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November 27, 1963

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COMPARATIVE REACTOR FLUX SPECTRA

by

R. C. Gumprecht

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COMPARATIVE REACTOR FLUX SPECTRA

INTRODUCTION

This document is explanatory in nature and is intended to clarify certain questions about reactor neutron flux spectra in various AEC production facilities. Simplified models are used to illustrate neutron "temperature", spectral "hardening", and the so-called "Westcott R."

SUMMARY

There is no single neutron flux spectrum which uniquely characterizes any particular reactor in the AEC complex. (The spectra shown in reference (2) refer only to conditions within several typical or "standard" fuel loadings. Within limits, each reactor is capable of generating a wide variety of flux spectra by variations of the following factors:

1. Core power density
2. Fuel configuration and composition
3. Type of fissile material in fuel (with respect to absorption and fission cross sections, resonances, and neutrons released per fission)
4. Configuration, composition, and location of target material
5. Moderator temperature
6. Other factors including process channel size and material, cladding type and thickness, etc.

Some broad, general comparisons of flux spectra can be drawn between D₂O-moderated reactors and graphite-moderated reactors, but detailed comparisons applicable to irradiation studies for any particular target material must be made from calculations or experiments performed for the particular fuel-target core load design in question.

Several sample graphical representations of flux spectra are presented to illustrate the significance of the various factors enumerated above; fine structure has been eliminated for the sake of clarity and brevity.

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- (1) "Flux Spectrum Comparison of D₂O Moderated Lattice with Hanford Lattices", A. T. Gifford to A. B. Greninger, ORRGR, 10-8-63.
 - (2) Hanford Reactor and Separations Facilities Advantages, HW-78100, June 27, 1963. (SECRET).

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DISCUSSION

A. Explanation of Logarithmic Energy Interval Plot

The familiar process by which high energy neutrons slow down to low energies by successive collisions with moderator nuclei causes the existence, at any moment in time, of neutrons of widely varying energies. The distribution of the relative abundance of neutrons over the energy range of interest constitutes a "spectrum." A flux spectrum describes the distribution of flux (neutrons/cm²/sec) over energy. The usual symbol for flux as a function of energy, $\phi(E)$, represents the derivative of the total flux at any energy, E (i.e., the number of neutrons per square centimeter per second having energies lying between E and $E + dE$ divided by dE , or - the flux per unit energy interval). In the low range of neutron energies, where the process of slowing down is complete, the neutrons are essentially at thermal equilibrium with the moderator, hence the name "thermal" neutrons. In this energy range, the curve of $\phi(E)$ assumes the shape of a Maxwellian distribution whose mean energy can be described by the neutron "temperature." Above this energy range, neutrons are still in the process of slowing down and $\phi(E)$ is approximately directly proportional to the reciprocal of neutron energy (hence the description "Maxwellian with a $1/E$ tail").

Ordinary graphical plots of $\phi(E)$ versus E , if extended over a fairly broad range of energies, convey little useful information since the Maxwellian portion of the curve usually reaches its peak at a quite small fraction of one electron volt. A much more useful form is obtained by plotting $E \phi(E)$ versus $\log E$. Mathematically, $E \phi(E)$ is the derivative of the total flux with respect to $\log E$ (or flux per unit logarithmic energy interval). In this form the " $1/E$ tail" becomes level; the amount of flux between any two energies E_1 and E_2 is directly proportional to the area under the curve between $\log E_1$ and $\log E_2$. Examples of the latter form of plot are shown in Figures 1-3. Each of these curves is defined as follows:

$$E \phi(E) = M \left(\frac{E}{kT} \right)^2 e^{-\frac{E}{kT}} + F \Delta \quad (1)$$

where: M is the total Maxwellian flux,
 F is the epi-Maxwellian or "slowing down" flux per unit logarithmic energy interval,
 Δ is the cut-off function which joins the slowing down and Maxwellian portions of the flux,
 k is the Boltzmann constant,
 T is the neutron temperature in degrees Kelvin,
 E is the neutron energy (usually in electron volts).

The neutron "temperature" is linked to the flux-weighted mean neutron energy in the Maxwellian portion of the curve through the relation:

$$\bar{E} = 2 kT \quad (2)$$

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The epithermal flux level, F , is established by the rate at which neutrons are slowing down to thermal energies and the amount and kind of moderator atoms present. Thus, F is directly proportional to the power density per unit volume of reactor core, directly proportional to the number of fast neutrons released per fission, inversely proportional to the slowing down power per unit volume, and directly proportional to the "resonance escape probability."

B. Spectral Hardening Effects

1. 1/v Absorbers

In the vicinity of a strong 1/v absorber, the Maxwellian portion of the spectrum becomes distorted due to preferential absorption of the lower energy neutrons so that the residual flux has a higher mean energy than it would otherwise have. This increase in mean energy (or temperature) is commonly called "spectral hardening." For simplicity, the Maxwellian portions of the sample curves given in Figures 1-3 have been constructed as undistorted curves with mean energies equivalent to the respective hardened fluxes.

The absorption-type reaction rates of the majority of all known isotopic nuclei in neutron fluxes are governed by the so-called "1/v" cross section. (The probability that any particular nucleus in a unit volume of material will absorb a neutron travelling through this material is inversely proportional to the velocity of the neutron.) For such an isotope the reaction rate, per nucleus of the material, is closely approximated by:

$$RR = (M + F) \sigma_0 \sqrt{\frac{\pi T_0}{4 \pi}}$$

where:

M , F , and T are defined as in equation (1),

$T_0 = 293 \text{ K}$,

σ_0 = absorption cross section measured for a neutron velocity of 2200 meters per second.

In this terminology the conventional "flux" usually stated in reference to kinetics calculations, etc. is given by: "Flux" = RR/σ_0 .

2. Treatment of Non-1/v Absorbers - Westcott Formulation

Some isotopes, notably those of heavier weight, have absorption cross sections which deviate markedly from this simple 1/v relationship, not only in the Maxwellian range of energies but also in the epithermal region where large increases or "resonances" occur. Various

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procedures have been used in reaction rate calculations for the latter type isotopes. One of the most convenient systems has been proposed by C. H. Westcott⁽³⁾ wherein each isotope is given an equivalent $1/v$ cross section, within limits, by the following simple formula:

$$\hat{\sigma} = \sigma_0 (g + r s)$$

where: g is a measure of the "non- $1/v$ -ness" of the isotope in the Maxwellian portion of the spectrum and is tabulated as a function of neutron temperature,
 s is a measure of departure from "1/v-ness" in the epi-Maxwellian region (also tabulated as a function of temperature),
 r is a measure of the proportion of epi-Maxwellian neutrons in the spectrum and is closely approximated by $F/(M + F)$ from equation (1),
 σ_0 has the same meaning as in equation (3).

In this terminology, a pure $1/v$ absorber has a g factor of 1 and an s factor of 0; a pure Maxwellian spectrum has an r factor of 0; an isotope having large epi-Maxwellian resonances and a small σ_0 has large s factors. Isotopes having low-lying resonances at the upper edge of the Maxwellian spectrum have g factors which increase markedly with neutron temperature (for example, Pu^{239}).

C. Application + Examples of Parameter Variations

Since several isotopes are usually involved in heavy isotope transmutations (fissile isotope, fertile isotope, product isotope, contaminant isotopes, etc.) and since each of these has its own set of g and s factors, it is obvious that the efficiency of the operation, in terms of side reactions, product losses, etc., is dependent on the character of the flux spectrum used. It also follows that a flux spectrum which is ideal for one type of transmutation may not be ideal for another type of transmutation. For example, improved efficiency in the conversion of U^{238} to Pu^{239} is usually accompanied by an increase in the value of the Westcott R factor brought about by an increase in the ratio of the number of uranium atoms to the number of moderator atoms; on the other hand, in the conversion of Th^{232} to U^{233} , one of the members of the side chain producing U^{232} contaminant, Pu^{231} , has a cross section which increases markedly with r (has large s factors). Consequently, in addition to other effects such as $n, 2n$ reactions and γ, n reactions (high energy reactions also involving solid angle dependencies), and $\text{Th}^{230} n, \gamma$ captures, a high r factor would tend to promote the growth of U^{232} contaminant in the U^{233} product.

In order to illustrate the influence of each of the lattice parameters on the flux spectrum, the series of curves shown in Figures 1-3 were constructed from data generated by the 7090 Code MOFDA.

(3) Westcott, C. H. Effective Cross Section Values for Well-Moderated Thermal Reactor Spectra, CRRP-960, Atomic Energy of Canada. November 1, 1960. 3rd Edition. (UNCLASSIFIED).

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1. Parameter Variations within Hanford Lattice

Figure 1 shows four curves for four different conditions in a Hanford, graphite-moderated, light-water-cooled, 8 3/8" x 8 3/8" lattice, each of which was calculated for approximately 1450 kilowatt maximum tube power in 23 1/2 feet of fuel charge.

Curve A, Figure 1, shows the flux spectrum inside green natural uranium, internally-externally cooled fuel with graphite temperatures about 500 C. Including the effects of spectral softening by the light water and spectral hardening due to absorptions in the fuel, the neutron temperature in the fuel is calculated to be about 680 K, $M = 6.46 \times 10^{13}$, $F = 0.296 \times 10^{13}$.

Curve B is a duplicate of A in all respects except that the graphite temperature has been lowered artificially to 100 C. Note that the epithermal flux remains essentially unchanged while the Maxwellian flux shifts to lower energies and a lower level at the same total fission rate. $T = 433$ K, $M = 4.97 \times 10^{13}$, $F = 0.293 \times 10^{13}$. The displacement between the right sides of the Maxwellian curves in A and B is in the area of a significant Pu^{239} capture resonance.

Curve C is also a duplicate of A except that the enrichment level is set at 0.947 per cent U^{235} . F is slightly reduced because an adjacent target column is a nonpower producer. A Maxwellian flux lower than Curve A is required to maintain the same fission rate since more U^{235} atoms are present. $T = 694$ K, $M = 4.85 \times 10^{13}$, $F = 0.252 \times 10^{13}$.

Curve D is for the same column power and same graphite temperature as for Curve A except that the fuel element is about twice as large, enriched to 1.3 per cent U^{235} and in an overbored process channel. F is reduced as a result of a lower resonance escape probability. Again the Maxwellian flux is greatly depressed because of the much larger number of U^{235} atoms present. The neutron temperature is appreciably higher due to an increase in spectral hardening. $T = 746$ K, $M = 1.654 \times 10^{13}$, $F = 0.224 \times 10^{13}$.

2. Example Parameter Variations in a Heavy Water Lattice

Figure 2 shows curves for two types of fuel in a D_2O cooled and moderated lattice. These curves are calculated to be approximately average conditions for 14-foot charges operating at 3965 kw with a moderator temperature of about 99 C and an average coolant temperature (midpoint of charge) of about 70 C.

Curve E is for natural uranium in the Mark VB design. Note that the epithermal flux level is appreciably higher than for the four Hanford examples. The power density (per unit volume) of this core is roughly an order of magnitude higher than for Hanford but the slowing down power, or

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Varied Flux Spectra in Hanford Lattices

- A Natural Uranium - Hot Graphite
- B Natural Uranium - Cold Graphite
- C 0.947 Enriched Uranium - Hot Graphite
- D 1.3 Enriched Uranium - Hot Graphite - Overbore

Figure 1

Figure 2
Varied Flux Spectra in Heavy-Water Lattices
E Mark VB Fuel - Natural Uranium
F Mark VIB - Orallooy Lithium Assembly

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Figure 3

Sample Flux Spectra in Target Assemblies
Graphite and Heavy-water Lattices

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"thermalizing" power of D₂O is much larger than that of graphite. Spectral hardening in the rather massive fuel assembly brings the calculated neutron temperature to about 433 K. $F = 0.86 \times 10^{13}$, $M = 6.42 \times 10^{13}$.

Curve F is an estimate of the average flux spectrum in the or alloy driver portions of the Mark VI assembly operating at the same tube power, moderator and coolant conditions as for Curve E. The epithermal flux is appreciably higher than for Curve E due to the removal of U²³⁸ resonance capture. The Maxwellian portion of the curve is depressed due to the over-all thermal "blackness" of the assembly. $T = 414$ K, $F = 1.15 \times 10^{13}$, $M = 3.25 \times 10^{13}$.

3. Sample Thorium Target Lattice Comparisons

Figure 3 illustrates sample flux spectra calculated for two possible, but not necessarily prototypical, thorium target designs.

Curve G is for rather low density thorium oxide in a Hanford lattice. $T = 620$ K, $F = 0.288 \times 10^{13}$, $M = 8.03 \times 10^{13}$.

Curve H is for a more massive element of thorium metal in a Savannah River lattice. $T = 407$ K, $F = 0.89 \times 10^{13}$, $M = 9.89 \times 10^{13}$.

As pointed out in the previous discussion, this figure would be useful in predicting U²³³ buildup rates but would indicate only one of the considerations in the U²³² contaminant buildup problem.

CONCLUSIONS

The curves given in Figures 1-3 show that significant variations in flux spectra can be created in either graphite-moderated or heavy-water-moderated nuclear reactors. In general, the Maxwellian portions of the Hanford fluxes tend to peak at higher energies due to the higher moderator temperatures; epi-Maxwellian fluxes reach higher levels in heavy-water reactors due to their higher core power densities per unit volume of core.

ACKNOWLEDGEMENTS

My thanks are due to E. R. Perl for calculating and constructing the curves shown in the figures.

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