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The following experiments require a neutron source, which may be any alpha-beryllium source such as plutonium-beryllium, polonium-beryllium, or radium-beryllium. The first two have no source gammas and are easier to handle; the outer two have long half lives and need not be replaced annually; the latter two can be procured in greater strength. The experiments can be performed with a source strength greater than several million neutrons per second, and each can easily be performed in one afternoon with a source strength of ten million neutrons per second.

The other equipment is of two types: the neutron containment facility and the neutron detection materials with associated equipment. There are no detailed summaries of the first, nor suggested lists of equipment for the second that apply to neutron work. General references of one sort or another are:

Nuclear Reactor Theory by S. Glasstone and M. C. Edlund, (van Nostrand, New York, 1952)

Reactor Physics Laboratory Manual, TID-5262, available from the Office of Technical Services, Washington, D. C.

Pile Neutron Research by D. J. Hughes (Addison-Wesley, Cambridge, Mass., 1953)

Experimental Nucleonics by Ernest Bleuler and George J. Goldsmith, (Rinehart, New York, 1952)

Miscellaneous Physical and Chemical Techniques of the Los Alamos Project, edited by A. C. Graves and D. K. Froman (McGraw-Hill, New York, 1952)

Proceedings of the International Conference on the Peaceful Uses of Atomic Energy, August, 1955, Vol. 5, Sessions 19A, 21A (United Nations, New York, 1956)

Conference of the Academy of Sciences USSR, July, 1955; English translation AEC-TR-2435, Part I (Superintendent of Documents, Washington, D. C.)

In addition, a detailed laboratory account of experimental work in the Argonne school (ISNSE) will be published early in 1958, and a treatise on reactor theory and its experimental basis by A. M. Weinberg and E. P. Wigner will be published late in 1958.

The counting equipment depends to some extent on the available neutron source strength. Foils of 25 cm² require a G-M tube, a gas-flow windowless beta-proportional counter, or a minimum three inch diameter scintillation crystal and photomultiplier tube. Smaller foils can be counted with end-window beta-proportional counters or scintillation counters. My recommendation would be the three inch diameter anthracene crystal and photomultiplier tube; this supposes a good electronic maintenance. For minimum maintenance the G-M tube will give as good results but it requires more handling of foils and is less flexible in the choice of irradiation schedules. Rectangular foils are used with the G-M tube, with one dimension being the circumference of the tube and the other dimension dependent on the tube design; the size may be about 4 cm x 6 cm. The foil is wrapped around a mandrel, then slipped into a tube 6 cm long, and the tube and foil then slipped over the G-M tube to its central part. With the anthracene crystal the foils are laid flat on

the crystal. A NaI (Tl) crystal for gamma counting is also good but has a higher background rate. In either case all foils must be put in identically the same positions to be counted with the same efficiency.

The other tool for neutron counting is the boron-trifluoride counter. Several sizes will suffice; say 1/2 in. diam x 1 in. long, 1 in. x 6 in., and perhaps also 2 in. x 12 in. It may be necessary to construct a shield for the counters; a shield is principally five inches of paraffin around the counter with boron carbide or cadmium immediately adjacent to the tube. For foils or BF₃ tubes, pulse counting equipment is needed, consisting of a high voltage unit, pre-amplifier and amplifier, scaler and register, and timer. This equipment can be procured as a single unit except for the pre-amplifier.

Because the statistical precision of a count is the square root of the count, it is common to count a foil or BF₃ counter for 10 000 counts to gain 1% precision. If five indium foils, with a half life of 54 min, are to be counted the minimum rate should be 1000 c/min. The maximum rate depends on the counter; for a G-M counter the coincidence loss is 5% at a counting rate of 10 000 c/min, but the proportional counters and scintillation counters can be used to 50 000 c/sec. The minimum acceptable rate is affected by the background counting rate of the unit; for a G-M counter this is 20 to 30 c/min, for the BF₃ counter it can be lower, but for the crystal scintillation counter it will be one to several hundred c/min. Indium foils irradiated overnight reach the saturation rate, but most of these experiments will activate the indium sufficiently in half an hour with a neutron source of 10⁷ n/sec; cadmium covered foils are a general exception. With the BF₃ counter, the counting rate is immediately evident, and a quick decision can be made to collect 10 000 counts or only 1000 counts in the interest of speed; for indium foils only a test foil can establish the irradiation times for the experiment.

For rules of thumb, a 5 mil (0.12 mm) thick foil of indium absorbs 8% or 9% of the neutrons falling on it; the efficiency for counting the indium betas is from 30% to 50%. A one inch diameter BF₃ detector (enriched boron at 12 cm Hg pressure) will absorb 4% or 5% of the neutrons passing along a diameter. Neutron fluxes are usually expressed as neutrons per cm² per second, and so the area of the foil or BF₃ counter enters into the total counting rate in a known flux. Cadmium depresses the resonance neutron flux; the cadmium covered foil value is multiplied by 1.09 (for 0.5 mm Cd) in taking the cadmium difference.

For neutron containment, there are four items. One produces slow neutrons for use in thermal neutron experiments, the other three study the properties of graphite, of water, and of a graphite and uranium system (or of a water-uranium system if preferred).

Slow neutrons are easily produced by slowing-down in a neutron howitzer of water or paraffin or wood. Because of the difficulty of making a water-tight box with re-entrant tubes, paraffin is more often used; wood is nearly as efficient. The neutron howitzer at the Oak Ridge School of Reactor Technology is a cube of $3/4$ inch plywood with 2 inch diameter aluminum tubes for the source and the foils, and the remainder filled with paraffin; were I to make a new one it would be 18 inches (45 cm) on a side instead of the present 24 inches (60 cm). A cylinder is equally good, but with wooden sides a cube is easier to construct. Paraffin cools slowly and shrinks considerably, so that several days are needed to fill such a box. The aluminum tubes are filled with sliding paraffin or wooden plugs. A slot in the plug can hold foils; one plug has a small aluminum can that fastens on with a quarter twist lock for placement of the neutron source. Tubes that look directly at the source should have about two centimeters of paraffin between their ends and the source; a smaller ratio of fast neutrons to slow neutrons can be had by making the projection of the tube not include the source position.

A pile of graphite (sigma pile) is generally built according to the size of available graphite. If new graphite is bought from National Carbon Company in 51 inch lengths, the sigma pile will be economical to build as a square prism 68 inches (equals 1.33×51) on a side and about eight feet high. In metric measure, 178 cm x 178 cm x 240 cm. At a large laboratory that must make absolute neutron flux determinations, a permanent sigma pile and Ra-Be source can be used to standardize foils for neutron counting. Otherwise it is a simple matter to design the pile so that some graphite rods are removable and can be replaced with uranium rods to form an exponential pile (graphite-uranium system). One graphite bar, about 40 cm from the base, is cut $1/32$ inch undersize so that it can be slid in or out; the neutron source is deposited in a hole in the bar which is on the vertical axis of the pile. Other bars at various elevations are also undersize; a slot is milled into each bar large enough to hold the tray and the foil within it, or to hold a graphite block when no foil is there. Provision should be made for a foil position every five or six centimeters along the axis, although not more than six or eight foils are irradiated simultaneously, and for foil positions every 10 or 15 cm in a horizontal direction. Only one bar need be milled for the horizontal work, and the spacing will depend on the spacing of fuel channels in the exponential pile work.

For water and a high energy neutron source, the only precise experiment is performed with a water tank at least one half meter in diameter and one meter long or high. For teaching purposes, but not for a precision measurement, a 22 liter or 50 liter round bottom flask can be used for another experiment. In this, as elsewhere in this report, experiments are limited to those relevant to neutron measurements and nuclear reactor properties. The neutron source in the paraffin howitzer or in the water tank can be used to irradiate vanadium for a radiochemistry experiment, or to produce carrier-free radio-manganese by the Szilard-Chalmers process; the neutron source alone can be used for gamma rays from an excited state of carbon; but by themselves such works are not relevant to our purpose.

A. Experiments with a Neutron Howitzer

Experiment 1. Use of counting equipment. Half life for growth and decay.
Purpose: To become familiar with the handling of foils and the computations of saturated activity; the saturated activity of a foil is proportional to the neutron flux in which it was irradiated.
Method: Silver foils, 5 cm x 5 cm x 0.01 cm, are irradiated in the neutron howitzer. Irradiation times of 1, 2, 4, 6, and 10 minutes at a common flux give the growth curve; the 10 min foil, followed for 20 min gives the decay curve. The 23 sec activity may be neglected, and only the 2.3 min activity studied.
Reference: TID-5262, Experiment I
Bleuler and Goldsmith, Experiment 4

Experiment 2. Neutron detection.
Purpose: To study the operation of a BF_3 counter and the influence of hydrogen and cadmium on the counting rates.
Method: A one inch diameter BF_3 tube is placed in the external beam from the neutron howitzer, and the optimum voltage and amplifier gain are determined. Hydrogen at the back and sides of the tube increases the counting rate; cadmium between the tube and hydrogen decreases the rate. The transmission of cadmium in the direct beam is apparently increased by following it with a thin sheet (3 mm) of hydrogenous material. Hydrogen may be paraffin or plastic such as the methacrylates.
Reference: TID-5262, Experiment 8

Experiment 3. Transmission cross section for slow neutrons.
Purpose: The simplest, and hence the most extensively reported, is the total cross section measured in transmission. Transmission measurements are for beams, not for isotropic distributions.
Method: A BF_3 counter, preferably shielded from side radiation, is in the direct beam from the howitzer. The transmission ratio of a weighed sample of iron or lead or aluminum, measured with the sample in the beam and out of the beam, leads to the total cross section per atom of material. There are "good" and "bad" geometry variations, and cadmium differences must be used.
Reference: TID-5262, Experiment 8
J. R. Dunning, G. B. Pegram, G. A. Fink and D. P. Mitchell,
Phys. Rev. 48, 265, (1935)

Experiment 4. Scattering cross section for slow neutrons.

Purpose: One of the principal processes for slow neutrons.

Method: For geometrical reasons, this experiment requires a stronger neutron source or longer counting times than a transmission experiment, and so the background counting rate must be kept low. The neutron beam of the howitzer falls on a sample at 45° to the beam, such that all neutrons pass through the sample (that is, the sample is 90° to the position of a mirror which would reflect light at 90°). The BF_3 detector is at 90° to the beam and so measures the right angle scattering. For slow neutrons, measured by a cadmium difference, scattering is isotropic and one angle is as good as another, except that for 90° the absorption and thickness can be allowed for in the calculations. The scattering cross section is measured by comparison with a standard, which can be graphite.

Reference: TID-5262, Experiment 8

M. Goldhaber and G. H. Briggs, Proc. Roy. Soc. A162, 127 (1937)

Experiment 5. Albedo.

Purpose: To measure the reflection coefficient (albedo) of paraffin or water.

Method: Three large indium foils (or silver foils) are irradiated for equal times at one point in the howitzer or in a tank of water. The foils must be larger in two dimensions than the diffusion length in the medium, about 3 cm in water or paraffin. One foil is covered with cadmium on both sides (Z), one foil has cadmium on one side (Y), and one is bare (X). In practice aluminum or other weak absorber is used to keep the foils clean when not otherwise covered. From the three counting rates, the albedo is computed: $\text{albedo} = 1 - \frac{2(Y-Z)}{X-Z}$. A correction may be needed for the thickness of the foil.

Reference: TID-5262, Experiment 17

E. Amaldi and E. Fermi, Phys. Rev. 50, 910 (1936)

Experiment 6. Angular distribution of slow neutrons.

Method: Diffusion theory of neutrons gives a simple formula for the angular distribution of neutrons leaking from a diffusing material. A small hole, say 2 cm in diameter, is cut in a piece of cadmium used to shield one face of the howitzer. The large BF_3 detector is used end on to increase its counting efficiency, and is mounted in a boron or cadmium shield which also serves as a collimator. The detector is swung around to various angles using the hole in the cadmium as the center of motion.

Reference: E. Amaldi and E. Fermi, Phys. Rev. 53, 1020 (1938)

Experiment 7. Temperature of neutrons.

Method: A sample of known boron concentration must be used. The boron should be in the range of 100 to 200 mg/cm², with the middle of the range being optimum. Pyrex plates, boron carbide or boron oxide packed between two thin sheets of aluminum, will serve; be careful of purity because boron compounds, elemental boron especially, are quite impure and can be used only if carefully analyzed. The scattering and absorption cross sections are known at 2200 m/sec from the "Neutron Cross Sections, BNL-325"; the transmission cross section for a beam from the howitzer differs from the lower 2200 m/sec value by the lower absorption of the boron at the higher neutron velocities of the beam, that is, its temperature.

Reference: Hughes, Chapter 3.4 and Experiment 4 (Appendix 4)
J. Pelser, Physica 21, No. 1, page 22 (1955)

Experiment 8. Resonance energies by the boron absorption method.

Method: Boron plates, of known boron content or calibrated in a mono-energetic neutron beam, are measured in transmission with an indium detector. If the beam is filtered by cadmium, almost all of the activation of the indium is by neutrons of 1.46 ev, the resonance peak. This energy value can be determined from the transmission-measured cross section of boron and the $1/v$ law for neutron absorption in boron.

Reference: H. H. Goldsmith and F. Rasetti, Phys. Rev. 50, 328 (1936)

B. Experiments with a Tank of Water

Experiment 9. Migration length in water for Po-Be neutrons.

Purpose: To find the second moment of the neutron distribution, the single number most used to characterize the distribution. For Ra-Be or Po-Be sources, the attenuation length for 5 Mev neutrons is found as well.

Method: The neutron source is placed at least 15 cm from the base and sides of the large water tank. The BF₃ counter is used to measure the slow neutrons to at least 35 cm from the source. If A is the counting rate at each distance r, then the graph of $\text{Log}Ar^2$ versus r can be extrapolated as a straight line to find A for very large distances; the slope of the line gives the attenuation length for first collisions, a number needed in shielding calculations. Numerical integration of Ar^4 and Ar^2 leads to the mean squared distance a neutron travels from birth to death, r^2 . The migration area of reactor theory (albeit for fission neutrons and not for Po-Be neutrons) is $M^2 = r^2/6$. If the source is strong enough to make the foil measurements with cadmium covers, the mean square distance from birth energy to the indium resonance energy at 1.46 ev is found; adding the increments for 1.46

ev to thermal and for thermal diffusion (Barkov and Mukhin) should give the value for M^2 .

References: TID-5262, Experiment 20

J. W. T. Spinks and G. A. R. Graham, Canad. J. Res. A28, 60 (1950)

W. Baer, J. Appl. Phys. 26, 1235 (1955)

J. Tittman, Phys. Rev. 90, 256 (1953)

L. M. Barkov and K. N. Mukhin, J. Nuclear Energy 4, 91 (1957)

J. E. Wilkins, Jr., et al, Proceedings Geneva 1955, Vol. 5, P/597.

Experiment 10. Mean life of thermal neutrons in water.

Method: The $1/v$ law of thermal neutron capture can also be stated that the chance of absorption of a neutron is proportional to the time a neutron stays in the vicinity of a nucleus. If the neutron density is measured in water before and after the addition of a known amount of absorber such as boron, we can compute the mean life of neutrons in water, the ratio of the absorption cross section of hydrogen to that of boron, and the absolute strength of the neutron source if we have a counter of known efficiency (frequently called the manganese bath method). The size of the water flask or tank depends on the neutron source strength; a source of 10^7 n/sec can be used in a one meter diameter tank.

References: Bleuler and Goldsmith, Experiment 7

B. Hamermesh, G. R. Ringo, and S. Wexler, Phys. Rev. 90, 603 (1953)

G. C. Hanna and O. J. C. Runnalls, Canad. J. Phys. 34, 959 (1956)

J. DeJuren and J. Chin, J. Res. Nat. Bur. Stand. 55, 311 (1955)

C. Experiments with the Sigma File

For a moderating material, there are three numbers of interest; the slowing down length for fission neutrons, the thermal diffusion length, and the mean transport path (or the diffusion coefficient). If only a Po-Be neutron source is available, then the slowing down length for such neutrons (average energy about 4 or 5 Mev) is found. Further, in graphite we can measure the Fermi age, the parameter of the gaussian distribution of neutrons, or we can measure the spatial distribution of neutrons and compute the second moment which is six times the Fermi age.

Experiment 11. Slowing down (Fermi age) of neutrons in graphite.

Method: The indium foils, or other resonance absorbers, are irradiated in cadmium covers. A graph of log flux (or log saturated activity) versus the square of the distance from the source gives a Fermi age for graphite; fission neutrons have a different age from Po-Be neutrons. Closer to the source than 25 cm, the high energy resonances of indium

change the slope of the curve; at distances greater than 60 cm, the uncollided neutrons of initial source energy change the slope. At least six foils can well be used in the range from 30 to 60 cm.

References: TID-5262, Experiment 19

L. D. Roberts, J. E. Hill, and G. A. McCammon, Phys. Rev. 80, 6 (1950)

Glasstone and Edlund, Section 6.139 - 6.147.

Experiment 12. Diffusion length of thermal neutrons in graphite.

Method: Although the Po-Be neutron source is a point source, a plane in the graphite pile more than 60 or 70 cm from the source acts as a plane source of neutrons. Beginning about 70 cm from the point source, irradiate four indium foils at 20 cm intervals. The slope of the line on the graph of log flux (or log saturated activity) versus distance from the source gives the relaxation length for the sigma pile; a simple calculation for lateral neutron leakage based on the geometrical size of the pile, gives the diffusion length for the thermal neutrons. Properly a cadmium curtain at 60 cm from the source should be used for a cadmium difference. The effect of the epithermal neutrons can be calculated from the Fermi age of the source, but for teaching purposes this is a small effect that may be neglected.

References: TID-5262, Experiment 18

H. G. Hereward, G. C. Laurence, H. R. Paneth, and B. W. Sargent, Canad. J. Res. A25, 15 (1947)

Glasstone and Edlund, Section 5.67 - 5.91

Hughes, Experiment 2 (Appendix 4)

Experiment 13. Mean transport path of thermal neutrons in graphite.

Method: The extrapolated boundary of a pile is related to the mean transport path in the medium. If the neutron source is placed about 120 cm from the upper boundary of the sigma pile to give a large flux at the boundary, the flux no longer varies as a single exponential, as in the diffusion length measurement, but as a hyperbolic sine where the distance from the extrapolated boundary enters into the argument. Measure the flux at 10 cm intervals going down from the upper boundary. Curve fitting, helped by use of the relaxation length measured in the previous experiment, in the graphical manner to find the extrapolation distance.

References: TID-5262, Experiments 21 and 22

S. W. Kash and D. C. Woods, Phys. Rev. 90, 564 (1953)

P. Auger, A. M. Munn, and B. Pontecorvo,

Canad. J. Res. A25, 143, (1947)

Hughes, Experiment 2, (Appendix 4)

D. Experiments With the Natural Uranium and Graphite Exponential Pile

For large reactors it is possible to find the values of many parameters by studying a fraction of the system. If the exponential pile is large compared with the lattice cell length of the critical heterogeneous reactor and large compared to the slowing down and diffusion lengths, then the neutron spectrum is unaffected by the leakage through the sides and is characteristic of the reactor and not of the neutron source in the base. If the source is at the base, then the flux measured vertically varies as the hyperbolic sine. Measured horizontally the flux varies as the sine; near the source there are higher harmonics and for ease of analysis one must look at the flux at least one meter distant to find the fundamental mode alone. Most reports of exponential pile work include data of the variation of the material buckling, calculated from the flux variation in the vertical direction, versus changes in the lattice size or uranium slug size. Other parameters measured in the exponential pile have been less frequently reported.

References: TID-5262, Experiment 22

Proceedings, Geneva, 1955, Vol. 5 Session 21A; (see especially E. R. Cohen, P/605, and D. E. Davenport, P/599)

Glasstone and Edlund, Sect. 9.90 - 9.111

L. V. Groshev et al, Conference Acad. Sci. USSR, July, 1955, page 39.

A. B. Smith, "An Experimental Study of Heterogeneous Exponential Assemblies", June, 1955, (Available from Argonne National Laboratory); this is the manual used in the School of Nuclear Science and Engineering

Experiment 14. Axial buckling and critical size.

Method: The source is in the pedestal of the exponential pile on the vertical axis, and measurements of the flux are made along the axis. More than say 60 cm from the source the flux varies as the hyperbolic sine with only a small harmonic content. From the analysis of the flux shape the material buckling of the system is determined and from that number the critical size is calculated. The flux should be measured at corresponding points in the lattice cell; for instance always in the plane of the fuel channels, or at $a/4$ and $3a/4$ (where a is the cell height) since those two points are symmetrically located and have the same neutron spectrum.

Experiment 15. Intracell flux distribution.

Purpose: The average flux in a cell of a heterogeneous pile is related to the thermal utilization. The disadvantage factor in the fuel includes the depression of flux at the fuel surface as well as the flux shape in the interior, but with a Po-Be source, only the depression at the surface can be seen.

Method: The graphite bars between two fuel channels are made removable. A 5 cm diameter hole is bored through them at right angles to the channels. The hole is generally filled with discs of graphite, but indium foils (or dysprosium foils) can be sandwiched between the discs. Because of the interaction of the foils, only one or two can be irradiated at one time. A foil cemented to a uranium slug gives the surface flux. Smaller foils are desirable, but just how small they can be depends on the available source strength. The thermal utilization can be computed from the integral of the neutron flux in the moderator and the estimate from diffusion theory of the flux within the uranium slug.

Experiment 16. Mean macroscopic absorption cross section and thermal utilization.

Purpose: The cadmium ratio of indium is a simple ratio to measure. Measured only in the exponential pile, the cadmium ratio leads to the mean macroscopic pile absorption cross section. If the cadmium ratio of the same foil is also measured in a sigma pile with known Fermi age and diffusion length, the thermal utilization is found as well.

Method: The foils are irradiated and counted in the usual way. Care must be taken to irradiate them in the "average" positions of the lattice cell.

Experiment 17. Lateral flux traverses.

Method: The sinusoidal variation of flux in a horizontal direction is verified by measuring from one edge to a point past the center; the flux should be measured only at a corresponding point of each cell to avoid intracell variations. The flux measured at the lowest plane (nearest the source) can be decomposed into the first three or four odd harmonics; the flux measured one meter higher shows only the fundamental mode and its attenuation rate between the two levels must be the same as the attenuation rate measured in Experiment 14.

Discharge one row of uranium slugs and replace with a cadmium rod. The rod may be made by wrapping cadmium around an aluminum or steel tube and protecting the soft cadmium with some cover. Repeat the lateral traverse at the higher level and see qualitatively the changed flux near a control rod.