ 10^{-24} cm^3	Total Activation	Relative Production
^{235}U	2.442	577.1	0.9739	4.968×10^{-5}	7.510	0.512
^{239}Pu	2.898	740.6	1.0545	1.6964×10^{-4}	8.956	3.439
^{241}Pu	3.01	950.0	1.0494	1.09×10^{-6}	8.956	0.029

TP' = 3.980

D. Total Thermal Absorption - Poisoned Case

Material	$(1+a)g\sigma_f, b$	σ_a, b	Concentration, Atoms/ 10^{-24} cm^3	Volume, Vol. Fuel	Thermal Activation	Relative Absorption
^{235}U	661.5	---	4.968×10^{-5}	1.0	7.208	0.237
^{239}Pu	1157.4	---	1.6964×10^{-4}	1.0	8.729	1.714
^{241}Pu	1305.0	---	1.09×10^{-6}	1.0	8.729	0.012
^{240}Pu	---	290	1.332×10^{-5}	1.0	7.232	0.028
^{234}U	---	95	1.09×10^{-6}	1.0	7.232	0.001
^{238}U	---	2.73	2.118×10^{-2}	1.0	7.232	0.418
Zr	---	0.188	4.1418×10^{-2}	0.2545	7.435	0.015
Graphite	---	0.0038	8.2121×10^{-2}	78.988	8.358	0.205

 $A'_2 = 2.650$

E. Absorption in Copper Poison

σ_a, b	Concentration, Atoms/ 10^{-24} cm^3	Volume, Vol. Fuel	Thermal Activation	A'_{2a}	Epithermal Activation	pA'_{1a}
380	3.4907×10^{-2}	0.7191	8.518	0.813	0.562	0.049

The function $g(\Delta\phi, \Delta\phi^+)$ is related to the flux mismatch between the fluxes in the test cavity and the equilibrium fluxes characteristics of the medium. The correction due to flux mismatches was set equal to zero in this analysis because measurements indicated the mismatch was small. The leakage correction term, $\tau^2 B^4$, was evaluated to be 0.033 using MOFDA⁽⁷⁾ to obtain L^2 and τ , and age diffusion theory to obtain B^2 .

Analysis of Experimental Results

Reactivity measurements were obtained from reactor period data, and reaction rate measurements were inferred from volume and flux weighted foil activation averages.

The data for the poisoned technique measurement and values of the quantities in Equation (3.1) are presented in Table 3.II. The microscopic cross sections, σ_f and σ_a , and neutrons per fission, ν , are the values from BNL-325.⁽⁴⁾ Westcott g factors⁽⁵⁾ for a temperature of 28.5 °C were used to correct the 2200 m/sec fission cross sections. Activations were measured for copper absorption, ^{235}U fission, and Pu fission relative to a thermal spectrum. The activation of ^{239}Pu and ^{241}Pu were assumed to be the same. The absorption of ^{235}U , ^{239}Pu , and ^{241}Pu were calculated from the fission foils using values of α obtained from calculations using THERMOS.⁽⁶⁾ Absorptions in all other cell components were assumed to be proportional to the measured copper activation. An approximate value of 0.91 was determined for p with the computer

code MOFDA.⁽⁷⁾ The value of k_∞ is not sensitive to this number since A_{1a} is small.

By using the unpoisoned technique the value of k_∞ was determined using Equation (3.9). The quantities used are tabulated in Table 3.II. The total thermal absorption in the cell is taken from Table 3.II and renormalized for the total volume instead of unit fuel volume. The value of k_∞ using the unpoisoned technique agrees with the value from the poisoned measurements to within much less than the minimum estimated uncertainty of 1% in $(k_\infty - 1)$.

TABLE 3.III. Quantities Used in Unpoisoned Technique k_∞ Calculation

$\Delta\rho^{\text{cell}}$	$58.5 \pm 0.1\%$
$-\Delta\rho^{\text{Cu}}$	$0.092 \pm 0.001\%/g$
σ^{Cu}	3.8 b
V^{cell}	$2.0975 \times 10^{-4} \text{ cm}^3$
A_{2a}	12.842
A_{1a}	0.757
p	0.914
$(\int \phi V)^{\text{cell}}$	$(3.828) (259.47) = 993.25$
M^2	641.5 cm^2
τ	354 cm^2
B^2	$5.27 \times 10^{-4} \text{ cm}^{-2}$

That is,

$$\phi_{th_0} = \frac{V_a \int_0^{0.64} \sum_a (E) \phi^{s'}(E) dE}{V_a \int_{0a}} = \frac{A_{2a}^s}{V_a \int_{0a}} \quad (3.6a)$$

where $\sum_o$ is the 2200 m/sec cross section. For a $1/v$ detector,

$$\phi_{th_o} \equiv v_o \int_o^{0.64} n^{s'}(E) dE \quad (3.6b)$$

and Equation (3.5) can now be rewritten as

$$\begin{aligned} & \left(-\frac{\Delta \rho^{cell}}{\Delta \rho^a} \right) \frac{(A_{2a} + pA_{1a})}{\phi_{th_o} S} \\ &= \frac{1}{\phi_{th}^+ \phi_{th_o}} \sum_{i=1}^m \phi_i^+ J_i \quad (3.7) \end{aligned}$$

Equation (3.7) defines an adjoint weighted neutron leakage from the cell per unit surface per unit thermal flux. The excess production cross section is the total leakage from the cell divided by the volume of the cell.

$$\begin{aligned} \langle \sum_p \rangle_o &= - \left(\frac{\Delta \rho^{cell}}{\Delta \rho^a} \right) \frac{V_a \sum_{oa}}{V_{cell}} \frac{(A_{2a} + pA_{1a})}{A_{2a}^s} \\ &= \left(\frac{S}{V} \right) \frac{1}{\phi_{th}^+ \phi_{th_o}} \sum_{i=1}^m \phi_i^+ J_i \quad (3.8) \end{aligned}$$

Equation (3.8) contains two quantities which equal $\langle \sum_p \rangle_o$. One quantity is capable of experimental determination, and the other can be obtained from calculations. The subscript on $\langle \sum_p \rangle$ is a reminder that this cross section is normalized to the 2200 m/sec cross section of the poison, $\sum_{oa}$. The same normalization must be used in calculations. No multigroup calculations of $\langle \sum_p \rangle$ have been obtained as

yet for comparison, and the values in Table 3.I. are intended as an invitation for those who are so inclined to test the theoretical models used in the various codes.

An expression for k_∞ using the unpoisoned technique can be obtained after adopting a two-energy group model,⁽³⁾

$$\begin{aligned} k_\infty &= 1 + \frac{\langle \sum_p \rangle \phi_{th} V_{cell}}{(\sum \phi V)_{th}^{cell}} \\ &+ g(\Delta \phi, \Delta \phi^+) + \tau^2 B^4 \quad (3.9) \end{aligned}$$

General Remarks

This paper presents a first attempt to compare results of an unpoisoned technique measurement with results of a poisoned technique measurement. Measurements of reactor lattice parameters in the High Temperature Lattice Test Reactor (HTLTR) would be much simplified by use of an unpoisoned technique because of the high temperatures anticipated. Consequently, there is considerable interest in doing unpoisoned experiments and improving techniques for future application in HTLTR, as well as PCTR, measurements.

The agreement of the values for k_∞ in Table 3.I. is encouraging, but final evaluation of the comparison must be delayed until similar comparisons are available for other lattices. It is also of interest to make multigroup calculations of $\langle \sum_p \rangle$ for the development of experiment-theory correlations in the near future.

APPENDIX

The Theoretical Model Used for k_{∞} in the Poisoned Technique

Identification of k_{∞} with k_{eff}

The following development is based on the definition of the effective neutron multiplication factor (k_{eff}) as the total neutron production divided by the sum of the total loss due to absorption and leakage. Such a definition is in accord with standard multigroup calculations of k_{eff} and leads to the usual (and physically meaningful) definition of the neutron lifetime. The values of k_{∞} calculated from these experiments are consistent with this definition of k_{eff} .

Using a two-energy group formulation, the neutron balance equations can be written as

$$D_1 \nabla^2 \phi_1 - \sum_1 \phi_1 \sum_{1 \rightarrow 2} \phi_1 + [(\nu \sum_f)_1 \phi_1 + (\nu \sum_f)_2 \phi_2] = 0 \quad (A-1a)$$

and

$$D_2 \nabla^2 \phi_2 - \sum_2 \phi_2 + \sum_{1 \rightarrow 2} \phi_1 = 0 \quad (A-1b)$$

The notation can be simplified if we rewrite these two equations as

$$-L_1 A_1 P_{21} + P_{11} + P_{12} = 0 \quad (A-2a)$$

and

$$-L_2 -A_2 + P_{21} = 0 \quad (A-2b)$$

The symbols A_i and L_i refer to absorption and leakage, respectively, in the energy group i . The symbol P_{ij} refers to the production of neutrons into energy group i due to reactions in energy group j .

The total production of neutrons (TP) and the total loss of neutrons due to absorption (TA) and leakage (TL) can be written as follows:

$$TP = P_{11} + P_{12} \quad (A-3)$$

$$TA = A_1 + A_2 \quad (A-4)$$

and

$$TL = L_1 + L_2 \quad (A-5)$$

Simultaneous solution of Equations (A-2a) and (A-2b) for the total production of neutrons gives

$$TP = P_{11} + P_{12} = A_1 + A_2 + L_1 + L_2 \quad (A-6)$$

The standard form⁽⁸⁾ of the equations relating k_{eff} and k_{∞} in two group theory can be written as

$$k_{eff} = k_{\infty} \left(\frac{\text{nonleakage probability from Group 1}}{\text{nonleakage probability from Group 2}} \right)$$

or

$$k_{eff} = k_{\infty} \left(\frac{1}{1 + L_1/A_1 + P_{21}} \right) \times$$

$$\left(\frac{1}{1 + L_2/A_2} \right) \quad (A-7)$$

The two group equation for k_{∞} becomes

$$k_{\infty} = \frac{A_1 + A_2 + L_1 + I_2}{A_2} \times \left(\frac{P_{21}}{A_1 + P_{21}} \right) \quad (A-8)$$

A small amount of algebra using Equations (A-6), (A-7), and (A-8) leads to the original definition of k_{eff} .

$$k_{\text{eff}} = \frac{TP}{TA + TL} \quad (A-9)$$

The Evaluation of the Experimental k_{∞}

The PCTR method determines the k_{∞} of an infinite unpoisoned lattice by poisoning the lattice so that the infinite medium neutron multiplication factor of the poisoned lattice, k'_{∞} , is unity. In the following development the unprimed symbols refer to quantities in the unpoisoned lattice, and the primed symbols refer to the corresponding quantities in the poisoned lattice.

Equation (A-8) can be used to write expressions for the infinite neutron multiplication factor for the respective poisoned and unpoisoned lattices.

$$k_{\infty} = \frac{A_1 + A_2 + L_1 + L_2}{A_2} \times \left(\frac{P_{21}}{A_1 + P_{21}} \right) \quad (A-10a)$$

$$k'_{\infty} = 1 = \frac{A'_1 + A'_2 + L'_1 + L'_2 + A'_{2a} + A'_{1a}}{A'_2 + A'_{2a}} \times \left(\frac{P'_{21}}{A'_1 + A'_{1a} + P'_{21}} \right) \quad (A-10b)$$

The symbols A'_{2a} and A'_{1a} are the respective subcadmium and epicadmium absorptions in the poison.

If we define neutron escape probabilities in the unpoisoned and poisoned lattices, respectively, as

$$p = \frac{P_{21}}{P_{21} + A_1} \quad \text{and} \quad p' = \frac{P'_{21}}{P'_{21} + A'_1} \quad (A-11)$$

and if we make use of Equation (A-6), Equations (A-10a) and (A-10b) become

$$k_{\infty} = \frac{TP}{A_2} p \quad (A-12)$$

and

$$k'_{\infty} = 1 = \frac{TP'}{A_2 + A_{2a}} p' \times \left(\frac{1}{1 + A'_{1a} / \left(A'_1 + P'_{21} \right)} \right) \quad (A-13)$$

The unpoisoned k_{∞} can be written as Equation (A-12) divided by Equation (A-13)

$$k_{\infty} = \frac{TP}{A_2} \frac{A'_2 + A'_{2a}}{TP'} \frac{p}{p'} \times \left[1 + \frac{A'_{1a}}{A'_1 + P'_{21}} \right] \quad (A-14)$$

Equation (A-14) is almost the expression used to calculate k_{∞} from the experimental data. Some algebra is required to obtain the final result together with the assumption that the addition of the poison does not change the escape probability (i.e., $p=p'$, if adding the poison does not change the neutron moderating properties of the lattice).

After the escape probabilities are canceled, Equation (A-14) can be written as

$$k_{\infty} = \frac{TP}{TP'} \left[\frac{A'_2 + A'_{2a}}{A_2} + \frac{(A'_2 + A'_{2a})}{A'_1 + P'_{21}} \times \frac{A'_{1a}}{A_2} \right] \quad (A-15)$$

In an infinite system,

$$P'_{21} = A'_2 + A'_{2a} \quad (A-16)$$

and

$$\frac{A'_2 + A'_{2a}}{A'_1 + P'_{21}} = p = p' \quad (A-17)$$

The expression used to evaluate the experimental k_{∞} values is obtained by substituting Equations (A-17) into (A-15). The result is

$$\begin{aligned} k_{\infty} &= \frac{TP}{A_2} \frac{A'_2 + A'_{2a} + pA'_{1a}}{TP'} \\ &= \left[1 + \frac{A'_{2a} + pA'_{1a}}{A'_2} \right] \times \left(\frac{TP}{A_2} \right) \left(\frac{A'_2}{TP'} \right) \end{aligned} \quad (A-18)$$

If the addition of the poison does change the slowing down properties of the lattice so that $p \neq p'$, the result is expressible in the form

$$k_{\infty} = \frac{TP}{A_2} \left[\frac{(p/p') (A'_2 + A'_{2a}) + pA'_{1a}}{TP'} \right]$$

These expressions were used to evaluate k_{∞} by the poisoned technique as given in Table 3.I. Thus, the theory-experiment correlation for these values are directly applicable for multigroup calculations with one note of caution: if the calculations are made using a

basic cross section set other than the one used to analyze the experiment, a review should be made to assure that the cross sections are consistent. However, the value of k_{∞} derived from the experiment is less sensitive to the absolute cross sections since in essence $(k_{\infty} - 1)$ is the cross section dependent value. If $k_{\infty} = 1$, cross section information is not required to interpret the experiment.

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THE APPLICATION OF MULTIGROUP NEUTRON
TRANSPORT THEORY TO PHOENIX FUEL CORES

C. M. HEEB

The most frequently envisioned Phoenix fuel environment is a small, strongly reflected, very hard spectrum reactor. Using the results from approach-to-critical experiments⁽¹⁾, an investigation of the applicability of various standard calculational methods to this neutronic system is currently underway. A very strong group structure dependence has been found for transport theory models of this system. The experimental assembly will be reviewed, and the results of the transport calculations will be given.

The experimental system consists of a 19 can bundle of 2-in. nominal diameter aluminum cans loaded with 20 wt% Pu-Al alloy fuel discs alternated with polyethylene moderator wafers. The entire 19 can bundle is immersed in water which provides an effectively infinite neutron reflector. The cans are in the close-packed hexagonal array with sides touching. The principal moderator is therefore the polyethylene with the water in the reflector and only in the interstitial positions of the core. The proportion of metal and polyethylene is chosen so that the overall metal-to-water ratio of the assembly is close to one. The plutonium used has 8% ²⁴⁰Pu. The H-to-Pu ratio is close to 200, which yields a computed fast-to-thermal flux ratio greater than 6 for the pure core. It therefore admirably meets the requirements for first generation Phoenix core neutronics.

With the use of 17 group, one-dimensional diffusion theory with standard cross sections, the experimental subcritical multiplication can be calculated to within less than 1%. Four-group diffusion theory yields an over-estimation of k_{eff} by 2%. An attempt has been made to calculate k_{eff} for the system with multi-group neutron transport theory, consisting of the four double angle approximation (S_4), using four energy groups⁽²⁾, and isotropic scattering. The calculated value of k_{eff} is 11% above the comparable diffusion theory calculational results. The source of this surprisingly large discrepancy turns out to be the calculation of core leakage. By comparing diffusion theory results with transport theory for various hypothetical systems, it is clear that the magnitude of the discrepancy increases as the core size decreases, as the system is reflected, and as the isotropic approximation is made. The small strongly-reflected Phoenix core seems cunningly designed to provide the largest possible discrepancy.

To provide accurate estimates of Phoenix system multiplications using multigroup transport theory one must attempt to estimate the core leakage energy dependence more accurately. This has been done by increasing the number of energy groups in the transport calculation. At 17 energy groups the difference between experiment or diffusion theory and S_4 transport theory is reduced to 3% above the experimental result. This remaining disagreement with experiment must be ascribed to spectrum mixing at the

core reflector interface and to possible chemical binding effects above the upper energy of the single thermal group (0.683 eV). These effects are not included in the multigroup cross sections. The spectrum averaged multigroup cross sections were obtained from HRG* and THERMOS.⁽³⁾ The detailed computation results are displayed in Table 3.III.

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*HRG is a site revision of the PI Multigroup Slowing Down Code GAM

TABLE 3.III. Comparison of Various Transport and Diffusion Theory Results

Case Description	No. of Groups	Transport ^(a) $k_{eff}^{(b)}$	Diffusion $k_{eff}^{(b)}$
1. 19 can core, 30 in. high, water ref.	4	1.08	0.97
2. Core, no leakage	4	1.66	1.66
3. Core, axial leakage only	4	1.59	1.59
4. Core, with radial leakage only	4	1.12	1.01
5. Core, with radial leakage only, S ₈ transport approx.	4	1.12	----
6. Core, radius increased 6 cm at const. ref. thickness	4	1.29	1.21
7. Core, radius decreased by 6 cm at const. ref. thickness	4	0.85	0.73
8. Unreflected core	4	0.85	0.75
9. 19 can core, 30 in. high, water ref., anisotropic scattering in lab. sys.	4	1.07	----
10. 19 can core, 30 in. high, water ref., anisotropic approximation	17	0.98	0.95

(a) Unless indicated specifically, all transport calculations utilize the S₄ approximation, and the isotropic scattering approximation in the laboratory system. Transport results are computed by using DTF-IV,⁽²⁾ and diffusion theory results are obtained from HFN.⁽⁴⁾

(b) Estimated experimental k_{eff} is 0.95

PRE-EXPERIMENTAL CALCULATIONS OF THE
PRCF - PHOENIX CORE

G. J. Busselman and D. N. Adrian

Calculations of the PRCF-Phoenix Core have been underway for some time. Table 3.IV. is a summary of these calculations to date. The 2-D diffusion code EXTERMINATOR is used in all cases as is the 1965 Sher data for ^{239}Pu . Thermal cross sections are obtained from the THERMOS portion of the HAMMER code, and the epithermal values are from the HRG code. The plutonium composition used is 20 wt% plutonium in aluminum with the following isotopic distribution: 76.6/19.3/3.5/0.6 for $^{239}\text{Pu}/^{240}\text{Pu}/^{241}\text{Pu}/^{242}\text{Pu}$. Although the "as fabricated" isotopic analysis is not yet available, no serious departure from the above data is expected.

These same methods were used to calculate the CAF-Phoenix experiment and resulted in an over-prediction of the reactivity by between 2.0 and 3.5% in Δk . A 17 group diffusion theory calculation of the CAF-Phoenix core matches the experiment to less than 1 mk. Similar calculations will eventually be carried out for the PRCF-Phoenix cores.

The earliest calculations were carried out before the beryllium reflector of the PRCF experiment was stacked in place; hence, the MTR reflector con-

ditions were assumed. After the reflector was in place, the water content was estimated to be 10 vol%. A measurement of this quantity will be attempted after completion of the experiment. The critical loading prediction was about 11 boxes. At this point a difference between the assumed and actual mockups of the PRCF reflector was discovered. The simulated aluminum beam tubes were too close to the core and the core beryllium interface was not a very good representation of the MTR situation. After all of the near core water gaps had been included in the calculation, the predicted critical loading was about 13 boxes. On the final day of the experimental approach, the heterogeneous aspects of the beam tubes were examined with a side view calculation of the core. As much as 4% reactivity could be tied up in this effect.

The above calculations were carried out for symmetric core loadings of 6, 9, 12, and 27 fuel boxes and the predictions were made by reading curves. After the critical size was established at 16.5 fuel boxes, the number 5 calculation of Table 3.IV. was repeated with refined water gap dimensions. The resulting k was 1.055. If the CAF error and the maximum beam tube heterogeneity effect are additive and correct for this system, the predicted multiplication would be between 0.995 and 0.980.

TABLE 3.IV. Summary of Phoenix Core Calculations

No. of Boxes to be Critical	Calculated k at Measured Critical, i.e., 16.5 Boxes	Calculated k Adjusted by CAF Experience	Condition of Reflector
1. 10.2-10.8	1.132	1.112-1.097	2% H ₂ O in Be, i.e., MTR condition
2. 10.6-11.2	1.124	1.104-1.089	10% H ₂ O in Be app. PRCF condition
3. 10.8-11.5	1.120	1.100-1.085	Above plus Be partially removed and Al beam tubes added in a homogenized representation
4. 11.5-12.3	1.102	1.082-1.067	Above plus water gas adjacent to core caused by interstices of Be tube reflectors
5. 12.7-13.8	1.075	1.055-1.040	Above plus water gap at control sheet position
6. Case 5 = 0.04, if looking at the end of the core rather than the top Δk for smeared Al tubes to heterogeneous Al. This case was run the day the system went critical.			
6'. 15.2-16.5	1.035	1.015-1.000	Cases 5 and 6 combined.

INVESTIGATION OF POWER PEAK SUPPRESSORS IN THE MTR-PHOENIX CORE

W. W. Porath

Preliminary investigation of the Hx-Pu core⁽¹⁾ to be installed in the MTR led to the discovery of large power peaks below the core in the follower fuel when the control rods were inserted. The peaking was found to be a minimum when the control rods were completely withdrawn, and a maximum when the control rod tips were flush with bottom of the stationary fuel.

To counteract this effect and thereby raise the operating power level, two methods of reducing the thermal flux spikes were considered; borated and fittings below the core, and borated side plates on that side of the stationary fuel next to the beryllium reflector. All of the calculations

were performed using the two-group, three-dimensional diffusion code WHIRLAWAY.⁽²⁾

Two concentrations of boron were used in each case in addition to the normal, or unborated, condition. The concentrations used and the effects on the power peaking are given in Table 3.V. In each condition investigated, the control rods were partially inserted into the core with the top of the follower fuel 45.2 cm below the top of the stationary fuel.

There was, however, a considerable change in the control rod model between calculations of the two types of flux suppressors. The poisoned sections of the hollow rods were made thinner and considerably blacker. An error in the normalization of the total control rod worth was also corrected. As can be seen from Table 3.V., this results in

an increase in rod strengths of about 2.5%. A further consequence of these changes and perhaps of better flux convergence in the side plate calculations, the point of maximum power shifted from below the core in the fuel followers to the center of the core at the extreme bottom of the fixed fuel. There is still a large but not maximum power peak below the core in the followers. The conclusions drawn are valid if one considers each type of suppressor separately.

The conclusions drawn from this investigation are that the borated end fittings are the best method of reducing the power peak below the core. The optimum amount of boron necessary to reduce the peak while not adversely affecting the core life may lay between the values examined.

The borated side plates will reduce the peaking along the edge of the core, but they adversely affect the peaking in the area of its maximum value. Thus, more selective location of flux suppressors will be needed if they are to be used to provide an optimum benefit for the least reactivity loss.

References

1. W. W. Porath. Preliminary 3-D, 2-Group Analysis of a Full Pu, MTR-Phoenix Configuration, Pacific Northwest Laboratory, Richland, Washington October, 1966.
2. T. B. Fowler, M. L. Tabias. WHIRLAWAY - A Three Dimensional, Two-Group Neutron Diffusion Code for the IBM-709 Computer, ORNL-3150, Oak Ridge National Laboratory, August, 1961.

TABLE 3.V. Effects of Boron Concentrations

Case	Boron Conc, N/b-cm	Boron in Al, wt%	$\hat{p}/\bar{p}$, Borated End Fittings (a)	Borated Side Plates (b)
1	0.0	0.0	4.83	3.38
2	3.0×10^{-5}	0.02	4.23	3.41
3	3.0×10^{-4}	0.20	3.27	3.52

Case	k_{eff}		Point/p Along Be-Core Interface (Bottom center of core)	
	Borated End Fittings (a)	Borated Side Plates (b)	Borated End Fittings (a)	Borated Side Plates (b)
1	1.058	1.032	4.11	1.65
2	1.039	1.026	4.00	1.63
3	0.996	0.984	3.08	1.50

(a) Old control rod model

(b) New control rod model

PRCF-PHOENIX FUEL EXPERIMENT

R. I. Smith, L. D. Williams, and
E. C. Davis

Installation and functional testing of the mechanical structure for the Phoenix fuel core in the Plutonium Recycle Critical Facility (PRCF) was completed on schedule. A plane view of the reactor configuration is shown in Figure 3.1.

The initial critical loading was achieved with 16-1/2 fuel elements, in a 3 x 5 1/2 box array. No shim assemblies were included in this clean, critical core. An irradiation was performed to determine thermal flux distributions from copper pin activations and to determine power distributions by scanning fission product activities in the fuel plates. The measured distributions and the flux

FIGURE 3.1. PRCF - Phoenix Core Configuration

and power results from a 2-D calculation using the code EXTERMINATOR⁽¹⁾ are compared in Figures 3.2 and 3.3

Following the shim-free critical measurements, the fuel elements in the eight shim locations were replaced with the fuel-follower shim assemblies. The reactivity worth of a shim assembly, when installed with the fuel follower fully up in the core, was about two-thirds of the reactivity worth of the fuel element it replaced. A critical loading was achieved with a 3 x 9 array (19 elements, 8 shims), with the eight shims withdrawn 65%.

The reactivity worths of the safety sheets, located in the beryllium reflector, were measured in the 16 1/2 element clean core and in the 3 x 9

shimmed core. The worths were determined by comparing the rate of flux decay following a sheet drop to that calculated for various step changes in reactivity. The worth of all six sheets was about 15 mk in the 16 1/2 element core and about 14 mk in the full 3 x 9 core. The worth of a center sheet in the 16 1/2 element core was about 5 mk and in the 3 x 9 core was about 3.5 mk. There appeared to be a small interaction between the safety sheets since the measured worth of all sheets was slightly less than the sum of the measured worths of the individual sheets.

Power distributions were measured in the plates of a fuel follower of a shim which was withdrawn 65%. The fission

FIGURE 3.2. Midplane Core Traverse in North-South Direction 16 1/2 Element Clean Core

FIGURE 3.3. Midplane Core Traverse in East-West Direction 16 1/2 Element Clean Core

product activities in the fuel plates were measured using the Gamma Scanning facility at PRCF. The relative power distributions inferred from the plate scans are shown in Figure 3.4. The power peaking on the outside surfaces of the fuel follower is quite significant and may prove to be the limiting factor in power operation of a Phoenix core at MTR.

During PRCF operation with the full core in place, a steady loss of reactivity with time ($\sim 2\text{¢/hour}$) was observed.

It is presently believed that this phenomena was caused by formation of small bubbles of hydrogen from an aluminum-water reaction on the surfaces in the core. The fuel assemblies will be treated in hot H_2O for 6 to 8 hr to produce a good aluminum-oxide film on all surfaces to inhibit the bubble formation.

The core has been unloaded to permit restacking of the beryllium reflector in the region between the core tank wall and the safety sheet thimbles.

FIGURE 3.4. Axial Power Profile in Shim Fuel Follower Full Core - Banked Shims at 65% Withdrawn Scan of Fission Product Activities

As initially stacked, this region contained about 30% H_2O . The restacked region will present a nearly solid beryllium region at the core-reflector interface and will significantly improve the similarity between the PRCF-Phoenix core and a planned MTR-Phoenix core.

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BURNUP DATA FROM LOW EXPOSURE PLUTONIUM-ALUMINUM FUEL*

D. E. Christensen, E. B. Reppond,
W. A. Reardon

A compilation of all the burnup data obtained by destructive analysis from aluminum-1.8 wt% plutonium fuel

* *Summary of a report issued as
BNWL-CC-905.*

irradiated in the PRTR is completed. The initial ^{240}Pu isotopic concentration was 6.25 at.%. The data extends to 50% depletion of the initial plutonium. The compilation makes the burnup data, burnup equations, and burnup parameters used in data analysis, e.g., ^{137}Cs yield, etc., available to those who are interested in further analysis. It does not present any conclusions concerning the data analysis except to indicate bad data points.

BURNUP DATA FROM HIGH EXPOSURE PLUTONIUM-ALUMINUM FUEL*

D. E. Christensen, E. B. Reppond,
W. A. Reardon

A compilation of all the burnup data taken by destructive analysis from aluminum-2.6 wt% plutonium fuel irradiated in the PRTR is completed. The

initial ^{240}Pu isotopic concentration was 16.45 at.%. The data extends to 52% depletion of the initial plutonium. The compilation makes the burnup data available to those who are interested in further analysis. It does not present any conclusions concerning the data analysis except to indicate bad data points.

ANALYSIS OF PLUTONIUM-ALUMINUM FUEL BURNUP DATA

R. P. Matsen

A least squares analysis of four sets of plutonium-aluminum burnup data has been completed. Each set included plutonium isotopic concentrations obtained from thirty to forty irradiated PuAl fuel samples. The parameters determined using a subroutine written for a least squares fitting program⁽¹⁾ are the cross section ratios $\hat{\sigma}_c^{49}/\hat{\sigma}_a^{49}$,

$\hat{\sigma}_a^{40}/\hat{\sigma}_a^{49}$, and $\hat{\sigma}_c^{41}/\hat{\sigma}_a^{41}$. Detailed numerical results of these analyses will be presented in a forthcoming document.

The following comments are a result of the completed analysis.

- The plutonium burnup equations will fit the isotopic concentration data over a substantial range of values for a particular cross section ratio. Thus, a change in the value of a given cross section ratio is compensated by appropriate adjustments in the remaining ones.

A unique set of cross section ratios is obtained when one of them is preset and then held fixed during the iterative fitting process. For this purpose, $\hat{\sigma}_a^{41}/\hat{\sigma}_a^{49} = 1.143$ was held fixed because of the several cross section ratios it is the one that can be most reliably calculated from basic cross section data.

- The cross section ratio $\hat{\sigma}_a^{42}/\hat{\sigma}_a^{49}$ is poorly determined from the fitting process. A calculated value of 0.07 is used in the fitting equations.
- An excellent representation of the data is obtained by assuming that all cross section ratios except $\hat{\sigma}_a^{40}/\hat{\sigma}_a^{49}$ are constant as a function of exposure. When $\hat{\sigma}_a^{40}/\hat{\sigma}_a^{49}$ has the form $A + B/N^{40}$ (A and B are constants and N^{40} is the ^{240}Pu concentration), the value of the Mean Square Misfit obtained is a factor of two smaller than the Mean Square Misfit obtained with $\hat{\sigma}_a^{40}/\hat{\sigma}_a^{49}$ held constant.
- The results of the least squares

A FITTING FORMULA FOR CRITICAL MASS DATA H₂O MODERATED, CYLINDRICAL ARRAYS OF RODS

W. P. Stinson

A method of accurate interpolation of critical mass data and of determining the moderator-to-fuel ratio for maximum buckling or minimum critical mass is presented in this report. Some results of curve fitting for this type of data have been reported.^(1,2,3) The inadequacy of the current methods has led to the determination of an empirical equation which is fit to the experimental data. In order to increase

fits must be corrected for ^{241}Pu decay. Averaged over all data samples, ^{241}Pu loss by decay is approximately 5% of the ^{241}Pu loss by neutron absorption.

- Once the fit has been obtained, burnup (i.e., depletion) of the plutonium is accurately predicted from the cross section ratios of the fit. Experimentally, the burnup of each sample is determined by ^{137}Cs analysis.
- Reliable cross section ratio results could be obtained from a small number (say 8 to 12) of uniformly spaced (by exposure) samples.

Reference

1. B. H. Duane. "Machine Analysis Abstract -- Maximum Likelihood Nonlinear Vector Fields -- Battelle Northwest Program Likely," *Physics Research Quarterly Report*, Jan., Feb., March, 1966 pp 1-3, BNWL-284, 1966.

the confidence of the empirical method, an equation of similar form is established with a theoretical basis.

Empirical methods were used to develop the formula

$$\frac{1}{B^2} = \frac{A_1}{X - A_2} + \frac{A_3}{A_4 - X} \quad (3.1)$$

where $1/B^2$ is the inverse buckling, X is the moderator-to-fuel ratio, and A_1, A_2, A_3, A_4 are constants determined for each set of data. This equation represents the sum of two hyperbolas having A_2 and A_4 as asymptotes. The A_2 is the minimum lattice spacing or

moderator-to-fuel ratio for which the fuel geometry and composition can become critical and conversely, A_4 is the maximum lattice spacing or moderator-to-fuel ratio for critical.

By combining the two terms on the right of Equation (3.1) and inverting, we obtain an equation for the buckling

$$B^2 = \frac{X^2 - (A_2 + A_4)X + A_2 A_4}{(A_1 - A_3)X - (A_1 A_4 - A_2 A_3)} \quad (3.2)$$

The critical number of fuel rods (N) is related to the buckling through the equation

$$B^2 = \left(\frac{J_0}{R + \lambda} \right)^2 + \left(\frac{\pi}{H + 2\lambda} \right)^2 \quad (3.3)$$

where $R^2 = \frac{N\lambda^2\sqrt{3}}{2\pi}$, and λ is the lattice spacing and N is the number of fuel rods. Also, since the moderator-to-fuel ratio is proportional to the square of the lattice spacing, Equation (3.1) with the proper choice of parameters A_1, A_2, A_3 , and A_4 , will fit data representing.

$1/B^2$ vs X

$1/B^2$ vs λ

N vs X

N vs λ

Equation (3.2) is a good representation of experimental data. However, if a similar equation could be derived based on theoretical considerations, the use of Equation (3.2) for interpolation and slope evaluations would be more justified.

Fuel loadings, even though they may be small, will have a critical equation given by

$$k_{\text{eff}} = \frac{k_{\infty} e^{-B^2 \tau}}{1 + L^2 B^2} \quad (3.4)$$

where

k_{∞} = multiplication factors

B^2 = Buckling

L^2 = diffusion length

τ = age to thermal.

For small assemblies infinitely reflected by H_2O , the probability of thermal neutron leakage ($1 + L^2 B^2$) is small. The nonleakage probability of the slowing down neutrons is given by $e^{-B^2 \tau}$. The critical equation may be written in the form

$$k_{\text{eff}} = \frac{\eta \epsilon p f e^{-B^2 \tau}}{1 + L^2 B^2} = \text{CPf} \quad (3.5)$$

where

$$C = \frac{\eta \epsilon}{1 + L^2 B^2} \text{ and } P = p e^{-B^2 \tau}$$

η = number of fission neutrons per neutron capture

ϵ = fast fission factor

p = resonance escape probability

f = thermal utilization.

The P is the probability that a neutron will escape resonance capture while slowing down and will not leak from the system. C is a slowly varying parameter.

Equations (3.4) and (3.5) can be combined such that

$$\frac{k_{\infty} e^{-B^2 \tau}}{1 + L^2 B^2} = \text{CPf} \text{ or } B^2 = \frac{k_{\infty} e^{-B^2 \tau} - \text{CPf}}{L^2 \text{CPf}} \quad (3.6)$$

If we let

$$C_1 = \frac{C}{k_{\infty} e^{-B^2 \tau}} = \frac{n \epsilon}{(1 + L^2 B^2) k_{\infty} e^{-B^2 \tau}}$$

$$L^2 = L_o^2 \left(1 - \frac{C_2}{X}\right)$$

$$f = \frac{1}{C_3 X}$$

$$P = \left(1 - \frac{C_4}{X}\right),$$

then

$$B^2 = \frac{X^2 - \frac{C}{C_3} X + \frac{C_1}{C_3} C_4}{L_o^2 \frac{C_1}{C_3} X - L_o^2 \frac{C_1}{C_3} \left(C_2 + C_4 - \frac{C_2 C_4}{X}\right)} \quad (3.7)$$

Equation (3.7) can be compared directly with Equation (3.2) if the variable X in the term $\frac{C_2 C_4}{X}$ in Equation (3.7) compensates for the slowly varying parameters C_1 , C_2 , C_3 , and C_4 .

The validity of using the slowly varying parameters with a compensating

factor in Equation (3.7) is shown by using this equation to calculate B^2 for lattices composed of EBWR fuel rods,⁽⁴⁾ 48 in. long, 3/8 in. diam, clad in 0.027 in. Zircaloy-2, and containing 826.6 gm $\text{PuO}_2\text{-UO}_2$. The $\text{PuO}_2\text{-UO}_2$ is 1.50 wt% PuO_2 in UO_2 depleted to 0.16 wt% ^{235}U . Values of C_1, C_2, C_3 , and C_4 were determined using the value of ϵ calculated with the code GAM-TECH,⁽⁵⁾ and values of $B^2, k_{\infty}, L^2, \tau, P, f$, and n calculated with the code HRG.⁽⁶⁾ The values of $1/B^2$ calculated with Equation (3.7) are shown in Table 3.VI for comparison to values calculated from Equation (3.1).

Equation (3.1) was used to fit experimental data from the EBWR fuel rod assemblies using the generalized least squares fitting Program Learn.⁽⁷⁾ The four parameters for Equation (3.1) determined from the fit are listed in Table 3.VII. Calculated values obtained using the parameters and Equation (3.1) are listed in Table 3.VIII with the experimental data. The comparisons show that the fitting error of Equation (3.1) is less than the experimental error on the data.

Differentiating Equation (3.1) results in

TABLE 3.VI. Inverse Bucklings for EBWR Fuel Assemblies Calculated from Equation (3.7) Compared to Values Calculated from Equation (3.1)

ℓ	H/Pu	Values Calculated from Equation (3.7)	Values Calculated from Equation (3.1)	
		$1/B^2 (\text{cm}^2)$	$(1/B^2 \text{ vs } \ell)$	$(1/B^2 \text{ vs } H/\text{Pu})$
0.55	230.3	199.3	199.5 ± 2.2	199.7 ± 2.3
0.60	326.2	147.8	146.0 ± 2.0	146.3 ± 2.2
0.71	566.8	121.1	120.9 ± 1.3	121.4 ± 1.5
0.80	793.6	127.0	129.2 ± 1.7	129.5 ± 1.8
0.90	1077.4	166.0	165.6 ± 1.4	165.6 ± 1.4
0.93	1169.1	185.4	186.4 ± 1.9	186.8 ± 2.0

TABLE 3.VII. Calculated Values for the Parameters of Equation (3.1) to Fit Data for EBWR Fuel

$\underline{Y}$	$\underline{X}$	A_1	A_2	A_3	A_4
			For Experimental Data		
N	ℓ	76.2 $\pm$ 0.4	0.4955 $\pm$ 0.0003	48.5 $\pm$ 1.	1.085 $\pm$ 0.003
N	H/Pu	131,600. $\pm$ 760.	132.6 $\pm$ 0.5	215,800. $\pm$ 3,800	1,767 $\pm$ 10.
1/B ²	H/Pu	17,120. $\pm$ 160.	105. $\pm$ 1.	119,710. $\pm$ 810.	1,894. $\pm$ 5.
1/B ²	ℓ	11.5 $\pm$ 0.1	0.4738 $\pm$ 0.0007	34.7 $\pm$ 0.3	1.153 $\pm$ 0.002
For Data Calculated by Equation (3.7)					
1/B ²	ℓ	10.9 $\pm$ 1.3	0.4756 $\pm$ 0.0076	30.3 $\pm$ 3.0	1.116 $\pm$ 0.017
1/B ²	H/Pu	16,500. $\pm$ 2,400	106. $\pm$ 15.	103,000. $\pm$ 10,000	1,771. $\pm$ 56.

TABLE 3.VIII. Experimental Data From EBWR Fuel Compared to Values Calculated from Equation (3.1)

Experimental Data				Values Calculated from Equation (3.1)			
ℓ	H/Pu	N	1/B ² , cm ²	N	N	1/B ²	1/B ²
				N vs ℓ	N vs H/Pu	1/B ² vs H/Pu	1/B ² vs ℓ
0.55	230.3	1487 $\pm$ 3	208.3 $\pm$ 0.09	1487 $\pm$ 1	1487 $\pm$ 1	208.3 $\pm$ 0.2	208.3 $\pm$ 0.2
0.60	326.2	829 $\pm$ 7	153.6 $\pm$ 1.8	829 $\pm$ 1	829 $\pm$ 1	153.7 $\pm$ 0.1	153.8 $\pm$ 0.2
0.71	566.8	484 $\pm$ 1	127.4 $\pm$ 0.5	484.5 $\pm$ 0.7	482.9 $\pm$ 0.6	127.2 $\pm$ 0.1	127.0 $\pm$ 0.1
0.80	793.6	420 $\pm$ 1	133.5 $\pm$ 0.5	420.5 $\pm$ 0.8	420.8 $\pm$ 0.8	133.6 $\pm$ 0.1	133.6 $\pm$ 0.1
0.90	1077.4	452 $\pm$ 2	164.2 $\pm$ 0.5	450.8 $\pm$ 0.7	452.2 $\pm$ 0.6	164.2 $\pm$ 0.1	164.4 $\pm$ 0.1
0.93	1169.1	488 $\pm$ 2	181.2 $\pm$ 0.7	489 $\pm$ 1	487.8 $\pm$ 0.9	181.2 $\pm$ 0.1	181.1 $\pm$ 0.2

$$\frac{d(1/B^2)}{dX} = \frac{A_3}{A_4 - X} - \frac{A_1}{X - A_2} \quad (3.8)$$

The moderator-to-fuel ratio for maximum buckling and for the minimum critical mass can be easily obtained from Equation (3.8) as well as the temperature coefficient of buckling resulting from a change in moderator density or lattice spacing.

The sensitivity of Equation (3.7) to small changes in its parameters makes it difficult to compare bucklings calculated with Equation (3.7) to the experimental values. For example, a 1% change in $P(C_4)$ at a low H/Pu ratio will cause an 80% change in B^2 . B^2 is more sensitive to the parameters C_1 and C_3 and much less sensitive to changes in L^2 (C_2).

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FERTF ZERO-POWER TESTS IN THE PRTR

J. W. Kutcher and J. H. Lauby

A number of zero-power measurements have been performed in the Plutonium Recycle Test Reactor (PRTR), using the Fuel Element Rupture Test Facility (FERTF). Most of these measurements were performed with the FERTF element located in the central channel (#1550); however, Channels 1849 and 1847 were also used. These channels are at radii 56.30 and 60.96 cm respectively. In addition to the FERTF element, the core loading consisted of 54 standard Mark I-R 19-rod clusters of UO_2 -2 wt% PuO_2 in D_2O coolant, with 21 wppm Boron-10 in the D_2O moderator.

The FERTF element used in channels 1847 and 1849 consisted of six UO_2 -2 wt% PuO_2 rods in a stainless steel basket tube, cooled with H_2O . One experiment in channel 1550 utilized a FERTF element consisting of six natural UO_2 rods in a stainless steel basket tube. Another set of experiments in channel 1550 utilized an element consisting of eight UO_2 -2 wt% PuO_2 rods in a stainless steel basket tube. In the latter set, a one inch diameter rod of aluminum or stainless steel was placed in the center of the element to reduce the volume of H_2O in the assembly and to depress the thermal flux locally.

The effect of the loss of H_2O coolant was measured for several

FERTF configurations. The results of these measurements are listed in Table 3.IX. In all these cases, the loss of coolant produced a positive change in reactivity.

Power generation rates in fuel rods in the FERTF element and in the first ring of elements in the Batch Core have been measured by gamma scanning of these rods after a low power PRTR irradiation. These measurements determined the power ratio between rods in the FERTF and the maximum rods in a 19 rod cluster in the first ring of elements in the Batch Core.

Relative power generation rates for various configurations on the FERTF have been calculated with the one-dimensional diffusion theory code HFN,⁽¹⁾ using cross sections calculated by THERMOS⁽²⁾ and HRG.⁽³⁾ The results of these measurements and corrected calculations are listed in Table 3.IX.

Critical tests have also been performed with a 19 rod cluster of UO_2 -2 wt% PuO_2 in channel 1550 with both H_2O and D_2O coolant. Gamma scanning was used to measure power ratios between rods in the element

in channel 1550 and the highest power rods in an element in the first ring of the Batch Core. The results of these measurements and of corresponding calculations are shown in Table 3.X.

TABLE 3.X. Relative Power Generation Rates (Channel 1550/First Ring Channel)

Coolant in Channel 1550	Measured	Calculated
H_2O	1.09	1.13
D_2O	0.99	1.02
Ratio: Power (H_2O)/Power (D_2O)	1.10	1.11

It can be seen from Table 3.X that replacing D_2O coolant with H_2O in channel 1550 produces a local power increase of about 10%.

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TABLE 3.IX. Relative Power Generation Rates

Fuel Type	No. of rods in FERTF Element	FERTF Location	Center of FERTF Element	Power Ratio, Measured	FERTF/1st Ring F.E. Calculated	H_2O Void Worth mk
UO_2	6	1550	H_2O	0.62 ± 0.05	0.55	+ 4.8
UO_2 -3 wt% PuO_2 (a)	6	1550	H_2O	1.76 ± 0.03	1.70	-
UO_2 -2 wt% PuO_2 (a)	6	1550	H_2O	1.53 ± 0.01	1.48	+ 2.1
UO_2 -2 wt% PuO_2	6	1849	H_2O	0.99 ± 0.02	0.98	+ 1.15
UO_2 -2 wt% PuO_2	6	1847	H_2O	0.84 ± 0.02	0.88	-
UO_2 -2 wt% PuO_2	8	1550	A1	1.26 ± 0.02	1.18	-
UO_2 -2 wt% PuO_2	8	1550	SS	1.17 ± 0.02	1.08	+ 1.3 m k

(a) These results were reported in a previous Quarterly Report.⁽⁴⁾

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CORRELATIONS OF EFFECTIVE CROSS SECTION RATIOS OF PLUTONIUM ISOTOPES

R. C. Liikala and W. L. Purcell

Calculations of neutron spectra were performed to obtain ratios of effective cross sections as a function of exposure for an Al-1.8 wt% Pu 19-rod cluster which was irradiated in the PRTR. Initially the plutonium contained 6 wt% ^{240}Pu . The calculations were made utilizing several consistent sets of basic cross sections.⁽¹⁻⁴⁾ The cross section ratios resulting from these calculations are compared to values obtained from a least squares analysis of the burnup data.⁽⁵⁾ The purpose of the comparison is to determine which set or sets of basic cross sections describe the data better.

Isotopic concentrations⁽⁶⁾ measured for various exposures are used in the computation of the neutron spectrum within a single cell of the reactor. The neutron spectrum is computed using the transport theory codes HRG⁽⁷⁾ and THERMOS.⁽⁸⁾⁽⁹⁾ The various cross section sets differed because of normalization of the cross sections for the thermal energy group at 2200 m/sec.

The comparison of calculated values of the ratios of effective cross sections to values deduced from analysis of the burnup data show that: the ratios of effective cross sections calculated using the Leonard⁽¹⁾ normalization for the thermal ^{239}Pu and ^{241}Pu

cross sections agree best with experimental values; the ratios obtained using the Westcott⁽⁴⁾ sets are about one standard deviation away from experimental values; the Sher⁽²⁾⁽³⁾ values for the 2200 m/sec constants of ^{239}Pu result in ratios approximately two standard deviations from the experimental data. The calculated values for the ratio $\hat{\sigma}_a^{40}/\hat{\sigma}_a^{49}$ of the effective absorption cross section for ^{240}Pu to that for ^{239}Pu are all about 20% lower than the experimentally derived values. It is expected that the method of calculating the ^{240}Pu resonance cross section needs to be improved. Other schemes for computing the effective cross section are being evaluated. The method most successful in correlating the ^{240}Pu ratio will be used for studies for which the initial concentration of plutonium in the fuel was 17 wt% ^{240}Pu .

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PHOTONEUTRON CALCULATIONS FOR D₂O SYSTEMS

U. P. Jenquin

A series of calculations of photoneutron production for various fueled geometries in the PRTR⁽¹⁾ have been completed. The number of delayed neutrons occurring because of the (γ -n) reaction in D₂O significantly affects the kinetics behavior of these systems. Experimentally measured data of the fraction of neutrons produced by a (γ ,n) reaction from a point source of gamma rays in an infinite medium of D₂O have been reported previously.⁽²⁻⁴⁾ These data can be combined with the calculated results to determine the delayed photoneutron fraction for each reactor system.

The number of neutrons produced from the (γ ,n) reaction in D₂O have been calculated for systems containing 19 rod clusters of PuAl and PuO₂-UO₂ in D₂O, and for an infinite medium of D₂O containing a point source of gamma rays. The ratio of photoneutrons produced in each reactor loading relative to the number produced in the pure D₂O system

containing the point source is defined as the "photoneutron production efficiency."

To calculate the efficiencies, the space-energy distribution of the fission gamma rays was determined using the transport theory code, S-XIII.⁽⁵⁾ In these calculations a unit lattice cell for an 8 in. hexagonal lattice spacing was represented geometrically as concentric cylinders. Isotropic and first order anisotropic cross sections were computed using Program GAMMAX.⁽⁶⁾ Values for the (γ ,n) cross section of deuterium were obtained from Reference 7. The gamma ray energy spectrum was assumed to be the same for both systems.

The calculated efficiencies for producing photoneutrons for the two types of 19-rod clusters are listed in Table 3.XI.

Since both systems contain 1/2 in. diam fuel rods, the differences in calculated efficiencies are a result of differences in composition only.

These values can be used along with the measured photoneutron fractions⁽²⁻⁴⁾

TABLE 3.XI. Efficiency for Producing Photoneutrons

Type of Fuel	Percent Efficiency	
	20 °C Coolant and Moderator	265 °C Coolant and 65 °C Moderator
Al-1.8 wt% Pu	34.8	33.0
UO ₂ -2 wt% PuO ₂ -UO ₂ (88% TD)	22.2	21.5

to obtain the effective delayed photo-neutron fraction for each system. (8)

The PuO₂-UO₂ fueled system contains two fissionable elements, uranium and plutonium. A weighting function which accounts for the fraction of the total fissions occurring in each element must be applied in obtaining the effective delayed fraction.

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THE DANCOFF CORRECTION AND ITS EFFECT ON RESONANCE INTEGRALS AND REACTIVITY

G. L. Simmons

The calculation of resonance reaction rate parameters in lattices of absorber rods requires the consideration of the net reduction of the flux incident on the surface of an absorber

rod due to the shadowing effect of adjacent rods. Alternatively one might consider the effect of the fictive reduction in the surface area of the absorber rod due to the adjacent rods in the lattice. The correction for this effect was first proposed by Wigner and was calculated in the geometry of neighboring infinite, parallel cylindrical

rods by Dancoff and Ginsburg,⁽¹⁾ hence the name Dancoff correction. Although the Dancoff correction for the surface resonance absorption term is only an approximation, it has been demonstrated by experiment⁽²⁾ and by comparison with Monte Carlo calculations⁽³⁾ that the application of such a correction gives reasonably good results. Sauer⁽⁴⁾ and Levine⁽⁵⁾ also have suggested a simple method of calculating the Dancoff correction.

All of the above methods for calculating the Dancoff correction have been programmed in FORTRAN IV for the UNIVAC 1107.* This code has been used to calculate values of the mean chord length, ℓ , for 6.6 wt% PuO-UO₂ fuel⁽⁶⁾ used in three different square arrayed lattices, the spacings being 0.52, 0.56, and 0.735 in. The code HRG⁽⁷⁾ has been used to calculate the neutron slowing down spectrum in the cell for these three lattices. The calculations were made using various values for the mean chord length, ℓ , which were obtained via the various schemes for computing the Dancoff correction. The values of these Dancoff corrections and their effect on the effective resonance integrals for ²³⁸U and ²⁴⁰Pu are shown in Table 3.XII.

It is noteworthy that the use of the Dancoff-Ginsburg formulation for calculating the Dancoff correction results in a higher value of the resonance absorption integral than does any of the Sauer approximate methods, the largest difference being about 6.5% in I_{eff} . Consequently, one would expect to see similar differences in calculated reactivities.

TABLE 3.XII. The Effect of Dancoff Corrections on Reactivity, k_{∞} for 6.6 wt% PuO₂-UO₂-H₂O Lattices

Lattice Spacing, in.	k_{∞}	
	Dancoff $(1-C_i)^{n_i}$	Sauer's No Clad Correction
0.52	1.367	1.406
0.56	1.417	1.449
0.735	1.504	1.523

This effect on reactivity may be seen by examining the values of the infinite multiplication factor, k_{∞} . The HRG and THERMOS codes were used to compute cell average cross sections. These values are used in the following expression to determine k_{∞} values.

$$k_{\infty} = \frac{\nu \bar{\Sigma}_{f1} \left(\frac{\bar{\Sigma} a_2}{\bar{\Sigma}_{\text{out } 1 \rightarrow 2}} \right) + \nu \bar{\Sigma}_{f2}}{\Sigma a_1 \left(\frac{\bar{\Sigma} a_2}{\bar{\Sigma}_{\text{out } 1 \rightarrow 2}} \right) + \bar{\Sigma} a_2}$$

A calculated difference of as much as 3% in the values of k_{∞} is shown in Table 3.XII.

By using Sauer's approximation, we obtain consistently higher values of k_{∞} than those obtained by using Dancoff-Ginsburg's formulation. This observation is consistent with the trends in the calculated resonance integrals in Table 3.XIII.

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* See computer code abstracts from this quarterly.

TABLE 3.XIII. Comparison of Effective Resonance Integrals, I_{eff} (a) for ^{238}U and ^{240}Pu

L.S.	Dancoff-Ginsburg Method (No Clad Correction)		$I_{eff}(^{238}\text{U})$			
			Using C_{eff}		Using C_{eff}	
	n_i		Neglect Clad	Include Clad	Neglect Clad	Include Clad
	$(C-C_i)$	$1-\Sigma n_i C_i$				
0.52	19.5619	19.3890	18.8163	18.6027	18.6115	18.3861
0.56	19.9982	19.8890	19.4726	19.2809	19.3087	19.1044
0.735	20.9463	20.9360	20.7616	20.6373	20.6979	20.5629

L.S.	Dancoff-Ginsburg Method (No Clad Correction)		$I_{eff}(^{240}\text{Pu})$			
			Using C_{eff}		Using C_{eff}	
	n_i		Neglect Clad	Include Clad	Neglect Clad	Include Clad
	$(1-C_i)$	$1-\Sigma n_i C_i$				
0.52	2151.66	2134.26	2076.38	2054.72	2055.61	2032.69
0.56	2194.46	2184.52	2142.68	2123.36	2126.16	2105.54
0.735	2290.05	2289.02	2271.68	2259.31	2265.35	2251.89

Comparison of Dancoff Correction Values, C_{eff}

L.S.	Dancoff-Ginsburg Method (No Clad Correction)		Sauer's Method			
			C_{eff}		C_{eff}	
	n_i		Neglect Clad	Include Clad	Neglect Clad	Include Clad
	$(1-C_i)$	$1-\Sigma n_i C_i$				
0.52	0.81965	0.80375	0.75194	0.73296	0.73374	0.71390
0.56	0.85939	0.85008	0.81143	0.79388	0.79642	0.77784
0.735	0.95140	0.95038	0.93336	0.93132	0.92718	0.91413

(a) The sum of resolved and unresolved resonance contributions.

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POWER PEAKING IN UO_2 -2 wt% PuO_2 - H_2O CORES

V. O. Uotinen and S. Kobayashi

Calculations have been performed of power peaking around a water hole and near the core-reflector boundary in two H_2O -moderated cores containing UO_2 -2 wt% PuO_2 fuel rods. These calculations provide analytical results for comparison with results of recent PRCF experiments.⁽¹⁾ The results are summarized in Table 3.XIV.

Cell-averaged cross sections were generated for four energy groups of neutrons with the codes THERMOS,⁽²⁾ TEMPEST,⁽³⁾ and HRG.⁽⁴⁾ These cross sections were used in a one-dimensional diffusion theory calculation (HFN⁽⁵⁾ code) of effective multiplications and spatial distributions of flux and power. To represent a water hole in the center of the lattice, the lattice constants of the central cell were assumed to be identical to the lattice constants for the reflector. Calculated power peaking factors around a central water hole are high by

5 to 6%, whereas near the core-reflector boundary they are low by 4 to 5%. The gradient of the power density is very large at both of these positions. Therefore, the calculated power peaking factors depend strongly on how the physical radius is related to the radius used in obtaining the calculated value. The radii that were chosen to represent positions near the water hole and near the core-reflector boundary are the radii at which the experimental values were obtained. The sources of experimental uncertainties are being investigated to determine probable errors for the measured values.

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TABLE 3.XIV. Comparison Between Measured and Calculated Power Peaking Factors

Definition of Power Peaking Factor	8 wt% ^{240}Pu , Core I			24 wt% ^{240}Pu , Core II		
	Exp.	Calc.	Calc. Exp.	Exp.	Calc.	Calc. Exp.
Avg. Power in Fuel Rod Adjacent to Central H_2O Hole	1.450	1.517	1.046	1.556	1.646	1.058
Avg. Power in Core						
Avg. Power in Fuel Rod at Core-Reflector Boundary	1.093	1.041	0.952	1.002	0.955	0.953
Avg. Power in Core						
Avg. Power in Fuel Rod Adjacent to Central H_2O Hole ^(a)	1.144	1.144	1.000	1.127	1.144	1.015
Avg. Power in Same Fuel Rod with no H_2O Hole Present						

(a) Normalized so that average power in core is unity in both cases.

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CALCULATIONS FOR H₂O-MODERATED
LATTICES OF UO₂-4 wt% PuO₂ FUEL RODS
 S. Kobayashi and V. O. Uotinen

A series of experiments in H₂O-moderated lattices containing UO₂-4 wt% PuO₂ fuel rods is being planned. The experimental program will include approach-to-critical experiments in the CAF and critical experiments in the PRCF.

Preliminary calculations have been performed to help predict critical masses and various lattice parameters for these experiments.

The fuel rod size was assumed to be the same as for existing rods which contain 2 wt% PuO₂ in UO₂: 0.5075 in. in diameter, 36 in. long, and clad with 0.030 in. of Zircaloy-2.

Calculations were performed for three hexagonal lattice pitches (0.75, 0.80, and 0.85 in.) and for two isotopic compositions of plutonium. The plutonium in the two types of fuel contained the following weight percent:

²³⁹Pu-91.62; 71.76
²⁴⁰Pu-7.65; 23.50
²⁴¹Pu-0.70; 4.08
²⁴²Pu-0.03; 0.66

The density of the mixed oxide fuel was assumed to be 9.6536 g/cm³, which corresponds to 1152.6 g of UO₂-PuO₂ mixture per fuel rod.

The computer code THERMOS⁽¹⁾ was used to compute average thermal (0 to 0.683 eV) cross sections for a cylinder-ized lattice cell; the TEMPEST⁽²⁾ code was used to compute the thermal transport cross section; the HRG⁽³⁾ code was used to compute cell-averaged cross sections for three fast groups. Lattice constants generated in this way were used in a one-dimensional diffusion theory calculation (HFN⁽⁴⁾ code) of the effective multiplication.

Some of the preliminary results are shown in Table 3.XV and Figure 3.5. A report is being prepared which contains additional results and more detail. The calculated critical masses are compared in Table 3.XVI with

TABLE 3.XV. Calculated Results for 4 wt% $\text{PuO}_2\text{-UO}_2$ (Nat.)- H_2O Cores

	7.65 wt% ^{240}Pu			23.5 wt% ^{240}Pu		
Lattice Pitch, in.	0.75	0.80	0.85	0.75	0.80	0.85
Volume Ratio V_m/V_f	1.1567	1.4891	1.8423	1.1567	1.4891	1.8423
No. of Fuel Rods at Critical	384.67	261.86	197.77	657.88	415.52	300.58
Radius at Critical (cm)	19.613	17.265	15.938	25.655	21.748	19.649
$\bar{\eta} f_{(\text{thermal})}(\text{a})$	1.7660	1.7544	1.7392	1.6794	1.6669	1.6508
$\bar{\eta}_{(\text{thermal})}(\text{a})$	1.8333	1.8417	1.8484	1.7477	1.7554	1.7613
$f_{(\text{thermal})}(\text{a})$	0.9633	0.9526	0.9409	0.9609	0.9496	0.9372
Disadvantage Factor, D.F.	1.8562	1.8283	1.8161	1.7616	1.7365	1.7263
$P(^{238}\text{U})(\text{b})$	0.6638	0.7184	0.7598	0.6636	0.7185	0.7600
$P(^{240}\text{Pu})(\text{b})$	0.8424	0.8688	0.8883	0.7963	0.8259	0.8491

$$\bar{\eta} f = \frac{(\bar{\nu} \bar{\Sigma}_f)_{\text{cell}}}{(\bar{\Sigma}_a)_{\text{cell}}}, \quad \bar{\eta} = \frac{(\bar{\nu} \bar{\Sigma}_f)_{\text{fuel}}}{(\bar{\Sigma}_a)_{\text{fuel}}}, \quad f = \bar{\eta} f / \bar{\eta}$$

$$\text{(b)} \quad P = 1 - \frac{(\bar{\Sigma}_a)_{\text{cell}}}{(\bar{\Sigma}_r)_{\text{cell}}}, \quad P(^{238}\text{U}) \text{ for } 1.17 \times 10^4 \text{ eV} \sim 2.38 \text{ eV}$$

$$P(^{240}\text{Pu}) \text{ for } 2.38 \text{ eV} \sim 0.683 \text{ eV.}$$

TABLE 3.XVI. Critical Number of Rods, N_c , for 2 wt% and 4 wt% $\text{PuO}_2\text{-UO}_2\text{-H}_2\text{O}$ Lattice

	Lattice Pitch, in.	N_c for 2 wt% $\text{PuO}_2\text{-UO}_2$	N_c for 4 wt% $\text{PuO}_2\text{-UO}_2$	N_c (4 wt%) N_c (2 wt%)
8 wt% ^{240}Pu	0.75	499 (a)	385	0.77
	0.80	319 (b)	262	0.82
	0.85	251.5 (c)	198	0.79
24 wt% ^{240}Pu	0.75	944 (a)	658	0.70
	0.80	524 (b)	416	0.79
	0.85	394 (c)	301	0.76

(a) Estimated from subcritical experiments at other lattice pitches

(b) Subcritical measurement

(c) Critical measurement

FIGURE 3.5. Thermal and Resonance Parameters in 4 wt% $\text{PuO}_2\text{-UO}_2$ (Nat) Lattices

critical masses that were measured in lattices containing 2 wt% $\text{PuO}_2\text{-UO}_2$ rods. The critical mass decreases by roughly 20 to 30% when the plutonium enrichment increases from 2 to 4 wt%.

References

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4.0 FAST REACTORSEFFECT OF BeO ON FTR NEUTRONICS

R. B. Kidman and W. W. Little

In large fast reactors, one can often increase the Doppler coefficient and ameliorate the sodium void coefficient by adding a moderator, such as BeO, to the reactor core. A moderator increases the Doppler coefficient by enhancing the resonance flux, and decreases the sodium void coefficient by leveling the adjoint flux spectrum.

A brief parametric survey has been performed to investigate the effect of BeO on the FTR neutronics. Items considered include the Na coefficient, Doppler coefficient, critical mass, and flux spectrum.

All calculations are based on the parameters listed in Table 4.I. The principal variables are the volume percent of SS, fuel ($\text{PuO}_2 + \text{UO}_2$), and BeO.

Figure 4.1 gives the critical ($k = 1.02$) ^{239}Pu inventory for various concentrations of BeO. The corresponding core-averaged uranium to plutonium ratio and spectrum indices are shown in Figures 4.2 and 4.3 respectively. All calculations were performed using the DTF-IV transport code (cylindrical geometry) and a modified version of the Russian 26-group cross sections.

The Doppler coefficient as a function BeO concentration is given in Figure 4.4. As predicted, note that BeO enhances the Doppler coefficient in cores containing a reasonable percent of fertile fuel.

The central sodium perturbation worth in terms of σ_p and δk is given in Figures 4.5 and 4.6 respectively for various BeO loadings. The total sodium void worth (for Case 5), as a function of the fraction of core voided, is described in Figure 4.7. The latter curve was generated using the DTF-IV code in spherical geometry.

It is clear from the preceding figures that BeO has a beneficial effect on both the Doppler and the

TABLE 4.I. Parameters Used in BeO Survey

800 liter core

L/D = 0.9

2 Equal-volume radial zones

Pu: 85% ^{239}Pu , 15% ^{240}Pu

F.P. Corresponding to 2 at.% burnup

 $k_{\text{eff}} = 1.02$

40 vol% Na in core

Case	vol% SS	vol% (Fuel + BeO)	vol% BeO
1	40	20	0
2	↓	↓	5
3	↓	↓	10
4	30	30	0
5	↓	↓	5
6	↓	↓	10
7	20	40	0
8	↓	↓	5
9	↓	↓	10

FIGURE 4.1. The Change in ^{239}Pu Inventory (kg) as a Result of Varying the Volume Percent of Fuel + BeO.

sodium void coefficient. The variations, however, are not pronounced. Furthermore, BeO may have a detrimental side effect on other reactor characteristics (fuel and thermal performance). These effects will have to be investigated in detail before the question of BeO addition to the FTR is resolved.

FTR CONTROL RODS

R. A. Harris

Poison control rods located in both the reflector and core are presently envisioned for the FTR. Calculations were completed to determine the reactivity worths of representative reflector rod arrangements. The reactor model for the calculations was

FIGURE 4.2. The Change in U/Pu Ratios as a Result of Varying the Volume Percent of Fuel + BeO

an 800 liter, mixed oxide core containing 91 core positions. Core height is 93.8 cm and core diameter 104.2 cm, with 30 cm thick axial and radial reflectors. The core was subdivided into two equal volume radial zones, with the fuel density in each adjusted to equalize the volume averaged power density. Composition

FIGURE 4.3. The Change in Energy Spectrum as a Function of the Volume Percent of Fuel + BeO

FIGURE 4.5. The Central Na Perturbation Cross Section as a Function of the Volume Percent of Fuel + BeO

FIGURE 4.4. Doppler Effect as a Function of Volume Percent Fuel + BeO

of each of the regions is shown in Table 4.II. Neutron leakage in the axial direction was simulated by a buckling term for a 117.8 cm high bare core.

For the reflector control rod calculations, the Russian 26 group cross section set was collapsed to 5 groups using spectra characteristic of each region obtained from one dimensional transport theory calculations. The worth of a full ring of rods on the core periphery based on the 5 group set differed by less than 1.7% from the worth using the 26 group set; thus, the collapsed set was used in the two dimension (R- θ) transport

FIGURE 4.6. Central Na Worth as a Function of Volume Percent (Fuel + BeO)

FIGURE 4.7. Reactivity Change as a Function of Percent Core Voided

TABLE 4.II. Reactor Composition, vol%

Material	Core		Control Rod	Rod Follower	Radial Reflector	Axial Reflector
	Zone 1	Zone 2				
PuO ₂	5.15	7.15				
²³⁸ UO ₂	19.85	17.85				
Na	45.00	45.00	30.00	40.00	20.00	70.00
304 SS	25.00	25.00	20.00	60.00		30.00
B ₄ C			50.00			
Inconel					80.00	
Void	5.00	5.00				

theory calculations to obtain worths of several peripheral control rod patterns. Each position in the control rod ring was represented by a sector of a cylindrical shell which could contain either poison or reflector material. The arrangements of control rods considered are shown in Figure 4.8. Patterns were chosen to separate the rods to reduce interactions, and with grouped patterns to determine the interaction effect. Results for the fully inserted patterns are shown in Figure 4.9 together with one dimensional results in which the poison material was homogeneously distributed throughout the annular rodded region. From the figure it is obvious that the one dimensional calculations overestimate the worth of the rod pattern. This is expected, since every neutron that enters the rod row must go through a boronated region. The two dimensional calculations indicate a more realistic linear variation of worth versus number of rods. The two values shown for the 18 rod case indicate the magnitude of the reduced effectiveness caused by ganging the rods.

FIGURE 4.8. Control Rod Patterns for FTR ($R-\theta$ Calculations)

FIGURE 4.9. Reflector Control Rod Worths

ANALYSIS OF REACTOR TRANSIENTS INDUCED BY FUEL REARRANGEMENT

W. W. Little

Since the FTR will test high risk fuel, the consequences of fuel failure and relocation (slumping) must be carefully considered. Such considerations could lead to restrictions on the location and size of high risk test assemblies.

Let us consider accidents involving fuel relocation in the central channel

of an 800 liter, 2 zone FTR core. First, the total worth of fissile fuel in the central channel (averaged over the channel length) is

$$\delta k_{\text{channel}} \sim 1.0 \frac{\delta M}{M_0},$$

where $(\delta M/M_0)$ is the fraction of fuel in the channel. (See Figures 4.10 and 4.11 for detailed fuel worth curves.)

FIGURE 4.10. ^{239}Pu Reactivity Coefficient as a Function of Radial Position

FIGURE 4.11. ^{239}Pu Reactivity Coefficient as a Function of Axial Position

As our model for fuel rearrangement, we shall assume that the top 1/3 of the central assembly falls into a position adjacent to the middle third. The change in reactivity caused by this relocation is

$$\delta k_{\text{melt}} \approx 0.21 \left(\frac{\delta M}{M_0} \right).$$

If we now assume a 0.25 sec fall time (approximates gravity fall), the average insertion rate, in units of \$/sec ($\beta = 0.003$), is

$$R \text{ ($/sec)} \approx 280 \left(\frac{\delta M}{M_0} \right).$$

Kinetics calculations using the K-III kinetics code were performed for a variety of input and shutdown coefficients. Figures 4.12-4.14 give the temperature response as a function of insertion rate for total reactivity changes of \$0.75, \$1.00, and \$1.25. Figures 4.15-4.17 show the

effect of the Doppler coefficient on transients induced by step changes of \$0.75, \$1.00, and \$1.25. The temperature shown is the center-line temperature in an average pin.

FIGURE 4.13. Temperature Response for Truncated Ramp Insertion

FIGURE 4.12. Temperature Response for Truncated Ramp Insertion

FIGURE 4.14. Temperature Response for Truncated Ramp Insertion

FIGURE 4.15. Reactivity Step Insertion in a Hot Core

FIGURE 4.17. Reactivity Step Insertion in a Hot Core

FIGURE 4.16. Reactivity Step Insertion in a Hot Core

If we assume that scram rods will not be effective before ~ 0.3 sec, and the maximum allowable temperature rise is approximately 300 to 400 °K, we then can make the following tentative observations:

- Reactivity changes of \$1.25 are unacceptable.
- The maximum permissible reactivity insertion probably lies between \$0.75 and \$1.00, depending principally on the input rate.
- Variations in the Doppler-coefficient in the range $-0.004 < (TdK/dT) < -0.006$ are of secondary importance for the particular incident considered here.

CONSIDERATIONS ON THE FUEL-ROD PLUG CONCEPT

A. E. Waltar

Because mixed oxide fuels are known to exhibit radial densification

as a function of burnup, it is of interest to determine whether or not the resultant void in the center of a fuel rod presents a significant safety problem. Should melting occur in a rod containing a central void,

fuel from a low reactivity worth region could flow unimpeded through the void into a high worth region creating a more reactive configuration. One method of restricting such fuel movement is to insert plugs at selected axial positions along the fuel rod. Two aspects of this problem have been considered: the reactivity consequences of fuel melting with central voids present, and the effect on the Doppler coefficient of inserting plugs in the fuel rods.

Two idealized models were chosen to estimate reactivity changes associated with the melting of partially voided fuel rods. The first model (Case I) assumes an axial void along the entire length of the fuel as the initial condition. The worst possible condition in this model is for the entire fuel rod to melt and compact

symmetrically toward the center of the core. The second model (Case II) assumes that initially only the upper two-thirds of the rod contains a void, and that only this portion of the rod melts in the accident. Fuel movement is then assumed to take place under the force of gravity. Plugs were located symmetrically above and below the fuel rod center line in Case I; whereas, they were located only in the top half of the fuel in Case II. Figures 4.18 and 4.19 illustrate the two models chosen.

For the selected plug locations, the calculated results can be approximated by:

$$\text{CASE I: } \delta k = \frac{0.7 (\text{V.F.}) (\delta M/M_0)}{1 + N}$$

$$\text{CASE II: } \delta k = \frac{0.5 (\text{V.F.}) (\delta M/M_0)}{1 + N}$$

FIGURE 4.18. Case I for 20 vol% Void

$$\delta k = 0.0007 \rightarrow 0.23$$

for the case of no plugs. One plug in the top half (two plugs total) would cut this value in half.

Since the introduction of plugs into the fuel rods would be done primarily to reduce the consequences of this type of accident, it is important to note the effect of the plugs on other parameters. Of paramount importance is the effect of plugs on the Doppler coefficient.

Four substances were considered as possible candidates for plug materials, tantalum, tungsten, SS and BeO. For the physics calculations, it was assumed in each case that the candidate plugs replace an equal volume percent of UO_2 in the FTR core. The calculated effect on the Doppler coefficient of homogeneous additions of each of the above materials is illustrated in Figure 4.21.

Based upon the models chosen for this investigation, the inclusion of plugs in the fuel rods would significantly reduce the reactivity insertion accompanying a fuel melting accident of the type postulated; however, the postulated accident with no plugs at all would not be very severe. Further, the effect on the Doppler coefficient of including such plugs is substantial, especially for tantalum and tungsten.

FIGURE 4.21. Effect of Plugs on the Doppler Coefficient

In addition, the tantalum and tungsten plugs would adversely affect the sodium void coefficient and increase the critical mass. To the extent considered in this study, only BeO plugs would make a consistently favorable contribution to the safety problem.

ANALYSIS OF SEFOR CENTRAL SODIUM VOID EXPERIMENT

L. L. Maas

The analysis of the central sodium void experiment performed on the SEFOR mockup in ZPR-III has been undertaken to check calculational techniques used for the FTR.

SEFOR has an $L/D \sim 1$ and a core volume of 527.6 liters. A shape factor of 0.95 was used to convert the cylindrical reactor volume to a spherical model for the sodium void calculations. The radial and axial reflectors of

the cylindrical reactor were smeared into a volume-weighted spherical reflector composition. All calculations were done with 26-energy group cross sections in spherical geometry using transport theory.

In the experiment, the central 5, 10, or 15 vol% of the core was voided of sodium; the reactivity effects measured for voiding these central volumes are shown in Figure 4.22. The calculated curve is also shown. Although the fractional error between the experimental and calculated points is large, the maximum

FIGURE 4.22. *Sefor Measured and Calculated Na Loss Reactivity*

error over the limited experimental range is about 15%. Both the experimental and calculated results show that

the maximum sodium void effect occurs with about 10 vol% of the core voided.

DOPPLER COEFFICIENT STUDIES

R. E. Schenter

Crystalline Binding Effects in $^{238}\text{UO}_2$

Introduction and Summary

The change in the Doppler coefficient due to crystalline binding effects has been studied for ^{238}U in a UO_2 lattice for fast reactor systems. Use has been made of the "weak binding," or the short compound nucleus lifetime (SCNL) approximation,⁽¹⁾ where the ^{238}U atoms in the UO_2 lattice are treated as a free gas with an effective temperature.* Calculations were made to determine the effective temperature versus temperature dependence using the experimental phonon distribution of Dolling, et al.⁽²⁾ and comparisons were made with results obtained assuming a Debye model for the crystal. It was found that for temperatures above 300 °K the effective temperature dependence was well represented by the Debye model ($\theta_D \sim 620$ °K). This result implies that the crystalline binding effects will be important only for temperatures less than or equal to the Debye temperature θ_D . Inclusion of the

binding effects then leads to a reduction in the calculated value of the Doppler coefficient in the lower temperature region, which is of the order of 30% at room temperature (~ 300 °K). The temperature dependence of the reactivity was calculated for the ZPR-III Assembly 47 using the FCC fundamental mode code⁽³⁾ with a modified version of the Russian 26-group data compilation as input.⁽⁴⁾ Incorporation of the crystalline binding effects in this calculation caused the Doppler coefficient ($\frac{dK}{dT}$) to diverge from the " $\frac{1}{T}$ " rule in the low temperature region in qualitative agreement with the experimental results of Reynolds and Stewart.⁽⁵⁾

Effective Temperature Model

The effective temperature T_{eff} in the SCNL approximation is given in terms of $\bar{E}$ the average energy per mode of oscillation as⁽¹⁾

$$kT_{\text{eff}} = \bar{E} = \int_0^\infty d\nu g(\nu) \left(\frac{h\nu}{2} \right) \coth \left(\frac{h\nu}{2kT} \right), \quad (4-1)$$

where ν is the phonon frequency and $g(\nu)d\nu$ is the phonon frequency distribution, number per unit frequency interval. The results of using Equation (4.1) are given in Figure 4.23 for the Dolling⁽²⁾ phonon

* This should be a good approximation for neutrons with energies that are important for Doppler coefficient calculations in fast reactors (~ 2 KeV)

FIGURE 4.23. Effective Temperature Versus Temperature for the Dolling Phonon Distribution and the Debye Distribution ($D_0 = 620$ °K)

distribution (Figure 4.24) and for the Debye distribution, where $g(\nu)$ is given as ⁽⁶⁾

$$g_D(\nu) = 3 \frac{\nu^2}{\nu_D^3} \text{ for } \nu \leq \nu_D \quad (4-2a)$$

$$= 0 \quad \text{for } \nu > \nu_D, \quad (4-2b)$$

and the Debye temperature θ_D is defined as

$$k\theta_D = h\nu_D \quad (4-3)$$

The description of a crystalline substance is very often given in terms of its Debye temperature so that crystalline effects in a fast reactor are sometimes characterized by a single number θ_D . This description is approximate, since in general the crystal does not have a Debye phonon distribution. One can, however, define an effective Debye temperature which varies with the medium temperature and gives the correct answer for a given physical property. The results

FIGURE 4.24. Dolling Phonon Distribution for ^{238}U in a UO_2 Lattice

of this type of prescription is shown in Figure 4.25, where the effective Debye temperature versus temperature is given using the average energy $\bar{E}(T)$ and the specific heat $C_V(T)^*$. The results show (Figures 4.23 and 4.25), that to a good approximation, for temperatures above 300 °K, ^{238}U in a UO_2 lattice is described by a Debye model with a Debye temperature of 620 °.

* The specific heat curve ($C_V(T)$) is given to indicate how the effective Debye temperature varies with a particular physical property. Also, the specific heat dependence is the experimental quantity which is usually used to determine the Debye temperature of a given substance.

Doppler Coefficient Dependence

In order to explicitly show the crystalline binding effects on the Doppler coefficient, a $1/T$ dependence in the free gas model has been assumed so that

$$\left. \frac{dK}{dT} \right|_{\text{FG}} = \frac{C}{T}, \quad (4-4)$$

where C is a constant. Integrating Equation (4.4) yields

$$k_{\text{FG}}(T) = A + C \ln T. \quad (4-5)$$

Using the effective temperature model then gives

FIGURE 4.25. Effective Debye Temperatures Versus Temperature for the Dolling Phonon Distribution

$$k_{\text{Bound}}(T) = k_{\text{FG}}(T_{\text{eff}}) \quad (4-6a)$$

$$= A + B \ln T_{\text{eff}} \quad (4-6b)$$

Finally, differentiating Equation (4.6) and substituting in Equations (4.4) and (4.5) produces the desired expression for the bound Doppler coefficient in terms of the free gas coefficient

$$\left. \frac{dK}{dT} \right|_{\text{Bound}} = \left\{ \frac{T}{T_{\text{eff}}} \frac{dT_{\text{eff}}}{dT} \right\} \left. \frac{dK}{dT} \right|_{\text{FG}} \quad (4.7)$$

Results (using the above equations) showing the temperature dependence of the deviation from the free gas model

due to crystalline binding effects are given in Figures 4.26 and 4.27.

Comparison of Cross Section Sets

Temperature dependent self-shielding factors, which are used in Doppler coefficient calculations, have been calculated for ^{238}U in the resolved resonance region ($5 \text{ eV} < E \leq 3.92 \text{ KeV}$). Zero degree cross-sections were obtained using program UNICORN (NAA-SR-11980) and the resonance parameters for ^{238}U given in BNL-325. These resonance parameters are essentially the same as those given in the Evaluated Nuclear Data File B

FIGURE 4.26. Crystalline Binding Effects on the ^{238}U Doppler Coefficient

FIGURE 4.27. Reactivity Change Versus Temperature for the APR-III Assembly 47 Including Crystalline Binding Effects

(ENDF/B)⁽⁶⁾. The temperature dependent self-shielding factors were then obtained using programs BROAD and FFAC. Typical results are shown and compared with those given by the Russian (Bondarenko) set in Table 4.III where f_c , f_t , f_e are the capture, total and elastic scattering self-shielding factors, respectively, and σ_o represents the contribution to the total cross section from all the other isotopes in the reactor calculation.

Using the above self-shielding factors, calculations were made of the Doppler coefficient for the ZPR-III Assembly 48. Values of $T \frac{dk}{dT}$ (300 °K

→ 2100 °K) of -0.00729 were found using a fundamental mode model and compared very close to the value of -0.00727 given by the Russian set.

The contribution to the Doppler coefficient from ^{239}Pu has also been calculated using a fundamental mode model for ZPR-III Assemblies 47 and 48. Use was made of temperature dependent fission cross-sections of ^{239}Pu obtained by Doppler broadening directly the data obtained from Petrel Bomb Test.⁽⁷⁾ Values for $T \frac{dk}{dT}$ (300 °K → 2100 °K) of 1.9×10^{-4} and 1.5×10^{-4} for Assemblies 47 and 48, respectively, were calculated.

TABLE 4.III. Self Shielding Factors- ^{238}U -Group 14(1.0-2.15 KeV)

		f _c at σ _o equal to							
T°K		10 ⁴	10 ³	10 ²	10	0			
ENDF/B	300	0.979	0.851	0.529	0.303	0.246			
	900	0.986	0.894	0.606	0.357	0.278			
	2100	0.990	0.922	0.672	0.416	0.325			
"RUSSIAN"	300	0.99	0.89	0.63	0.34	0.29			
	900	1.00	0.93	0.74	0.43	0.36			
	2100	1.00	0.97	0.82	0.51	0.42			
		f _t at σ _o equal to				f _e at σ _o equal to			
		10 ³	10 ²	10	0	10 ³	10 ²	10	0
ENDF/B		0.807	0.585	0.464	0.284	0.878	0.683	0.558	0.479
		0.846	0.613	0.483	0.337	0.904	0.714	0.580	0.510
		0.876	0.642	0.504	0.396	0.923	0.744	0.603	0.541
"RUSSIAN"		0.91	0.74	0.64	0.50	0.95	0.84	0.74	0.60
		0.92	0.78	0.65	0.51	0.97	0.86	0.77	0.63
		0.93	0.82	0.68	0.53	0.99	0.91	0.82	0.66

These values are much smaller (~ 0.10) than those given by the current modified version of the Russian (Bondarenko) cross-section set.

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DEVELOPMENT OF PERT, A 2-D PERTURBATION CODE

R. Wayne Hardie

The two-dimensional perturbation theory code, 2-D PERT, is now operational. The primary purpose of the code is to calculate reactivity coefficients in cylindrical geometry using either two-dimensional fluxes and adjoint fluxes from 2DF, or one-dimensional axial and radial fluxes and adjoint

fluxes from DTF-IV. Other options include the calculation of the effective delayed neutron fraction, neutron generation time, and activity traverses. Finally, the code can be used to compute an initial flux guess for 2DF; that is, 2-D PERT will combine the radial and axial flux dumps from DTF-IV to form a two-dimensional initial flux guess for 2DF.

The code will be used for driver fuel and test management studies.

FTR PHASE-A CRITICAL EXPERIMENTS AND PRELIMINARY ANALYSES

R. A. Bennett, S. L. Engstrom,
Q. L. Baird

FTR Phase-A Critical Experiments consisted of measurements of the

worths of in-core and reflector control rods, measurements of spatial distributions of small sample worth, and reaction rate distributions. Rod worths were also measured in two and three rod arrays to investigate interactions. Heterogeneity and enrichment

effects were also determined. This work was done in support of the design of the FTR control and safety systems. The results are being used in experiment-theory correlations to verify calculation models used in FTR neutronics design. This report summarizes the experiment-theory correlations completed to date. Complete descriptions of the experiments are given elsewhere.^(1,2,3)

Summary

Worths of heterogeneous rods at in-core positions can be calculated to within ~8% using relatively simple two-dimensional transport models. This is demonstrated in the comparisons of experimental and calculated results presented in Table 4.IV. Refinements of the model are expected to improve

the agreement and are discussed in the body of this report.

Calculated worths of reflector rods and their interactions appear to be in better agreement (2 to 4%) than for the in-core rods as shown in Table 4.V. However, the calculations completed so far do not correctly account for axial leakage nor rod heterogeneity. These effects do tend to cancel, however, and it is expected that the reflector rod calculations are at least as good as the in-core rod calculations. Heterogeneity and leakage effects are discussed in more detail in the main body of this report.

Experiments

All experiments were performed in ZPR-3 in either Assembly 48 or in Assembly 48A, which was a modified

TABLE 4.IV. Preliminary Experiment - Theory Correlation, In-core Rods

Rod Position	Rod Description	Rod Worth		E/C
		Experimental (E)	Calculated (C)	
Inches Inserted	Rod No.	% $\Delta k/k$	% $\Delta k/k$	
8	1 (enr)	0.0029	-	
12		0.0268	0.0265 ^(a)	1.011
19		0.2870		
27		0.8920		
35		1.484		
42		1.720		
54		1.738		
54	1 (nat)	0.746	0.779	0.958
54	2 (enr lump)	1.528		
54	3 (enr)	1.734	2.023	0.857

(a) Experimental uncertainties are $\pm 2\%$

TABLE 4.V. Enriched Reflector Control Rod Worths

Rod Locations ^(a)	Experimental ^(b) % $\Delta k/k$	(E)	Calculated ^(c) % $\Delta k/k$	(C)	Correlation E/C
P9	0.417				
P23	0.412				
P9, P23	0.821		0.838		0.980
L10	0.392				
T10	0.382				
L10, T10	0.754				
L10, T10, P23	1.188				
L10, T10, P9	1.097				
N9	0.350				
R9	0.350				
N9, R9	0.674				
N9, R9, P23	1.092				
N9, R9, P9	0.939		0.977		0.961

(a) Rod locations are shown in Figure 4.34

(b) Experimental uncertainties are $\pm 2\%$

(c) Computational (numerical) precision is better than $\pm 0.005\% \Delta k/k$.

version of Assembly 48. Assembly 48 as used for the in-core rod experiments is shown in Figures 4.28, 4.29 and 4.30. A cross section of a typical core drawer is shown in Figure 4.31. Cross sections of distributed and lumped rod assemblies are shown in Figures 4.32 and 4.33. Two heterogeneities that complicate the analysis of these experiments are to be noted in Figures 4.28 and 4.29. First, the safety rods are core drawers enriched in plutonium that are asymmetrically located and the asymmetry is different in each half of the reactor. Second, the distribution of materials in a unit cell of the core is asymmetric, which in effect shifts the neutronic center

of the core off of the geometric center of the system. These effects may account for some of the disagreement between theory and experiment. The safety rod asymmetry has been evaluated experimentally⁽⁴⁾ and is discussed in more detail later on.

The reflector rods were measured in a modified version of Assembly 48 (48A) that had two diametrically opposite 100° sectors of the blanket replaced with a nickel-steel-sodium reflecting mixture. These reflector sectors were used to provide a neutron spectral environment for the reflector rod experiments that resembled that of an FTR more closely than would the spectrum of the ²³⁸U blanket. A cross sectional diagram of Assembly

Unit Cell Area = 4.7524 in²
= 30.6606 cm²

Asymmetries in Assembly 48
Control Rod Test Location
Safety Rods
Control Rod

FIGURE 4.28. Critical Assembly 48 (Half #1)

- Asymmetries in Core 48
- Control Rod Test Location
- Safety Rods
- Control Rod

FIGURE 4.29. Critical Assembly 48 (Half #2)

FIGURE 4.30. Horizontal Cross Section of Assembly 48 with a Central Sodium Stainless Steel Column

48A is shown in Figure 4.34. The horizontal section is similar to Figure 4.30 with the inconel sectors extending over the core and end blankets. It should be noted that while safety rod asymmetry remains, here it is the same in both halves.

The locations for experimental rods are indicated in Figure 4.34 by letter-number combinations as used in Table 4.V. These locations were selected to provide varying degrees of rod interaction.

Core Drawer

Control Drawer

Safety Drawer

FIGURE 4.31. Assembly 48 Drawer Loadings

FIGURE 4.32. Cross Section of Distributed Control Rod Drawer, Rod No. 1

- This rod (Rod # 2) is to be compared to a rod similar to Rod 1 (Figure 4.32) with the central sodium plate replaced by aluminum with 45% density (Rod #3).
- In this figure only the drawer contents are shown; the drawer and matrix tube are as shown in Figure 4.32.

FIGURE 4.33. Cross Section of Lumped Control Rod Drawer Contents, (a, b,) Rod No. 2

FIGURE 4.34. Critical for Interaction
Experiments, Assembly 48A (Halves 1 & 2)

Calculations

Calculations of the worths of both in-core and reflector rods were made with the transport codes DTF IV⁽⁵⁾ and 2DF⁽⁶⁾, which are one and two-dimensional models, respectively. Five group cross section sets were collapsed from a modified version of the Russian 26 group set⁽⁷⁾ using fundamental mode diffusion calculations of spectra in the code FCC⁽⁸⁾. Cross section sets were averaged over 26 group spectra for each region of the assemblies as indicated in Table 4.VI.

TABLE 4.VI. Region Cross Section Sets

Region	Spectrum	Criticality Condition
Core	Core	$k_{\text{eff}} = 1$
Blanket	Blanket	$B^2 = 0$
In-core Rods	Core	$k_{\text{eff}} = 1$
Reflector	Reflector	$B^2 = 0$
Reflector Rods	Core	$k_{\text{eff}} = 1$

The energy bounds of the five groups of the spectrum are 10.5, 0.8, 0.1, 0.01, 0.001 MeV, and 0.025 eV. The 26 group spectra are shown in Figure 4.35 with these energy bounds illustrated.

In-Core Rod Calculations

Both R-Z and x-y calculations of the worths of in-core rods have been made to obtain some measure of both the axial leakage effects and heterogeneity effects. The safety rod asymmetries were removed by replacing them with the normal core composition. The corresponding necessary increase in core volume was added at the core-reflector interface. The additional amount of core required was established in a series of reactivity experiments performed earlier.⁽⁴⁾

For the R-Z calculations the system was cylindricized as illustrated in Figure 4.30, which also shows the radial and axial mesh structure. The distributed rod of Figure 4.32 was homogenized by conserving material as were the other zones of the assembly. These homogenized atom densities are presented in Table 4.VII.

The worth of this homogenized rod was deduced from the difference between the calculated effective multiplications of the assembly containing, first, a sodium-steel column and, second, the rod mixture. The value of 2.023% $\Delta k/k$ is quite large compared to the measured value 1.734% $\Delta k/k$. This difference is in part due to decreased self-shielding introduced by homogenizing the rod. A x-y heterogeneous calculation, to be discussed below, illustrates this.

FIGURE 4.35. Calculated Spectra for Assembly 48A

TABLE 4.VII. Compositions for Core 48 and 48-A Calculations

		Atom Densities - Atom/b-cm						
		Core	Safety Rod	Hom. Enr. Rod	Hom. Nat. Rod	Inconel 1, Zone 6	Inconel 2, Zone 7	Inconel 3, Zone 8
1	^{239}Pu	0.001645	0.003183					
2	^{240}Pu	0.000106	0.000152					
3	^{241}Pu	0.000011	0.000014					
4	^{235}U	0.000016	0.000009					
5	^{238}U	0.007427	0.004682					
6	C	0.020767	0.020766	0.006350	0.005244			
7	Fe	0.009985	0.013611	0.026775	0.026775	0.013914	0.013921	0.013788
8	Cr	0.002681	0.003655	0.007189	0.007189	0.003684	0.003686	0.003650
9	Ni	0.001330	0.001798	0.003537	0.003537	0.047514	0.47514	0.047514
10	Mo	0.000206						
11	Al	0.000109	0.001682					0.000361
12	Na	0.006231	0.004154	0.006263	0.006263	0.004176	0.004152	0.003897
13	^{10}B			0.015486	0.003736			
14	^{11}B			0.001584	0.015736			

		Blanket	SS Mesh	Hom. Engr. Rod W/Al	Het. Enr. ^(a) Rod W/Al	Het. Nat. ^(a) Rod	Het. Enr. ^(a) Rod	SS-Na Column
1	^{239}Pu							
2	^{240}Pu							
3	^{241}Pu							
4	^{235}U	0.000083						
5	^{238}U	0.039798						
6	C			0.006350	0.030162	0.024909	0.030162	
7	Fe	0.004444	0.004710	0.025831	0.050996	0.051826	0.051826	0.013080
8	Cr	0.001193	0.001264	0.006936	0.013693	0.013915	0.013915	0.003512
9	Ni	0.000587	0.000622	0.003413	0.006737	0.006846	0.006846	0.001728
10	Mo							
11	Al			0.002263	0.021499			
12	Na			0.004175	0.023498	0.023498	0.023498	0.016701
13	^{10}B			0.015486	0.073556	0.017746	0.073556	
14	^{11}B			0.001584	0.007524	0.074746	0.007524	

(a) For distributed or lumped heterogeneous rods.

To perform a good two-dimensional (xy) calculation of a three-dimensional system, one must properly account for axial leakage, which has spatial dependence. In the calculations reported here a single non-spatially dependent approximation has been used. This was established by adjusting the effective height of a one-dimensional (radial) model of the above R-Z configuration. This axial leakage was then applied directly in the xy calculations. The model for the xy calculations is shown in Figure 4.36. The square core was chosen to give approximately the same k_{eff} as the unrodded cylindrical core and is considered adequate for these central rod calculations. The calculational models and mesh detail for the distributed and lumped rods are shown in Figures 4.37 and 4.38.

To test the axial leakage approximation the homogeneous rod calculation was repeated and the value of 2.001% $\Delta k/k$ was obtained, which is in good agreement with the R-Z result. Values of 1.843 and 1.645% $\Delta k/k$ were found for the distributed and lumped rods, respectively, which illustrates the effect of heterogeneity. While these values still overestimate the rod worths, they are reasonably close and the differences of 6 and 7%, respectively, can be used as correction factors in design calculations.

Further, while these heterogeneous calculations produce significant improvements in the experiment-theory correlation over that obtained from

homogeneous models, the lack of agreement remains undesirably large. Inspection of the calculation model reveals three areas where improvement can be sought with additional effort:

1. A five group energy model of the neutron spectrum has been employed. While this has been shown to be reasonably accurate on fairly homogeneous full-reactor problems, no attempt has been made to demonstrate the same with respect to extremely heterogeneous problems. Group structure can be increased to 26 groups.
2. An S2 transport scattering model has been used, which is known to be, at best, a weak approximation. Recent efforts of others⁽⁹⁾ indicate considerable sensitivity to the order of the S_n approximation. Again, the order of the approximation can be extended to even higher approximation (e.g., S_4).
3. The choice of cross section averaging schemes is not too well established. Cross sections for a given zone of a heterogeneous system are a function of the energy distribution of neutrons within and outside the zone in question. While this problem is not new, it has been considered a second order problem in fast reactor analysis until recently. Since, as indicated, one is now interested in calculational precision of better than a few percent, this problem must again be

* See Figures 4.37 and 4.38 for detail

FIGURE 4.36. Calculational Model--Central Control Rod Experiments (two dimensional x-y geometry)

FIGURE 4.37. Calculational Model of Distributed Control Rod (No. 3)

FIGURE 4.38. Calculational Model of Lumped Control Rod (No. 2)

faced in fast reactor analysis as it has been faced in thermal reactor analysis.

A first step in such a refinement would be a fine group (26) calculation of the neutron spectrum in an idealized cylindrical model of the experimental rods and reactor. Using the spectra from the various zones of materials, one could then obtain average (five group) cross sections more closely approximating those required.

Reflector Rod Calculations

The calculations of the worths of reflector rods and combinations of rods is following essentially the same pattern as that established for the in-core rods. The xy geometric model and mesh detail is shown in Figure 4.39. It is to be noted that the safety rod must be included in these calculations because no known method of increasing the core radius exists that will not displace the rods.

FIGURE 4.39. Calculational Model, 1/4 Core of ZPR-III

Therefore, the safety rod for the 1/4 and 1/2 core calculations has been placed at the average safety rod position; the asymmetry has been ignored. This should introduce only a small error in the absolute multiplication and spatial flux distribution and only a second order error in rod worths.

Core and blanket compositions remain the same as for the in-core calculations. The reflector compositions are tabulated by zone in Table 4.VII.

Worths of the reflector rods are again deduced from k_{eff} calculations with and without rods present. With the rods out, the rod locations were filled with reflector materials. For

the two rod combinations computed to date, the agreement is quite good. However, the rods were homogenized and leakage was taken to be that of the unrodded core. Hence, the actual lengths of the rods have essentially been ignored. Heterogeneity and rod length effects have been investigated and found to be approximately 10%; opposite in sign; hence, tend to cancel. Therefore, one may conclude that this model for calculating reflector rod worths is as adequate as that used for the in-core calculations.

Axial effects of reflector rods are presently being investigated using three-dimensional 2 group diffusion models.

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FTR SHIELDING STUDIES

D. R. Marr, C. A. Mansius, and
W. L. Bunch

The attenuation characteristics of a number of alternate shield arrangements have been compared analytically to provide a basis for the conceptual design of the Fast Test Reactor. The removal-diffusion code MAC was used for these calculations. In all cases, the reactor was assumed to consist of an 800 liter cylindrical core with oxide fuel operating at 400 MW. Important considerations in evaluating a shield arrangement are: biological exposure rates in regions to be entered by operating personnel, radiation damage rates in structural components, heat generation rates in shield components, activation rates in the primary and secondary coolant, and radiation environment for sensors in the reactor instrument system.

Table 4.VIII presents the thickness and composition of four radial shield

arrangements. Results of the calculations are summarized in Figures 4.40 and 4.41. The first arrangement provides a base case in which the gas-cooled neutron shield is assumed to consist of graphite encased in steel tubes. The calculations indicate this arrangement to be inadequate because of excessive leakage of activation flux to the secondary sodium located outside the neutron shield. In the second case, it was assumed that the graphite contained 1 wt% boron. Although the boron was calculated to depress the thermal neutron flux and gamma generation rate throughout the neutron shield region, total neutron leakage to the secondary sodium was still considered to be excessive because the effect of the boron was negligible on the intermediate and higher energy components of the spectrum. In the third case, it was assumed that the graphite in the neutron shield region was replaced with serpentine, a mineral containing

TABLE 4.VIII. Composition and Thickness of Postulated Radial Shield Regions

Region Number	Name	Thickness, cm	Composition, vol%
Base Case 1			
1	Core 1	27.1	PuO ₂ = 5.657, UO ₂ = 19.343, 304 SS = 25.0 Na = 45.0, void = 5.0
2	Core 2	27.1	
3	Skewed Reflector	17.3	304 SS = 55.7, Ni = 26.2, Na = 18.1
4	Sodium Pool 1	10.0	Na = 100
5	Sodium Pool 2	35.0	Na = 100
6	Tube Reflector	60.0	Na = 19.4, 304 SS = 17.6, Inconel X = 62.9
7	Sodium Pool 3	5.0	Na = 100
8	Vessel	7.0	304 SS = 100
9	Pipe Space	62.0	Void
10	Neutron Shield	152.0	Steel = 25.7, C = 54.1 void = 20.2
11	Steel Wall	1.0	Steel = 100.0
12	Secondary	60.0	Na = 100

Boronated Case 2, same as base case except for following:

10	Neutron Shield	152.0	Steel = 25.7, 1% boronated graphite = 54.1, void = 20.2
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Serpentine Case 3, same as base Case 1 except for following:

10	Neutron Shield	152.0	Steel = 25.7, Serpentine = 54.1, void = 20.2
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Pool Case 4, same as base Case 1 except for following:

9	Pipe space	62.0	Na = 100
10	Neutron Shield	152.0	Steel = 25.7, C = 54.1, Na = 20.2

FIGURE 4.40. Neutron Flux Density as a Function of Distance into Shield

FIGURE 4.41. Gamma Energy Flux Density as a Function of Distance into Shield

magnesium oxide, silicon dioxide and bound water of hydration. For the thickness of neutron shield considered, this arrangement would satisfactorily suppress activation in the secondary system. The fourth case simulates a pool concept in which the canned graphite neutron shield of the base case would be submerged in sodium. Although sodium is not an efficient shield material, its effect is quite apparent when used to fill the void regions in the shield.

Base case calculations have also been completed for the upper and lower axial shield regions. For these

calculations it was assumed that skewed guide tubes would be used to position fuel within the core. Material densities in the shield regions are average values based on a skew angle of 2.75° and a triangular pitch. The lower and upper shield arrangements shown in Tables 4.IX and 4.X are based on dimensions for concept PNL-III. Results of the neutron calculations for these two cases are summarized in Figures 4.42 and 4.43, where "fast" neutrons are those with $E > 0.06$ MeV. Additional calculations are in progress to determine the effect of alternate compositions.

TABLE 4.IX. Composition and Thickness of Postulated Lower Axial Shield Regions

Shield Region	Name of Region	Region Width, cm	Region Materials and Effective Densities (g/cc)			
			Sodium	Chromium	Iron	Nickel
1	Fuel Support	20.3	0.739	0.22	0.80	0.13
2	Stainless Tubes and Sodium	20.3	0.792	0.115	0.421	0.066
3	Stainless Tubes and Sodium	40.6	0.80	0.102	0.376	0.059
4	Stainless Tubes and Sodium	40.6	0.807	0.088	0.326	0.051
5	Stainless Tubes and Sodium	40.6	0.815	0.075	0.278	0.044
6	Stainless Tubes and Sodium	33.0	0.83	0.055	0.192	0.030
7	5 in. Support Plate	12.7	0.17	1.15	4.23	0.66
8	Stainless Tubes and Sodium	33.0	0.80	0.073	0.27	0.042
9	5 in. Support Plate	12.7	-	1.45	5.34	0.84
10	Sodium	38.1	0.84	-	-	-
11	3 in. Stainless Plate	7.6	-	1.45	5.34	0.84
12	Sodium	82.5	0.84	-	-	-
13	2 in. Stainless Vessel	5.08	-	1.45	5.34	0.84
14	Sodium	53.3	0.84	-	-	-
15	1 in. Primary Shell	2.5	-	1.45	5.34	0.84

TABLE 4.X. Composition and Thickness of Postulated Upper Axial Shield Regions

Shield Region	Name of Region	Region Width, cm	Region Materials and Effective Densities g/cm ³			
			Sodium	Chromium	Iron	Nickel
1	Fission Gas Plenum	61.0	0.42	0.426	1.57	0.25
2	Process and Hanger Tubes	30.5	0.601	0.387	1.426	0.224
3	Process and Hanger Tubes	71.1	0.634	0.329	1.212	0.190
4	Process and Hanger Tubes	200.7	0.720	0.177	0.651	0.102
5	Tubes and Nozzles	76.2	0.70	0.218	0.801	0.126
6	Gas Plenum Above Pool	50.8	-	0.203	0.75	0.118
7	Steel Ball Shield	76.2	-	0.87	3.2	0.50
8	Gas Plenum Above Ball Shield	73.7	-	0.116	0.43	0.067
9	304 SS Plate	5.08	-	1.45	5.38	0.84

Heat generation rates in shield regions have been estimated based on the gamma distributions generated in the MAC calculations. These estimates are summarized in Table 4.XI together with the composition and thickness of the shield regions employed in the calculations. Since MAC is a one

dimensional code, the geometric effect is not included in these estimates and the tabulated values should be conservative. Additional calculations are planned for alternate shield arrangements and methods of refining the code to incorporate the geometric effect are being investigated.

FIGURE 4.42. Lower Axial Neutron Flux
for PNL-III Concept

FIGURE 4.43. Upper Axial Neutron Flux
for PNL-III Concept

TABLE 4.XI. Calculated Heat Generation Rates in the FTR Shield

<u>Region (Vol% Material)</u>	<u>Thickness, cm</u>	<u>Heat Generation Rate, MW</u>
Top Axial:		
Reflector (70 SS, 30 Na)	60	3.9
Plenum (3 SS, 97 Na)	140	0.3
Tube Sheet (97 SS, 3 Na)	7.5	0.04
Upper Pool (8 SS, 92 Na)	260	0.01
Bottom Axial:		
Reflector (70 SS, 30 Na)	60	3.9
Plenum (3 SS, 97 Na)	132	0.31
Intermediate Tube Sheet (97 SS, 3 Na)	12.8	0.03
Inlet (100 Na)	37.0	0.00
Lower Tube Sheet (100 SS)	12.8	0.00
Radial:		
Skew Reflector (56 SS, 26 Ni, 18 Na)	17.32	12.4
Sodium (100 Na)	45.0	1.3
Tube Reflector (18 SS, 63 Ni, 19 Na)	60.0	7.8
Sodium (100 Na)	5.0	0.03
Pressure Vessel (100 SS)	7.0	0.04

GAMMA INTENSITY FROM FTR SODIUM

R. E. Shaver, C. A. Mansius, and
W. L. Bunch

An estimate was made of the gamma intensity at selected locations within two different closed loop cell concepts to provide a basis for evaluating the effect of the radiation environment on the placement of components whose lifetimes are affected by radiation. Activity of the sodium was based on earlier calculations for a 5MW loop. The computer program ISOSHLTD was used to calculate the geometric and attenuation effects. Because of the complex arrangement, it was necessary to take sums and differences of real and hypothetical

segments of the loop and loop components to achieve the desired results. Results are presented in Table 4.XII. In the base cell, the hot trap, cold trap, and heater are located in a shielded compartment separate from the primary pump, heat exchanger, and expansion tank. In the unitized cell, all components are within the same cell. The gamma intensity in the unitized cell is calculated to be about an order of magnitude higher because of the presence of the traps and heater. The tabulated range indicates the effect of possible variations in loop flow parameters; the actual activity will also be strongly influenced by the design of the FTR core and reflector.

An estimate was made of the effect of fission products in the sodium on the gamma activity levels. It was assumed that fission products from fuel that had been generating 150 kW for 200 days were instantaneously distributed homogeneously throughout the closed loop system. This quantity of fission products would essentially double calculated radiation levels. Following shutdown of the system, the ^{24}Na activity would decay with a 15 hr half life and the fission products would become the predominant source of activity.

Heat generation in structural concrete was calculated for two possible coolant system arrangements to compare cooling requirements. In the first arrangement the coolant external to the reactor would be contained within four parallel piped loops connecting the heat exchangers and primary pumps. In the second arrangement the sodium from the reactor

would pass through one of the heat exchangers before entering a large open pool. The four primary pumps would be submerged in this pool of sodium. It was assumed that the activation rate in the sodium would be the same in both systems, but the specific activity of the pool system would be less because of the quantity of material present.

Gamma heating due to the decay of the sodium was calculated using the computer program ISOSHL. In the pool concept the calculated heat generation rate would be relatively uniform over a large portion of the surrounding concrete structure, while in the loop concept the calculated heat generation rate would be realized only adjacent to each loop. In both cases the heating rates are sufficiently low to permit the use of structural concrete. Results are presented in Table 4.XIII.

TABLE 4.XII. *Calculated Gamma Intensity at the EM Fill and Recirculation Pump Blower Motor in Conceptual Loop Cells*

Cell Arrangement	Gamma Intensity	
	MeV/cm ² sec	R/hr
Base cell	0.8 - 1.3 x 10 ⁹	1.3 - 2.1 x 10 ³
Unitized cell	1.0 - 1.7 x 10 ¹⁰	1.6 - 2.6 x 10 ⁴

TABLE 4.XIII. *Calculated Heat Generation Rate in Concrete from Radioactive Sodium*

FTR Concept	Gamma Flux, MeV/cm ² sec	Heat Generation, W/cm ²	Surface Rate, W/cm ³
Loop	3.1 x 10 ¹⁰	4.9 x 10 ⁻³	0.27 x 10 ⁻³
Pool	5.2 x 10 ⁹	8.3 x 10 ⁻⁴	0.46 x 10 ⁻⁴

5.0 CRITICAL MASS PHYSICS

BASIC CRITICALITY EXPERIMENTS WITH PLUTONIUM NITRATE SOLUTIONS IN SLAB GEOMETRY

R. C. Lloyd, E. D. Clayton, and
L. E. Hansen

Criticality experiments were continued with the adjustable slab assembly to determine critical thicknesses of bare and reflected "infinite" slabs of plutonium nitrate solution.⁽¹⁾ Measurements were made with the 42 x 42 in. vessel bare, water reflected, and with the vessel reflected on its sides with a one in. thick layer of Lucite. Slab thicknesses were in the range of 4.8 in. to 8.0 in. The plutonium concentration was about 58 g Pu/liter at acid molarities of 2.3 and 5. The plutonium contained 4.6 wt% ²⁴⁰Pu. Measurements were also made to evaluate the effect of the stainless steel vessel walls, lattice reinforcement, and nearby hood walls on criticality.

The results of the measurements are presented in Figures 5.1 and 5.2. The 1/16 in. tank wall was evaluated by determining the critical height with and without an added 1/16 in. of steel plate. From these measurements it was possible to estimate the height for zero thickness steel. From Figures 5.1 and 5.2 where the critical height is plotted against slab thickness, the difference in height due to the vessel wall could be determined in terms of slab thickness. Similar

measurements were made to determine the effect of the support gridwork, which is 1/8 in. material on a 4 in. square lattice made 4.5 in. deep. The effect of the closest hood wall was evaluated by adding a similar wall on the opposite side and again extrapolating on the critical height, thus obtaining the effect in terms of height. The results of these evaluations are given in Table 5.1.

TABLE 5.1. Evaluations of Tank Walls,
Support Structure and Hood Wall

	<u>5M</u>	<u>2.3M</u>
Tank Wall - Bare	0.10 (a)	0.08 (a)
Support Structure - Bare	0.14	0.10
Hood Wall - Bare	0.03	0.03
Tank Wall - Reflected	0.15	0.14
Support Structure - Reflected	0.02	0.02

(a) Evaluation corrections are given
in inches of slab thickness

Final chemical analyses of the solutions have not been completed by the Analytical Laboratory.

Further data analyses will be made to determine extrapolation lengths, bucklings, and infinite system slab thicknesses for clean systems on completion of chemical analyses.

An attempt was made to compute the experimentally determined critical dimensions of the 2.3M nitrate solution. The calculations were made

FIGURE 5.1. Criticality of $\text{Pu}(\text{NO}_3)_4$ in Slab Geometry

FIGURE 5.2. Criticality of $\text{Pu}(\text{NO}_3)_4$ in Slab Geometry

using HFN⁽²⁾ with multigroup constants (18 energy groups) derived from the GAMTEC-II⁽³⁾ code.

The computed eigenvalues are in very poor agreement with the experimental results. The calculations strongly indicate that the effect of the stainless steel and other extraneous reflectors has been underestimated. The values of k_{eff} computed for bare and stainless steel reflected systems are consistently low, and for water reflected systems they are consistently high. The analysis is continuing, and it is felt that the calculations may be improved by homogenizing the stainless steel and water reflector region, thereby increasing the poison effect of the steel. Likewise, a low density steel region external to the steel vessel wall in

the bare cases may improve the reflection properties of the steel and therefore raise the computed eigenvalues.

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CRITICALITY EXPERIMENTS WITH CASKS OF PLUTONIUM METAL

R. C. Lloyd and E. D. Clayton

A series of criticality experiments was performed with United Kingdom nut type casks (Figure 5.3.) containing plutonium metal, utilizing the Remote Split-Table Machine. The experiments provided data for nuclear safety guidance in handling, storing, and shipping this material.

Critical approach type experiments were carried out with the nine available casks in bare and Lucite reflected arrays. The casks are cadmium plated steel nuts weighing 9.39 kg and contain-

ing three cylinders of plutonium metal ($\sim 1\frac{1}{4}$ in. diam; 2 in. long), totaling about 6.6 kg of plutonium per cask. The plutonium contains 14.9 wt% ²⁴⁰Pu. Reflected experiments were also made with 1/4 and 1/2 in. of Lucite moderator placed between the casks. The effect of moderation removal and the effect of the cadmium plating was determined qualitatively from changes in neutron multiplication. Data from the experiments are presented in Tables 5.II and 5.III.

The bare close-packed 3 x 3 square array was quite subcritical. The conservative plots extrapolated to greater than 13 casks for criticality, while

FIGURE 5.3. United Kingdom Casks of Pu Metal Reflected by Lucite

TABLE 5.II. Criticality of Casks of Plutonium Metal

Exp. No.	Reflector	Center-to-Center Spacing, in.	Extrapolated No. of Casks for Criticality	Density Analog Estimate	Comments	Estimated No. of Casks With Cd Coating Removed
1	Bare	4-3/8 sq. array	19	11	Close Packed	
2	Bare	4 hex array	20	10	Close Packed	
3	Lucite	4 x 4-3/8	10.5	6.3	Close Packed	9.5
5	Lucite	4-1/4 x 4-5/8	11.5		1/4 in. Lucite Moderator	
	Lucite	4-1/4 x 4-5/8	11.75		Moderator Removed	
6	Lucite	4-1/2 x 4-7/8	14		1/2 in. Lucite Moderator	11
	Lucite	4-1/2 x 4-7/8	13.5		Moderator Removed	12.5

TABLE 5.III. Neutron Counting Data

Exp. No.	Lucite Moderator	Count Rate in Counter 1	Count Rate in Counter 2	Count Rate in Counter 3	Average	Effect on No. of Casks For Criticality
4	Close Packed Array					
	none	67880	54768	67911	63520	0.05
	With Lucite in Two Holes, 1-3/4 in. x 2 in.	67213	53695	66819	62576	
5	1/4 in.	53040	40318	51745	48368	1/4
	none	49314	37489	48228	45010	
5a	Close Packed Array Comparison Counts to Exp. No. 5					
	none	77288	57305	75354	69982	
6	1/2 in.	36607	27090	35618	33105	1/2
	none	39197	29146	38377	35573	
	Cadmium Coating Removed From Center Cask					x 9 = 3
	1/2 in.	38382	28296	37494	34724	x 9 = 1
	none	39997	29502	39046	36182	

5.5

BNWL-472

the more accurate plots extrapolated to 19 casks. The Lucite reflected, close-packed assembly gave extrapolations of about 10-1/2 casks for criticality. The reflected assembly with 1/4 and 1/2 in. Lucite moderator gave extrapolations of 11-1/2 and 14 units, respectively.

The count rate with 1/4 in. of moderator was about 7 1/2% greater than for the same system without moderator; conversely, the system without moderator gave a 7 1/2% higher count rate than that with the 1/2 in. moderator, thus indicating that the reactivity would remain unchanged for a moderator thickness of about 3/8 in. Filling two of the holes between the casks with Lucite, as was done in experiment No. 4, causes a decrease of about 1.5% in count rate, indicating that flooding a close packed assembly would be less reactive than merely reflecting it.

Neutron counting data taken with and without cadmium coating on the center cask of the 3 x 3 array indicated that removal of the coating, with and without the 1/2 in. Lucite moderator, made only 5% and 2% increase in the count rate, respectively.

Further analysis of the count rate

changes to determine their effects in terms of critical number of casks was made. These values are included in Tables 5.II. and 5.III. The figure shows the critical number of casks versus edge of cask to edge of cask spacing. The criticality estimates were determined by comparing these count rates to those of similar experiments where critical values were measured and curves of percent of critical mass versus count rate could be established. These estimates would not be as accurate as if obtained by a critical approach experiment, but do give indications of magnitude useful for consideration in nuclear safety guidance.

The removal of the 1/4 in. moderator in Exp. No. 5 increased the critical number of casks by about 1/4 of a cask. The removal of the 1/2 in. moderator in Exp No. 6 decreased the critical number of casks by about 1/2 cask. Multiplying the effect seen on removal of the Cd coating on the center cask by nine for the cases with and without the 1/2 in. Lucite moderator gave a critical number of casks of 11 and 12.5, respectively.

CRITICAL MASS AND BUCKLING MEASUREMENTS FOR 2.1 wt% ENRICHED URANIUM FUEL TUBES IN LIGHT WATER

L. E. Hansen

A series of critical mass and buckling measurements were completed to provide nuclear safety guidance for the handling and processing of 2.1 wt% enriched U fuel tubes. The fuel being

considered was 1.375 in. OD, 0.633 in. ID, and was in the form of elements 7 in. in length. To perform the inverse multiplication and exponential experiments, the fuel elements were stacked in 49 in. columns contained in aluminum tubes having a 1.395 in. ID and a 55 mil wall thickness.

Five triangular lattice spacings were examined, thereby providing data

over a range of moderation ratios. Critical masses, bucklings, and extrapolation lengths are given in Table 5.IV. at the respective water-to-uranium volume ratios.

The effects of mixing 0.95 and 2.1 wt% enriched fuel tubes in the 2.2 in. lattice was also examined. The two fuel arrangements and the results obtained are given below:

1. Alternate Tubes of 2.1 wt% and 0.95 wt% Enriched Uranium, With the Central Element Being 2.1 wt% ^{235}U

The total critical number of fuel tubes was 70.7 (34.85 tubes of 0.95 wt% and 35.85 of 2.1 wt%).

2. Alternate Rings of 2.1 wt% and 0.95 wt% Fuel, With the Innermost Seven Being 2.1 wt%

In this case, the total number of tubes for criticality was 71.0 (36.0 tubes of 0.95 wt%, and 35 tubes of 2.1 wt%).

It is interesting to note that the alternate tube and alternate ring arrangement yielded the same results within the experimental uncertainty.

This effect should be examined further, particularly at other water-to-uranium volume ratios.

Also examined were the effects of adding seven Pb-Cd (~4% Cd) poison columns to the 2.2 in. lattice of 2.1 wt% enriched fuel. The experimental critical number of fuel tubes for the two poison arrangements are given below:

1. Seven Poison Elements Grouped About the Center of the Assembly With Two Fuel Elements Separating Each Poison Element

The critical number of fuel tubes was estimated to be 66.3.

2. Seven Poison Elements Grouped About the Center, With One Fuel Element Separating Each Poison Element

The critical number of fuel tubes for this system was 75.4. The minimum critical mass derived from the experiments was 338 kg U; this was for a water-to-uranium volume ratio of approximately 2.86. The maximum material buckling was 96.75 m^{-2} , at a water-to-uranium volume ratio of 2.10.

TABLE 5.IV. Critical Masses, Bucklings, and Extrapolation Lengths

Lattice Spacing, in.	Vw/VU	Uranium Height, in.	Estimated Critical No. of Fuel Tubes	Extrapolation Length, cm	Material Buckling, m^{-2}
1.95	1.57	49.0	49.1	7.15	96.67
2.10	2.04	49.0	42.3	6.80	96.69
2.20	2.36	49.0	38.8	6.68	96.64
2.40	3.04	49.0	37.5	5.86	93.84
2.60	3.78	49.0	42.6	5.73	78.70

CRITICALITY OF POISONED PuO_2 FUELED ASSEMBLIES

L. E. Hansen, R. C. Lloyd, and
S. R. Bierman

A series of criticality experiments were performed, utilizing the Remote Split Table Machine, in support of the Gas Cooled Fast Breeder Reactor (GCFR) program. These experiments are being performed for General Atomic, and are designed to check basic neutron data and methods of calculation employed in the design analysis of the GCFR.

The PuO_2 -polystyrene fuel has an H:Pu ratio of 5, and 12 isotopic percent ^{240}Pu . The resultant spectrum is similar to that which could be encountered as a result of water entry into the GCFR core. In the event of such an occurrence, it is necessary to know the effects of various resonance and thermal poison materials which could be employed for control. With this in mind, material worth measurements are being made using dysprosium, gadolinium, and hafnium. These materials are in the form of oxide powders which, for the purpose of obtaining simple geometric experiment, have been compacted in stainless steel containers whose dimensions are 6 x 6 x 1/4 in.

The basic geometry used is a rectangular parallelepiped with base dimensions of 12 x 18 in. Inverse multiplication measurements were made to determine the critical height of a bare unpoisoned assembly, and assemblies containing three (1/4 in.) layers of each poison centered in the assembly with approximately 1 1/2 in. of fuel separating each poison layer (see Figure 5.4.). Experiments were also performed using layers of Plexiglas next to the poison plates to determine the effects of the poison in a more thermal spectrum. A similarly moderated unpoisoned assembly was also examined. Figure 5.5. shows the arrangement of fuel, Plexiglas, and poison which was used.

Since there was insufficient 12% ^{240}Pu (H/Pu = 5) material available to obtain criticality, it was necessary to complete the critical experiments using 2.2% ^{240}Pu (H/Pu = 15) PuO_2 polystyrene. The measured critical dimensions of the assemblies are given in Table 5.V.

Integral neutron spectral density measurements are also planned which will yield information for comparison to the computed neutron spectrum in the vicinity of the poison plates.

FIGURE 5.4. Assembly Arrangement

FIGURE 5.5. Assembly Arrangement with Plexiglas

TABLE 5.V. Critical Dimensions of the Assemblies

Experiment Number			Width, in.	Length, in.	Height of 12% ^{240}Pu Fuel Plus Plexiglas and Poison, in.	Critical Height of 2.2% ^{240}Pu Fuel, in.	Critical Mass of Pu, kg
12-005-011	Polystyrene	Dy_2O_3	12.36	18.28	10.16	4.43	92.30
12-005-012	Polystyrene	Unpoisoned	12.36	18.26	9.30	1.84	81.84
12-005-014	Polystyrene	Gd_2O_3	12.38	18.26	10.24	3.81	89.87
12-005-015	Polystyrene	HfO_2	12.38	18.30	10.25	3.90	90.22
12-005-016	Polystyrene-Plexiglas	Unpoisoned	12.37	18.24	12.05	None	68.10
12-005-017	Polystyrene-Plexiglas	Dy_2O_3	12.39	18.27	13.88	5.56	96.68
12-005-018	Polystyrene-Plexiglas	Gd_2O_3	12.38	18.27	13.93	5.20	95.34

MINIMUM CRITICAL ENRICHMENT OF HYDROGENOUS URANYL NITRATE SYSTEMS

S. R. Bierman and G. M. Hess

In the recovery of uranium from spent fuel elements and uranium bearing wastes, the uranium is converted to the nitrate form by dissolution in a nitric acid solution. It remains

in this chemical form throughout many of the processing steps. For low enriched homogeneous mixtures, criticality is strongly dependent on the enrichment and the degree of moderation. Thus, to permit safe and efficient operation during the processing and handling of low enriched uranyl nitrate, it is necessary to

know over what range of moderation criticality is possible for a given enrichment and to know at what minimum enrichment criticality is possible with optimum moderation.

An experimental program to provide data for determining the minimum critical enrichment of hydrogenous uranyl nitrate systems and to provide information in support of calculational models used to establish criticality controls has been completed in the Physical Constants Test Reactor (PCTR). Driver fuel elements were loaded into the PCTR to obtain a characteristic neutron spectrum across a 7 in. cubic test cell located in the center of a 21 in. cube of buffer material of the same or similar composition as the test cell material. A typical loading, with the 21 in. cube positioned in the center cavity of the PCTR is shown in Figure 5.6. Reactivity measurements were made on twenty-three different arrangements of buffer material and test cells to obtain data from which k_{∞} as a function of moderation could be determined. The measurements were made over the region of optimum moderation and at two different ^{235}U enrichments (2.14 and 2.25 wt%). The results of these measurements along with previous measurements⁽¹⁾ on 3.04 wt% enriched uranyl nitrate will permit determin-

ing the minimum critical enrichment for hydrogenous uranyl nitrate systems. The measurements also provide additional experimental values of k_{∞} as a function of moderation which are useful in checking the validity of a given calculational model.

Data reduction and analysis are currently in progress; however, some of the experimental results as obtained are shown in Table 5.VI. The H/U ratio shown is the current best estimate, since the chemical analyses are not yet complete. The reactivity response of the PCTR to a given material undergoing measurement is shown as $\Delta\rho_{\text{cell}}$ and is the observed difference in the reactivity with the center test cell filled with helium and with the material. Although these values are preliminary and do not include any of the neutron poison reactivity measurements, they do indicate a minimum critical enrichment slightly over 2.1 wt%. Most calculational models yield a minimum critical enrichment slightly under 2.1 wt%.

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Neg 0670839-1

FIGURE 5.6. Minimum Critical Enrichment of Hydrogenous Uranyl Nitrate Systems

TABLE 5.VI. Reactivity Response of PCTR to Various Test Cell and Buffer Region Loadings - Raw Data

Enrichment, wt% ^{235}U	H/U (a) Test Cell Atomic	Δk $\rho_{\text{cell}} - \rho_{\text{void}},$ ¢	Comments
2.14	3.44	-5.439	
2.14	6.37	+0.198	"Soft" loading
2.14	6.37	+0.141	"Hard" loading
2.14	10.34	-0.535	(Three differing
2.14	10.34	-0.942	(PCTR driver
2.14	10.34	-0.879	(loading arrangements
2.14	8.08	+1.219	Cd R on graphite, boundary was 2.76
2.14	8.08	+1.205	Cd R on graphite, boundary was 3.73
2.14	8.08	+1.233	Buffer H/U at 7.24
2.14	7.26	+0.831	Cd R on graphite boundary was 2.83
2.14	7.23	+0.463	Buffer H/U at 7.24
2.14	9.25	+1.304	Buffer H/U at 7.24
2.26	8.25	+5.026	Cd R on graphite boundary was 2.72
2.26	7.26	+4.375	Buffer H/U at 8.39
2.26	9.39	+5.333	Buffer H/U at 8.39
2.26	9.82	+5.381	Buffer H/U at 8.39
2.26	8.25	+5.568	Test cell at 6.2 °C Reactor at 27.432 °C
2.26	11.00	+4.914	Buffer H/U at 11.27
2.26	11.77	+3.795	Buffer H/U at 11.27
2.26	9.82	+5.308	Buffer H/U at 11.27

(a) Buffer region has the same H/U ratio unless noted otherwise.

ROSSI-ALPHA MEASUREMENT

G. M. Hess

A system for Rossi-alpha measurement has been assembled and is outlined in Figure 5.7. A scintillator and photomultiplier tube are used, the output of the latter is amplified, dis-

criminated, and fed to the linear gate. If the linear gate has been enabled, the pulses will pass the gate, and the first pulse will trigger the first delay generator. At the end of the period of time set by this generator, the second delay generator is triggered and enables the scaler for

a period set by this second generator. During this time the scaler will count the output of the discriminator.

The first delay generator then allows one to go out in time from an initial detection. The second delay generator sets the width of the counting interval at the end of this delay.

The second delay generator, at the end of the counting interval, enables the linear gate, the gate width being set large enough, using the average count rate, to allow another trigger. The number of intervals counted is recorded using the preset scaler count-

ing the input to the second delay generator. When the preset limit is reached, the scaler is disabled. The system is started by enabling the linear gate externally.

Data is collected for a number of delay times over the range of interest using a counting interval consistent with the neutron lifetime. As many as 20 data points may be necessary to perform a meaningful least-squares fit, but the time required for the collection of this data would certainly be less than that required using multichannel equipment with its relatively large deadtime problems.

FIGURE 5.7. Rossi-Alpha Measurement System

P₀ METHOD OF NOISE ANALYSIS

G. M. Hess

The probabilities of observing zero counts in various sized intervals can be related to the prompt neutron decay constant in a reactor assembly.⁽¹⁾

$$P_0 = \left(\frac{\overline{c^2} - (\overline{c})^2}{\overline{c}} \right) - \frac{\overline{c}}{\overline{c^2} - (\overline{c})^2} - 1 \quad (5.1)$$

and

$$\frac{\overline{c^2} - (\overline{c})^2}{\overline{c}} = 1 + \frac{\epsilon \frac{v(v-1)}{(v)^2} (1-\beta)^2}{\left[1 - \frac{1 - e^{-\frac{\beta-P}{\Delta} \tau}}{\frac{\beta-P}{\Delta} \tau} \right]} \quad (5.2)$$

where

P_0 = probability of observing zero counts in interval of time τ ,

$\frac{\overline{c^2} - (\overline{c})^2}{\overline{c}}$ = variance-to-mean ratio,

ϵ = detector efficiency,

and the other symbols have their usual meanings.

A system has been assembled to measure these P_0 probabilities and is outlined in Figure 5.8.

A scintillator is used in conjunction with a photomultiplier tube, the output of which is amplified, discriminated, and fed to the flip-flop. If the flip-flop has been reset by the pulse generator, a pulse will be input to Scaler No. 1. The same pulse that resets the flip-flop, triggers the delay generator, which enables Scaler

No. 1 for the amount of time desired, τ . Thus, the pulse that has passed the flip-flop will be recorded by Scaler No. 1 if it occurs in the interval of time set by the delay generator. A maximum of one pulse may be recorded since the pulse from the discriminator disables the flip-flop. The pulse generator then resets the flip-flop and the process continues until the preset scaler, which is counting the pulses from the pulse generator, has been satisfied. At this time, Scaler No. 1 and Scaler No. 2 are disabled and the counts are recorded. Scaler No. 2 has been counting the total count rate directly from the discriminator so that the average count rate is known.

The P_0 probability is then the difference of the total number of intervals counted (Preset Scaler) and the number of intervals which has at least one count in them (Scaler No. 1), over the total number of intervals (Preset Scaler).

This procedure is repeated for various values of τ , the variance-to-means are calculated from Equation (5.1), and a two parameter least squares fit are made to Equation (5.2.), giving a value for $\frac{\beta-P}{\Delta}$, or α .

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FIGURE 5.8. System for Measuring P_o Probability

PLUTONIUM CRITICALITY STUDIES - UNDER-MODERATED REGION

S. R. Bierman, L. E. Hansen, R. C. Lloyd and E. D. Clayton

A new series of critical experiments were begun at the Critical Mass Laboratory with PuO_2 -polystyrene compacts and the Remote Split-Table Machine. These experiments are a continuation of the basic research program to provide data needed for evaluating the effects of moderation and ^{240}Pu on intermediate neutron spectra plutonium systems. Previous experiments⁽¹⁾ were with fuel having a H/Pu atomic ratio of 15, and plutonium densities of 1.12 and 1.05 g/cm³ with ^{240}Pu isotopic contents of 2.2 and 8 wt%, respectively. The fuel used in the current experiments has a H/Pu atomic ratio of 5 and a plutonium density of 2.30 g/cm³ with a ^{240}Pu isotopic content of 11.5

wt%. A summary of the pertinent experimental data to date is shown in Table 5.VII. (This fuel was also used to evaluate the worth of various resonance absorbers. The results of these measurements are covered elsewhere in this report.)

Initially, the PuO_2 -polystyrene compacts were canned in aluminum for contamination containment; however, this resulted in about a 15% poorly defined void fraction being introduced into the system. Consequently, the aluminum cans were replaced with a 7 mil thick pressure sensitive tape capable of withstanding temperatures above 100 °C. To determine the effect of this 7 mil cladding tape, experiments were performed with additional layers of tape surrounding each fuel block. The critical mass remained essentially unchanged, over the thickness range of interest, although

Experiment Number	Reflector	Base Size, in.	Critical Height, in.	Critical Mass, kg	Comments
12-005-003	Full Lucite	10.69 x 10.41	8.27 $\pm$ 0.01	31.9	A1 canned fuel blocks
12-005-004	Full Lucite	10.19 x 10.19	7.75 $\pm$ 0.01	28.5	Horizontal Buckling Measurement - fuel clad in 7 mil tape
12-005-005	Bare	14.41 x 14.22	11.28 $\pm$ 0.02	81.0	1.4 kg of 15 H/Pu, 8.0% ^{240}Pu as driver fuel used to reach criticality
12-005-020	Bare	14.25 x 14.50	11.33 $\pm$ 0.01	77.5	11 mils of tape on all blocks 5.9 kg of 15 H/Pu, 2.2% ^{240}Pu as driver
12-005-021	Bare	14.31 x 14.56	11.54 $\pm$ 0.02	77.8	22 mils of tape on all blocks 6.2 kg of 15 H/Pu, 2.2% ^{240}Pu as driver
12-005-023	Full Lucite	12.19 x 12.30	5.82 $\pm$ 0.01	30.9	Horizontal Buckling Measurement
12-005-024	Full Lucite	16.25 x 16.40	4.34 $\pm$ 0.01	41.2	Horizontal Buckling Measurement
12-005-025	Full Lucite	20.31 x 20.50	3.69 $\pm$ 0.01	54.8	Horizontal Buckling Measurement

TABLE 5.VII. *Plutonium Criticality Studies*

the critical size increased slightly as a result of the decrease in plutonium density. The negative reactivity effect of the density reduction was offset in this undermoderated system by the addition of moderator contained in the tape.

Because of the relatively high plutonium concentration at H/Pu of 5 and the poor thermal conductivity inherent in this type of fuel, core heating experienced from α -decay was much greater than that observed in the previous 15 H/Pu fuel. Temperature increases of as much as 40 °C were observed during some of the approaches-to-criticality. However, neutron multiplication measurements at various temperatures showed that for a 30 °C change in the core central temperature of a bare prism, only a 0.5% change is observed in the

extrapolated critical size. For the reflected prism the temperature effect was small and not observable by these methods. In the reflected system, the reflector temperature lags the core temperature by several degrees.

Measurements were also completed on four Lucite reflected rectangular prisms having different horizontal bucklings. From these measurements, it will be possible to determine the critical thickness and the extrapolation length of a reflected infinite slab of this 5 H/Pu fuel. This data is currently being analyzed.

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6.0 NEUTRON PHYSICSCOMPLETION OF FAST-NEUTRON TOTAL-CROSS-SECTION MEASUREMENTS

D. G. Foster, Jr. and D. W. Glasgow

The final block of measurements for the total-cross-section program⁽¹⁾ was taken in October and November, 1966. The run encompassed 48 samples and provided approximately 1/5 of the entire body of raw data. Data-reduction has now been completed, and the results are ready for publication and distribution.

The complete set of final results includes neutron total cross sections from 2.25 to 15 MeV for the following samples:

- Seventy-eight naturally-occurring elements (the only omissions are the noble gases and the highly-radioactive elements heavier than bismuth);
- Fourteen isotopically-enriched or artificially-produced samples: D, ^6Li , ^7Li , ^{42}Ca , ^{44}Ca , ^{53}Cr , ^{99}Tc , ^{147}Pm , ^{182}W , ^{186}W , ^{206}Pb , ^{233}U , ^{235}U , and Pu; the results for ^6Li , ^{42}Ca , ^{44}Ca , ^{53}Cr , ^{182}W , ^{186}W , ^{206}Pb , ^{233}U , and ^{235}U have been further enriched algebraically by combining the results from more than one isotopic mixture;
- Algebraic enrichments of 5 isotopes which dominate their natural isotopic mixtures: ^{40}Ca , ^{52}Cr , ^{184}W , (to 68 at.%), ^{208}Pb (to 70a/o), and ^{238}U ; and,
- Two common abnormal isotopic mixtures, radiogenic lead (89 at.% ^{206}Pb) and enriched uranium (93 at.% ^{235}U).

Thus, there are 99 records in the final list, comprising 24,112 data points.

Three severe instrumental problems which were present in the apparatus at the end of 1965, were eliminated before the final run. The first of these, electronic loss of stored data in memory or during output, was eliminated by replacing the home-made, 400 channel analyzer with a PDP-8 computer. The second, unreliable punching of the paper-tape output, was eliminated by replacing the old tape perforator with a Tally P-120 perforator with a parity-checking attachment (however, the entire 1966 run was punched out without a single parity fault!). The third, a counting-rate-dependent instability in the zero point of the time-of-flight scale, was not positively diagnosed until the end of 1965, when the overall stability of the apparatus became good enough to permit unambiguous diagnosis. It was cured (albeit rather reluctantly) by replacing the vernier chronotron with a combination of a commercial time-to-pulse height converter and a commercial analog-to-digital converter. The latter had been furnished with the computer. The former was purchased from Edgerton, Germeshausen, and Grier along with the necessary auxiliary modules to provide logical control. The new apparatus was interfaced to the computer as a single system, using the modules manufactured by Digital Equipment Corp., which also manufactured the PDP-8 computer and built the interface equipment for the analog-to-digital converter.

A supervisory program (TOFDAD) for taking time-of-flight data on the PDP-8 was prepared by modifying D. N. Samsky's Dual One-Dimensional Analysis and Display program (DODAD).⁽²⁾ The major modifications included:

- Inclusion of the necessary commands for the time-of-flight interface circuits;
- Conversion to 500 channels of double-precision storage in quadrants 2 and 3 of the 4096-word memory, with memory protection of the fourth quadrant (which contained the overflow of the program; DODAD is written entirely in the first quadrant);
- Modification of the display routines to accommodate double-precision words from 63 counts to 8195 counts full-scale;
- Punchout with parity checking and error correction on the P-120 tape perforator, in Friden 8-level code, with automatic run-number incrementation and inclusion of the special characters required for reading by the paper-tape-to-card program TCS;
- Incorporation of a routine to calculate centroids of peaks in double precision including subtraction of background; and,
- Zero suppression in the decimal printout routines.

In addition, two independent utility programs were stored in the fourth quadrant, a test program for the tape perforator (written by E. N. Greenwell), and a Flexowriter-simulation program (originally written by R. E. Draper) to convert the paper-tape output into typewritten pages.

The performance of the new equipment amply justified the effort expended in making the changes. There was, however, one major instrumental problem which seriously threatened the run. A minor instability in the performance of the detector, which we have suspected for several years but never identified, became conspicuous when the pulse-shape discrimination of the detector abruptly disappeared. After a lengthy search, the cause was traced to some unstable parameter in the 58AVP photomultiplier, which could not be detected in any operating characteristic except the manner in which space charge in the collector region is formed and dissipated following a large pulse. Thus, it affected the pulse-shape discrimination which rejects photon-induced background without changing the pulse-height discrimination which determines the threshold of the detector. The problem was cured by replacing the faulty photomultiplier.

After rebuilding the detector, the overall stability of the time-of-flight system justified the performance of another experiment, following the completion of the scheduled transmission measurements. The additional experiment⁽³⁾ consisted of gathering further (probably conclusive) evidence for the nonexistence of a supposed level in the ^{10}B nucleus at 2.86 MeV. If such a level exists, it should produce a discrete neutron group in the time-of-flight spectrum of the $^9\text{Be}(\text{d},\text{n})^{10}\text{B}$ reaction, which we use for calibrating the time scale of our equipment. Accordingly, we spent several hours optimizing the time resolution and background

rejection of the apparatus, and removing all possible extraneous scattering material from the neighborhood of the beryllium target, and then accumulated the spectrum from a target 89 keV thick until the height of the ground-state peak reached 10,000 counts/channel. Four successive measurements were so nearly identical in time scale that they could be combined into a single measurement without degradation of the resolution. The peak/background ratio relative to the ground-state group was 650/1. We found no evidence for the 2.86 MeV level, and set an upper limit for the intensity of a transition to this level of one-third percent, relative to the two transitions which flank it in energy. A paper describing this work has been published.

Processing of the transmission data to obtain total cross sections followed the sequence devised earlier in the program. The punched paper tape was converted to punched cards (program TCS), and the cards used for entry into the primary data-reduction program (BIG NED), usually by way of a magnetic tape coded in NTRAN records for the UNIVAC 1107 computer. The output from BIG NED consists of total cross sections for a particular sample, using all of the data acquired under purportedly identical experimental conditions. A second program (LI'L NED) then serves to prepare combined results for several measurements taken under different conditions. BIG NED contains a routine for compensating the counting-rate-dependent shift in time zero. This procedure removes most of the resulting spurious structure in the cross sections above 11 MeV, but only

at the expense of introducing some covariance into the results. The covariance is enhanced when runs are combined, so both BIG NED and LI'L NED must use fully covariant calculations.

The amount of residual spurious structure which remains depends partly on the severity of the original instability, which varied with time, and partly on the slope and curvature of the cross section in the critical energy range, which varies with mass number. Since the total cross section is given by

$$\sigma_T = C \ln R,$$

where C depends on the thickness of the sample and $1/R$ is its transmission, the error in the total cross section caused by a small shift in time zero is given by

$$d \sigma_T = -C \frac{dR}{R},$$

and, hence, is proportional to the relative slope, dI/I , of the source spectrum. Since the relative slope is easily determined (it depends only on deuteron energy and effective resolution width), corrections for the residual error can be made by subtracting from the apparent cross section a suitable multiple of the relative slope.

The actual procedure used was to subtract estimated corrections from the data above 11.5 MeV, using the contaminant-subtraction routine in LI'L NED, and generate punched cards with the resulting adjusted cross sections. These cards were then entered into the short covariant-least-squares fitting program COVFT, which was originally written for the analysis of the deuterium cross section

near the di-neutron threshold,⁽⁴⁾ in order to determine the chi-square for fitting a second-degree polynomial to the cross section above 11.5 MeV. The behavior of chi-square near the correct value of slope-error contamination is quadratic, so that the correct amount of contamination to subtract can be determined from the uncorrected data and two estimated corrections. A correction determined in this manner was applied over the entire energy range to approximately 1/3 of the elements; for the others the amplitude of the spurious structure was already less than 1.5% and no additional correction was deemed necessary. The shape of the relative-slope error is so characteristic that this correction scheme works even for the light elements—such as nitrogen and carbon—in which there is genuine structure in the cross section at these energies.

A few miscellaneous errors were also removed during the final stages of data processing. Several samples, which had been prepared in powdered form, were found to exhibit the characteristic structure of the oxygen cross section. By comparing these cross sections to those of neighboring elements (or of other isotopes of the same element) as templates, it was easy to determine whether the actual contamination was due to elemental oxygen (as in ¹⁸⁶W sample) or to water vapor (which appeared as a fixed contaminant in Te, but increased with time in arsenic and rhenium), and to determine the magnitude of the contamination within a few tenths atom-percent. Comparison between our older measurements and those on new - and more carefully prepared - samples of lithium, sodium, and potassium revealed (as anticipated)

errors in the densities ascribed to the older samples. Similarly, our first measurements on carbon, sulfur, and calcium, all had errors in energy scale, which can be traced to the fact that our earliest work used the interval between the upswing and downswing bursts in the beam-pulsing system to calibrate the time scale. We then assumed that this interval was equal to one-half the period of the deflection oscillator, instead of calibrating against the full period of the oscillator as in all of our later work. The length of the two half-cycles frequently differs by as much as 1%. In these three cases we used our newer measurements to determine the correct average channel width in the older measurements by matching the pronounced fine structure in the respective cross sections, and repeated the original data reduction using the corrected channel width.

The remaining steps before publication of this work are relatively routine, but must be performed systematically because of the amount of data involved. Comparison of our results with the existing literature can be performed conveniently with LLNL NED, using a magnetic-tape copy of the Livermore library prepared by R. J. Howerton, supplemented with recent work and a few overlooked publications located by our own search of the literature. Tabulations of the actual measurements will be prepared as a BNW formal report, since they are too bulky to be published in the standard archive journals. We are exploring the feasibility of using the Benson-Lehner plotter to prepare graphs for publication. Following our custom in the past with our preliminary data, we

are submitting numerical copies of the results to the international cooperative neutron data compilation system, via the Sigma Center at Brookhaven. Copies are also being sent, on request, to the local compilation groups at the major AEC laboratories.

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