[redacted]

ORNL-1820
Progress
c. 5

AEC RESEARCH AND DEVELOPMENT REPORT

DECLASSIFIED

Classification Changed To:
By Authority Of: E. J. Murphy 8-4-61
By: S Beidel 6-13-62

PHYSICS DIVISION

SEMIANNUAL PROGRESS REPORT

FOR PERIOD ENDING SEPTEMBER 10, 1954

OAK RIDGE NATIONAL LABORATORY
OPERATED BY
CARBIDE AND CARBON CHEMICALS COMPANY
A DIVISION OF UNION CARBIDE AND CARBON CORPORATION

UCC

POST OFFICE BOX P
OAK RIDGE, TENNESSEE

[redacted]

[redacted]

[redacted]

ORNL-1820

This document consists of 24 pages.

Copy 5 of 190 copies. Series A.

Contract No. W-7405-eng-26

PHYSICS DIVISION

SEMIANNUAL PROGRESS REPORT

for Period Ending September 10, 1954

A. H. Snell, Director

Edited by
E. P. Blizard, Associate Director

DATE ISSUED

JAN 6 1955

OAK RIDGE NATIONAL LABORATORY
Operated by
CARBIDE AND CARBON CHEMICALS COMPANY
A Division of Union Carbide and Carbon Corporation
Post Office Box P
Oak Ridge, Tennessee

[redacted]

[redacted]

[redacted]

[redacted]

ORNL-1820
Progress

INTERNAL DISTRIBUTION

1. C. E. Center
2. Biology Library
3. Health Physics Library
4-6. Central Research Library
7. Reactor Experimental Engineering Library
8-13. Laboratory Records Department
14. Laboratory Records, ORNL R.C.
15. C. E. Larson
16. L. B. Emlet (K-25)
17. J. P. Murray (Y-12)
18. A. M. Weinberg
19. E. H. Taylor
20. E. D. Shipley
21. E. J. Murphy
22. F. C. VonderLage
23. A. H. Snell
24. J. A. Swartout
25. A. Hollaender
26. K. Z. Morgan
27. F. L. Culler
28. M. T. Kelley
29. E. M. King
30. E. O. Wollan
31. D. W. Cardwell
32. A. S. Householder
33. L. D. Roberts
34. R. B. Briggs
35. R. N. Lyon
36. W. C. Koehler
37. W. K. Ergen
38. E. P. Blizard
39. M. E. Rose
40. S. Bernstein
41. G. E. Boyd
42. Lewis Nelson
43. R. W. Stoughton
44. G. E. Winters
45. W. H. Jordan
46. E. C. Campbell
47. D. S. Billington
48. M. A. Bredig
49. R. S. Livingston
50. C. P. Keim
51. J. L. Meem
52. C. E. Clifford
53. G. H. Clewett
54. C. D. Susano
55. C. J. Borkowski
56. D. D. Cowen
57. P. M. Reyling
58. W. M. Good
59. G. C. Williams
60. E. P. Wigner (consultant)
61. John Dunning (consultant)
62. K. Lark-Horovitz (consultant)
63. J. L. Fowler
64. R. W. Johnson
65. J. R. McNally, Jr.
66. E. C. Smith
67. R. G. Cochran
68. C. L. Storrs
69. P. M. Steir
70. J. H. [illegible]iler
71. H. G. MacPherson (consultant)
72. A. D. Callihan
73. ORNL Document Reference Library (Y-12 Branch)
74. M. J. Skinner

EXTERNAL DISTRIBUTION

75. AF Plant Representative, Burbank
76. AF Plant Representative, Seattle
77. AF Plant Representative, Wood-Ridge
78. ANP Project Office, Fort Worth
79-86. Argonne National Laboratory
87. Armed Forces Special Weapons Project (Sandia)
88. Army Che[redacted]
89-91. Atomic Energy Commission, Washington
92. ~~Battelle Memorial Institute~~
93-96. Broo[redacted]er)

[redacted]

[redacted]

97. Bureau of Ships
98. Carbide and Carbon Chemicals Company (C-31 Plant)
99-100. Carbide and Carbon Chemicals Company (K-25 Plant)
101. Chicago Patent Group
102. Chief of Naval Research
103. Columbia University (Havens)
104. Commonwealth Edison Company
105. Department of the Navy – Op-362
106. Detroit Edison Company
107. Directorate of Research (WADC)
108-110. duPont Company, Augusta
111. Foster Wheeler Company
112-114. General Electric Company (ANPD)
115-118. General Electric Company, Richland
119-120. Goodyear Atomic Corporation
121. Hanford Operations Office
122. Iowa State College
123. Kirtland Air Force Base
124-127. Knolls Atomic Power Laboratory
128-129. Los Alamos Scientific Laboratory
130. Monsanto Chemical Company
131-133. Mound Laboratory
134. National Advisory Committee for Aeronautics, Cleveland
135. National Bureau of Standards
136. Naval Medical Research Institute
137-138. Naval Research Laboratory
139. New Brunswick Laboratory
140-141. New York Operations Office
142-144. North American Aviation, Inc.
145. Nuclear Development Associates, Inc.
146. Patent Branch, Washington
147-150. Phillips Petroleum Company
151. Pratt & Whitney Aircraft Division (Fox Project)
152. Rand Corporation
153. Sandia Corporation
154. USAF Headquarters
155. U. S. Naval Radiological Defense Laboratory
156. UCLA Medical Research Laboratory (Warren)
157-161. University of California Radiation Laboratory, Berkeley
162-164. University of California Radiation Laboratory, Livermore
165-166. University of Rochester
167-168. Vitro Corporation of America
169. Walter Kidde Nuclear Laboratories, Inc.
170-173. Westinghouse Electric Corporation
174. Yale University
175-189. Technical Information Service, Oak Ridge
190. Division of Research and Medicine, AEC, ORO

[redacted]

[redacted]

Reports previously issued in this series are as follows:

ORNL-51	March, April, and May 1948
ORNL-159	June, July, and August 1948
ORNL-228	September, October, and November 1948
ORNL-325	December, January, and February 1948-49
ORNL-365	Period Ending June 25, 1949
ORNL-480	Period Ending September 25, 1949
ORNL-576	Period Ending December 15, 1949
ORNL-693	Period Ending March 15, 1950
ORNL-781	Period Ending June 15, 1950
ORNL-864	Period Ending September 20, 1950
ORNL-939	Period Ending December 20, 1950
ORNL-1006	Period Ending March 20, 1951
ORNL-1107	Period Ending June 20, 1951
ORNL-1175	Period Ending September 20, 1951
ORNL-1216	Period Ending December 20, 1951
ORNL-1374	Period December 20, 1951 to June 20, 1952
ORNL-1421	Period Ending September 20, 1952
ORNL-1477	Period Ending December 20, 1952
ORNL-1630	Period Ending September 10, 1953
ORNL-1715	Period Ending March 10, 1954

Unclassified work for the period between November 1948 and March 10, 1954 is reported in the following reports: ORNL-325 supplement, ORNL-366, ORNL-481, ORNL-577, ORNL-694, ORNL-782, ORNL-865, ORNL-940, ORNL-1005, ORNL-1092, ORNL-1164, ORNL-1278, ORNL-1289, ORNL-1365, ORNL-1415, ORNL-1496, ORNL-1620, and ORNL-1705.

 [redacted]

[redacted]

CONTENTS

[redacted]

[redacted]

PHYSICS DIVISION SEMIANNUAL PROGRESS REPORT

SUMMARY

SHIELDING RESEARCH

Since the contribution of the scattered flux to the total measured flux in experiments at the Tower Shielding Facility must be known for a variety of conditions, much of the work of the Shielding Analysis Group has been directed toward calculations of neutron ground and air scattering. These calculations are being compared with similar calculations at other installations. In addition, a calculation of the fast-neutron leakage from the Bulk Shielding Reactor was performed, and agreement with leakage flux measurements made with the recoil proton spectrometer indicates that the reactor power density determinations are correct. A calculation by the Monte Carlo method of the number and energy of slant incident photons that penetrate or are reflected from a finite thick slab of infinite extent was also made.

At the Lid Tank Shielding Facility, the air-duct experimentation has been continued with several arrays of GE-ANP helical ducts. Other experiments at the LTSF included: measurements of a carbon neutron removal cross section as 0.750 barn for a continuous carbon medium, and a study which indicated little difference between the secondary gamma-ray production in lead and that in bismuth.

Some gamma-ray spectral and angular distribution measurements at the Bulk Shielding Facility were used in calculations of the dose to be expected in the crew compartment of a divided shield. The design of the shield presented by the 1950 Shielding Board was used, and the results were compared with values calculated by the Board. The total dose inside the crew compartment is somewhat lower than that calculated by the Board, although the direct beam is considerably higher. In another investigation at the BSF, an attempt was made to determine the amount of visible light given off by reactors, and it was estimated that a typical nuclear-powered aircraft would have a brightness equivalent to that of a 10-w incandescent light bulb. Other BSF experiments included: radiation measurements behind various thicknesses of graphite placed against the reactor, the determination of the thermal-neutron flux perturbation by gold foils in water, and the development of a method to determine the power of the ARE.

The power of the MTR-type reactor at the TSF was determined by a calorimetric method. The temperature rise of the water surrounding the reactor was measured while the reactor was being operated at a nominal power of 400 kw. The actual power was found to be 1.20 ± 0.01 w for a nominal power of 1 w. When the power measurements are combined with anticipated flux measurements, this experiment should provide a good, new value for the energy per fission in U^{235}. In another experiment it was indicated that the contribution of ground-scattered neutrons to the total scattered intensity measured at the maximum altitude (~ 195 ft) was 2 to 5%. A series of experiments with the use of a mockup of the GE-ANP R-1 shield design have been started.

CRITICAL EXPERIMENTS

Critical assemblies for two proposed aircraft reactors, the reflector-moderated reactor and the supercritical water reactor, have been studied. The RMR assembly consisted of two regions, a 15-in.-dia spherical core containing 11.067 kg of U^{235} and a surrounding beryllium reflector. The entire sphere was 41 in. in diameter. The SCWR assembly consisted of an aqueous solution of enriched UO_2F_2 contained in stainless steel tubes distributed in an organic liquid ($C_5H_4O_2$). The liquid simulated the nuclear properties of supercritical water and also served as a neutron reflector on the lateral surface of the cylindrical tube bundle. Criticality was reached with 11.48 kg of U^{235}.

Two basic reactor studies were carried out. In one experiment the interrelation of the uranium concentration and the liquid height (aqueous solution of uranium oxyfluoride, 93.2% enriched U^{235}) in spherical containers was determined for critical conditions. In another experiment the critical parameters of hexagonal latticed arrays of enriched U-Al slugs in water were investigated.

[redacted]

THEORETICAL PHYSICS

A systematic investigation of orbits in a high-energy fixed-frequency cyclotron is being carried out by use of the ORACLE. Previous calculations seem to indicate that a proton energy of at least 350 Mev is attainable, but it is hoped that with the present calculations this energy limit can be increased considerably. The complete calculations will also yield design specifications for the machine.

ACTIVATION RESONANCE INTEGRALS OF U^{238} AND Th^{232}

Resonance activation integrals of 278 ± 20 barns for U^{238} and 67.3 ± 4.7 barns for Th^{232} were found by comparing induced beta activities with Au^{197}, the resonance integral for the gold being taken as 1583 barns.

---

1. SHIELDING RESEARCH

E. P. Blizard

SHIELDING ANALYSIS

E. P. Blizard

F. H. Murray   C. D. Zerby

H. E. Stern[1]   J. E. Faulkner

Several calculations of scattered neutron radiation for various situations have been carried out in an attempt to interpret the experimental data at the Tower Shielding Facility, and these calculations have been compared with similar work at other installations. A calculation of the fast-neutron leakage from the Bulk Shielding Reactor has also been made. In addition, a Monte Carlo method has been used to determine the number and energy of photons that penetrate or are reflected from a finite thick slab of infinite extent.

Fast-Neutron Ground- and Air-Scattering Calculations

In an effort to understand the air-scattered neutron intensities which have been measured at the TSF, a number of calculations have been carried out with a variety of conditions and assumptions. The predominant difference between the current work and that which was done previously is that the interference of the ground is taken into account. The ground scattering is then calculated separately. In an attempt to obtain a better fit to the data, such aspects as anisotropy of scattering and slant penetration at the detector tank (crew shield) were taken into account in some of the work. Furthermore, measurements were made of the angular distribution of the radiation leaving the reactor shield, and these data were used in some of the calculations.

The importance of the multiply scattered neutrons is unknown; however, considerable effort has been made to extend the calculations beyond the singly scattered case.

The various ORNL calculations have been compared with some calculations at other installations, and a reference list is presented in Table 1.1. A more detailed comparison will be published separately, along with gamma calculations.[2]

Fast-Neutron Leakage of the Bulk Shielding Reactor

J. E. Faulkner

Because of the large degradation in energy per collision to be expected for very fast neutrons in the hydrogen-moderated BSR, it is possible to obtain a reasonably accurate estimate of fast flux ($E \sim 8$ Mev) even though only one or two collisions are considered. Such a calculation has been performed for comparison with leakage fluxes measured on the recoil proton spectrometer. The calculation including twice-scattered neutrons does not differ greatly from that including only virgin and singly collided neutrons, indicating that the former is probably a good estimate of the total. Agreement of this calculation with the spectrometer measurements indicates that the reactor power-density determinations are accurate; the results will be published.[3]

[1] Consolidated Vultee Aircraft Corporation.

[2] *Index of Some Fast-Neutron and Gamma-Ray Air- and Ground-Scattering Calculations*, ORNL CF-54-11-2 (to be published).

[3] J. E. Faulkner, *Fast Neutron Leakage of the Bulk Shielding Reactor*, ORNL CF-54-8-101 (to be published).

TABLE 1.1. FAST-NEUTRON GROUND- AND AIR-SCATTERING CALCULATIONS

Type of Scattering	Extent of Medium	Source[a]	Detector	Range of Parameters[b]	Remarks	Reference[c]
			Air Scattering			
Single, anisotropic	Air in presence of ground	Anisotropic (GTR,[d] assumed 3.25 Mev)	Isotropic, unshielded	$h = 12.5$ to 72.5 ft, $D = 30$ to 90 ft	First leg attenuation; clouded crystal ball	MRN-53
		1 (isotropic) $\cos \alpha$ $\cos^2 \alpha$ $\cos^3 \alpha$	Shielded, blacked out on five sides	$h/D = 0 \longrightarrow 2$	No air attenuation	CF-54-8-104, ORNL-1771
Single, isotropic	Infinite air	Isotropic (0.3, 1.0, 3.0 Mev)	Isotropic, unshielded	$D \times \Sigma_0 = 0 \longrightarrow 1$	First leg attenuation; clouded crystal ball	MRN-5
	Air in presence of ground	Isotropic	Isotropic, unshielded	$2h/D = 0.5 \longrightarrow \infty$	No air attenuation	CF-54-8-96, ORNL-1771
				$D = 64$ ft	Air attenuation	BAC D-14624
	Infinite air	Isotropic	Isotropic, unshielded	$D = 64$ ft	With and without air attenuation	BAC D-14625
Multiple, anisotropic	Infinite air	Isotropic	Isotropic, unshielded	$D = 64$ ft	Scattering is $1 + 0.7 \cos \theta$; with and without air attenuation	BAC D-14625
		Anisotropic	Any		No calculations, method only; includes air attenuations	ORNL-1771, CF-54-11-83
		Isotropic	Isotropic	$D/\lambda = 0 \longrightarrow 5$	No air attenuation	MRN-1
Multiple, isotropic	Infinite air	Isotropic	Isotropic	$D = 64$ ft	Air attenuation	BAC D-14625
				$\lambda = 130$ m, $D = 64$ ft	Results reported for first and second scatterings	ORNL-1771

TABLE 1.1 (continued)

Type of Scattering	Extent of Medium	Source[a]	Detector	Range of Parameters[b]	Remarks	Reference[c]
			Ground Scattering			
Single	Infinite concrete	Isotropic (3 Mev)	Isotropic	$D = 65$ ft, $h = 12\frac{1}{2}$ ft	No air attenuation; first leg attenuation in concrete; clouded crystal ball	MRN-29
	Infinite ground	Isotropic	Isotropic	$D = 64$ ft	Air and ground attenuation	BAC D-14624
		Any	Isotropic		Method only; also developed for albedo	ORNL-1771, CF-54-8-103
Multiple	Infinite ground	Isotropic	Isotropic	$D = 64$ ft	Air attenuation; based on experimental albedo of 0.12 half isotropic and half cosine reradiation	BAC D-14625
		Isotropic	Isotropic	$D/h = 0 \rightarrow 6$	No air attenuation; isotropic albedo	MRN-58
		Anisotropic (GTR)	Isotropic	$D/h = 0 \rightarrow 6$	No air attenuation; isotropic albedo	MRN-58

[a] Monoenergetic.

[b] h = source-detector height above the ground;
D = source-detector separation distance;
Σ_0 = macroscopic cross section;
λ = mean free path.

[c] ORNL, CF: Oak Ridge National Laboratory reports;
BAC: Boeing Airplane Co. reports;
MRN: Consolidated Vultee Aircraft Corporation reports.

[d] Ground Test Reactor, Consolidated Vultee Aircraft Corporation.

Penetration of Composite Slabs by Slant Incident Gamma Radiation

C. D. Zerby

A straightforward Monte Carlo method was used to determine the number and energy of photons that penetrate or are reflected from a finite thick slab of infinite extent. The Klein-Nishina differential cross-section formula was used to calculate the scattering behavior of the photons, while the absorption probabilities were determined from an empirical equation fit to published data. The incident radiation was considered to be parallel, monoenergetic gamma rays incident at an oblique angle with the normal. The slab was alternately considered to be polyethylene, lead, or a composite slab with a thick section of polyethylene followed by a thin section of lead. For the lead slabs (2.5, 5, 7.5, and 10 cm in thickness) the initial energy of the photons was 2.45 m_0c^2, incident at 0, 30, 60, 75, and 85 deg with respect to the normal. For the polyethylene slabs (3, 9, 15, and 30 in. in thickness) the initial energy of the photons was 2.45 m_0c^2, incident at 0 to 60 deg. For the composite slab the polyethylene was varied in thickness from 3 to 15 in., while the lead varied from 0 to ½ in. The incident photons on the composite slab had energies of 2 and 6 m_0c^2 at 0, 30, and 60 deg.

The results of these calculations were reported in a separate memorandum.[4]

LID TANK SHIELDING FACILITY

G. T. Chapman
J. M. Miller D. K. Trubey
J. B. Dee[5]

The air-duct experimentation and removal-cross-section measurements have been continued at the LTSF, and a comparison of the secondary gamma-ray production in lead and bismuth was made. In addition, several preliminary tests for a more extensive investigation of the reflector-moderated reactor and shield have been performed; these tests were reported previously.[6]

Duct Tests

J. M. Miller

The air-duct experimentation at the Lid Tank has continued with several arrays of GE-ANP helical ducts. The ducts were fabricated from flexible steel conduit (3-in.-dia) shaped around a 9-in. core. After removal of the core, the ducts were stiffened by means of Fiberglas wrapping. The projected length of each duct along the z axis was 46.5 in., and duct arrays were arranged so that there was 5 in. between the duct center lines.

Fast-neutron dose measurements at the end of an array of three ducts (5 in. from center to center) was a factor of approximately 1.4 greater than the dose at the end of a single duct. However, the measurements were made in a low-flux region, and the statistics of the experiment were considered to be poor. The thermal-neutron measurements were considerably higher at the end of the three-duct array. These indications of the interactions of adjacent ducts are of considerable importance to the air-cooled reactor projects. No adequate theory is available for prediction of this effect at present.

Thermal-neutron measurements beyond an array of 35 ducts (5 in. from center to center) show that the flux is increased by a factor of approximately 300 over the flux in plain water. It has not been determined whether this difference is due to the reduced density of the medium or to streaming in the ducts. Future measurements will be made with the duct array in a medium of iron Raschig rings and borated water to enhance the gamma attenuation. Lead will also be placed close to the array to determine the effect of additional gamma shielding near the duct region.

Some of these measurements were presented previously.[7]

Removal Cross Sections

Measurements of the removal cross section of carbon have been made in a continuous carbon medium obtained by dissolving sugar ($C_{12}H_{22}O_{11}$) in water. The solution (density = 1.312 ± 0.001 g/cc) was contained in a large tank that had a ⅛-in.-thick Inconel window on the source side. Since the medium had 95% as much hydrogen and

[4]C. D. Zerby, *Preliminary Report on the Penetration of Composite Slabs by Slant Incident Gamma Radiation*, ORNL CF-54-9-120 (Sept. 21, 1954).

[5]Pratt and Whitney Aircraft Division.

[6]J. B. Dee, D. K. Trubey, and W. Steyert, *ANP Quar. Prog. Rep. Sept. 10, 1954*, ORNL-1771, p 164.

[7]J. M. Miller, *ANP Quar. Prog. Rep. Sept. 10, 1954*, ORNL-1771, p 166.

oxygen as does plain water and since these elements were in the same ratio as in water, no geometric correction was made in calculating the effective neutron-removal cross section, σ_r. The average σ_r for the range of 90 to 140 cm from the source was 0.750 barn. This is to be compared with a value of 0.81 ± 0.05 barn from the LTSF measurement behind a slab of graphite that contained 51.3 g of carbon per square centimeter, which corresponds to the sugar-water solution at 145 cm.

A removal cross section for carbon has also been calculated by use of measurements behind 1-, 2-, and 3-ft thicknesses of graphite at the Bulk Shielding Facility. The resulting cross sections are 0.82, 0.84, and 0.80, respectively.

The latest values of all removal cross sections calculated from LTSF measurements are given in Table 1.2. Both of the above measurements were reported in detail in a current report.[8]

TABLE 1.2. EFFECTIVE NEUTRON-REMOVAL CROSS SECTIONS

Material	σ_r (barns)
Measured Values	
Al	1.31 ± 0.5
Be	1.07 ± 0.06
Bi	3.49 ± 0.35
C	0.81 ± 0.05
Cu	2.04 ± 0.11
Fe	1.98 ± 0.08
Ni	1.85 ± 0.10
Pb	3.53 ± 0.30
U	3.6 ± 0.4
W	2.51 ± 0.55
B_4C	4.3 ± 0.4 5.1 ± 0.4
B_2O_3	4.30 ± 0.41
C_2ClF_3	6.66 ± 0.8
C_7F_{16}	26.3 ± 0.8
$(CH_2)_n$	2.84 ± 0.11
$C_{30}H_{62}$	80.5 ± 5.2
LiF	2.43 ± 0.34
Derived Values	
B	0.97 ± 0.1 0.67 ± 0.25
Cl	1.2 ± 0.8
F	1.29 ± 0.06
H	1.00 ± 0.05
Li	1.01 ± 0.04*
O	0.99 ± 0.10

*Value obtained subsequent to the period covered by this report.

Secondary Gamma-Ray Study

D. K. Trubey

Gamma-ray dose measurements were made behind 6 in. of lead or bismuth placed in a tank of borated water (1.1 wt % boron) at various distances from the source plate. As had been anticipated from previous experiments, the dose in the ordinary water behind the tank decreased monotonically, as the distance (d) between the source and the metal increased, becoming essentially constant for $d \geq 47$ cm. The dose at $d = 47$ cm was therefore considered to be entirely from gammas originating at the source and was subtracted from the doses for $d < 47$ cm to give the secondary gamma production. The resulting plot of the secondary gamma doses had a slope characteristic of neutron curves, and little difference could be seen between the lead and bismuth secondary gamma-ray production. The data for this experiment were reported previously.[9]

BULK SHIELDING FACILITY

F. C. Maienschein   R. G. Cochran
G. M. Estabrook   K. M. Henry
J. D. Flynn   E. B. Johnson
M. P. Haydon[10]   T. A. Love
R. W. Peelle

Calculations of the gamma-ray dose in a standard, divided shield design have been made by using

[8] D. K. Trubey, *ANP Quar. Prog. Rep. Sept. 10, 1954*, ORNL-1771, p 164; R. G. Cochran *et al.*, *ANP Quar. Prog. Rep. Sept. 10, 1954*, ORNL-1771, p 168.

[9] D. K. Trubey, *ANP Quar. Prog. Rep. June 10, 1954*, ORNL-1729, p 118.

[10] Part-time employee.

earlier gamma-ray spectral and angular distribution measurements around the mockup of a reactor shield. Other investigations have included: a quantitative measurement of the light to be given off from a nuclear-powered airplane, radiation dose and flux measurements beyond various thicknesses of graphite placed against the BSR, a determination of the perturbation of the neutron flux by the presence of gold foils in water, and the development of a method for the power determination of the ARE.

Gamma-Ray Air-Scattering Calculations

F. C. Maienschein

The introduction and gradual acceptance of a divided shield for nuclear-powered aircraft have increased the difficulty of determining the radiation dose received by the aircraft crew, particularly the dose from air-scattered radiation. A series of calculations, in which experimental results from the BSF were used, have been made for determining the gamma-ray dose in the crew compartment of the divided shield presented by the 1950 Shielding Board,[11] and the results of these calculations have been compared with values calculated by the Shielding Board. The experimental data used in the calculations included measurements of the gamma-ray spectra and angular distributions at several positions along the shield boundary of a mockup of the reactor shield section of the ANP-53 divided shield.

The gamma-ray flux reaching the outside edge of the crew shield was first calculated. These calculations were then extended to determine the dose inside the crew shield. For this calculation the ANP-53 crew shield was replaced by a simple cylinder of roughly the same dimensions. In addition, a calculation was made of the direct (unscattered) gamma radiation reaching the crew position through the rear of the crew shield.

Results of the air-scattering calculation indicate that although the contribution of the radiation escaping around the edge of the reactor shadow shield to the flux at the outside of the crew shield is extremely small compared with the total flux, the dose due to the radiation from the edge of the shadow shield amounts to over one-third the total dose inside the crew compartment.

The total dose inside the crew compartment from air-scattered gamma rays (see Table 1.3) is somewhat lower than that calculated by the 1950 Shielding Board;[11] however, the agreement is good if it is considered that for the Shielding Board calculations it was assumed that all the gamma rays were of 3-Mev energy and that all were radially emitted from the reactor shield.

The direct (unscattered) beam dose is considerably higher than that calculated by the Shielding Board. This is *attributable primarily to the gamma* radiation that escapes around the edges of the shadow shield – a factor which was unforeseen at the time of the Shielding Board calculation.

[11] *Report of the Shielding Board for the Aircraft Nuclear Propulsion Program,* ANP-53 (Oct. 16, 1950).

TABLE 1.3. GAMMA-RAY FLUX AND DOSE AT THE CREW COMPARTMENT

Method	Flux Outside Crew Compartment, ϕ/P (photons/sec/w)	Dose Inside Crew Compartment, D (r/hr/w)
Radial scattered	4.5	0.36×10^{-9}
Skew scattered	0.016	0.19×10^{-9}
Direct	16.0	130×10^{-9}
ANP-53 scattered*		1.2×10^{-9}
ANP-53 direct*		1.2×10^{-9}
Skyshine**		1.14×10^{-9}

*No correction made for differences in reactor leakage.

**H. E. Hungerford, *The Skyshine Experiments at the Bulk Shielding Facility,* ORNL-1611 (July 14, 1954).

The details of the experiments and the calculations have been published.[12,13]

Reactor Air Glow

F. C. Maienschein R. W. Peelle
R. G. Cochran K. M. Henry
T. A. Love

The effective production of visible light in air by reactor radiation has been experimentally measured in the BSR. The detector used was a photomultiplier with a spectral response equivalent to that of the average human eye, and the light standard was an incandescent light bulb. The light production expressed as watts of visible light per watt of gamma radiation dissipated in air was equal to about 6×10^{-5}. This would indicate that a typical nuclear-powered aircraft (dose external to reactor shield = 1.4×10^6 r/hr) would have a brightness equivalent to that of a 10-w incandescent light bulb.[14]

Reactor Radiations Through Slabs of Graphite

R. G. Cochran J. D. Flynn
G. M. Estabrook K. M. Henry
E. B. Johnson

Fast-neutron dose, gamma-ray dose, and thermal-neutron flux measurements were made beyond 1-, 2-, and 3-ft thicknesses of graphite which were placed against the BSR. The spectrum of fast neutrons through the 1-ft thickness was also measured in the 1.3- to 10-Mev energy range.

Calculations with the use of these data corroborated the neutron removal cross section value for carbon calculated from LTSF data (cf. "Removal Cross Sections," this report). The experiment also resolved uncertainties concerning the degree of thermalization of neutrons in graphite. These data were of particular interest in the design of a reactor that will be used for therapeutic purposes at Brookhaven.

Details of the experiment were reported previously.[15]

Thermal-Neutron Flux Perturbation by Gold Foils in Water

E. B. Johnson

It is well known that the flux is perturbed when a detector that is not infinitely thin is introduced into a neutron flux. The magnitude of the perturbation depends upon the absorption cross section of the detector and its physical dimensions and upon the properties of the medium into which it is introduced. It is customary at the BSF to use the flux of the ORNL Standard Graphite Pile[16] as the reference flux for the calibration of foils and counters. However, when a foil is used for thermal-neutron flux measurements in water, the flux perturbation caused by the presence of the foil is different from that obtained when the foil is in graphite, since the slowing-down properties of water differ from those of graphite. Klema and Ritchie[17] arrived at a semiempirical correction for this effect for the 25-cm^2, 5-mil-thick indium foils normally used for flux measurements in water. The presence of these foils results in a 22% depression of the flux from its unperturbed value. Frequently it would have been desirable to use gold foils for flux determinations in water, but no value for the flux depression has been available. Therefore an experiment has been completed at the BSF that was designed for measuring this effect.

Thermal-neutron flux measurements were made with both the indium and the gold (1-cm^2, 5-mil-thick) foils at five positions along the north-south center line of the reactor. The perturbation factor of 1.22 was applied, as usual, to the indium foil data, but no such correction was made on the gold foil data. The results are shown in Fig. 1.1. If it is assumed that a probable error of ±2% is associated with each measurement, there is no discernible difference between the thermal-neutron flux measurements made with indium foils, using the perturbation correction, and those with gold foils, using no correction. It is recognized that the presence of any absorbing material must result in a local depression of the flux; therefore, it seems either that this effect is within the probable errors of measurement for small gold foils in a

[12] F. C. Maienschein, *ANP Quar. Prog. Rep. June 10, 1954*, ORNL-1729, p 121.

[13] F. C. Maienschein, F. T. Bly, and T. A. Love, *Gamma Radiation in a Divided Aircraft Shield*, ORNL-1714 (Sept. 14, 1954).

[14] See, also, F. C. Maienschein *et al.*, *Measurements of Reactor-Induced Air Glow*, ORNL CF-54-9-1 (to be issued).

[15] R. G. Cochran *et al.*, *Reactor Radiations Through Slabs of Graphite*, ORNL CF-54-7-105 (July 30, 1954).

[16] E. D. Klema, R. H. Ritchie, and G. McCammon, *Recalibration of the X-10 Standard Graphite Pile*, ORNL-1398 (Oct. 17, 1952).

[17] E. D. Klema and R. H. Ritchie, *Preliminary Results on the Determination of Thermal Neutron Flux in Water*, ORNL CF-51-4-103 (April 24, 1951).

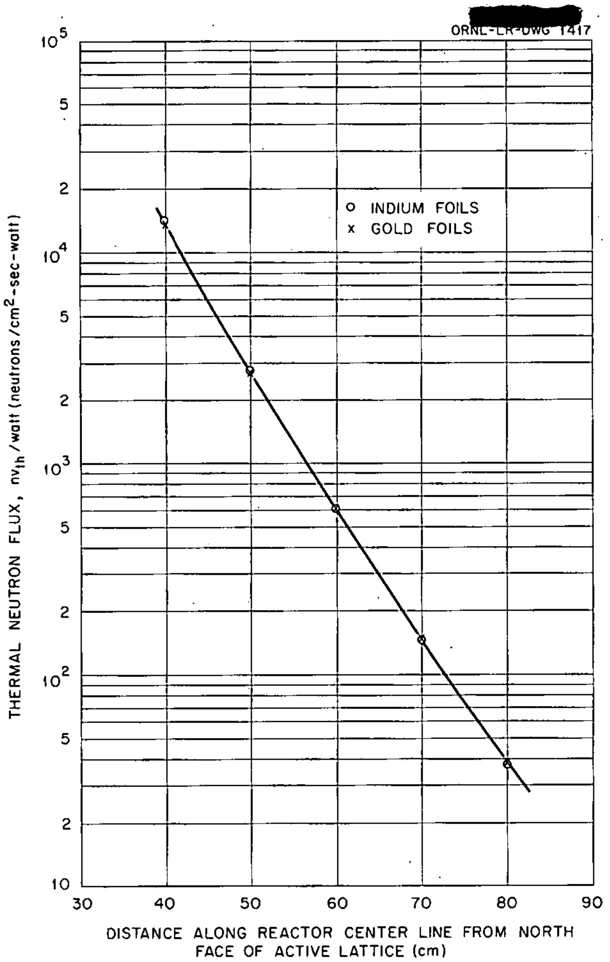

Fig. 1.1. Comparison of Indium and Gold Foil Measurements in Water Surrounding BSF Reactor.

water medium or that the perturbation correction used for the indium foils is insufficient. The latter appears to be more probable. These results substantiate those obtained by Uthe[18] from a similar experiment performed in the water tank at the thermal column of the ORNL Graphite Reactor.

Fuel Activation Method for Power Determination of the ARE

E. B. Johnson

One method for determining the power of the ARE will be a comparison of the activation of fuel samples. The activation of a sample which has been exposed in the known flux of the BSR will be compared with the activation of a fuel sample which will be withdrawn from the ARE after it has been operated at a nominal power level of approximately 1 w for 1 hr. The operating power of the ARE can be determined from decay curves of the two samples. For the power comparison it will be necessary to know the uranium content of the sample and of the ARE, as well as the energy per fission.[19]

TOWER SHIELDING FACILITY

C. E. Clifford

T. V. Blosser — M. F. Valerino[21]
L. B. Holland — J. Van Hoomissen[22]
D. L. Gilliland[20] — J. L. Hull

F. N. Watson

The experimental program at the TSF is well under way, and much of the work has been reported in recent ANP quarterly progress reports.[23,24] The power of the TSR was determined by a calorimetric method. Preliminary experiments to determine the contribution of ground- and air-scattered neutrons to the total flux were performed, and the experimentation on the GE-ANP R-1 divided shield mockup has begun.

Permission was granted by the Advisory Committee on Reactor Safeguards to operate the TSR at a power of 500 kw.

Calorimetric Power Determination of the Tower Shielding Reactor

L. B. Holland — J. L. Hull

The shielding measurements at the TSF must be normalized in terms of reactor power, requiring a new measurement with a reasonable degree of accuracy for each reactor loading. A procedure has been developed and tested so that the power can be determined from calorimetric measurements.[25]

[18]P. M. Uthe, private communication.

[19]For details of this method, see E. B. Johnson, *Fuel Activation Method for Power Determination of the ARE*, ORNL CF-54-7-11 (July 31, 1954).

[20]GE-ANP.

[21]NACA.

[22]Boeing Airplane Co.

[23]*ANP Quar. Prog. Rep. June 10, 1954*, ORNL-1729, p 136.

[24]T. V. Blosser *et al.*, *ANP Quar. Prog. Rep. Sept. 10, 1954*, ORNL-1771, p 175.

[25]See, also, C. E. Clifford *et al.*, *Calorimetric Power Determination of the Tower Shielding Reactor*, ORNL CF-54-8-105 (to be issued).

The TSR resembles the BSR, the fuel elements for both reactors being slightly modified MTR elements. Eighteen U-Al fuel plates (clad with 2S aluminum) in each element contain a total of 140 g of U^{235}. The aluminum-to-water volume ratio for the reactor proper is about 0.7. The elements are held in the reactor assembly by an aluminum grid plate which will accommodate 30 elements in a 5 by 6 array. An extension which can be attached to the grid will allow a 5 by 7 array. The entire reactor assembly is immersed in a large volume of water in a movable 12-ft-dia tank (essentially a cylinder with a hemispherical bottom) as shown in Fig. 1.2. The reactor is operated at a maximum power of 500 kw and is cooled by means of a convective water flow. For this experiment it was

UNCLASSIFIED
DWG. 2-01-056-0-T2

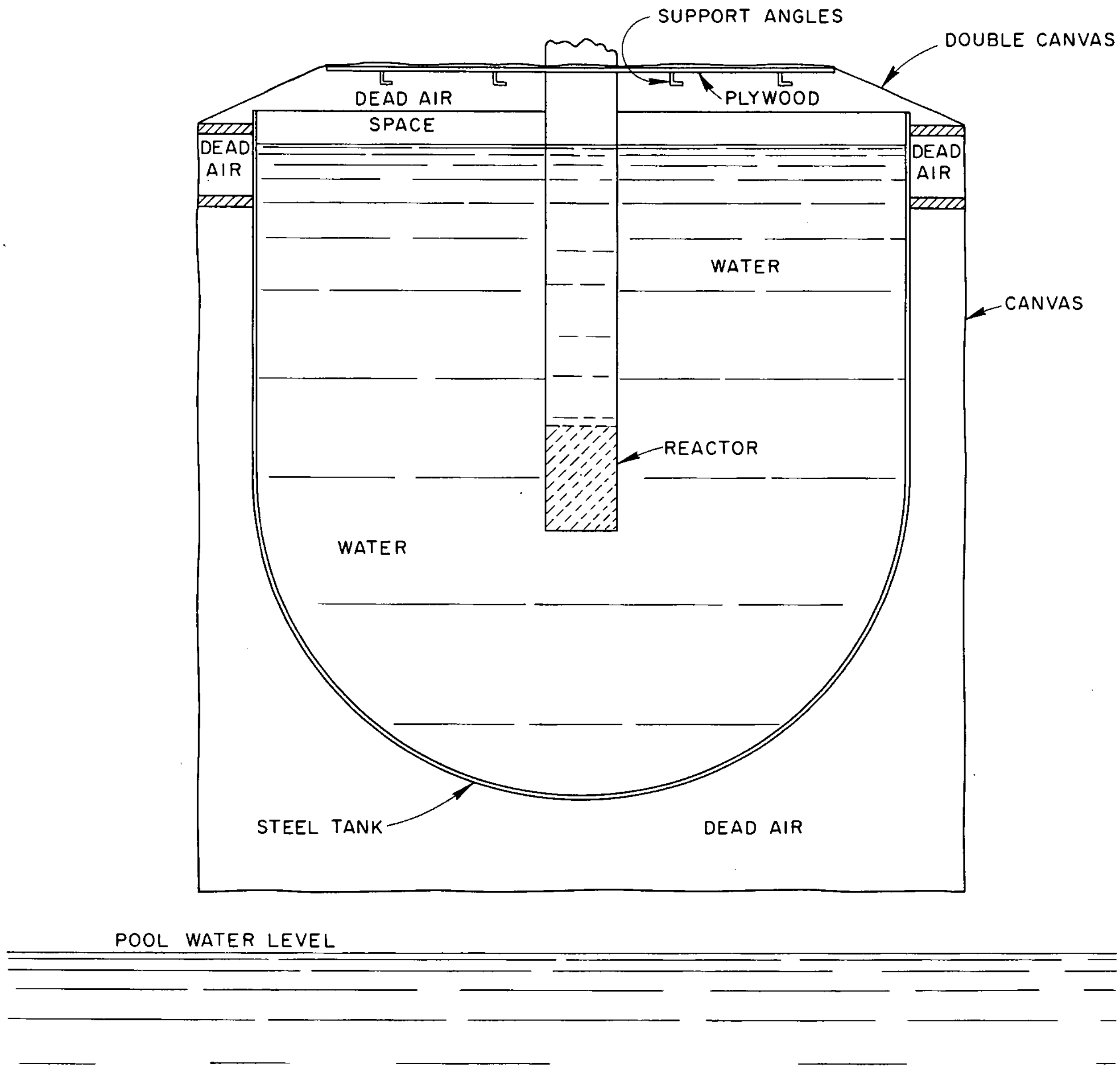

Fig. 1.2. Diagram of TSR Tank in Pool Showing Insulation Around Tank.

loaded with a 5 by 7 element lattice configuration as shown in Fig. 1.3.

The power of the reactor is equivalent to the rate of heat input:

$$P = \frac{dQ}{dt} = k \frac{dT}{dt} \quad ,$$

where k is the product of the heat capacity of reactor and tank and the heat equivalent factor. The power can then be determined by measuring k and dT/dt if the reactor is run at a constant power. For this determination the TSR was run at a nominal power of 400 kw.

Measurements of the tank were used to calculate the actual mass of water and steel in the determination of a value for k. As a check on the mass of water, the dilution of a known mass of chromate in the water was determined, and the two methods agreed within 0.7%. The aluminum and uranium in the reactor core were neglected, and the value of k for the water and steel was calculated to be 3.878×10^4 whr/°C.

The value of dT/dt was determined by means of thermocouple measurements in the water surrounding the reactor. The reactor was placed nearly in the center of the reactor tank, which was suspended in the drained handling pool. For best insulation, sections of plywood were placed 18 in. above the water on the reactor support frame. A double layer of canvas was then placed on the plywood and draped around the tank. This arrangement created a dead air space above the water in the tank as well as around the tank.

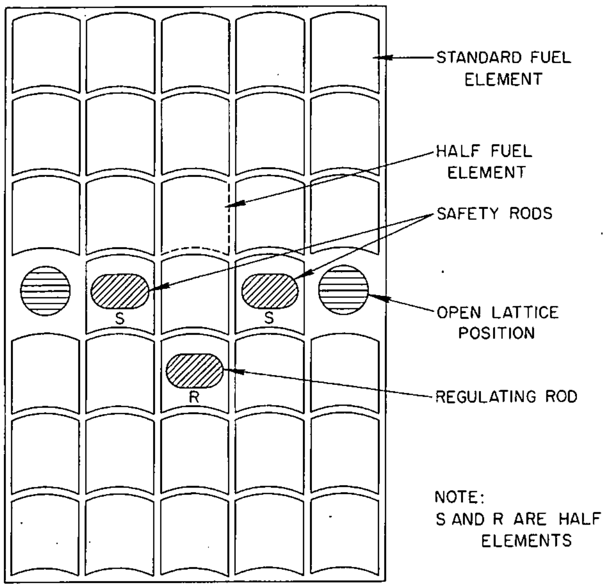

Fig. 1.3. Tower Shielding Reactor Loading No. 2.

Four electric mixers were mounted within the tank to give what appeared to be the best mixing of the water, with a minimum amount of power input to the water from the agitation.

Twelve iron-constantan thermocouples were mounted on a cross constructed of aluminum. The assembly was placed in the tank so that measurements were made along a vertical line that was midway between the reactor and the tank wall and along a horizontal line that coincided with the reactor center line. There were no variations in the readings when the thermocouple assembly was repositioned for duplicate runs.

Before the reactor was started, ice was added to the water in the tank in order to lower the water temperature below the ambient air temperature. After the ice had melted, thermocouple readings were taken for 1 hr to determine the rate of heat leakage into the water. The readings were continued while the reactor was raised to power, during reactor operation, and for a period after the reactor was shut down. The reactor was run until the temperature in the tank was as far above ambient air temperature as it was below ambient before the run. The readings were then continued for 1 hr after shutdown. The temperature was plotted in terms of the thermocouple readings vs time (Fig. 1.4).

Referring to Fig. 1.4, the rate of temperature change (dT/dt) of the system was calculated as follows:

$$\frac{P}{k} = \frac{dT}{dt}(R) - \frac{dT}{dt}(M) \pm \frac{dT}{dt}(\text{off}) + \frac{dT}{dt}(\beta,\gamma) \quad ,$$

where

$dT/dt\,(R)$ = rate of temperature change of system with reactor on,

$dT/dt\,(M)$ = rate resulting from water agitation by mixers,

dT/dt (off) = rate resulting from heat leakage into or out of the system from surroundings,

$dT/dt\ (\beta,\gamma)$ = rate resulting from energy production due to long-life fission products not yet in equilibrium at the time of measurement.

The rate of heat leakage into and out of the tank over the temperature range used was self-compensating, and the maximum was 0.03°C/hr. Since the effect on the slope was so small, dT/dt (off) was neglected in the final calculation. The other components of dT/dt were calculated as:

$$dT/dt\ (R) = 12.20°\text{C/hr},$$

$$dT/dt\ (M) = 0.02°\text{C/hr},$$

$$dT/dt\ (\beta,\gamma) = 0.15°\text{C/hr}.$$

Therefore

$$P = k\frac{dT}{dt} = (3.878 \times 10^4\ \text{whr/°C})[(12.20 + 0.15 - 0.02)\ \text{°C/hr}] = 478.2\ \text{kw} \quad .$$

As described above, the error in determining $k(H_2O)$ was ±0.7%. Since the error in $dT/dt\ (R)$ was extremely small (as shown by a least-square fit to a straight line) and the other components of dT/dt were only 1.0% (approx) of the total, a value of ±0.5% was arbitrarily assumed for all errors other than $k(H_2O)$. The error, then, was

$$\pm\sqrt{(0.7)^2 + (0.5)^2} = \pm 0.86\% \quad .$$

Thus the actual power for a nominal power of 1 w was 1.20 ± 0.01 w.

A power calibration run was also made for a 5 by 6 element reactor loading, but the data has not been completely analyzed. Preliminary data indicate that the power level was 1.02 w for a nominal power of 1 w.

Fast-Neutron Ground- and Air-Scattering Measurements

T. V. Blosser — J. Van Hoomissen
D. L. Gilliland — F. N. Watson

Measurements of ground- and air-scattered fast neutrons were made at the TSF as a function of the reactor-detector height. The configuration in which the measurements were made resembled a portion of an aircraft divided shield. The detector tank (essentially a 5-ft cube of water representing a shielded crew compartment forward of the reactor) was located a horizontal distance of 64 ft from the 12-ft-dia reactor tank, and the reactor was positioned in the reactor tank at an azimuthal angle of 330 deg from the common center line of both tanks. The reactor-detector altitudes were varied between 0 and 195 ft, and the distribution of the thermalized neutrons in the detector tank was measured with a BF_3 counter along a line normal to, and in the vicinity of, the right side of the detector tank. Preliminary analysis of the data indicates that the ground-scattered contribution at the maximum height was 2 to 5% of the total scattered neutrons. The details of this experiment have been published.[26]

GE-ANP R-1 Divided Shield Mockup Tests

T. V. Blosser — M. F. Valerino
D. L. Gilliland — J. Van Hoomissen

The test program on the GE-ANP R-1 divided shield mockup began with measurements around the reactor shield section in the TSF handling pool. In general, these measurements agreed with similar measurements made at the BSF.[27] The tests with the mockup suspended in air have begun with the reactor shield section located a horizontal distance of 64 ft from the TSF detector tank. A 5-in. thickness of lead (five 1-in. slabs) was placed in the tank 1 ft from the rear (reactor side). The reactor-detector altitude was 195 ft. Thermal-neutron measurements were made with a BF_3 counter in the detector tank, and relaxation lengths were determined as follows:

Near rear of tank (between lead and rear face)	~5.0 cm
Between lead and center of tank	~5.0 cm
Near right, left, bottom, and top sides of tank	4.2 cm for the first 25 cm
Near front of tank	3.5 cm (av)

Further measurements in the detector tank with the 5 in. of lead remaining in position will include: (1) fast-neutron measurements with a recoil proton

[26] C. E. Clifford *et al.*, *Preliminary Study of Fast Neutron Ground and Air Scattering at the Tower Shielding Facility*, ORNL CF-54-8-95 (Aug. 23, 1954).

[27] H. E. Hungerford, *Bulk Shielding Facility Tests on the GE-ANP R-1 Divided Shield Mockup*, ORNL CF-54-8-94 (to be issued).

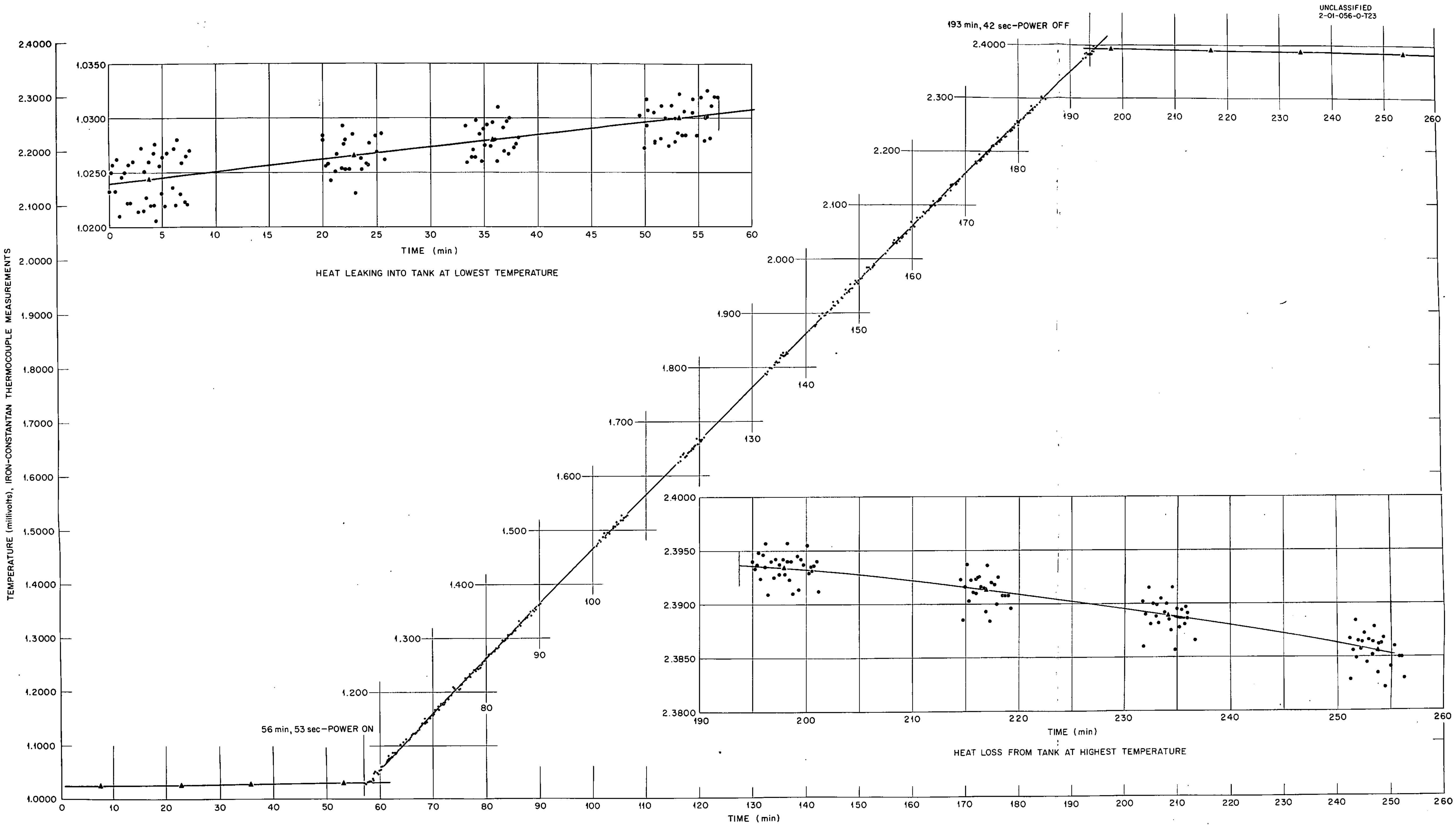

Fig. 1.4. Temperature Rise in Reactor Tank vs Time During TSR Power Calibration; TSR Loading No. 2.

dosimeter; (2) gamma-ray measurements in both plain and borated water; and (3) gamma-ray measurements in borated water with 0.2, 0.4, and 0.7 in. of lead shielding the detector by means of the lead absorber wheel (leadometer) described previously.[28] The effect of removing the slabs of lead from the borated water one at a time will also be determined.

[28]T. V. Blosser, *Phys. Semiann. Prog. Rep. March 10, 1954*, ORNL-1715, p 10.

Fast-neutron and gamma-ray dose rates and the thermal-neutron flux will also be measured with counters suspended in air at various points from the reactor shield. The detector tank will then be replaced by the crew compartment mockup for the final determination of the dose rate the pilot of an airplane could receive.

---

2. CRITICAL EXPERIMENTS

A. D. Callihan

AIRCRAFT REACTOR STUDIES

Reflector-Moderated Reactor

D. Scott[1]	B. L. Greenstreet[1]
J. S. Crudele[2]	J. W. Noaks[2]
R. M. Spencer[3]	J. J. Lynn[4]

It was previously reported[5] that the first step of the present RMR critical experiment program would be the construction of a small two-region reflector-moderated reactor to provide experimental data on a system of simple geometry and materials for use in checking the present ANP calculational methods. The assembly, which has now been constructed, is approximately spherical and has a 41-in. OD. It has a center, or core region (containing 11.067 kg of U^{235}), about 15 in. in diameter surrounded by a beryllium reflector. The core consists of alternate sheets of enriched uranium metal (4 mils thick) and Teflon, $(CF_2)_n$, so that it is possible to vary the uranium loading, within the specified dimensions, in order to make the system critical. The Teflon and the uranium have been cut to various shapes necessary to form the subassemblies required to conveniently construct the core.[6]

Measurements of the neutron flux were made in a vertical direction along a radius by using indium foils (with and without cadmium covers). A relative power, or fission-rate, distribution was made through the core along the same radius by observing the activity of the fission fragments retained in aluminum foils placed in contact with uranium foils. The fission rates were measured on opposite sides of three uranium foils near the reflector, and in each case the rate was higher on the side toward the beryllium. The details of the constructed assembly and the results of all the measurements are presented in another report.[7]

Measurements will be made of the reactivity coefficients, as a function of the radius, of a few metals such as Inconel, nickel, and cadmium. Upon completion of the experiments on this assembly, it is planned to construct a larger reactor of the same shape that will consist of three regions – the beryllium island and reflector separated by the fuel annulus. The diameters of the beryllium island and the fuel annulus will be about 10.4 and 20 in., respectively. Initially, the assembly will have no structural materials such as Inconel core shells etc. The purpose of this assembly will be to provide a further check of the ANP calculational methods.

[1]ARE Division.

[2]Pratt and Whitney Aircraft Division.

[3]USAF.

[4]Physics Division.

[5]D. Scott and B. L. Greenstreet, *Phys. Semiann. Prog. Rep. Mar. 10, 1954*, ORNL-1715, p 18.

[6]A more complete description is given by D. Scott and B. L. Greenstreet, *Reflector-Moderated Critical Assembly Experimental Program*, ORNL CF-54-4-53 (April 8, 1954).

Supercritical Water Reactor

J. S. Crudele   J. W. Noaks

Although the experiments on the preliminary critical assembly of the Supercritical Water Reactor (SCWR) have been interrupted in order to investigate the Reflector-Moderated Reactor, it is appropriate to report some of the results obtained. The

[7]D. Scott *et al.*, *ANP Quar. Prog. Rep. Sept. 10, 1954*, ORNL-1771, p 44.

assembly has been described previously[8] as consisting of an aqueous solution of enriched UO_2F_2 contained in stainless steel tubes which are distributed in an organic liquid in a pattern designed to give a uniform radial thermal-neutron flux. The liquid, furfural ($C_5H_4O_2$), which also serves as a neutron reflector on the lateral surface of the cylindrical tube bundle, has a hydrogen density approximately that of water under the temperature and pressure conditions designed into the SCWR. It simulates the nuclear properties of supercritical water as well as may be determined without further knowledge of the effect of binding energies on diffusion and slowing-down lengths. Stainless steel was inserted in the fuel tubes to represent the reactor structure.

The loading of the critical assembly was calculated[9] by use of a four-group diffusion-theory method to be 15.34 kg of U^{235} and 261 kg of stainless steel in a 76.2-cm equilateral right cylinder with a side reflector 6.5 cm thick.

In the experimental study, a quantity of UO_2F_2 aqueous solution containing uranium enriched to 93.14% in U^{235} was distributed among the tubes at a U^{235} concentration of 0.505 g/cc. The number of tubes required for criticality was measured as a function of the furfural height as increasing quantities of stainless steel were inserted and, subsequently, as the solution was diluted. In this manner a stepwise approach was made to a loading of uniform concentration which would be critical at the designed linear dimensions of both the fuel solution and the furfural and which would contain the prescribed mass of stainless steel. These design conditions have not been achieved because the last dilution was overestimated, and a large part of the solution is at a concentration somewhat lower than that required for criticality, a deficiency compensated for by locating about 50 tubes of higher fuel concentration in one peripheral section. The configuration is described in Table 2.1.

Some neutron flux measurements were made by using indium foils (with and without cadmium covers). The data were obtained along a radius 36.7 cm from the bottom of the core and along a longitudinal traverse 2.4 cm from the cylinder axis. Although the results are preliminary, it is believed that the apparent nonuniformity of the radial thermal flux is probably greater than the experimental uncertainty. A few measurements show that the radial importance of U^{235} also decreases with distance from the center. These measurements are presented in another report.[10]

In one experiment for the evaluation of the furfural reflector, annular sheets of aluminum were inserted adjacent to the wall of the reactor tank, thereby reducing the reflector thickness from 10.0 to 3.6 cm. The effect of the aluminum, indicated by the critical height of the furfural, was a slight increase in the reactivity.

TABLE 2.1. DATA FOR SCWR CRITICAL EXPERIMENT

	Experimental	Design
Number of tubes*	215	215
Height of UO_2F_2 solution, cm	71.2	76.2
Height of furfural, cm	69.5	76.2
Mass of U^{235}, kg**	11.48	15.34
Thickness of reflector, cm	10.0	6.5
Mass of stainless steel, kg	220.4	261

*Expressed as equivalent number of tubes 1 in. in diameter; the array contains 193 which are 1 in. in diameter, 8 which are ¾ in., and 25 which are ½ in.

**Of this loading, 86.8% was in a solution having a U^{235} concentration of 0.196 g/cc, and 13.2% was in a solution having a U^{235} concentration of 0.505 g/cc.

BASIC REACTOR STUDIES

UO_2F_2 Critical Experiments

J. K. Fox J. T. Thomas
E. R. Rohrer

Experiments have been performed to determine the interrelation of uranium concentration and liquid height in spherical containers at critical conditions. Aqueous solutions of uranium oxyfluoride (93.2% enriched U^{235}) of various concentrations were made critical in a partly filled sphere 12.6 in. in diameter. The capacity of the sphere is 17.02 liters, and it was completely surrounded with a light-water reflector. Table 2.2 and Fig. 2.1 summarize the critical conditions

[8] E. L. Zimmerman *et al.*, *Phys. Semiann. Prog. Rep. Mar. 10, 1954*, ORNL-1715, p 18.

[9] Private communication from G. Chase, Fox Project, Pratt and Whitney Aircraft Division.

[10] D. Scott *et al.*, *ANP Quar. Prog. Rep. Sept. 10, 1954*, ORNL-1771, p 44.

[redacted]

TABLE 2.2. CRITICAL CONDITIONS FOR AQUEOUS SOLUTIONS OF UO_2F_2 IN A PARTLY FILLED SPHERE

U^{235} Concentration (g/liter)	$H:U^{235}$ Atomic Ratio	Void (liters)	Void Fraction	U^{235} Critical Mass (kg)
50.3	515.0	0.00	0.00	0.856
52.1	496.5	0.33	0.02	0.870
53.1	487.6	0.50	0.03	0.877
125.4	203.1	5.52	0.32	1.444
177.4	143.1	6.46	0.38	1.873
345.3	71.5	7.47	0.44	3.301
507.8	47.0	7.67	0.45	4.772
125.2*	203.5	0.00	0.00	2.129

*Unreflected.

measured, including the concentration and mass required in the absence of the reflector and with no void present.

Critical Conditions of Uranium-Aluminum Slug Lattices in Water

J. K. Fox — C. Cross
L. W. Gilley — V. G. Harness

The investigation of the critical parameters of hexagonal latticed arrays of enriched U-Al slugs in water has continued.[11] The number of slugs in a critical array 12 in. (one slug) high was formerly determined to be 134. Arrays two slugs (24 in.) high have now been built with the alloy enclosed in a snugly fitting aluminum tubing having 0.03-in.-thick walls and with one fuel space near the center of the lattice left empty. The spacing for maximum reactivity has been found to be $1\frac{1}{16}$ in. edge-to-edge, and 181 ± 1 is the minimum number of slugs (containing a total of 3.99 kg of U^{235}) required for criticality. The removal of the aluminum tubing reduced the critical mass about 10%. The reactivities of fuel elements in various positions in the lattice and of two proposed poison control rods have been obtained from reactor period measurement.

[11] J. K. Fox and J. H. Marable, *Phys. Semiann. Prog. Rep. Mar. 10, 1954*, ORNL-1715, p 13.

[redacted]

ORNL-LR-DWG 4358

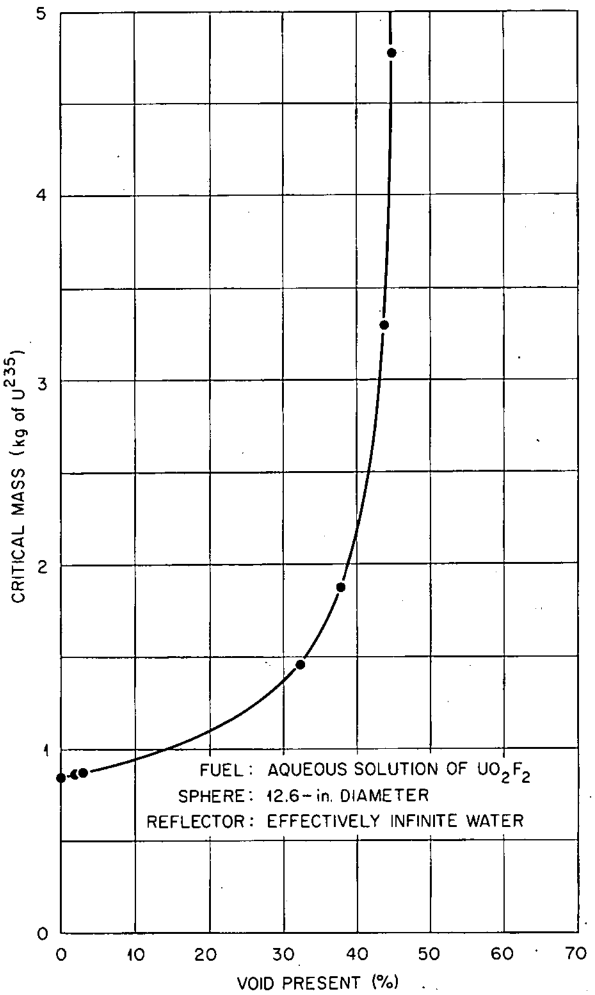

Fig. 2.1. Critical Mass of a Partly Filled Spherical Container of U^{235} Solution, Completely Water Reflected.

An additional experiment[12] has been done with alloy slugs, 8 in. long by 1.35 in. in diameter, each of which contains 38 g of U^{235}. It has been shown that a closely packed hexagonal array 16 in. high, with slug axes vertical, requires more than 1000 slugs for criticality.

[12] A. D. Callihan *et al.*, *Critical Mass Studies, Part VI*, Y-801 (Aug. 8, 1951).

[redacted]

[redacted]

3. THEORETICAL PHYSICS

ORBITS IN THE THOMAS CYCLOTRON[1]

D. S. Falk[2] T. A. Welton

A systematic investigation, by the use of the ORACLE, has been undertaken of orbits in the high-energy, fixed-frequency cyclotron of the type suggested by L. H. Thomas and E. M. McMillan. Previous calculations, performed by hand at the University of California Radiation Laboratory, were necessarily very slow and could cover only a relatively small part of the possible machine configurations. The UCRL calculations, performed by D. L. Judd for a fairly special class of machines, seem to indicate that a proton energy of at least 350 Mev is attainable. It is hoped that the rapid and much more complete calculations possible with the ORACLE will allow this energy limit to be extended upward by a sizable amount and at the same time yield design specifications for a machine that will achieve proton energies of great interest at present in high-energy physics. It is anticipated that a beam current very much larger (factor of 10^3) than that in present machines will become available.

The calculation proceeds by assuming a family of "nested" steady orbits in the median plane of the cyclotron. That is, a function $q(\beta,\theta)$ is assumed of the form

$$(1)\quad q(\beta,\theta) = q_0(\beta)[1 + q_1(\beta)\cos p\theta + q_2(\beta)\cos^2 p\theta + q_3(\beta)\cos^3 p\theta + q_4(\beta)\cos^4 p\theta + q_5(\beta)\cos^5 p\theta + q_6(\beta)\cos^6 p\theta] ,$$

where q_0, q_1, q_2, q_3, q_4, q_5, q_6 are assumed functions of β (the term β equals v/c, where c is the speed of light and v is the particle speed), θ is the azimuthal angle about the machine center in the median plane, and p is the number of times the field pattern repeats during one revolution. A steady, closed orbit of given β is specified by

$$(2)\quad r = q(\beta,\theta) ,$$

and the "nesting" condition is imposed that if $\beta_1 > \beta_2$, then the orbit for $\beta = \beta_1$ lies entirely *outside* that for $\beta = \beta_2$.

An ORACLE code has been largely completed which will do the following:

1. for given q_1, q_2, q_3, q_4, q_5, q_6, adjust q_0 to give fixed orbital frequency,
2. for given q_1, q_2, q_3, q_4, q_5, q_6, differentiate q to find the magnetic field at all points on the orbit,
3. compare this field with that obtained for the preceding value of β to obtain derivatives of the field with respect to radius,
4. integrate the equations for small vertical and radial deviations from the assumed steady orbit to find whether the orbit is stable or not.

With this code, an orderly exploration will be undertaken, first to maximize the proton energy obtainable and then to reduce the cost of the machine as much as is consistent with reasonable performance.

[1]This problem was coded for the ORACLE by T. Arnette.

[2]Summer visitor, Cornell University.

4. ACTIVATION RESONANCE INTEGRALS OF U^{238} AND Th^{232}

R. L. Macklin H. S. Pomerance

The activation resonance integrals of U^{238} and Th^{232} were measured relative to gold. Cutoff filters of gadolinium, cadmium (two thicknesses), and cadmium-rhodium were tried, giving cutoff energies of 0.2, 0.4, 0.8, and 2.0 ev, respectively. The neutron beam of hole 10 in the ORNL Graphite Reactor was used.

The effect of rhodium is contrary to expectations. Results of the work for the gadolinium and cadmium filters are 278 ± 20 barns for U^{238} and 67.3 ± 4.7 barns for Th^{232}, both values being obtained after the $1/v$ subtraction in which 1583 barns was used as the resonance integral for gold.

[redacted]