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Physics and Nuclear Energy

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1. The establishment of a chain reactor and thereby the release of nuclear energy was first achieved by physicists. Out of this laboratory experiment, consisting of a pile of graphite with uranium rods stuck into it, has developed the nuclear power industry, built up with extremely large capital investments and involving a great number of people.

It is a worthwhile effort to assess the soundness of our knowledge of the basic processes and physical data in this field since so much depends on them. This will be attempted in the notes below.

The advanced power reactor based on natural uranium or enriched fuel operating at high temperatures is complex and presents a number of problems which are not present in a zero power assembly; therefore, it leads to a clearer picture if one looks at the latter first. Since almost all power producing systems are based on fissions produced by low energy neutrons, such systems will mainly be considered in this note.

Before any reactor can be built it is essential to calculate the size, the neutron economy and the temperature coefficient of the reactivity, neutron spectra and spatial distribution of the neutron flux, reaction rates throughout the reactor, reactivity control and the time dependence of some of these quantities. The basic physical processes are known in principle and the mathematical theory necessary to make such calculations is available in the literature [1][1]). In order to be successful in this endeavour the following conditions have to be satisfied:

a) The general theory must cover all the relevant physical processes.

b) The approximations generally necessary in calculations must be sound to avoid introducing unknown errors.

c) The basic physical data used for the numerical evaluations must not introduce uncertainties.

Ideally one would like to compute a reactor ab initio; in principle, this could be done for some purposes with the Monte Carlo method, for example, by following the life history of a neutron in the assembly chosen for study and repeating the process very many times. Whilst the

[1]) The numbers in brackets refer to References, page 210.

computing effort for such an undertaking would be enormous, an attempt to follow such a programme on paper rapidly reveals that our present knowledge of the quantitative basic nuclear and other data is not yet known with sufficient precision and that our understanding of some of the fundamental processes of the energy transfer by neutrons to the moderator is quite inadequate. These deficiencies in our knowledge are also quite relevant since computer time based on uncertain input data is uneconomical and the results obtained are afflicted with errors very difficult to assess.

2. Basic nuclear data [2]

It would serve little purpose on this occasion to enumerate the basic physical constants which go into a reactor calculation and to tabulate the differing numerical values obtained by various laboratories. There is, in fact, considerable difference of opinion as to what is needed and how accurate the basic data have to be, though it seems established that with increasing reactor complexity the amount of information increases. It should also be noted that the nuclear information required by the reactor theoretician to enable him to calculate multiplying systems depends considerably on the details of the reactor, particularly on the neutron spectrum and the mean neutron energy.

2.1 *Thermal and epithermal reactors*

The information available in the thermal region, 0.025 e.v., is most reliable because here reaction cross sections are large, and do not, in general, vary widely, and neutrons for experiments in this energy region are amply available even in low power reactors; since monochromatic neutrons from crystal spectrometers and time-of-flight methods of fairly simple design are available, many laboratories with reactors have contributed to the collection of data such as fission cross section, total cross section, number of neutrons per fission. The equipment which the physicist has to use is therefore comparatively simple. For higher energies, say above 10 e.v., the crystal spectrometer becomes inoperable and much more sophisticated methods with higher flux reactors have to be used if good measurements have to be made of partial cross sections such as fission, scattering, capture cross sections. But even in the thermal energy region the number η of neutrons emitted per capture by U (233) has recently been under critical re-examination. As can easily be seen, $\eta - 2$ is the important quantity for breeding and U (233) is of particular interest for the thorium cycle. Different laboratories obtained 0.18 to 0.25 for this value. Since the lower value may well make breeding impossible after other unavoidable neutron losses have been included ($\eta_{\text{effective}} - 2 < 0$), a great deal of concern was felt by those depending

on the thorium cycle. This is an example showing how a 3 p.c. error in a measured quantity may influence our entire plans for reactors. Thus the physicists and chemists who jointly measure such data bear a great responsibility as their statements are of considerable implications economically and politically.

Another region where nuclear data are of great importance and not yet complete concerns the temperature coefficient of the reactivity dk/dt. It has now become clear that particularly the fission cross section, and consequently dk/dt, are affected in the case of thermal and epithermal reactors through the Doppler effect and its influence is felt up to quite high energies of the order of a few hundred electron volts. To calculate its influence upon the reactor reactivity the cross sections of all the resonances below that energy have to be known; this includes σ_0, the cross section at exact resonance and the total Γ and the various partial widths Γ_s, Γ_γ, Γ_f, for scattering, capture and fission. This represents a very considerable programme involving the expenditure of considerable means, investments of staff and computer time.

One of the most important components of the operating costs of a reactor is the length of the fuel cycle. Since chemical processing of the highly radioactive spent fuel and the refabrication of the fuel elements is slow and costly, there is every incentive to leave the fuel in the reactor as long as possible. Without going into the complex transformation which the fuel undergoes on prolonged neutron exposure (up to 3000, and hoping to attain 10,000 Megawatt days per ton of fuel) it is clear that information, for example on *Pu* (240) and *Pu* (241) and on fission product nuclei produced by multiple neutron capture, is of decisive importance in forecasting the reactor reactivity in its dependence on time. To obtain these data is certainly cumbersome, as is all work with highly active materials.

2.2 *Intermediate and fast reactors*

At low energies, neutron resonances are as a rule well isolated and the energy dependence of the reaction cross section is given by the dispersion formula of BREIT and WIGNER

$$\sigma_r(E) = \sum_i \frac{\pi \lambda^2 g_i \Gamma_{ni} \Gamma_{ri}}{(E - E_i)^2 + (\Gamma_i/2)^2}$$

(where E_i, λ, g_i stand for the resonance energy, wavelength and statistical spin factor and Γ_i, Γ_{ni}, Γ_{ri} for the total, neutron and reaction width of level i) provided the mean distance D between resonances is large compared to the mean total width of the resonance ($\overline{\Gamma_i} < D$). As the neutron width for interactions involving zero angular momentum only increases proportional to $\sqrt{E}$, the definition of isolated resonances becomes im-

possible at higher energies, since the width becomes equal or larger than the mean level distance and the levels overlap. Under these conditions the Breit-Wigner relation is inapplicable and is replaced by a mean reaction cross section $\bar{\sigma}_r$ averaged over a suitable energy interval, which, for angular momentum zero, reads as:

$$\bar{\sigma}_r = 2\pi^2 \lambda^2 g \frac{\Gamma_n \Gamma_r}{D(\Gamma_n + \Gamma_r)} ; \qquad S = \frac{\bar{\Gamma}_n^0}{D}$$

where Γ_n^0 is the mean reduced neutron width. ($\Gamma_n = \Gamma_n^0 \sqrt{E}$). This mean cross section is closely connected with the strength function S [3] which, on the other hand, can be estimated from theoretical considerations of nuclear models. This source of information is useful if experimental data in the higher energy region are not available, since in the region of overlapping resonances the neutron spectrum does not vary violently.

A fast reactor is obtained through reduction or complete elimination of the moderator. Such a system has a predominantly high energy spectrum, where the cross sections are smooth, and slowly varying functions of energy.

Though the fission cross sections in this energy range, for example, are now well established, the information on inelastic scattering, which becomes very important in this region, is only quite unsatisfactorily known. It is interesting to note that inelastic scattering of fast neutrons in thermally fissile elements is practically unknown and new methods will have to be developed to determine this quantity with the precision required.

We can expect that more information about data for fast reactor calculations will be asked for before very long if the fast reactor prototypes now under construction show satisfactory characteristics. The good breeding properties of fast systems are likely to make them popular in future in spite of the large cost in pure fissile material if dilution with fertile matter is resorted to to keep the energy production per unit volume at a technically possible level. If this should be the tendency, physicists will have to be relied upon to procure much information, since computation will play a preponderant part as realistic integral assemblies are very costly.

3. The neutron energy spectrum of reactors

In almost all reactor quantities there occurs the integral

$$I = \int \sigma_X(E)\ \Phi(E)\, dE$$

where $\sigma_X(E)$ refers to the process X (e.g. fission, neutron capture, etc.), $\Phi(E)$ to the neutron flux per unit energy interval, and the integral I to the reaction rate. Hence the interest in the energy composition of the neutron flux.

It has in the past been an acceptable procedure to assume in the moderator of a thermal reactor an energy distribution which is made up of the superposition of a Maxwellian and a slowing down spectrum of the form:

$$\Phi(e)\ dE \sim \left[\frac{E}{E_0^2}\, e^{-E/E_0} + \frac{\gamma}{E}\right] dE$$

$$E_0 = k\,T$$

This is quite a good approximation in large lattice systems of low absorption and with uniform temperature, since most neutrons make many scattering collisions before being removed. Under these conditions the validity of the first term of the above equation is approximately guaranteed and the second term accounts for the slowing down of the neutron with «free» moderator nuclei, where γ measures the relative incompleteness of thermalisation or the hardness of spectrum.

The problem becomes very much more complex when one now considers highly absorbing systems such as reactors fuelled with enriched or pure fissile materials, or systems with high temperature gradients, or both. To define the problem let us divide the neutron energy ranges into three classes:

Table of Various Stages of Neutron Moderation

Class	*Neutron energy range*	*Mechanism*	*Information required; other remarks*
I	High energy MeV–E_{max}	Energy loss only through free collision with stationary nuclei. Spectrum 1/E.	Scattering and reaction cross sections (E, E′, θ) of relevance in fuels; not effective in C, H_2O moderators; has to be considered in Be, D_2O.
II	Thermalisation $E_m \geqslant E \geqslant O$	The temperature motion of the target nuclei is taken into account; energy gains and losses important: in solids through emission and absorption of phonons; in the liquids H_2O, D_2O through molecular excitation and phonon processes.	Transfer function (scattering kernel). Models: H gas (Wigner-Wilkins) and many others.
III	Thermal equilibrium E ~ kT.	Gains and losses equal; in the absence of large absorption, σ_a, approximate equilibrium spectrum, which is independent of energy transfer mechanism. Energy distribution: $\frac{E}{E_0^2} \exp\left(-\frac{E}{E_0}\right)$.	The effective temperature increases with σ_a and the $1/E$ part grows at the expense of the Maxwell distribution.

Though the division of the total energy range of neutron moderation into three parts is necessarily crude, it helps to isolate the individual problems. E_m may be taken roughly as the energy where molecular processes set in in liquids or it can be set equal to the DEBYE temperature for solids. As an illustration of the range of energies involved we note: H_2 (gas); dissociation energy 4.5 e.v.; first rotational state $E_r = 0.0075$ e.v.; first vibrational state $E_r = 0.54$ e.v.; Debye temperatures of graphite T_1 normal to hexagonal plane 1,000° C, T_2 in hexagonal plane 2,500° C [4]; beryllium metal 900° C.

3.1 *The process of thermalisation* (see chart above) [5].

3.1.1 *The infinite system* (theoretical results)

Since the thermalisation process is in the forefront of interest at the moment and is as yet incompletely understood, it will be discussed somewhat more fully.

It is useful to start by writing down the balance for the passage of neutrons into and out of a certain energy interval dE situated at E, assuming the system to be infinite and that space dependence can be disregarded:

$$\left(\sum_a(E) + \sum_{sc}(E)\right) \Phi(E)\, dE = \int_0^{E_m} \Phi(E')\, \sigma_{sc}(E' \to E)\, dE' dE + S(E)\, dE$$

$$\sum_{sc}(E) = \int_0^{E_m} \sigma_{sc}(E \to E')\, dE', \qquad S(E) = \int_{E_m}^{E_{f.n.}} \sigma(E'' \to E)\, dE''$$

$E_{f.n.}$ = Energy of fission neutrons.

where Σ_a, Σ_{sc} refer to the macroscopic absorption and scattering cross section, out of the energy interval $E\, dE$ and $\sigma_{sc}\ (E' \to E)$ to the scattering out of $E'dE'$ into the energy $E\ dE$; $S\ (E)$ is a source term required to compensate for $\Sigma_a(E)$ to maintain a stationary distribution; $\sigma_{sc}\ (E' \to E)$ with $0 \leqq E' \leqq E_m$ is often referred to as transfer function or the scattering kernel of the integral equation.

The first and up till now most satisfactory mathematical attempt to calculate the neutron thermalisation of an infinite system on the basis of the above equation is due to WIGNER and WILKINS (1944) who used as moderator model a gas of hydrogen atoms, assuming an energy independent, isotropic scattering law for neutron-proton scattering and a $1/v$ absorption cross section for neutron capture. In spite of the physically very unrealistic model (gas instead of say liquid H_2O or D_2O) of WIGNER and WILKINS, experiments by Poole (see below) showed that the neutron spectrum in an extended mass of water agreed well with the calculated one.

This is a very surprising result: the mechanism of energy loss and gain in liquids like water and in solids and gases are so different that we would have expected the spectrum in the thermalisation region to be affected. In the case of water the quantised rotational and vibrational states can be excited [6], whilst in crystals the energy transfer is entirely different involving the emission and absorption of single or multiple phonons. Moreover, it has been found in calculations that the kernel of the integral equation has little effect on the neutron spectrum in an infinite medium of constant temperature. M. J. POOLE, M. S. NELKIN and R. S. STONE point out that this is partly a consequence of the requirement of neutron balance and of the asymptotic relation between slowing down density and flux at high energies which ensures that the spectrum for the case of vanishing absorption $\sigma_a = 0$ and source strength $S = 0$ transforms into a Maxwell distribution; it is, however, not clear why these restrictive conditions are so effective.

3.1.2 *Finite systems*

Though we cannot say that the thermalisation process in a system of infinite extent has been amenable to a realistic theoretical treatment, the difficulty becomes much worse in a finite system as one could expect where the favourable and simplifying conditions of an infinite medium no more apply. In view of the practical implications it is now quite essential to realise how far removed one is in one's ability to compute important effects applying under reactor conditions.

It is instructive to describe the importance of thermalisation with an example illustrating the difficulty arising.

Britain's power stations are based on natural uranium fuelled and graphite moderated systems. In the course of time some of the *U* (238) is converted into *Pu* (239), its concentration increasing approximately in proportion to the irradiation dose. The geometrical arrangement of fuel and moderator in an example is shown in Fig. 1.

Consider now the fate of a neutron emitted from the fuel rod; it will, as a fast neutron, pass through *S* into *M*, be slowed down, say to temperature T_M, and on its return to the fuel pass through the hot (T_S) sleeve *S* again. Because *Pu* (239) possesses a fission resonance at 0.3 e.v., it is quite essential to know how much the neutrons are 'heated up' on their passage through *S*, since this determines the ratio of the number of neutrons absorbed either in *Pu* (239) or in *U* (238) and other nuclei which do not contribute through fission to the reactivity. This ratio contributes a positive component to the temperature coefficient of the reactivity and also affects the *U*–*Pu* conversion in an important manner. Unfortunately, it is at present not known how to compute the energy transfer in such a case satisfactorily, because there can be no question

of approaching anything like an equilibrium distribution corresponding to the temperature of the sleeve T_S.

Though the above example may seem a slightly artificial illustration, similar cases appear in reactor problems as soon as detailed design calculations are attempted. Problems arising out of energy transfer kernels in a hot system are quite frequent, but even in cold reactors we are greatly at a disadvantage through the uncertainties in spectrum calculations. After all, any criticality calculation involves nuclear cross sections, which are dependent on the neutron spectrum.

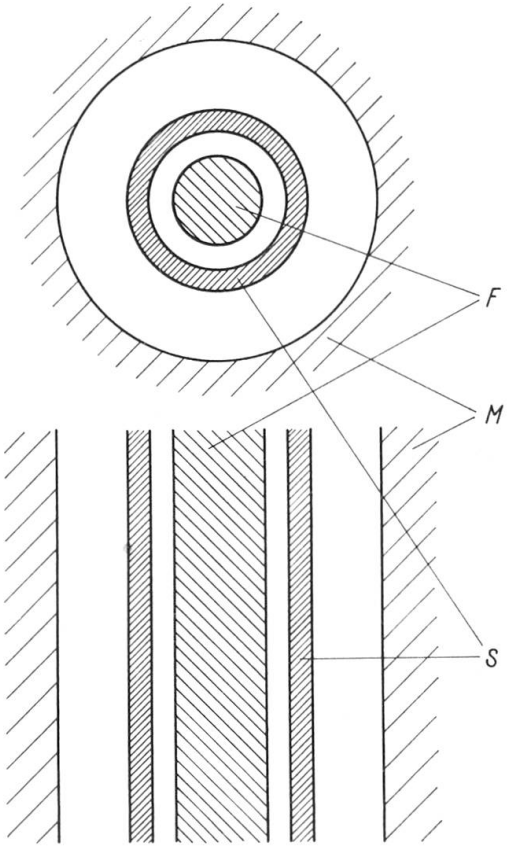

Figure 1
Fuel element (*U–Pu*) surrounded by hot sleeve. *F* fuel element, *S* sleeve at temperature T_S, *M* moderator at temperature T_M, both consist of graphite. $T_S > T_M$.

3.1.3 *The experimental approach*

The complexity of neutron thermalisation and the urgency to obtain the data required for practical purposes has led us at HARWELL to a three-fold attack on the problem of neutron thermalisation. Firstly, there is some work going on of a purely scientific interest, on the interaction of neutrons with crystals or liquids; secondly, measurements of neutron scattering in moderator materials of topical interest are carried out; and

thirdly, methods have been developed to measure neutron spectra directly in experimental assemblies. These recent developments are of great value and of considerable influence on reactor physics and some of the methods outlined below have found wide interest in the reactor field.

3.1.3.1 *The mechanism of energy exchanges of neutrons with the condensed phase*

Our knowledge of the energy interactions of neutrons with a crystal lattice has made spectacular progress since G. PLACZEK and L. VAN HOVE published in 1954 their paper on crystal dynamics and inelastic scattering of neutrons [7]. The results of these and subsequent papers by the same authors and others have provided equations which give cross sections for coherent and incoherent scattering involving momentum changes $k \rightarrow k'$ with emission or absorption of one phonon in terms of the frequency distribution of the normal vibrations of the lattice. In the case of incoherent scattering these relations permit the determination of the normal frequency distribution of the lattice from scattering experiments of very low energy neutrons by the crystal through measuring the distribution of the energy gain of the neutrons by the absorption of a phonon according to the equation:

$$\frac{\sigma_d}{d\Omega\, dE'} = \frac{\sigma_{inc}\,\hbar}{16\,\pi^2\, m}\; e^{-2W} \left|\frac{k'}{k}\right| \frac{(\bar{k}'-\bar{k})^2}{(E'-E)} \cdot \frac{e^{-\frac{h\,\omega}{k_0\,T}}}{1-e^{-\frac{h\,\omega}{k_0\,T}}}\; g(\omega)$$

where k, k_0 are scattered and incident wave vectors and W the Debye-Waller factor. $g(\omega)$ is the normal frequency distribution, which is the aim of the experiment.

On the other hand, $g(\omega)$ can be calculated if the interaction between the various atoms of the lattice is known. Neutron scattering therefore serves as a means to test any model of interaction. The only examples of a scatterer scattering completely incoherently are vanadium and hydrogen. Va has been studied by B. N. BROCKHOUSE and by R. S. CARTER, D. J. HUGHES and H. PALEVSKY with concordant results [8]. Not enough is known about the elastic constants of *Va* to compare $g(\omega)$ in detail with the theory by BORN-VON KARMAN, but it is stated that pure central forces are not compatible with the shape of $g(\omega)$ if neighbour and next neighbours only are included in the interaction. Similar studies, though naturally more complex through interference effects, have been made particularly extensively on aluminium [9], which scatters predominantly coherently. Agreement might be obtained by using at least 4 force constants for the transverse mode and at least 6 in total. It is clear from the above that crystal theory is not yet in a position to forecast neutron inelastic lattice scattering for practical purposes, but one can visualize a not too distant future when it will be possible to do so.

The interaction of neutrons with liquids is now becoming of great interest, partly because techniques have been developed for solid state studies suitable also for liquids, partly through theoretical investigations due to VAN HOVE, VINEYARD and others [10]. The observations [11] are very promising but not yet good enough to be reported extensively at this stage.

It has been noted that *low* energy neutrons are used to study their interactions with the normal vibration of lattices so as to restrict the physical processes possible to energy gain by the neutrons. As the sources of such work depend greatly on the intensity of the low energy neutron beam, a liquid hydrogen scatterer serving as a neutron radiator is placed inside the reactor in order to shift the neutron energy distribution from room to lower temperatures. Alternatively, one can use a liquid nitrogen cooled beryllium block, acting as a cut off filter (B.N.L.). In this manner a large increase can be achieved in neutron source strength for neutrons of long wavelengths.

From this short account it is evident that on a long term scale reactor physics will eventually be able to obtain the information required for reactor calculations from purely basic scientific data through advances in our knowledge of the interaction of neutrons with condensed matter.

3.1.3.2 *Scattering law determination* (P. A. EGELSTAFF, A.E.R.E.)

The second approach consists of a measurement of $(k_0 \rightarrow k)$ directly. To this end neutrons of momentum k_0 are allowed to fall on to a moderator element of the kind M at temperature T; the distribution of k after scattering in its dependence on angle and absolute value k is determined. These measurements have to be done for different k_0, moderator M and temperature T. The result will provide the complete information necessary for calculations requiring the scattering kernel of any scattering material at any required temperature.

A neutron beam from one of the high flux holes of the N.R.U. reactor at Chalk River is used as a source. Neutron pulses of a wave number interval Δk_0 are produced by rotating neutron shutters and allowed to be scattered by the specimen under investigation. The neutrons scattered are observed in 30 different directions by suitably placed counters and at each angular position a velocity analysis into 90 channels is performed by the time-of-flight method. All information is stored on magnetic tape, which is later fed into a computer for processing. The final result is expressed as scattering kernels $\sigma\ (E_0 \rightarrow E)$. These measurements will cover graphite, H_2O, D_2O, BeO, Be, and the energy of the incident neutrons will be varied from low values up to 1 e.v.

This is a project carried out jointly by the Atomic Energy Organisations of the U. K., Canada and the U.S.A.

3.1.3.3 *Direct spectrum measurements* (M. T. Pool, A. E. R. E.)

The third approach obtains the neutron spectrum $\Phi(E)$ for a chosen assembly directly by observing the distribution in energy of the neutrons emitted. To this end a collimator is inserted into the system (for example a small tank housing a lattice section) so that the neutrons emitted from the small area at the collimator bottom can be investigated. The actual energy distribution is measured by time-of-flight methods, using a pulsed neutron source to excite the lattice or alternatively arranging for a continuous influx of neutrons from a reactor into the lattice tank and chopping the beam issuing from the collimator to measure the neutron flight time to the detector. Such measurements can be carried out at different points between two lattice rods to determine how the spectrum changes from point to point, or can be used to obtain the anisotropy of the neutron current by studying the neutron emission parallel and perpendicular to fuel rods [12].

Neutron spectra have been observed with crystal spectrometers or time-of-flight methods as early as 1944, but the systematic study of the thermalisation and other processes of reactor lattice models to provide information for computations of designs are of quite recent origin and are only now spreading to many laboratories. For this reason it is useful to describe briefly some of the Harwell equipment which is used for reactor studies:

One experimental arrangement can be seen in fig. 2.

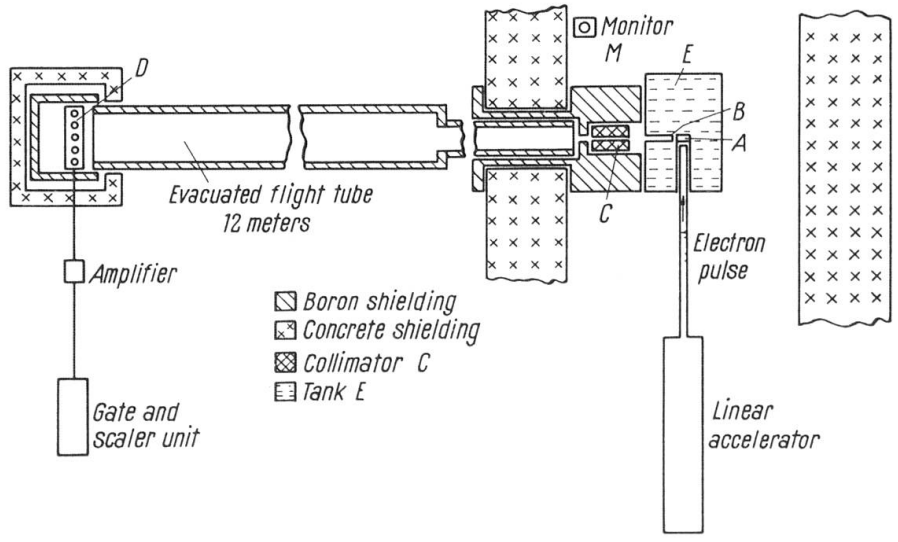

Figure 2

Experimental arrangement to study the neutron spectrum emitted from an assembly E (homogeneous system or lattice array) by time-of-flight neutron source: pulsed electron beam acting on composite mercury-uranium target (A). B source tube, C collimator, which is above not close to the target.

The pulsed neutron source is provided by a linear electron accelerator producing electron beam pulses of duration 0.2 to 2 μs and energy of about 28 M.V. The electrons yield gamma rays by slowing them down in a flowing mercury target whose gamma emission provides, from a surrounding uranium block, the neutrons through the photo-effect and photo-fission. Since the current in the electron pulse is at present of the order of 150–400 m.A. and the duty cycle about 10^{-3}, the target power is of the order 10 k.W. Under these circumstances the neutron production rate during the pulse is about 10^{16} sec^{-1}.

The measured neutron spectrum (Fig. 3) of a cadmium solution can serve as a representative result. It will be noticed that in the thermalisation region calculation and experiment disagree. Cadmium has a resonance at 0.178 e.v. The solid curve shows calculations based on the WIGNER-WILKINS model (Sofocate code programmed for IBM-704). The 1/E and Maxwell region agree with experiment (observer M. J. POOLE, A.E.R.E.).

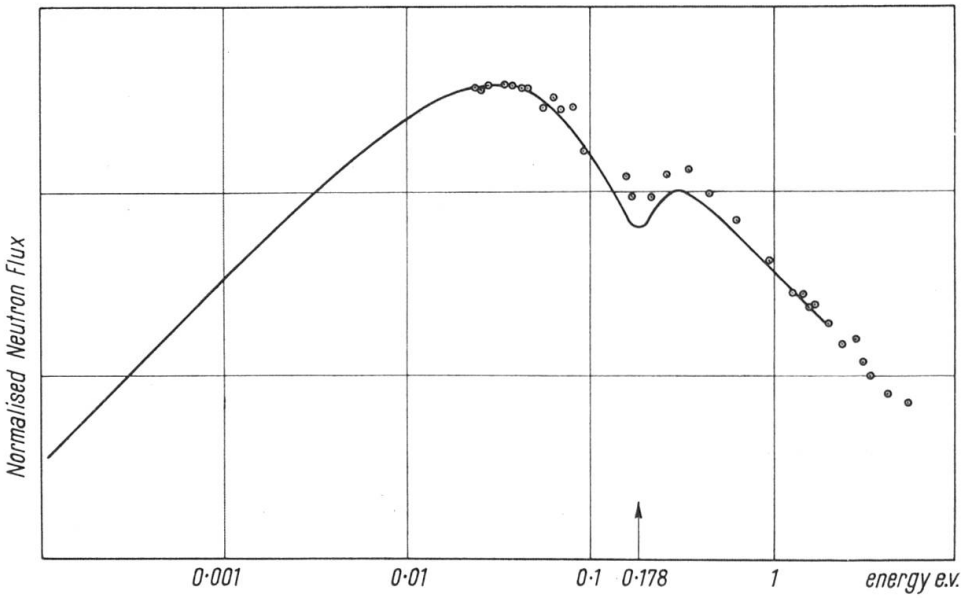

Figure 3
Comparison of observed spectrum of a cadmium nitrate solution with recent calculation. Ratio: 4.67 Cd atoms/100 H atoms

Not all assemblies are suitable for the pulse source method. Graphite lattices are at present excited with neutrons from the reactor Lido and the neutron beam is pulsed with a chopper to allow the neutron speed to be measured.

It has recently become clear that these methods are very powerful to study thermalisation, determination of mean neutron life for absorption, diffusion constants, lattice parameters. But the main value lies in the possibility of testing calculations and thus increasing our knowledge of selected physical processes and their models to a degree which will permit

calculations on chain reacting systems of any type with more confidence than can be done at present.

4. Integral experiments

Until more basic information is available, the reactor physicist is forced to perform 'integral' measurements, such as attempting to estimate the neutron spectrum with various absorbers of differing energy dependence of cross section and with threshold detectors for fast reactor fluxes. Other examples of such problems are: fine structure in fuel channels, lattice pitch, conversion factor, temperature effects on neutron spectra, pile oscillator estimates of effective cross sections. It has, however, to be borne in mind that the results of such observations depend on the energy spectrum of the neutrons, an uncertainty likely to introduce serious errors. For such studies quite large expensive assemblies have to be built, such as exponential piles or zero energy reactors. The information obtained in such work applies only to the particular reactor model, though one can learn a great deal about some features of assemblies by varying one parameter at a time such as the lattice pitch. All the large integral experiments are expensive, though, at the moment, necessary investigations. However, once the basic data and sufficiently powerful computer methods are available, integral experiments will still provide a most necessary test for the calculations performed, but some of them will then no longer be required saving capital expenditure and manpower.

Many problems which are of great importance for reactor design have not been touched upon, such as shielding. Here the basic information is mostly available, but the complications arising from the difficulties in satisfactorily computing the attenuation, the effects due to channels, or other tasks of the shielding expert, are still fairly serious.

It has been mentioned before that one of the key factors in the economics of nuclear energy lies in the behaviour of the fuel elements and in the stability of the moderators, such as graphite. Though chemical processes are responsible for some undesirable effects, there are two interesting phenomena worth mentioning, the COTTRELL effect and the WIGNER effect.

The former refers to the prediction of Dr. COTTRELL that irradiation of uranium can induce creep at lower stresses than in the absence of radiation. This has been verified experimentally as it was found that fuel elements in the vertical reactor channels bow and have to be supported by braces.

The WIGNER effect refers to the displacement by fast neutrons of carbon atoms from their lattice positions into interstitial locations; this process requires energy which is stored in the graphite lattice. The WIGNER effect causes the graphite to expand and change its electrical and thermal conductivity. The energy stored can be quite large and be

partly released by annealing if the temperature of the moderator is allowed to increase. In view of the possible hazard to the reactor if a sudden energy liberation takes place, a great deal of research has been carried out on the physics of disturbed lattices.

In conclusion, it must be stressed that the above remarks have not, owing to shortage of space, considered quite extensive topics such as health physics and instrumentation, both of which comprise large fields of great importance.

The main purpose of this report was to point out that the physical sciences still have a large part to play if we intend to place the fundament of the reactor science on a sound basis. We have to appreciate that in the past one was satisfied when a chain reacting system became divergent. It is quite a different matter if one aims at an economic, safe reactor with good characteristics which it maintains throughout its life. This cannot be done by engineers alone; the cooperation of physicists, metallurgists and chemists will have to provide the scientific basis to do so.

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