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A SERIES OF THERMAL, EPITHERMAL AND
FAST NEUTRON MEASUREMENTS IN THE MTR

By
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A Series of Thermal, Epithermal and
Fast Neutron Measurements in the MTR

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Summary

A series of neutron flux measurements using resonance reactions, np and n α reactions, and threshold energy fission reactions, were performed in one of the horizontal beam holes (HB-3) in the MTR. It was shown that the maximum flux above thermal energy is $(3.1 \pm 0.6) \times 10^{13}$ neutrons $\cdot \text{cm}^{-2} \cdot \text{sec}^{-1}$ and that the thermal flux maximum is $(2.3 \pm 0.4) \times 10^{14}$ neutrons $\cdot \text{cm}^{-2} \cdot \text{sec}^{-1}$.

Two separate instrument development programs are described. One is the design, construction and operation of a pneumatic facility for very rapid and safe transfer of materials into and out of the high level neutron field and the other is the development of a series of very small fission chamber probes using threshold fission reactions in very intense radiation fields. Attempts to express results as crude neutron spectra are described.

The threshold reaction $\text{Ni}^{58}(\text{np})\text{Co}^{58}$ was investigated for use as a radiation damage monitor for metals. It was concluded that more information is needed in regard to the reaction cross section before it can be approved for such service.

A SERIES OF HIGH ENERGY NEUTRON FLUX MEASUREMENTS IN THE MTR, HB-3 POSITION

Introduction

The early objectives of this project were pointed toward the establishment of a definite correlation between neutron energy distribution and radiation damage. (1) Although it was realized at the time that such an attempt would probably be successful only through the use of particle accelerators for obtaining monochromatic beams of neutrons with which to induce damage, it was nevertheless thought that a general type of correlation could be found even with the reactor. This was based upon the evidence that a thermal reactor in general contains two different spectral extremes. One of these is an almost pure fission type of spectrum and the other is a highly moderated spectrum containing an inverse energy proportionality which holds well upwards into the Mev region. (2) Since the moderated spectrum almost always contains orders of magnitude fewer neutrons it was required to choose a location such that a useful amount of radiation damage could be produced with a moderated spectrum in a short length of time. The reactor most suitable was the MTR and therefore the project was designed for the horizontal beam hole HB-3. The location of the experimental facility is shown in Fig. 1.

It was planned to measure first the flux characteristics of the hole with a series of small fission chambers which were developed especially for this purpose. These fission probes were designed for use with Np^{237} , U^{238} , Th^{232} and U^{236} to measure neutron energies from 0.5 Mev to above 2-1/2 Mev. As a corollary to this, threshold detectors were chosen for measuring neutrons with energies which lie

(1) J. B. Trice, et al., Solid State Quar. Prog. Rep. Aug. 31, 1953, ORNL-1606, p 23.

(2) D. J. Hughes, Pile Neutron Research, p 58, Addison-Wesley Publishing Company, Inc., Cambridge 42, Mass., 1953.

Fig. 1. Schematic Picture of MTR Lattice Loading During Experiments with Threshold Detectors.

within the range defined by detectors with effective thresholds from 2.4 to 8.1 Mev. With these two devices it was at least feasible in principle to determine the general characteristics of a neutron spectrum from a 0.5 Mev to 8 Mev. In addition to these two groups of detectors there was a third set - the so-called resonance reactions - used to determine the spectrum in the epithermal region.

One of the most important objectives in the beginning of the project was a measurement of the distribution in energy between ionization energy and knock-on energy which appears as a measure of radiation damage in solid materials such as metals bombarded by neutrons. The method by which this was to be undertaken employed two regions in the reactor, one of which contained a predominance of high energy neutrons produced directly from fission, and the other which contained the moderated spectrum. In the experiment to determine the difference in effects due to high energy neutrons and low energy neutrons, it was planned to perform radiation damage tests in each position and to monitor the incident neutrons both with respect to intensity and spectral distribution. A comparison between the amount of radiation damage induced in each of the tests was to be made to ascertain the differences caused by the two unlike spectra. Several difficulties encountered in attempting to do this pointed out the fact that this experiment as first conceived was perhaps too large in scope and too lengthy to be completed at the present time. If it had been possible to complete, however, it should have provided sufficient knowledge concerning the relationship between neutron energy and damage to make possible an intelligent search for a radiation damage monitor, that is, a single material which could be included with every experiment in order to define the radiation field attending the irradiated material. Perhaps the optimum way to obtain this objective at the present time would be to abandon reactors as instruments for making this particular fundamental investigation and instead to use monochromatic neutrons

produced by some reaction such as the $\text{Li}(\text{pn})\text{Be}$ reaction produced through the use of a Van de Graaf accelerator. Radiation damage produced by such monochromatic neutrons would appear as changes in such properties as elastic moduli, ⁽³⁾ electrical resistivity, ⁽⁴⁾ lattice parameters ⁽⁵⁾ and other properties. Changes in these properties could then be measured as a single function of neutron energy. From this it would then be possible to obtain a general relationship to describe the way in which incident neutron energy is imparted to a material with respect to both ionization and true radiation damage produced from the displacements of atoms by various orders of "knock-ons." Such an experiment might possibly show that all of the energy imparted to a solid by bombardment with reactor neutrons goes into radiation damage. The next experimental step would be to standardize some convenient monitor such as diamond or quartz which could be used to measure the amount of energy imparted to the test material. From such a monitor would follow unequivocally, therefore, the conditions under which the sample was irradiated. However, even if the results of the suggested Van de Graaf experiments were to show that there is a strong dependence of the ionization to knock-on ratio on incident neutron energy, the information would still be theoretically relevant to solid state physics.

Some of the possible monitors which have been considered for use with radiation damage experiments are diamond, which undergoes a change in lattice parameter as measured by means of x-ray diffraction; ⁽⁵⁾ copper, which undergoes

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- (3) D. O. Thompson, D. K. Holmes and T. H. Blewitt, Neutron Irradiation Effects Upon Young's Modulus and Internal Friction of Copper, Journal of Applied Physics 26, 9, 1188 (Sept. 1955).
 - (4) T. H. Blewitt, et al., Physics of Solids Institute Quar. Prog. Rep., July 31, 1951, ORNL-1128, p 55.
 - (5) M. C. Wittels and J. B. Trice, Solid State Division Quar. Prog. Rep., August 10, 1952, ORNL-1359.

a change in electrical resistivity under irradiation,⁽⁴⁾ and quartz, which enjoys a change in both optical properties and lattice parameters as measured by several different methods.⁽⁶⁾ One of the major difficulties with the actual use of such a monitor is the fact that radiation damage begins to anneal out appreciably at temperatures of the order of 100°C - 200°C.⁽⁷⁾ Since many experiments are performed in very high intensity radiation fields where the gamma heating makes cooling the sample very difficult, the use of such monitors would be prohibited except in specialized cases. It is therefore thought that a practical alternative for a monitor which actually measures the radiation field by means of induced radiation damage is complete information about the spectral characteristics of reactors together with a knowledge of the energy dependence of the radiation damage on the fraction of incident neutron energy which is deposited as "knock-on" energy. These facts would permit one to predict the conditions of irradiation for many in-pile engineering experiments to determine changes induced in such properties as creep, hardness, brittleness, and elastic moduli. It is possible, therefore, that the best monitor to use for radiation damage to metals work is one in which radioactive nuclei are produced in such a manner that a neutron flux measurement to include neutrons of all energies could be measured and the pertinent conditions under which radiation damage was induced could be obtained from empirical information.

The use of threshold detectors to measure the neutron spectral characteristics of the HB-3 hole in the MTR was fraught with many difficulties, the most serious

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- (6) R. W. Ditchburn, et al., The Optical Effects of Radiation Damage in Diamond and Quartz; Report of the Conference on Defects in Crystalline Solids, July 1954, p 92, The Physical Society, 1 Lawther Gardens, Prince Consort Road, London S. W. 7, 1955.
- (7) M. C. Wittels and F. A. Sherrill, Radiation Damage in SiO₂ Structures, Phys. Rev. 93, 1117, (1954).

of which was the lack of information in regard to the functional dependence of the cross section on neutron energy. A few of the energy dependence functions for threshold detectors were based on elementary concepts, many of the threshold functions were measured with very poor energy resolution and a few were measured either at only one or two points or were averaged over an entire fission spectrum. With such background data it was possible to determine whether or not the spectrum being measured was different from fission but it was difficult to estimate the errors involved in predicting the deviation of the beam hole spectra from fission. In regard to the use of two different locations in the reactor for measuring radiation damage, there were at least two principal difficulties: (1) as mentioned before gamma heat annealed out radiation damage to many of the materials used and (2) it was found that moderated spectra in most cases contained insufficient neutrons to impart measurable damage within an acceptable length of time. In spite of all the difficulties however, a rewarding amount of information was obtained in regard to both the conditions of irradiation and many of the nuclear characteristics of the materials used for measurements in the beam hole. Flux traverses with threshold detectors were run along the most important interval of the beam hole axis. Neutron spectra as measured with low resolution from a position adjacent to the lattice to about 3 feet away from the lattice indicated that very little deviation from a pure fission spectrum occurred throughout most of this region. In the low energy region however, the results from the tests with the resonance detectors showed that deviation from the inverse E proportionality predicted by elementary reactor theory was quite marked throughout the beam hole except in a position very close to the lattice.

The fact that threshold detectors were partially successful in measuring neutron spectra is encouraging especially since most of the measurements with

threshold reactions were based on scanty cross-section data. From the experience in the present experiment plus previous measurements it appears that the use of threshold detectors to measure neutron spectra could, if it were sufficiently developed, become a much more useful tool. It appears to be the only method capable of measuring neutron spectra in high flux intensity regions where it is desired to perturb the geometry of the irradiation as little as possible. Future programs designed to improve the method of threshold detectors should include cross-section studies for the most important nuclear reactions using monochromatic neutrons such as those produced from charged particle reactions with the 5-1/2 Mev ORNL Van de Graaf.⁽⁸⁾ Of especial interest in this connection would be the $\text{Al}^{27}(\text{n},\alpha)\text{Na}^{24}$ and the $\text{Ni}^{58}(\text{n},\text{p})\text{Co}^{58}$ reactions⁽⁹⁾ which have respectively effective thresholds of 8.1 Mev and 5 Mev. Both of these reactions are attractive candidates to consider for monitoring radiation damage experiments. The $\text{Ni}^{58}(\text{n},\text{p})\text{Co}^{58}$ reaction would be very useful since it has a 72 day half life. This would make possible an integrated flux measurement which for many experiments is more important than a measurement of the instantaneous fast neutron flux.

As mentioned previously, some of the most fruitful research pursuant to this project was the development of a series of very small fission chambers using several different threshold fissionable materials to probe reactor environs with high neutron fields. The chambers contained, in sequence, U^{238} , U^{236} , Np^{237} and Th^{232} and were successfully developed, tested and found to operate satisfactorily in regions in both the LTR and the MTR where instantaneous neutron fluxes exceeded

(8) H. B. Willard, J. K. Bair, and J. D. Kington, The Reactions $\text{T}^3(\text{p},\text{n})\text{He}^3$ and $\text{Li}(\text{p},\text{n})\text{He}^4$, Phys. Rev. 90, 865 (1953).

(9) T. O. Passell, R. H. Lewis and R. L. Heath, MTR Technical Branch Quarterly Report, June 30, 1955, IDO-16229, p 11. Also see - R. P. Shuman, et al., SO-11 (Sept. 5, 1952) p 6., Schenectady Operations Office, AEC.

10^{11} nv. The factor limiting their performance was the gamma ray ionization buildup which begins to saturate the chamber soon after the neutron intensity exceeds 10^{11} . It is imagined that further research could possibly develop a chamber operable in 10^{14} to 10^{15} (nv) range by minimizing the ionization volume accessible to gamma rays.

The advantages of the small fission chambers are (1) their compactness, since each one is only $3/4"$ O.D. and can therefore be inserted into many otherwise inaccessible regions (2) the simplicity for obtaining data since flux data can be evaluated within minutes after the probe is inserted into the reactor and (3) reliability with respect to the materials used since the cross-section data for fissionable materials is better known than for almost any other set of nuclear reaction data.

The results which have been obtained so far with threshold radioactivants, resonance detectors and fission chambers indicate that the fast flux maximum value in HB-3 is much lower than it had been thought to be prior to the experiment. The peak thermal flux is 2.3×10^{14} (nv) and the fast flux was expected to equal the thermal flux as a result of previous measurements in the LITR. This is not the case, the maximum fast flux being about 3.1×10^{13} or almost a factor of 10 under the thermal flux. Fig. 2 illustrates this quite well and presents the flux traverse for three positions in the beam hole. The dotted line shown in Fig. 2 are integral fission spectra⁽¹⁰⁾ normalized at 8.1 Mev to the experimental data as obtained with the $\text{Al}(n\alpha)\text{Na}^{24}$ reaction. Notice that at the position nearest the lattice the experimental data fit the fission shape quite well, and that the 2 points from the Np^{237} fission probe measurements coincide very well with the fission spectrum as measured with the $\text{Al}^{27}(n\alpha)\text{Na}^{24}$ reaction and several others.

(10) B. E. Watt, Energy Spectrum of Neutrons from Fissions Induced by Thermal Neutrons, LA-718 (Dec. 17, 1948).

Fig. 2. Fast-Neutron Flux Above Threshold Energies vs Threshold Energies.

Design and Development of the Pneumatic Facility

The fact that some of the materials used in this study have very short half lives necessitated the use of a rapid method for transferring the samples from the reactor to the counting room in order that there would be enough residual activity to measure in the gamma and beta ion chambers. Such materials as aluminum which has a 2.3 minute half life and vanadium which has a 3.7 minute half life made this necessary. Also, it is well known that if one retracts a material from a high intensity neutron field immediately without permitting time for very short lived activities to decay, the radiation emitted from such a sample can be extremely intense and therefore dangerous to handle. In order to permit operations with these very short lived activities it was decided at the beginning of the project to construct a pneumatic tube facility to operate in HB-3 in the MTR. The function of this machine was to insert samples into a predetermined position in HB-3, irradiate them for the desired period of time, and then remove them with sufficient haste to transfer them safely from the reactor facility to the counting room for analysis. The machine which was designed and built at ORNL for performing these services is shown in Figures 3 and 4. Samples can be loaded (Figure 5) in the train in each or all of the 16 cup positions shown. Containers for the samples are either magnesium or cadmium as desired. They are assembled in such a way that by actuating a small lever certain valves are opened and the four section train containing the cups and samples is pneumatically fired into the HB-3 hole. After the desired length of irradiation time the small lever is pushed again in the opposite direction and valves are opened which reject the assembly into the shielded receiver. By pushing the large levers which are visible in the Fig. 3 in the proper sequence the samples are ejected from the entire assembly into non-radioactive cups. These are then easily accessible for rapid transfer to the counting facilities. Transference

Fig. 3. The ORNL Pneumatic Facility at the MTR.

Fig. 4. Breech of the Automatic Pneumatic Facility Used for Fast Irradiations.

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Fig. 5. Two of the Jointed Shuttle Assemblies for Transportation of Samples Into the Reactor.

of a sample from the high intensity neutron field to the position of the first counting was effected in times as low as 30 sec. This rendered extremely reliable decay data on many of the very short lived activities.

Fig. 6 shows the fission probe and some of its accompanying electronic equipment. The probe electrode assembly is shown schematically in Fig. 7. The plates which hold the fissioning materials are easily detached in order that the entire series of fissionable elements may be used sequentially in one probe.

The major problems encountered during the developmental phase of the fission probes were the needs for (1) minimized capacitance of the 15-ft probe lead so that an acceptable pulse height plateau width could be obtained, (2) a probe construction to maintain high performance characteristics for long operation times (200 hrs) in flux fields greater than 10^{10} fast neutrons per cm^2/sec with correspondingly high intensity gamma fields, and (3) an accurate determination of the probe counter efficiency for each of the fissionable threshold materials to be used in the probe. The problem of probe lead capacitance was solved by varying both the geometry and materials of the probe signal lead system. A minimum distance in the reactor beam hole of 16 feet between the chamber and its preamplifier implied such a large capacitance ($430\mu\text{f}$) between the probe and the preamplifier that the signal from the probe was severely distorted by the time that it reached the preamplifier. It was recognized that the absence of a pulse height plateau was the result of this large capacitance. The capacitance was reduced to $180\mu\text{f}$ for 21 feet by using 10 mil nickel wires instead of 25 mil copper wire as the center conductor for signal lead and by increasing the inside diameter of the fused quartz insulators in the probe. Fig. 8 shows the pulse height plateau which was obtained with a potential difference of 360 volts across the probe plates. Operation of the probe

Fig. 6. Fission Probe and Associated Equipment.

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Fig. 7. Cross Section of Fission Chamber.

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Fig. 8. Typical Pulse Height Plateau Curve.

for 50 hrs in a field of 10^{11} fast neutrons/cm²/sec together with the accompanying gamma field (proportional to $\sim 10^{12}$ gammas/cm²/sec) proves that the probe possessed good radiation stability. Breakdown did occur at intensities of the order of 10^{12} fast neutron/cm²/sec but it was probably caused by the accompanying gamma rays and not by the neutrons. A convenient feature of the probe is that it can easily be handled after a day or so out of a neutron field of 10^{12} thermal neutrons/cm²/sec.

The probe was calibrated in the ORNL thermal column which provides a satisfactory plane source of thermal neutrons in water. The ratio of thermal to fast neutrons in this facility is approximately 10^6 . Calibration consisted of a comparison between the probe response and a gold foil flux measurement along the vertical traverse through the water. Use of the probe efficiency as measured with U^{235} and gold for other fissionable materials to be used in the probe seems to be justified. This was because the calculated chamber response for a known thermal flux agreed closely with the measured chamber response. The results from a vertical traverse calibration are shown in Table A-5 in which the flux measured with the probe (using a calculated chamber response determined from amount of fissionable material plus electrode geometry) is compared with the flux measured by a gold foil exposed to the same thermal flux as the probe electrode plates. Only after comparative measurements were made between a foil placed on the external surface of the probe and a foil placed between the electrodes was it assumed that the gold foil mounted on the side of the probe was exposed to the same flux as the probe electrodes. The two responses were identical within the limits of experimental accuracy (99%). Detection efficiency of the probe as determined by these measurements is $98 \pm 1.8\%$.

A major uncertainty in the use of gold to measure thermal neutron flux in water was in the so-called perturbation factor which is a measure of the change in flux caused by the presence of the foil. This factor was determined by a comparison in the ORNL thermal column of the gold with a known perturbation of indium in water. The result from gold was then compared with the flux measurement from gold in the ORNL Graphite Reactor in which the perturbation effect is quite well known. Here the thermal neutron flux is given by $\phi = 3.29 A$ where A is the saturated activity in counts per second for the beta counting rate in a certain beta counter (Physics Division's counter "Bob"). The number 3.29 is an experimentally determined number which includes counter geometry, nuclear and geometrical properties of gold, self absorption of both neutrons during irradiation and betas during counting, and the perturbation factor for a graphite medium.

The flux equation for water is $\phi = 3.29 A \alpha$ where α is a number which converts perturbation effects from graphite to those for water. Flux traverses made in the thermal column with gold were compared with indium measurements which were used as the standard of measurement for a water medium. The flux (ϕ) calculated by using indium and A , the saturated activity for gold, were substituted into the equation. $\alpha_{\text{au}} = \frac{(0.0753) A_{\text{indium}}}{(3.29) A_{\text{gold}}}$ to obtain α_{au} equal to 1.011. Since experimental errors of 5% are expected for these measurements it was decided that the 1% variation indicated by α_{au} was meaningless; hence α_{au} was called unity. The vertical flux traverse in the ORNL thermal column is shown in Fig. 9. Blueprints for the fission probe can be obtained from ORNL. They are listed as Q-1372-1, Q-1372-2, Q-1372-3, Q-1372-4, Q-1372-5.

Discussion of Data and Results

Characteristics of the (np) and (n α) reactions with respect to neutron activation are shown in Table B. These measure neutron flux within the range of

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Fig. 9. Vertical Traverse of Thermal Flux in ORNL Thermal Column.

energy from 2.4 Mev to 8.1 Mev and beyond as estimated with the effective threshold energies for fission spectra.

Each threshold activation test was monitored with a cobalt foil which upon measurement with a 4π gamma ray ionization chamber yielded a number proportional to the average reactor power during the irradiation time. Therefore, for nucleides which were irradiated at different reactor power levels it was possible to express the measurements from each experimental insertion in terms of a common reactor power. To assist normalization of each irradiation a traverse of the thermal neutron flux along HB-3 as measured with cobalt was obtained and expressed graphically. Then for each threshold detector measurement a cobalt monitor was included and placed in such a position that flux depression and perturbation effects from the cadmium enclosed threshold detector were very small. The thermal flux value from each monitor was then placed on the graph so that the flux ratios corresponding to reactor power ratios could be used to express all of the threshold detector measurements in terms of a common basis. This is depicted in Fig. 10. Also in Fig. 10 is shown a flux (pseudo) traverse at 30 mw obtained from cobalt foils enclosed in cadmium plus a few monitor foils also enclosed in cadmium along with the threshold detector. Actually, the use of epithermal flux monitors to track reactor power level is superior to the method of the thermal flux monitor because it is so much less sensitive to local variations in geometry, nuclear poisons, and control rod positions. For the present group of experiments it is fortunate that monitors were always included since it has been shown that neutron fluxes did not necessarily vary linearly with MTR recorded reactor power levels during this period.⁽¹¹⁾

(11) J. B. Trice, H. V. Klaus, J. F. Krause and R. H. Lewis, A Flux Depression Experiment in the Materials Testing Reactor, ORNL-CF-55-10-84 (Oct. 25, 1955).

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Fig. 10. Thermal and Epithermal Flux Plots in HB-3 Using Cobalt Monitors.

In Fig. 11 are shown data for the reaction $P^{31}(np)Si^{31}$ with a 2.4 Mev effective energy threshold. This reaction resulted from irradiation of $Mg_2P_2O_7$, a white powder which, following irradiation, was dissolved in dilute HNO_3 . Aliquots were pipeted from this for the preparation of mounts suitable for 4π geometry β - counting. Flux traverses from this material would appear to indicate that the spectrum deviates from fission with increasing distance from the reactor lattice. (See Figure 2). However, there is strong evidence that the apparent relative increase of neutrons in the region of 3 Mev in the fission spectrum is caused either by impurities in the $Mg_2P_2O_7$, unreported sensitivities to low energy neutrons, or faulty radio-chemical technique. The evidence that the neutrons retain a fission spectrum shape along the beam hole axis is twofold. First, comparison of the measurements of Np^{237} with other measurements at higher energies indicate that the fission shape is retained at low energies. (See Figure 2). Since the principal region of sensitivity for the reaction $P^{31}(np)Si^{31}$ lies between that of Np^{237} and reactions such as $Mg^{24}(np)Na^{24}$ and $Al^{27}(n\alpha)Na^{24}$ it is evident that this region in question is "bracketed" in such a way that deviation here from fission is unlikely. Secondly, a comparison of fission rates among Np^{237} , U^{238} and Th^{232} indicates that all three of these materials yield flux traverses along the hole axis which are almost identical in form. Since these three cover energies from about 0.6 Mev to 3 or 4 Mev and include the lower end of the energy interval of sensitivity for the $P^{31}(np)Si^{31}$ reaction it is almost certain that the deviation of the spectrum from fission with distance along the axis is very slight. Otherwise the three traverses for Np^{237} , U^{238} and Th^{232} normalized to unity adjacent to the reactor lattice would have diverged along the axis.

The $S^{32}(np)P^{32}$ traverse of Fig. 12 falls on the integral fission spectrum curve shown in Fig. 2. Flowers of sulfur were used to prepare the samples for this

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Fig. 11. Fast-Neutron Flux Above the Threshold Energy for $\text{P}^{31}(\text{n},\text{p})\text{Si}^{31}$ (2.4 Mev).

Fig. 12. Fast-Neutron Flux Above the Threshold Energy for $S^{32}(n,p)P^{32}$ (2.9 Mev) and $Ni^{58}(n,p)Co^{58}$ (5.0 Mev).

traverse. They were counted later using a $4\pi\beta^-$ counter. No chemistry was used for sulfur, the dry powder itself being spread out upon polystyrene mounts and placed in the counting chamber.

The (np) and (n α) reactions for Ni⁵⁸, Al²⁷, Si²⁸, and Mg²⁴ (see Figs. 12 through 16) were obtained by irradiation of the pure metals with the exception of Si²⁸ which was irradiated as Ottawa River sand. The resulting radio-nucleides from this group all have well known decay schemes and hence there was no difficulty determining a counter efficiency for each of them on the 4π gamma ray ionization chamber. Decay curves were plotted for each of this group except Ni⁵⁸ in order to identify the nucleide via its half-life. For the Ni⁵⁸(np)Co⁵⁸ reaction identification was obtained from analysis of the Co⁵⁸ with a gamma ray scintillation spectrometer after which the Co⁵⁸ was measured in the 4π gamma ray ionization chamber in order to obtain the absolute β^- disintegration rate. Examination of the two curves shown in Fig. 12, one for Ni⁵⁸(np)Co⁵⁸ and one for S³²(np)P³², shows that the Ni⁵⁸(np)Co⁵⁸ traverse is less steep than the S³²(np)P³² and thus it would appear to follow the lower energy groups of neutrons. Although the reaction Ni⁵⁸(np)Co⁵⁸ is exoergic by 0.6 Mev⁽¹²⁾ it nevertheless should behave as a threshold reaction because of its high coulomb barrier and should have an effective threshold energy of 5.0 Mev in a fission spectrum.⁽¹³⁾ The data for the Ni⁵⁸(np)Co⁵⁸ reaction are presented in Table B-4. The first column of flux numbers shows the fast flux above 8.1 Mev as measured with the Al²⁷(n α)Na²⁴ reaction by ORNL. The next column shows the flux (fission) above thermal energies as calculated from the Al²⁷(n α)Na²⁴ reaction based on the assumption that the spectrum is fission. The next column presents the total flux above thermal as measured by MTR. It is thought that the ORNL figures are probably correct since they are supported by other data described in this report.

(12) Based on the mass values given by Wapstra for the reaction Ni⁵⁸(np)Co⁵⁸. See Wapstra, A. H., *Physica* 21, 385 (1955).

(13) See Equation (2).

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ORNL-LR-DWG-9004

Fig. 13. Fast-Neutron Flux Above the Threshold Energy for $\text{Al}^{27}(n,p)\text{Mg}^{27}$ (4.6 Mev).

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ORNL-LR-DWG 9005

Fig. 14. Fast-Neutron Flux Above the Threshold Energy for $\text{Si}^{28}(n,p)\text{Al}^{28}$ (5.5 Mev).

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ORNL-LR-DWG-9006

Fig. 15. Fast-Neutron Flux Above the Threshold Energy for $\text{Mg}^{24}(n,p)\text{Na}^{24}$ (6.3 Mev).

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ORNL-LR-DWG-8645

Fig. 16. Fast-Neutron Flux Above the Threshold Energy for $\text{Al}^{27}(n, \alpha)\text{Na}^{24}$ (8.1 Mev).

The cross section values for the $\text{Ni}^{58}(\text{np})\text{Co}^{58}$ reaction as measured, using first the MTR flux figures, then the ORNL flux figures, are shown respectively in the next two columns. The 66.4 millibarn value based on ORNL data is adjacent to fuel and should represent the cross section averaged over a fission spectrum. Column 7 is merely a statement of the preceeding data in a different form. It is a list of the total flux-above-thermal values for each shuttle position and is based upon the two assumptions that (1) the spectrum of neutrons is fission, and (2) the cross section for the $\text{Ni}^{58}(\text{np})\text{Co}^{58}$ reaction is 66.4 millibarns. Thus, the total number of neutrons in the spectrum is given as 3.15×10^{13} from the $\text{Al}^{27}(\text{n}\alpha)\text{Na}^{24}$ reaction and the $\text{Ni}^{58}(\text{np})\text{Co}^{58}$ is normalized to this value of the flux. The most important fact which emerges from this is the deviation of the $\text{Ni}^{58}(\text{np})\text{Co}^{58}$ traverse from the other threshold traverses. In this particular case there is no reason to suspect impurities or faulty relative counting. The fact that the traverse tends to follow a lower energy group suggests that the cross section may possibly have intervals (such as resonances) of high sensitivity to low energy neutrons. However, measurements with cadmium covers have shown that it is not sensitive to thermal neutrons.

Figures 17, 18, and 19 exhibit the neutron flux traverse data obtained using the fission probes. Weight uncertainties made analyses with all materials except Np^{237} difficult and the best that could be done was to use the Np^{237} data and then to compare the shapes of the axial traverses of U^{238} and Th^{232} with Np^{237} . Table A-1 contains in summary form the weight data for the fissionable materials used. Only the Np^{237} weight determinations were completely satisfactory. Notice that the first weight determination for the Np^{237} is about a factor of ten less than the third weight determination. It has been shown since these determinations that the first weights were low by exactly a factor of 10. The third weight determinations

Fig. 17. Fast-Neutron Flux Above the Threshold Energy of Np^{237} (0.7 Mev) Compared with $Al^{27}(n,\alpha)Na^{24}$ (8.1 Mev).

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Fig. 18. Comparison of U²³⁸ Flux Traverse with Np²³⁷.

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Fig. 19. Comparison of Th^{232} Flux Traverse with Np^{237} .

are considered the best values since they are based on alpha counting which was performed many times on each plate with good reproducibility (98%). The purity of the Np^{237} is such that no significant counting error is introduced from fissionable impurities.

The remainder of the table is printed more as a matter of record than usefulness. The sources for each weight determination are listed. In some cases there were insufficient statistics on alpha counting and in other cases there were impurities to upset the determinations. Relative traverses for Np^{237} , U^{238} and Th^{232} however, are thought to be accurate. These materials did not contain significant amounts of impurities. Np^{237} is more sensitive to low energy neutrons ($E_{\text{eff}} = 0.6 \text{ Mev}$) than U^{238} ($E_{\text{eff}} = 1.4 \text{ Mev}$) and Th^{232} ($E_{\text{eff}} = 2.0 \text{ Mev}$) and the fact that the different flux plots can almost be made congruent points out the integrity of the fission spectrum along the axis of HB-3.

Data for the epithermal flux region is shown graphically in figures 20 through 27. The best insight into the way neutron energy distribution changes from an (E^{-1}) proportionality to an unknown distribution in the interval 100 ev to 10,000 ev can be gleaned from inspection of the curves plotted in Fig. 21. The drawn function for the several ($n\gamma$) reactions is inversely proportional to the cadmium ratio and in an (E^{-1}) epithermal spectrum it should not depend on the element used to obtain the function. That the four elements shown do produce the same function in close proximity to the lattice reveals the (E) $^{-1}$ nature of the reactor spectrum. The wide divergences away from the lattice are due in part to lack of measurement resolution but they do definitely indicate an energy sensitivity which is a function of the element used and therefore of the changing spectrum. The data for the region of close proximity to the lattice is more clearly revealed in Fig. 20 which shows the measured function (ϕ_0/E) obtained separately from the cadmium ratio data

SSD-B-1288
ORNL-LR-DWG-9121

Fig. 20. Epithermal Spectral Data for HB-3 Exhibited on an E^{-1} Spectral Plot.

SSD-B-1287
ORNL-LR-DWG-9008

Fig. 21. Traverses of a Function as Measured with Four Resonance Detectors to Indicate Deviations of Epithermal Spectrum from (E^{-1}) .

from each element. It is plotted at the resonance energy point obtained previously. The curves from the irradiation of NaCl, V and Al, used to calculate cadmium ratios, are shown in Figures 22 through 26. The data for cobalt, already discussed, are shown in Figure 10. Measured flux values from this group may not in some cases be entirely accurate in an absolute sense because of uncertainties in the decay scheme for some of the radioactive daughters. For this reason the ordinates are called saturated activities. Furthermore, the flux as measured by cadmium enclosed samples is merely proportional to the epithermal flux. In expressing these numbers, the thermal cross section is used in each case since this method yields numbers which can be divided to give cadmium ratios and for many of the bare samples the saturated activity is within a constant factor equal to the thermal flux value.

Errors

The errors are of two varieties; namely, those which resulted from faulty nuclear data and those which were introduced experimentally. Both kinds are difficult to estimate and both depend strongly upon the subjective attitude of the experimenter. Regarding the nuclear data, the cross section for the threshold reactions exist as plots of cross section versus energy, or as average values of the cross section in a fission spectrum. Some of these have attached to their values estimates of error and others do not. In general, it is probably fair to state that the values listed are rarely better than $\pm 10\%$ with the exception of the fissionable materials and rarely worse than $\pm 20\%$ for the best values. The $S^{32}(np)P^{32}$ reaction cross section is probably the most accurately known of the general non-fissionable group of threshold detectors and it has received the most attention from experimenters.

Examination of the data shown in Fig. 2 reveals an encouraging self consistency. Points along the traverse closest to the lattice indicate the presence of a fission spectrum with sufficient repetition to almost rule out the possibility of

Fig. 22. Saturated Activity Traverses for Bare and Cadmium-Covered Copper.

SSD-B-1278
ORNL-LR-DWG-8999

Fig. 23. Saturated Activity Traverses for Bare and Cadmium-Covered Sodium.

Fig. 24. Saturated Activity Traverses for Bare and Cadmium-Covered Chlorine.

SSD-B-1280
ORNL-LR-DWG-9001

Fig. 25. Saturated Activity Traverses for Cadmium-Covered Vanadium.

Fig. 26. Saturated Activity Traverses for Bare and Cadmium-Covered Aluminum.

SSD-B-1286
ORNL-LR-DWG-9007

Fig. 27. Traverses of the Function in Fig. 21 Using Chlorine and Copper.

coincidental errors which compensate just enough to yield a deceptive fission pattern. It is noteworthy on this graph that the measurements with Np^{237} represented at $E_{\text{eff}} = 0.6$ Mev agree with $\text{Al}^{27}(\text{n}\alpha)\text{Na}^{24}$ at 8.1 Mev E_{eff} . These two sets of measurements were taken at different times and even with different plug geometries in the hole although, of course, the comparisons are at the same positions. The measurements with Np^{237} are probably the most accurate group of all of those performed. The cross section is very well known,⁽¹⁴⁾ the probes were calibrated with meticulous care and the weights of fissionable material used were obtained through α counting.

Single measurements made on the $4\pi\gamma$ -ray ionization chamber are reproducible to within 2 to 5%,⁽¹⁵⁾ and, of course, these contribute very little error to the decay curve because of the large number of observations obtained for each curve. There is, however, an estimated calibration error of $\pm 10\%$ based on cobalt 60 standards.⁽¹⁵⁾ The errors associated with sensitivity of the chamber to energy are $\pm 5\%$. Estimated single error for the $4\pi\beta$ - counter is $\pm 7\%$.⁽¹⁵⁾

In the use of threshold detectors to derive spectra different from fission the procedure at present is to treat the cross section of each detector as though it were accurately represented by a step function whose threshold energy is the effective energy derived using a fission spectrum (see discussion or method). This introduces an error which depends on the shape of the modified spectrum. A complete error analysis for this approach has never been satisfactorily performed nor for that matter has a valid method been rigorously developed for measuring spectra from activation data. The present experiment does not concern itself with this variety of error since the evidence indicates that all spectra encountered here are fission above 0.6 Mev.

(14) D. J. Hughes and J. A. Harvey, Neutron Cross Sections, BNL-325, p 307.

(15) R. H. Lewis, Private Communication, 10/28/55.

The epithermal region is perhaps more easily charted than the threshold interval since cadmium ratio measurements do not require absolute counting. Here the most stringent requirements on counting are reproducibility of counter efficiency and geometry.

The major error in the resonance measurements depends on the original measurements made by the Argonne group in 1946.⁽¹⁶⁾ However, these detectors have been used subsequently to investigate epithermal spectra⁽¹⁷⁾ and the results from these indicate that data can be reproduced within 80% in regions of strong moderation and weak absorption.

In conclusion, it is thought that flux traverse absolute values based on Np^{237} were good to within 90% and that resonance and threshold (np and $\text{n}\alpha$) traverses were valid within 80% except for those materials which have already been excepted in the text.

Discussion of Method

A comprehensive description of threshold detector reactions together with a semi-quantitative method for predicting reaction rates to be expected in a reactor is given in Hughes File Neutron Research, Chapter 4.

A few pertinent definitions and explanations for the present report are made here for the sake of clarity and completeness.

A threshold detector as referred to in this report is ideally a nuclear reaction which has a zero cross section below its threshold energy and a constant value for all energies above the threshold energy. Such an ideal nuclear reaction does not exist in practice but a few reaction cross sections do actually approach this situation, especially the endothermic fission processes for thorium, actinium, uranium, and neptunium.⁽¹⁴⁾ Two non-fissionable materials, P^{31} and S^{32} , have

(16) S. M. Dancoff, et al., Activation Cross Sections by Boron Absorption, Argonne National Laboratory, CP-3781 (May 6, 1947).

(17) J. B. Trice, F. J. Muckenthaler, F. W. Smith and E. B. Johnson, Two Neutron Energy Measurements in the Bulk Shielding Facility Using Radioactants, ORNL-CF-53-5-139, Oct. 29, 1953.

(n,p) cross sections which approach this shape sufficiently well to serve as threshold detectors for pile neutron spectral studies. Others either deviate greatly or else have not been sufficiently well investigated. In general, lighter elements tend to show more pronounced resonance structure and heavier elements tend to approach more nearly the threshold detector form.

To calculate an effective energy threshold for capture of a neutron, followed by the emission of either an alpha particle or a proton, use is made of the equation

$$\int_0^{\infty} \sigma(E) nv(E) dE = \overline{\sigma}_{\text{fission}} \int_0^{\infty} nv(E) dE = \sigma_0 \int_{E_{\text{eff}}}^{\infty} nv(E) dE \quad (1)$$

where activation of a piece of material by means of a certain threshold reaction is proportional to any one of the three expressions for the activation integral. In the first integral the activation function is the product of the cross section times the neutron spectral function and is formed at each point in energy. In the second integral the total number of neutrons in the bombarding spectrum is expressed as the integral over neutrons of all energies and is multiplied by the cross section for the reaction averaged over the entire fission spectrum. This cross section, $\overline{\sigma}_{\text{fission}}$, is the one quoted by Hughes⁽¹⁸⁾ for many materials examined at the ANL. The third integral is expressed as the product of a constant cross section times the integral of all neutrons above an effective threshold E_{eff} . This is the artifice used to interpret actual cross sections as though they were truly threshold or step functions. All threshold type cross section data used in this report were converted to this ideal form. It is noteworthy that both highly resolved energy dependent expressions for a cross section and the cross sections

(18) See Reference 2, page 100.

measured with fission neutrons are expressed in this common form. For the case in which functional dependence on energy is listed σ_0 is chosen, somewhat arbitrarily, as being the "best" value of the function over the flat portion just following the initial rise region; that is, the region of penetration of the nuclear coulombic barrier. The value for E_{eff} can then be easily obtained graphically. In fact, this is illustrated in Figs. 28 through 32 for fission of Th^{232} , U^{236} , U^{238} , Np^{237} and for (np) reactions in S^{32} and P^{31} . When only $\overline{\sigma}_{\text{fission}}$ is listed E_{eff} is calculated by assuming that the cross section rise is proportional to the probability of the charged particles' penetration of the coulombic potential barrier. This is given by the expression⁽¹⁹⁾

$$P(E) = e^{-\frac{4z Ze^2}{vh}} \left[\arccos \sqrt{x} - \sqrt{x(1-x)}^{1/2} \right] \quad (2)$$

where

$z = 1$ for proton, 2 for alpha

$Z =$ nuclear charge of final nucleus

$e =$ electronic charge

$v =$ particle velocity

$x = E'/B$

$E' =$ particle energy $= (E - E_T)$

$B =$ Barrier height $= \frac{z Ze^2}{r}$ ergs

$E =$ neutron energy

$E_T =$ True threshold energy

(19) H. A. Bethe, Rev. Mod. Phys. 2, 69 (1937), p 163.

Fig. 28. The Activation Sensitivity Function $\sigma_f(E) n_{v_f}(E)$ vs E for Np^{237} .

Fig. 29. The Activation Sensitivity Function $\sigma_f(E) n \nu_f(E)$ vs E for U^{238} .

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Fig. 30. The Activation Sensitivity Function $\sigma_f(E) n v_f(E)$ vs E for Th^{232} .

Fig. 31. The Activation Sensitivity Function $\sigma_f(E)nv_f(E)$ for the Reaction $\text{P}^{31}(\text{n},\text{p})\text{Si}^{31}$.

Fig. 32. The Activation Sensitivity Function $\sigma_f(E) n v_f(E)$ for the Reaction $S^{32}(n,p)P^{32}$.

when this expression for the cross section is used one obtains

$$\int_0^{\infty} P(E) \, nv(E) dE = \int_{E_{\text{eff}}}^{\infty} nv(E) dE \quad (3)$$

from which one calculates the effective threshold energy E_{eff} . This equation is entirely equivalent to the expression

$$\int_0^{\infty} \sigma(E) \, nv(E) dE = \sigma_0 \int_{E_{\text{eff}}}^{\infty} nv(E) dE \quad (4)$$

where $\sigma(E)$ is a function which approaches σ_0 at saturation just as $P(E)$ approaches unity. In the cases where $\overline{\sigma}_{\text{fission}}$ is listed σ_0 is calculated from equation 4 where E_{eff} has already been obtained by use of equation 3.

If two threshold reactions have different effective threshold energies then the difference between the fluxes as measured by each one and based on their σ_0 's should be approximately the number of neutrons between the two threshold energies. Apparently, if one were given a large number of detectors with threshold energies sufficiently close one could measure the number of neutrons lying within any energy interval with high resolution and from this one could derive a very realistic neutron spectrum. However, errors which depend in some unknown way on the "effective" threshold energy in a spectrum differing from fission would probably invalidate such resolution. It is thought that small changes in the spectral function probably do not introduce a serious error in the values for E_{eff} used. Thus, for slightly modified spectra the presently described method is perhaps adequate. To simplify interpretation of the results in this report neutron fluxes are always expressed as the integral number of neutrons $\text{cm}^{-2} \cdot \text{sec}^{-1}$ above the E_{eff} as a function of E_{eff} .

For measuring epithermal neutron flux reference is made to the method described by Dancoff, et al.⁽¹⁶⁾ The neutron flux spectral function, if obeying $(E)^{-1}$ proportionality, is given by $\frac{\phi_0}{E}$ neutrons $\text{cm}^{-2} \text{sec}^{-1} \text{Mev}^{-1}$ where ϕ_0 is a constant describing the flux "amplitude" and is capable of measurement with any one of the resonance detectors listed in Table C. The expression for ϕ_0 is

$$\phi_0 = \frac{(nv)_{\text{th}} 1.43}{(CR - 1)(1 + \alpha)} \quad (5)$$

where

$(nv)_{\text{th}}$ = thermal neutron flux

CR = cadmium ratio

$$\alpha = \frac{\int_{0.4}^{\infty} \sigma_{(\text{res})}(E) nv(E) dE}{\int_{0.4}^{\infty} \sigma\left(\frac{1}{v}\right)(E) nv(E) dE} \quad (6)$$

The procedure used to measure whether or not the epithermal flux is contained in an $(E)^{-1}$ type of spectral function is to measure ϕ_0 at several different energies in the neutron field. This is accomplished by using resonance detectors with resonances located at the desired energies (See Table C). ϕ_0 is then determined for a particular resonance detector from its cadmium ratio as measured in the spectrum together with its value of α which is a measure of the resonance activation integral. The thermal flux $(nv)_{\text{th}}$ necessary to the computation is usually determined incidentally from one of the bare sample exposures made while obtaining the cadmium ratio.

If the values for ϕ_0 as measured with several different resonance detectors are equal it can be assumed that this value is the most probable one with which to describe the $(E)^{-1}$ part of the neutron spectrum.

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13. See Equation (2).
14. D. J. Hughes and J. A. Harvey, Neutron Cross Sections, BNL-325, p. 307
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18. See Reference 2, page 100.

19. H. A. Bethe, Rev. Mod. Phys. 2, 69 (1937), p. 163.

TABLE A
KEY TABLE OF FISSIONABLE MATERIALS USED

Material	Effective Threshold	Cross Section Saturation (10^{-24} cm ²)	$N_0\sigma$ (10^{-6})	Table Reference
Neptunium (237)	0.74	1.50	3.81	A-2
Uranium (238)	1.3 ^(a)	0.54	1.365	A-3
Thorium (232)	2.0	0.0598	0.155	A-4

Fission Chamber Ionization Volume Efficiency = 0.98

Literature Reference: See Report Reference No. 14.

(a) D. J. Hughes and J. A. Harvey, Neutron Cross Sections, BNL 325, p. 304

Table A-1 A Summary of Fissionable Materials Weight Determinations

Material	Identified	1st wt. (1)	Date	2nd wt. (2)	Date	Source	3rd wt. (3)
U ²³⁶	F	100.6 μ g	5-29-54	89.2 g	5-19-55	Sites & Gilp.	87.8 μ g
	G	10.2 μ g	-	-	-	-	9.84 μ g
	H	1.5 μ g	-	-	-	-	1.51 μ g
U ²³⁸	A	2.3 μ g	3-9-54	-	-	-	-
	B	11.3 μ g	-	-	-	-	33.8 μ g
Np ²³⁷	C	0.93 μ g	-	-	-	-	9.7 μ g
	D	9.30 μ g	-	-	-	-	96.1 μ g
Th ²³²	E	96.21 μ g	-	-	-	-	-
U ²³⁸	Plate 1	96 μ g	5-12-53	-	-	-	-
	Plate 2	- μ g	-	-	-	-	-
Np ²³⁷	-	853 μ g (d/m = 15223.07)	4-16-53	-	-	-	-
Th ²³²	Ni I	23 μ g	10-15-53	-	-	-	-
	II	26 μ g	-	-	-	-	-
U ²³⁸	-	82.0 μ g	-	58.1 μ g	5-19-55	Sites & Gilp.	242 μ g
U ²³⁵	A	100 μ g	5-19-55	96.6 μ g	5-19-55	-	89.0 μ g
	B	100 μ g	5-19-55	100.8 μ g	5-19-55	-	119.3 μ g
	C	100 μ g	5-19-55	95.1 μ g	5-19-55	-	97 μ g
	D	100 μ g	5-19-55	77.7 μ g	5-19-55	-	77.2 μ g

(1) Determined by H. O. Finley, N. M. Goodwin, and F. Kelly, Y-12 Technical Division, Special Assay Group.

(2) Determined by J. R. Sites, L. O. Gilpatrick, Y-12 Stable Isotope Research and Pr. Div., Mass Spectrometry Dept.

(3) Determined by J. B. Trice, ORNL, Solid State Division (Alpha Counting Analysis).

TABLE A-2

Np²³⁷ Fission Chamber Threshold Detector Data

Position (Inches from End of HB-3 Beam Hole)	File Power (kw)	File Power Normal Factor	Counts/min	Sample Weight (μ g)	Flux (Normal to 30 mw)
6.6"	590	50.9	275000	9.7	$6.4 \cdot 10^{12}$
12.6"	590	50.9	475000	9.7	$1.11 \cdot 10^{12}$
18.6"	590	50.9	10800	9.7	$2.52 \cdot 10^{11}$
24.6"	590	50.9	4170	9.7	$9.7 \cdot 10^{10}$
30.6"	590	50.9	1990	9.7	$4.64 \cdot 10^{10}$
36.6"	590	50.9	1156	9.7	$2.69 \cdot 10^{10}$
42.6"	590	50.9	721	9.7	$1.68 \cdot 10^{10}$
48.6"	590	50.9	473	9.7	$1.13 \cdot 10^{10}$
54.6"	590	50.9	355	9.7	$8.12 \cdot 10^9$
48.6"	30000	1.0	246000	96.1	$1.13 \cdot 10^{10}$
54.6"	30000	1.0	175000	96.1	$8.12 \cdot 10^9$
60.6"	30000	1.0	130000	96.1	$6.02 \cdot 10^9$
66.6"	30000	1.0	99400	96.1	$4.61 \cdot 10^9$
72.6"	30000	1.0	82000	96.1	$3.81 \cdot 10^9$
78.6"	30000	1.0	72400	96.1	$3.36 \cdot 10^9$
84.6"	30000	1.0	60000	96.1	$2.78 \cdot 10^9$
90.6"	30000	1.0	53000	96.1	$2.46 \cdot 10^9$
96.6"	30000	1.0	44400	96.1	$2.06 \cdot 10^9$

TABLE A-2 (Cont.)

Np²³⁷ Fission Chamber Threshold Detector Data

Position (Inches from End of HB-3 Beam Hole)	Pile Power (kw)	Pile Power Normal Factor	Counts/min	Sample Weight (μ g)	Flux (Normal to 30 mw)
102.6"	30000	1.0	37300	96.1	$1.73 \cdot 10^9$
108.6"	30000	1.0	33400	96.1	$1.55 \cdot 10^9$
114.6"	30000	1.0	29300	96.1	$1.36 \cdot 10^9$
120.6"	30000	1.0	26500	96.1	$1.23 \cdot 10^9$
126.6"	30000	1.0	22600	96.1	$1.05 \cdot 10^9$
132.6"	30000	1.0	20600	96.1	$9.57 \cdot 10^8$
138.6"	30000	1.0	18400	96.1	$8.55 \cdot 10^8$
144.6"	30000	1.0	17200	96.1	$8.0 \cdot 10^8$
150.6"	30000	1.0	15700	96.1	$7.30 \cdot 10^8$
156.6"	30000	1.0	16700	96.1	$7.75 \cdot 10^8$
162.6"	30000	1.0	18670	96.1	$8.66 \cdot 10^8$

TABLE A-3

U²³⁸ Fission Chamber Threshold Detector Data

Position (Inches from End of HB-3 Beam Hole)	Pile Power (kw)	Pile Power Normal Factor	Counts/min	Sample Weight (μ g)	Flux (Normal to 30 mw)
42.6"	30000	1.0	90000	242	$5.02 \cdot 10^9$
48.6"	30000	1.0	66000	242	$3.38 \cdot 10^9$
54.6"	30000	1.0	47600	242	$2.44 \cdot 10^9$
66.6"	30000	1.0	27300	242	$1.40 \cdot 10^9$
72.6"	30000	1.0	22000	242	$1.13 \cdot 10^9$
78.6"	30000	1.0	19000	242	$9.74 \cdot 10^8$
84.6"	30000	1.0	16200	242	$8.31 \cdot 10^8$
90.6"	30000	1.0	13500	242	$6.90 \cdot 10^8$
96.6"	30000	1.0	11600	242	$5.95 \cdot 10^8$
102.6"	30000	1.0	10100	242	$5.17 \cdot 10^8$
108.6"	30000	1.0	8980	242	$4.61 \cdot 10^8$
114.6"	30000	1.0	7980	242	$4.09 \cdot 10^8$
120.6"	30000	1.0	7220	242	$3.70 \cdot 10^8$
126.6"	30000	1.0	6500	242	$3.33 \cdot 10^8$
132.6"	30000	1.0	5800	242	$2.97 \cdot 10^8$
138.6"	30000	1.0	5260	242	$2.70 \cdot 10^8$
144.6"	30000	1.0	4888	242	$2.50 \cdot 10^8$
150.6"	30000	1.0	4500	242	$2.30 \cdot 10^8$
156.6"	30000	1.0	4150	242	$2.13 \cdot 10^8$
162.6"	30000	1.0	3420	242	$1.75 \cdot 10^8$

TABLE A-4
 Th^{232} Fission Chamber Threshold Detector Data

Position (Inches from End of HB-3 Beam Hole)	Pile Power (kw)	Pile Power Normal Factor	Counts/min	Sample Weight (μg)	Flux (Normal to 30 mw)
6.6"	300	1.0	71000	96.2	$3.24 \cdot 10^{10}$
12.6"	300	1.0	11600	96.2	$5.28 \cdot 10^9$
18.6"	300	1.0	2820	96.2	$1.28 \cdot 10^9$
24.6"	300	1.0	1152	96.2	$5.24 \cdot 10^8$
30.6"	300	1.0	569	96.2	$2.58 \cdot 10^8$
36.6"	300	1.0	339	96.2	$1.54 \cdot 10^8$
42.6"	300	1.0	216	96.2	$9.81 \cdot 10^7$
48.6"	300	1.0	128	96.2	$5.84 \cdot 10^7$
54.6"	300	1.0	115	96.2	$5.22 \cdot 10^7$

TABLE B
KEY TABLE OF (np) AND (n γ) THRESHOLD REACTIONS

Reaction	Thresh- old Energy	Gross Section (Satura- tion) (x 10 ⁻²⁴)	Half Life of Product	N σ (10 ⁻⁷)	Table	Type Counter	Effic.
P ³¹ (np)Si ³¹	2.4	0.075 ^(a)	2.62h	4.05	B-1	4 $\pi\beta^-$	0.98
S ³² (np)P ³²	2.9	0.300 ^(b)	14.3 d	53.6	B-2	4 $\pi\beta^-$	0.98
Al ²⁷ (np)Mg ²⁷	4.6	0.039 ^(c)	9.8 m	8.79	B-3	4 $\pi\gamma$	1.61 x 10 ^{6*}
Ni ⁵⁸ (np)Co ⁵⁸	5.0	1.23 ^(c)	72 d	125.0	B-4	4 $\pi\gamma$	1.47 x 10 ⁶
Si ²⁸ (np)Al ²⁸	5.5	0.080 ^(c)	2.3 m	7.35	B-5	4 $\pi\gamma$	1.2 x 10 ⁶
Mg ²⁴ (np)Na ²⁴	6.3	0.048 ^(c)	14.9 h	9.43	B-6	4 $\pi\gamma$	0.47 x 10 ⁶
Al ²⁷ (n α)Na ²⁴	8.1	0.111 ^(c)	14.9 h	24.7	B-7	4 $\pi\gamma$	0.47 x 10 ⁶

* Efficiencies for the γ -ray ionization chamber are in β^- disintegrations per volt.

- (a) D. J. Hughes and J. A. Harvey, Neutron Cross Sections, BNL 325, p. 105
- (b) D. J. Hughes and J. A. Harvey, Neutron Cross Sections, BNL 325, p. 108 (0.0 Mev to 3.5 Mev)
E. D. Klema and A. O. Hanson, Phys. Rev. 70; 106 (3.0 Mev to 6.0 Mev)
- (c) N. Goldstein et al., ANL Quar. Prog. Rep., July 26, 1946, CF-3574, p. 29.

TABLE B-1
REACTION: $P^{31}(np)Si^{31}$

Sample No. Position	Cont. Rods Avg. Pos. (in)	Date of Irrad.	Pile Power (MW)	Pile Power Norm. Factor	Volts/mg 10^{10} Scale	Irrad. Time (t)	Saturation Factor $1 - e^{-\lambda t}$	Flux ϕ (nv)
E1B-R11	18.857	8-26-54	30	1.0	$1.25 \cdot 10^{13}$	75m	$2.82 \cdot 10^{-1}$	$1.09 \cdot 10^{13}$
E1B-R14A	18.857	8-26-54	30	1.0	$3.26 \cdot 10^5$	75m	$2.82 \cdot 10^{-1}$	$2.86 \cdot 10^{12}$
E1B-R14B	18.857	8-26-54	30	1.0	$3.33 \cdot 10^5$	75m	$2.82 \cdot 10^{-1}$	$2.92 \cdot 10^{12}$
E1A-R14(1)	18.857	8-26-54	30	1.0	$4.17 \cdot 10^5$	1.51 hr	$3.29 \cdot 10^{-1}$	$3.13 \cdot 10^{12}$
E1A-R14(2)	18.857	8-26-54	30	1.0	$5.0 \cdot 10^5$	1.51 hr	$3.29 \cdot 10^{-1}$	$3.76 \cdot 10^{12}$
E1A-R1-42	18.857	8-26-54	30	1.0	$3.68 \cdot 10^5$	1.51 hr	$3.29 \cdot 10^{-1}$	$2.77 \cdot 10^{12}$
E1A-R2-4(1)	18.857	8-26-54	30	1.0	$8.82 \cdot 10^4$	1.51 hr	$3.29 \cdot 10^{-1}$	$6.64 \cdot 10^{11}$
E1A-R2-4(2)	18.857	8-26-54	30	1.0	$8.75 \cdot 10^4$	1.51 hr	$3.29 \cdot 10^{-1}$	$6.58 \cdot 10^{11}$
E1A-R34	18.385	8-25-54	30	1.0	$1.40 \cdot 10^4$	1 hr	$2.33 \cdot 10^{-1}$	$1.49 \cdot 10^{11}$

TABLE B-2

REACTION: $S^{32}(np)P^{32}$

Sample No. Position	Date of Irrad.	File Power (MW)	File Power Norm. Factor	Volts/mg 10^{10} Scale	Irrad. Time (t)	Saturation Factor $1 - e^{-\lambda t}$	Flux ϕ (nv)
B2-R11	6-17-55	30	1.0	$2.22 \cdot 10^3$	100 sec	$5.41 \cdot 10^{-5}$	$7.4 \cdot 10^{12}$
B2-R14	6-17-55	30	1.0	$2.50 \cdot 10^2$	100 sec	$5.61 \cdot 10^{-5}$	$8.33 \cdot 10^{11}$
B2-R22	6-17-55	30	1.0	$7.57 \cdot 10$	100 sec	$5.61 \cdot 10^{-5}$	$2.52 \cdot 10^{11}$
B4-R32	6-19-55	30	1.0	$2.67 \cdot 10$	300 sec	$1.68 \cdot 10^{-4}$	$2.97 \cdot 10^{10}$
B4-R41	6-19-55	30	1.0	8.59	300 sec	$1.68 \cdot 10^{-4}$	$9.56 \cdot 10^9$
B4-R44	6-19-55	30	1.0	4.96	300 sec	$1.68 \cdot 10^{-4}$	$5.52 \cdot 10^9$

TABLE B-3

REACTION: $\text{Al}^{27}(\text{np})\text{Mg}^{27}$

Sample No. Position	Cont. Rods Avg. Pos. (in)	Date of Irrad.	Pile Power (MW)	Pile Power Norm. Factor	Volts/mg 10^{10} Scale	Irrad. Time (t)	Saturation Factor $1 - e^{-\lambda t}$	Flux ϕ (nv)
D1A-R11	22.362	7-23-54	30	1.0	$1.64 \cdot 10^{-1}$	2m	$1.32 \cdot 10^{-1}$	$2.27 \cdot 10^{12}$
D1A-R14	22.362	7-23-54	30	1.0	$2.41 \cdot 10^{-2}$	2m	$1.32 \cdot 10^{-1}$	$3.36 \cdot 10^{11}$
D1A-R24	22.362	7-23-54	30	1.0	$1.94 \cdot 10^{-2}$	10m	$5.07 \cdot 10^{-1}$	$7.04 \cdot 10^{10}$
D1A-R44	22.362	7-23-54	30	1.0	$2.26 \cdot 10^{-3}$	40m	$9.41 \cdot 10^{-1}$	$4.4 \cdot 10^9$
D1A-R34	22.362	7-23-54	30	1.0	$3.12 \cdot 10^{-3}$	20m	$7.57 \cdot 10^{-1}$	$7.56 \cdot 10^9$
D1B-R11	22.805	7-24-54	30	1.0	$1.13 \cdot 10^{-1}$	1m	$6.83 \cdot 10^{-2}$	$3.02 \cdot 10^{12}$
D1B-R14	22.805	7-24-54	30	1.0	$5.85 \cdot 10^{-2}$	5m	$2.97 \cdot 10^{-1}$	$3.61 \cdot 10^{11}$
D1B-R24	22.805	7-24-54	30	1.0	$1.73 \cdot 10^{-2}$	11m	$5.44 \cdot 10^{-1}$	$5.82 \cdot 10^{10}$
D1B-R34	22.805	7-24-54	30	1.0	$3.65 \cdot 10^{-3}$	20m	$7.57 \cdot 10^{-1}$	$8.82 \cdot 10^9$
D1B-R44	22.805	7-24-54	30	1.0	$1.68 \cdot 10^{-3}$	40m	$9.41 \cdot 10^{-1}$	$3.27 \cdot 10^9$
D1C-R11	22.238	8-13-54	30	1.0	$6.32 \cdot 10^{-1}$	10m	$5.07 \cdot 10^{-1}$	$2.29 \cdot 10^{12}$
D1C-R14	22.802	8-14-54	30	1.0	$1.77 \cdot 10^{-1}$	33.3m	$9.05 \cdot 10^{-1}$	$3.59 \cdot 10^{11}$
D1C-R24	22.238	8-13-54	30	1.0	$1.27 \cdot 10^{-2}$	30m	$8.8 \cdot 10^{-1}$	$2.64 \cdot 10^{10}$
D1D-R11			30	1.0	$2.50 \cdot 10^{-1}$	10m	$5.07 \cdot 10^{-1}$	$9.03 \cdot 10^{11}$
D1D-R24			30	1.0	$2.44 \cdot 10^{-2}$	33.3m	$9.05 \cdot 10^{-1}$	$4.94 \cdot 10^{10}$
D1D-R41			30	1.0	$3.22 \cdot 10^{-3}$	30m	$8.8 \cdot 10^{-1}$	$6.69 \cdot 10^9$

TABLE B-4

DATA FOR THE REACTION $\text{Ni}^{58}(\text{n}, \text{p})\text{Co}^{58}$

Position	(nv) _f Above 8.1 Mev $\text{Al}^{27}(\text{n}, \alpha)\text{Na}^{24}$	(nv) _f Above Thermal from the reaction $\text{Al}^{27}(\text{n}, \alpha)\text{Na}^{24}$	(nv) _f Above Thermal from the reaction* $\text{Al}^{27}(\text{n}, \alpha)\text{Na}^{24}$	Cross Section for the reaction $\text{Ni}^{58}(\text{n}, \text{p})\text{Co}^{58}$ *		(nv) _f Above Thermal from the reaction $\text{Ni}^{58}(\text{n}, \text{p})\text{Co}^{58}$	(nv) _f Above Effec. Enrg. Thresh. (5.0 Mev) Based on $\sigma_{\text{fiss.}} =$ 66.4 mb	(nv) _f Above Eff. Enrg. Threshold (2.9 Mev) *
	($\times 10^{11}$)	($\times 10^{13}$)	($\times 10^{13}$)	(MTR) (Milli- barns)	(ORNL) (Milli- barns)	($\times 10^{13}$)		
R11	1.7	3.15	4.17	88.0	66.4	3.15	1.67×10^{12}	7.56×10^{12}
R12	0.85	1.57	2.16	83.6	60.7	1.72	9.12×10^{11}	
R13	0.46	0.851	1.28	72.5	48.2	1.17	6.2×10^{11}	
R14	0.24	0.444	0.634	71.4	49.6	.595	3.15×10^{11}	8.5×10^{11}
R21	0.135	0.250	0.426	53.6	31.5	.527	2.79×10^{11}	
R22	0.075	0.139	0.222	57.7	36.1	.256	1.36×10^{11}	2.59×10^{11}
R23	0.045	0.0834	0.124	57.4	38.5	.144	7.64×10^{10}	
R24	0.029	0.0538	0.0717	52.2	39.1	.0914	4.84×10^{10}	
R32	0.012	0.0222	0.026	57.5	49.1	.030	1.59×10^{10}	3.04×10^{10}
R33			0.0157					
R34	0.0055							
R41	0.0036	0.00666	0.0085	47.1	35.5	.0125	6.63×10^9	9.76×10^9
R42								
R44	0.0017	0.00315	0.00264	64.5	77(?)	.0022	1.44×10^9	5.63×10^9

*Data was obtained and analyzed by Phillips Petroleum Company (MTR).

TABLE B-5
 REACTION: $\text{Si}^{28}(\text{np})\text{Al}^{28}$

Sample No. Position	Cont. Rods Avg. Pos. (in)	Date of Irrad.	Pile Power (MW)	Pile Power Norm. Factor	Volts/mg 10^{10} Scale	Irrad. Time (t)	Saturation Factor $1 - e^{-\lambda t}$	Flux ϕ (nv)
I1A-R11	22.802	8-14-54	30	1.00	1.36	12m	0.973	$2.28 \cdot 10^{12}$
I1A-R14	22.802	8-14-54	30	1.00	$1.76 \cdot 10^{-1}$	12m	0.973	$2.95 \cdot 10^{11}$
I1A-R24	22.802	8-14-54	30	1.00	$2.15 \cdot 10^{-2}$	12m	0.973	$3.61 \cdot 10^{10}$
I1A-R32	22.802	8-14-54	30	1.00	$8.14 \cdot 10^{-3}$	12m	0.973	$1.37 \cdot 10^{10}$
I2A-R11	15.416	8-19-54	10	3.00	$4.11 \cdot 10^{-1}$	10m	0.973	$2.12 \cdot 10^{12}$
I2A-R14	15.416	8-19-54	10	3.00	$6.71 \cdot 10^{-2}$	10m	0.973	$3.46 \cdot 10^{11}$
I3A-R11	16.200	8-19-54	20	1.5	1.05	12m	0.973	$2.65 \cdot 10^{12}$
I3A-R14	16.200	8-19-54	20	1.5	$1.71 \cdot 10^{-1}$	12m	0.973	$4.31 \cdot 10^{11}$
I3A-R24	16.200	8-19-54	20	1.5	$2.07 \cdot 10^{-2}$	10m	0.973	$5.34 \cdot 10^{10}$
I3A-R32	16.200	8-19-54	20	1.5	$9.14 \cdot 10^{-3}$	10m	0.973	$2.35 \cdot 10^{10}$

TABLE B-6
REACTION $\text{Mg}^{24}(\text{np})\text{Na}^{24}$

Sample No. Position	Cont. Rods Avg. Pos. (in)	Date of Irrad.	Pile Power (MW)	Pile Power Norm. Factor	Volts/mg 10^{10} Scale	Irrad. Time (t)	Saturation Factor $1 - e^{-\lambda t}$	Flux ϕ (nv)
A1-R11	19.931	7-17-54	30	1.00	$1.31 \cdot 10^{-2}$	10m	$7.75 \cdot 10^{-3}$	$8.5 \cdot 10^{11}$
A1-R14	19.931	7-17-54	30	1.00	$2.11 \cdot 10^{-3}$	10m	$7.75 \cdot 10^{-3}$	$1.36 \cdot 10^{11}$
A1-R23	19.931	7-17-54	30	1.00	$3.82 \cdot 10^{-4}$	10m	$7.75 \cdot 10^{-3}$	$2.45 \cdot 10^{10}$
A1B-R11	20.703	7-19-54	30	1.00	$1.31 \cdot 10^{-2}$	10m	$7.75 \cdot 10^{-3}$	$8.4 \cdot 10^{11}$
A1C-R23	21.508	7-21-54	30	1.00	$1.74 \cdot 10^{-3}$	75m	$4.5 \cdot 10^{-2}$	$1.51 \cdot 10^{10}$
A1C-R32	21.508	7-21-54	30	1.00	$5.24 \cdot 10^{-4}$	75m	$4.5 \cdot 10^{-2}$	$4.60 \cdot 10^9$
A1C-R41	21.508	7-21-54	30	1.00	$1.71 \cdot 10^{-4}$	75m	$4.5 \cdot 10^{-2}$	$1.50 \cdot 10^9$
A1C-R44	21.508	7-21-54	30	1.00	$6.82 \cdot 10^{-5}$	75m	$4.5 \cdot 10^{-2}$	$5.98 \cdot 10^8$
A1D-R41	21.508	7-21-54	30	1.00	$1.78 \cdot 10^{-4}$	75m	$4.5 \cdot 10^{-2}$	$1.56 \cdot 10^9$
A1D-R44	21.508	7-21-54	30	1.00	$7.69 \cdot 10^{-5}$	75m	$4.5 \cdot 10^{-2}$	$6.75 \cdot 10^8$

TABLE B-7
REACTION $\text{Al}^{27}(\text{n}\alpha)\text{Na}^{24}$

Sample No. Position	Cont. Rods Avg. Pos. (in)	Date of Irrad.	Pile Power (MW)	Pile Power Norm. Factor	Volts/mg 10^{10} Scale	Irrad. Time (t)	Saturation Factor $1 - e^{-\lambda t}$	Flux ϕ (nv)
D1A-R11	22.362	7-23-54	30	1.0	$1.25 \cdot 10^{-3}$	2m	$1.54 \cdot 10^{-3}$	$1.55 \cdot 10^{11}$
D1A-R14	22.362	7-23-54	30 _g	1.0	$1.98 \cdot 10^{-4}$	2m	$1.54 \cdot 10^{-3}$	$2.45 \cdot 10^{11}$
D1A-R24	22.362	7-23-54	30	1.0	$1.37 \cdot 10^{-4}$	10m	$7.75 \cdot 10^{-3}$	$3.36 \cdot 10^9$
D1A-R44	22.362	7-23-54	30	1.0	$2.61 \cdot 10^{-5}$	40m	$3.05 \cdot 10^{-2}$	$1.63 \cdot 10^8$
D1A-R34	22.362	7-23-54	30	1.0	$4.42 \cdot 10^{-5}$	20m	$1.55 \cdot 10^{-2}$	$5.43 \cdot 10^8$
D1B-R11	22.805	7-24-54	30	1.0	$6.91 \cdot 10^{-4}$	1m	$7.7 \cdot 10^{-4}$	$1.71 \cdot 10^{11}$
D1B-R14	22.805	7-24-54	30	1.0	$5.14 \cdot 10^{-4}$	5m	$3.88 \cdot 10^{-3}$	$2.52 \cdot 10^{10}$
D1B-R24	22.805	7-24-54	30	1.0	$1.40 \cdot 10^{-4}$	11m	$8.52 \cdot 10^{-3}$	$3.12 \cdot 10^9$
D1B-R34	22.805	7-24-54	30	1.0	$5.70 \cdot 10^{-5}$	20m	$1.55 \cdot 10^{-2}$	$7.02 \cdot 10^8$
D1B-R44	22.805	7-24-54	30	1.0	$3.35 \cdot 10^{-5}$	40m	$3.05 \cdot 10^{-2}$	$2.09 \cdot 10^8$
D1C-R11	22.238	8-13-54	30	1.0	$7.03 \cdot 10^{-3}$	10m	$7.75 \cdot 10^{-3}$	$1.73 \cdot 10^{11}$
D1C-R14	22.802	8-14-54	30	1.0	$3.14 \cdot 10^{-3}$	33.3m	$2.55 \cdot 10^{-2}$	$2.35 \cdot 10^{10}$
D1C-R24	22.238	8-13-54	30	1.0	$3.1 \cdot 10^{-4}$	30m	$2.3 \cdot 10^{-2}$	$2.57 \cdot 10^9$
D1D-R11			30	1.0	$7.59 \cdot 10^{-3}$	10m	$7.75 \cdot 10^{-3}$	$1.87 \cdot 10^{11}$
D1D-R14			30	1.0	$4.12 \cdot 10^{-3}$	33.3m	$2.55 \cdot 10^{-2}$	$3.08 \cdot 10^{10}$
D1D-R24			30	1.00	$3.97 \cdot 10^{-4}$	33.3m	$2.55 \cdot 10^{-2}$	$2.97 \cdot 10^9$
D1D-R31			30	1.00	$2.19 \cdot 10^{-3}$	30m	$2.3 \cdot 10^{-2}$	$1.81 \cdot 10^{10}$
D1D-R41			30	1.00	$4.21 \cdot 10^{-5}$	30m	$2.3 \cdot 10^{-2}$	$3.49 \cdot 10^8$
D2C-R14	16.810	8-20-54	20	1.5	$5.69 \cdot 10^{-3}$	60m	$4.54 \cdot 10^{-2}$	$3.57 \cdot 10^{10}$
D2C-R24	16.810	8-20-54	20	1.5	$4.79 \cdot 10^{-3}$	60m	$4.54 \cdot 10^{-2}$	$3.02 \cdot 10^{10}$
D2C-R11	16.810	8-20-54	20	1.5	$3.12 \cdot 10^{-2}$	60m	$4.54 \cdot 10^{-2}$	$1.97 \cdot 10^{11}$

TABLE C
LIST OF NUCLEAR RESONANCE REACTIONS

Reaction $X^2(n,\gamma)X^{+1}$	Resonance Energy (ev)	Resonance Activation Integral (α_2)	Chemical Form	Ref. in This Report
Cobalt	120		Metal	C-1, C-7
Copper	570	0.933	Metal	C-2
Sodium	1710	0.238	NaCl	C-3
Chlorine	1800	0.079	NaCl	C-4
Vanadium	3000	0.134	Metal	C-5
Aluminum	9100	0.159	Metal	C-6

TABLE C-1

REACTION $\text{Co}^{59}(\text{n}\gamma)\text{Co}^{60}$

Sample No. Position	Cont. Rods Avg. Pos. (in)	Date of Irrad.	Pile Power (MW)	Pile Power Norm. Factor	Volts/mg 10^{10} Scale	Irrad. Time (t)	Saturation Factor $1 - e^{-\lambda t}$	Flux ϕ (nv)
A-R1-1	24.422	7/27/54	30	1.0	2.50×10^{-2}	1m	2.54×10^{-7}	2.2×10^{14}
A-R1-3	24.422	7/27/54	30	1.0	1.42×10^{-2}	1m	2.54×10^{-7}	1.25×10^{14}
A-R2-2	24.422	7/27/54	30	1.0	3.40×10^{-3}	1m	2.54×10^{-7}	2.99×10^{13}
A-R2-4	24.422	7/27/54	30	1.0	1.63×10^{-2}	10m	2.54×10^{-6}	1.45×10^{13}
A-R3-2	24.422	7/27/54	30	1.0	8.27×10^{-3}	10m	2.54×10^{-6}	7.34×10^{12}
A-R3-4	24.422	7/27/54	30	1.0	5.36×10^{-3}	10m	2.54×10^{-6}	4.76×10^{12}
A-R4-2	24.422	7/27/54	30	1.0	3.79×10^{-3}	10m	2.54×10^{-6}	3.36×10^{12}
A1-R3-1	19.931	7/17/54	30	1.0	1.03×10^{-4}	10.5m	2.67×10^{-6}	8.63×10^{10}
A1-R1-1	19.931	7/17/54	30	1.0	1.29×10^{-2}	10.5m	2.67×10^{-6}	1.08×10^{13}
A1C-R1-1	21.508	7/21/54	30	1.0	9.03×10^{-2}	75m	1.90×10^{-5}	1.06×10^{13}
A1C-R3-1	21.508	7/21/54	30	1.0	8.49×10^{-4}	75m	1.90×10^{-5}	9.98×10^{10}
A1D-R3-1	21.508	7/21/54	30	1.0	8.49×10^{-4}	75m	1.90×10^{-5}	9.98×10^{10}
A1D-R1-1	21.508	7/21/54	30	1.0	8.82×10^{-2}	75m	1.90×10^{-5}	1.18×10^{13}
A1B-R3-1	21.117	7/20/54	30	1.0	1.34×10^{-4}	10m	2.54×10^{-6}	1.18×10^{11}
D1C-R12	22.802	8/14/54	30	1.0	1.59×10^{-1}	10m	2.54×10^{-6}	1.37×10^{14}
D1C-R42	22.802	8/14/54	30	1.0				2.67×10^{12}
D1D-R12			30	1.0	6.28×10^{-1}	31.6m	8.01×10^{-6}	1.75×10^{14}
D1D-R32			30	1.0	2.08×10^{-2}	33.3m	8.44×10^{-6}	5.52×10^{12}
E1B-R12	18.857	8/26/54	30	1.0				6.39×10^{12}
E1B-R13	18.857	8/26/54	30	1.0				2.75×10^{12}
I1A-R12	22.802	8/14/54	30	1.0	1.88×10^{-1}	12m	3.05×10^{-6}	1.38×10^{14}
I2A-R43		8/19/54	10	3.0				2.51×10^{12}
I3A-R12	16.200	8/19/54	20	1.5	1.57×10^{-1}	12m	3.05×10^{-6}	1.73×10^{14}
I3A-R21	16.200	8/19/54	20	1.5	4.0×10^{-2}	12m	3.05×10^{-6}	4.4×10^{13}

TABLE C-1 (Cont.)

REACTION: $\text{Co}^{59}(\text{n}\gamma)\text{Co}^{60}$

Sample No. Position	Cont. Rods Avg. Pos. (in)	Date of Irrad.	Pile Power (MW)	Pile Power Norm. Factor	Volts/mg 10^{10} Scale	Irrad. Time (t)	Saturation Factor $1 - e^{-\lambda t}$	Flux ϕ (nv)
A2B-R1-1	21.907	7/22/54	30	1.0	2.88×10^{-2}	1m	2.54×10^{-7}	2.53×10^{14}
A2B-R3-1	21.907	7/22/54	30	1.0	9.34×10^{-4}	1m	2.54×10^{-7}	8.22×10^{12}
A2A-R1-1	21.117	7/20/54	30	1.0	1.40×10^{-1}	5m	1.27×10^{-6}	2.47×10^{14}
A2A-R3-1	21.117	7/20/54	30	1.0	4.73×10^{-3}	5m	1.27×10^{-6}	8.33×10^{12}
D2C-R12	16.810	8/20/54	30	1.0	8.81×10^{-1}	60m	1.52×10^{-5}	1.30×10^{14}
D2C-R32	16.810	8/20/54	30	1.0	3.13×10^{-2}	60m	1.52×10^{-5}	4.6×10^{12}
D2C-R42	16.810	8/20/54	30	1.0				2.93×10^{12}
H2-R1-2	23.360	7/25/54	30	1.0	1.73×10^{-2}	1m	2.54×10^{-7}	1.52×10^{14}
H2-R2-1	23.360	7/25/54	30	1.0	4.14×10^{-3}	1m	2.54×10^{-7}	3.64×10^{13}
H2-R3-1	23.360	7/25/54	30	1.0	9.38×10^{-4}	1m	2.54×10^{-7}	8.26×10^{12}
H2-R4-3	23.360	7/25/54	30	1.0	2.6×10^{-4}	1m	2.54×10^{-7}	2.29×10^{12}
B2-R22		6/17/55	30	1.0		100s		9.31×10^{11}
B4-R22		6/19/55	30	1.0		300s		1.41×10^{12}

TABLE C-2

REACTION: $\text{Cu}^{63}(\text{n}\gamma)\text{Cu}^{64}$

Sample No. Position	Cont. Rods Avg. Pos. (in)	Date of Irrad.	Pile Power (MW)	Pile Power Norm. Factor	Volts/mg 10^{10} Scale	Irrad. Time (t)	Saturation Factor $1 - e^{-\lambda t}$	Flux ϕ (nv)
A1B-R43	20.703	7/19/54	30	1	3.43×10^{-4}	10m	9.02×10^{-3}	6.33×10^{11}
A1B-R34	20.703	7/19/54	30	1	9.27×10^{-4}	10m	9.02×10^{-3}	1.71×10^{12}
A1B-R31	20.703	7/19/54	30	1	2.31×10^{-3}	10m	9.02×10^{-3}	4.26×10^{12}
A1B-R22	20.703	7/19/54	30	1	9.37×10^{-3}	10m	9.02×10^{-3}	1.73×10^{13}
A1B-R13	20.703	7/19/54	30	1	5.94×10^{-2}	10m	9.02×10^{-3}	1.10×10^{14}
A1C-R34	21.508	7/21/54	30	1	5.62×10^{-3}	75m	8.29×10^{-2}	1.13×10^{12}
A1C-R43	21.508	7/21/54	30	1	2.74×10^{-3}	75m	8.29×10^{-2}	5.5×10^{11}
A1-R13	19.931	7/17/54	30	1	6.27×10^{-2}	10.5m	9.46×10^{-3}	1.10×10^{14}
A1-R31	19.931	7/17/54	30	1	2.03×10^{-3}	10.5m	9.46×10^{-3}	3.57×10^{12}
A2A-R22	21.117	7/20/54	30	1	3.94×10^{-1}	5m	4.51×10^{-3}	1.45×10^{15}
A2A-R31	21.117	7/20/54	30	1	1.18×10^{-1}	5m	4.51×10^{-3}	4.36×10^{14}
A2A-R34	21.117	7/20/54	30	1	5.98×10^{-2}	5m	4.51×10^{-3}	2.21×10^{14}
A2A-R43	21.117	7/20/54	30	1	3.67×10^{-2}	5m	4.51×10^{-3}	1.35×10^{14}
A2A-R13	21.508	7/21/54	30	1	7.87×10^{-1}	5m	4.51×10^{-3}	2.90×10^{15}
A2B-R43	21.907	7/22/54	30	1	7.34×10^{-3}	1m	9.0×10^{-4}	1.36×10^{14}
A2B-R34	21.907	7/22/54	30	1	1.24×10^{-2}	1m	9.0×10^{-4}	2.29×10^{14}
A2B-R31	21.907	7/22/54	30	1	2.48×10^{-2}	1m	9.0×10^{-4}	4.58×10^{14}
A2B-R22	21.907	7/22/54	30	1	7.86×10^{-2}	1m	9.0×10^{-4}	1.45×10^{15}
A2B-R13	21.907	7/22/54	30	1	3.2×10^{-1}	1m	9.0×10^{-4}	5.92×10^{15}
A2C-R11		6/9/55	30	1	1.39	3.3m	3.0×10^{-3}	7.7×10^{15}
A2C-R14		6/9/55	30	1	3.58×10^{-1}	3.3m	3.0×10^{-3}	1.98×10^{15}
A2C-R22		6/9/55	30	1	1.33×10^{-1}	3.3m	3.0×10^{-3}	7.32×10^{14}
A2C-R32		6/9/55	30	1	1.36×10^{-1}	8m	7.83×10^{-3}	2.89×10^{14}
A2C-R41		6/9/55	30	1	8.06×10^{-2}	8m	7.83×10^{-3}	1.71×10^{14}
A2C-R44		6/9/55	30	1	4.8×10^{-2}	8m	7.83×10^{-3}	1.02×10^{14}

TABLE C-3

REACTION: $\text{Na}^{23}(\text{n}\gamma)\text{Na}^{24}$

Sample No. Position	Cont. Rods Avg. Pos. (in)	Date of Irrad.	Pile Power (MW)	Pile Power Norm. Factor	Volts/mg 10^{10} Scale	Irrad. Time (t)	Saturation Factor $1 - e^{-\lambda t}$	Flux ϕ (nv)
A1-R12	19.931	7/17/54	30	1	2.5×10^{-1}	10.5m	8.13×10^{-3}	2.85×10^{12}
A1-R21	19.931	7/17/54	30	1	4.18×10^{-2}	10.5m	8.13×10^{-3}	4.77×10^{11}
A1R-33	19.931	7/17/54	30	1	2.93×10^{-3}	10.5m	8.13×10^{-3}	3.34×10^{10}
A1-R42	19.931	7/17/54	30	1	1.53×10^{-3}	10.5m	8.13×10^{-3}	1.75×10^{10}
A1B-R12	20.703	7/19/54	30	1	2.43×10^{-1}	10m	7.75×10^{-3}	2.91×10^{12}
A1B-R21	20.703	7/19/54	30	1	3.96×10^{-2}	10m	7.75×10^{-3}	4.74×10^{11}
A1B-R33	20.703	7/19/54	30	1	2.82×10^{-3}	10m	7.75×10^{-3}	3.37×10^{10}
A1B-R42	20.703	7/19/54	30	1	1.24×10^{-3}	10m	7.75×10^{-3}	1.48×10^{10}
A2A-R12	21.508	7/21/54	30	1	6.04	5m	3.88×10^{-3}	1.44×10^{14}
A2A-R21	21.508	7/21/54	30	1	1.77	5m	3.88×10^{-3}	4.23×10^{13}
A2A-R33	21.117	7/20/54	30	1	1.69×10^{-1}	5m	3.88×10^{-3}	4.04×10^{12}
A2A-R42	21.117	7/20/54	30	1	9.85×10^{-2}	5m	3.88×10^{-3}	2.35×10^{12}
A2B-R12			30	1	1.31	1m	7.7×10^{-4}	1.58×10^{14}
A2B-R21			30	1	2.16×10^{-2}	1m	7.7×10^{-4}	3.71×10^{13}
A2B-R33			30	1	3.21×10^{-2}	1m	7.7×10^{-4}	3.86×10^{12}
A2B-R42			30	1	3.08×10^{-1}	1m	7.7×10^{-4}	2.60×10^{12}

TABLE C-4

REACTION: $\text{Cl}^{37}(\text{n}\gamma)\text{Cl}^{38}$

Sample No. Position	Cont. Rods Avg. Pos. (in)	Date of Irrad.	Pile Power (MW)	Pile Power Norm. Factor	Volts/mg 10^{10} Scale	Irrad. Time (t)	Saturation Factor $1 - e^{-\lambda t}$	Flux ϕ (nv)
A1-R33	19.931	7/17/54	30	1.0	2.99×10^{-3}	10.5m	0.174	1.71×10^{10}
A1-R12	19.931	7/17/54	30	1.0	2.4×10^{-1}	10.5m	0.174	1.37×10^{12}
A1-R21	19.931	7/17/54	30	1.0	4.43×10^{-2}	10.5m	0.174	2.53×10^{11}
A2A-R42	21.117	7/20/54	30	1.0	1.87×10^{-1}	5m	0.087	2.14×10^{12}
A1B-R12	20.703	7/19/54	30	1.0	3.75×10^{-1}	10m	0.167	2.22×10^{12}
A1B-R21	20.703	7/19/54	30	1.0	6.86×10^{-2}	10m	0.167	4.07×10^{11}
A1B-R33	20.703	7/19/54	30	1.0	5.11×10^{-3}	10m	0.167	3.05×10^{10}
A2B-R33			30	1.0	5.68×10^{-2}	1m	0.0181	3.12×10^{12}
A2B-R21			30	1.0	4.76×10^{-1}	1m	0.0181	2.62×10^{13}
A1B-R42	20.703	7/19/54	30	1.0	2.25×10^{-3}	10m	0.167	1.34×10^{10}

TABLE C-5

REACTION: $V^{51}(n\gamma)V^{52}$

Sample No. Position	Cont. Rods Avg. Pos. (in)	Date of Irrad.	Pile Power (MW)	Pile Power Norm. Factor	Volts/mg 1010 Scale	Irrad. Time (t)	Saturation Factor $1 - e^{-\lambda t}$	Flux ϕ (nv)
HLB-R1-4	23.360	7/25/54	30	1.0	6.48	1m	0.169	1.01×10^{12}
HLB-R2-2	23.360	7/25/54	30	1.0	1.47	1m	0.169	2.29×10^{11}
HLB-R1-1	23.360	7/25/54	30	1.0	3.57×10	1m	0.169	5.57×10^{12}
H1A-R4-1	23.360	7/25/54	30	1.0	7.27×10^{-1}	5m	0.604	3.18×10^{10}
H1A-R4-4	23.360	7/25/54	30	1.0	3.05×10^{-1}	5m	0.604	1.33×10^{10}
HLB-R3-2	23.360	7/25/54	30	1.0	1.68×10	5m	0.604	7.34×10^{10}
HLB-R4-4	23.360	7/25/54	30	1.0	3.70×10^{-1}	5m	0.604	1.62×10^{10}
HLB-R4-1	23.360	7/25/54	30	1.0	6.44×10^{-1}	5m	0.604	2.81×10^{10}
H1A-R3-2	23.360	7/25/54	30	1.0	1.73	5m	0.604	7.56×10^{10}
H1A-R1-1	23.360	7/25/54	30	1.0	7.04×10	2m	0.309	6.02×10^{12}
H1A-R2-2	23.360	7/25/54	30	1.0	5.07	2m	0.309	4.34×10^{11}
H1A-R1-4	23.360	7/25/54	30	1.0	1.04×10	2m	0.309	8.9×10^{11}

TABLE C-6

REACTION: $\text{Al}^{27}(\text{n}\gamma)\text{Al}^{28}$

Sample No. Position	Cont. Rods Avg. Pos. (in)	Date of Irrad.	Pile Power (MW)	Pile Power Norm. Factor	Volts/mg 10^{10} Scale	Irrad. Time (t)	Saturation Factor $1 - e^{-\lambda t}$	Flux ϕ (nv)
D1A-R11	22.362	7/23/54	30	1.0	1.56×10^{10}	2m	0.452	8.78×10^{12}
D1A-R14	22.362	7/23/54	30	1.0	1.45	2m	0.452	8.14×10^{11}
D1A-R24	22.362	7/23/54	30	1.0	5.24×10^{-1}	10m	0.951	1.4×10^{11}
D1A-R34	22.362	7/23/54	30	1.0	1.19×10^{-1}	20m	1.0	3.02×10^{10}
D1A-R44	22.362	7/23/54	30	1.0	5.91×10^{-2}	40m	1.0	1.5×10^{10}
D1B-R11	22.805	7/24/54	30	1.0	7.32	1m	0.26	7.15×10^{12}
D1B-R14	22.805	7/24/54	30	1.0	2.57	5m	0.778	8.37×10^{11}
D1B-R24	22.805	7/24/54	30	1.0	5.81×10^{-1}	11m	0.963	1.53×10^{11}
D1B-R34	22.805	7/24/54	30	1.0	1.8×10^{-1}	20m	1.0	4.49×10^{10}
D1B-R44	22.805	7/24/54	30	1.0	6.99×10^{-2}	40m	1.0	1.77×10^{10}
D1C-R14	22.802	8/14/54	30	1.0	4.66	33.3m	1.0	1.18×10^{12}
D1C-R34	22.802	8/14/54	30	1.0	1.44×10^{-1}	30m	1.0	3.67×10^{10}
D1C-R44	22.238	8/13/54	30	1.0	6.66×10^{-2}	30m	1.0	1.69×10^{10}
D1C-R24	22.238	8/13/54	30	1.0	4.87×10^{-1}	30m	1.0	1.24×10^{11}
D1D-R11			30	1.0	2.10×10	10m	9.51×10^{-1}	5.59×10^{12}
D1D-R34			30	1.0	1.33×10^{-1}	33.3m	1.0	3.39×10^{10}
D1D-R24			30	1.0	5.75×10^{-1}	33.3m	1.0	1.46×10^{11}
D1D-R31			30	1.0	4.63×10^{-1}	30m	1.0	1.18×10^{11}
D1D-R44			30	1.0	7.29×10^{-2}	30m	1.0	1.85×10^{10}
D1D-R41			30	1.0	1.11×10^{-1}	30m	1.0	2.81×10^{10}
D2B-R11	22.805	7/24/54	30	1.0	2.63×10^2	1m	2.6×10^{-1}	2.56×10^{14}
D2B-R14	22.805	7/24/54	30	1.0	8.08×10	1m	2.6×10^{-1}	7.88×10^{13}
D2B-R24	22.805	7/24/54	30	1.0	4.31×10	5m	7.78×10^{-1}	1.41×10^{13}
D2B-R34	23.360	7/25/54	30	1.0	2.03	5m	7.78×10^{-1}	6.6×10^{11}

TABLE C-6 (Con't.)

REACTION: $\text{Al}^{27}(\text{n})\text{Al}^{28}$

Sample No. Position	Cont. Rods Avg. Pos. (in)	Date of Irrad.	Pile Power (MW)	Pile Power Norm. Factor	Volts/mg 10^{10} Scale	Irrad. Time (t)	Saturation Factor $1 - e^{-t}$	Flux ϕ (nv)
D2B-R44	23.360	7/25/54	30	1.0	7.41	5m	7.78×10^{-1}	2.42×10^{12}
D2A-R24	22.805	7/24/54	30	1.0	3.93×10	5m	7.78×10^{-1}	1.28×10^{13}
D2A-R34	22.805	7/24/54	30	1.0	1.60×10	5m	7.78×10^{-1}	5.23×10^{12}
D2A-R44	22.805	7/24/54	30	1.0	9.47	10m	9.51×10^{-1}	2.53×10^{12}

TABLE C-7

REACTION: $\text{Co}^{59}(\text{n}\gamma)\text{Co}^{60}$

$\alpha_2 = 1.10$

Position	CR	CR-1	$(\text{CR}-1)(1 + \alpha)$	$(\text{CR}-1)(1 + \alpha)^{-1}$
R11	23.4	22.4	47.0	0.0213
R12	35.1	34.1	71.6	0.0140
R13	46.3	45.3	95.1	0.0105
R14	56.4	55.4	116.0	0.00862
R21	64.3	63.3	133.0	0.0075
R22	65.4	64.4	135.0	0.0074
R23	78.3	77.3	162.5	0.00614
R24	80.5	79.5	167.0	0.00599
R31	85.0	84	176.0	0.00568
R32	85.7	84.7	178.0	0.00562
R33	85.3	84.3	177.0	0.00565
R34	87.5	86.5	182.0	0.00549
R41	94.4	93.4	196.0	0.0051
R42	133	132	277.0	0.00361
R43	243	242	508.0	0.00197

TABLE C-8

REACTION: $\text{Cu}^{63}(\text{n}\gamma)\text{Cu}^{64}$

$$\alpha_2 = 0.933$$

Position	CR	CR-1	$(\text{CR}-1)(1 + \alpha)$	$[(\text{CR}-1)(1 + \alpha)]^{-1}$
R12	19.2	18.2	35.2	0.0284
R13	26.4	25.4	49.1	0.0204
R14	33.7	32.7	63.2	0.0158
R21	41.4	40.4	78.1	0.0128
R22	50.0	49.0	94.7	0.0106
R23	63.0	62.0	120.0	0.00843
R24	75.0	74.0	143.0	0.0070
R31	87.2	86.2	166.5	0.0060
R32	107.0	106.0	205.0	0.00488
R33	120.0	119.0	230.0	0.00435
R34	142.0	141.0	272.0	0.00368
R41	163.0	162.0	313.0	0.00319
R42	190.0	189.0	365.0	0.00274
R43	219.0	218.0	421.0	0.00237

TABLE C-9

REACTION: $\text{Na}^{23}(\text{n}\gamma)\text{Na}^{24}$

$$\alpha_L = 0.238$$

Position	CR	CR-1	$(\text{CR}-1)(1 + \alpha)$	$[(\text{CR}-1)(1 + \alpha)]^{-1}$
R12	50.0	49.0	60.6	0.0165
R13	78.5	77.5	96.0	0.0104
R14	84.8	83.8	104.0	0.0096
R21	89.2	88.2	109.0	0.00916
R22	96.0	95.0	117.5	0.0085
R23	97.0	96.0	119.0	0.0084
R24	106.0	105.0	130.0	0.0077
R31	105.0	104.0	129.0	0.0072
R32	112.0	111.0	137.0	0.0073
R33	115.0	114.0	141.0	0.0071
R34	128.6	127.6	159.0	0.00628
R41	146.0	145.0	179.0	0.00558
R42	164.0	163.0	202.0	0.00496

TABLE C-10

REACTION: $\text{Cl}^{37}(\text{n}, \text{p})\text{Cl}^{38}$ $\alpha_{\text{Cl}} = 0.079$

Position	CR	CR-1	$(\text{CR}-1)(1 + \alpha)$	$[(\text{CR}-1)(1 + \alpha)]^{-1}$
R21	81.9	80.9	87.2	0.01146
R22	94.7	93.7	101.0	0.0099
R23	102.5	101.5	109.5	0.00913
R24	110	109	117.6	0.0085
R31	119	118	127.3	0.00786
R32	123	122	131.5	0.0076
R33	139	138	149.0	0.00671
R34	150	149	160.8	0.00622
R41	154	153	165.0	0.00606
R42	160	159	171.5	0.00583

TABLE C-11

DATA FOR RESONANCE REACTION $v^{51}(n\gamma)v^{52}$

Position	CR	CR-1	$(CR-1)(1 + \alpha)$	$[(CR-1)(1 + \sigma)]^{-1}$
R11	39.3	38.3	43.4	0.023
R12	58.2	57.2	64.8	0.0154
R13	68.8	67.8	76.8	0.0130
R14	75.5	74.5	84.4	0.01185
R21	77.6	76.6	86.8	0.0115
R22	76.7	75.7	85.8	0.01165
R23	75.0	74.0	83.9	0.0119
R24	75.6	74.6	84.6	0.0118
R31	78.2	77.2	87.5	0.0114
R32	86.8	85.8	97.3	0.0103
R33	94.5	93.5	106.0	0.00943
R34	103.6	102.6	116.3	0.0086
R41	111.5	110.5	125.3	0.00798
R42	120.6	119.6	135.5	0.00738
R43	131.7	130.7	148.2	0.00675
R44	141.8	140.8	159.7	0.00626

TABLE C-12

REACTION: $\text{Al}^{27}(\text{n}\gamma)\text{Al}^{28}$ $\alpha_2 = 0.159$

Position	CR	CR-1	$(\text{CR}-1)(1 + \alpha)$	$[(\text{CR}-1)(1 + \alpha)]^{-1}$
R11	37.0	36.0	41.7	0.024
R12	58.3	57.3	66.4	0.015
R13	72.3	71.3	82.6	0.012
R14	80.0	79.0	91.6	0.0109
R21	83.4	82.4	95.4	0.0105
R22	86.2	85.2	98.7	0.0101
R23	91.0	90.0	104.5	0.00956
R24	100.0	99.0	115.0	0.0087
R31	109.5	108.5	126.0	0.00793
R32	119.0	118.0	137.0	0.00729
R33	132.0	131.0	152.0	0.00657
R34	139.5	138.5	160.5	0.00623
R41	148.5	147.5	171.0	0.00584
R42	153.0	152.0	176.0	0.00568
R43	140.0	139.0	161.0	0.00621
R44	156.0	155.00	179.0	0.00558

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