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EUROPEAN ATOMIC ENERGY COMMUNITY - EURATOM

DEPOSITION OF FISSION ENERGY IN AN ORGEL-TYPE REACTOR

by

B. HAYTINK and E. SCHMIDT

1968

ORGEL Program
Joint Nuclear Research Center
Ispra Establishment - Italy
ORGEL Project

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SUMMARY

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KEYWORDS

ORGEL REACTOR
ORGANIC COOLANT
FISSION PRODUCTS
RADIOLYSIS
ENERGY
NUMERICALS
FISSION

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1. INTRODUCTION

The energy released by fission of atomic nuclei, and available in a nuclear chain reactor, arises in various forms, namely as kinetic energy of fission fragments and neutrons, as gamma-ray energy and as beta-ray energy. The mechanisms of interaction with mass are strongly different for these kinds of energy and, consequently, one has to establish different calculation techniques, in order to evaluate the deposition of the energy. The knowledge of the dissipation rate of fission energy to the individual constituents of an ORGEL-type reactor is, for three reasons, interesting.

Firstly, knowing the liberated energy per fission, one has to determine a certain efficiency factor which gives the fraction of the liberated energy by the fission process, available as useful thermal energy. This fraction of useful thermal energy may be identified with the energy deposited to fuel elements and coolant. Depending on the size of the reactor and its construction and composition, the amount of the useful thermal energy varies with the reactor type. It influences the needed quantity of fissions in order to generate the required thermal power and, by this, in a slight measure, the consumption of fissile material per unit of thermal energy.

Secondly, an estimate of the energy deposited to the moderator is needed, in order to be able to judge the necessary expense connected with the cooling system for the moderator.

Thirdly, owing to the application of organic coolants, a special problem arises in an ORGEL-type reactor: organic molecules are decomposed by radiation. The decomposition rate per unit of absorbed energy depends strongly on the kind of radiation. The decomposition effect causes a noticeable term in the power generation cost for this reactor type. This fact leads to the necessity to evaluate the damaging rate of the coolant as accurate as possible.

In the scheme of the present work, this task is reduced to the computation of the energy fractions of each kind of radiation, which are dissipated to the coolant.

In so far as the work is referred to the computation of the energy balance in the reactor, it must be emphasized that thermal effects as, e.g., heat conduction, are not investigated here.

The aim of the present report shall be to summarize the source strengths for the individual kinds of energy liberated by the fission process, to propose a calculation model for their deposition to the reactor materials and to give the numerical results for the energy balance for the 250 MWe-ORGEL prototype reactor. For reasons of its strong decomposing effect, it has been tried to treat the neutron energy deposition particularly exactly.

2. ENERGY LIBERATED IN FISSION

2.1. Neutron energy

The average kinetic neutron energy E_n appearing in the prompt fission neutron spectrum may be related to the average number of fission neutrons per fission, ν, as follows (Ref. 1):

$$E_n \ [\text{Mev}] \simeq 0{,}78 + 0{,}62 \sqrt{\nu+1}$$

For the four most common thermally fissionable nuclei, the "World Consistent" thermal values of ν are taken from (Ref.2):

U-233:	2.51
U-235:	2.47
Pu-239:	2.80
Pu-241:	3.06

Neglecting the difference in E_n for the small part of delayed neutrons, the total kinetic neutron energy $E_{n,k}$ for the above-mentioned k isotopes, released per thermal fission, is equal to the product of v times E_n and amounts to:

4.9 Mev for k = U-233
4.8 Mev for k = U-235
5.8 Mev for k = Pu-239
6.2 Mev for k = Pu-241

The variation of v with the energy of the incident neutron may be derived by the relation of:

$$V(E) = V(\text{thermal}) + \frac{dv}{dE} \cdot dE,$$

where the $\frac{dv}{dE}$ - values lie between 0.1 and 0.15 Mev^{-1} for the above-given isotopes. Introducing the so-calculated v-values in formula (1), one can see that, e.g., for U-235 $E_{n,k}$ is increased for about 0.04 Mev for each 1-Mev increase of bombarding energy. The smallness of this correction and the fact that the greatest fraction of fissions in a thermal reactor happens at thermal energy, justifies the approximation that the released energy per fission in the above-cited fissile isotopes is equal to the energy released by thermal fission.

In order to define quite generally the mean kinetic neutron energy E_n arising per fission in the reactor, one has still to investigate the fast fission effects in the fertile materials U-238 and Th-232 and the additional secondary effects, as the (n, 2n)- and (γ, n)- reaction in heavy water. A calculation for a typical ORGEL lattice cell, containing D_2O as moderator, HB-40 as coolant, SAP as canning, Zircalloy as pressure tube and calandria tube, and slightly enriched uranium as fuel, showed that only about one tenth of the neutrons generated by fast fission of U-238 is contributed by the (n, 2n) process in deuterium. Therefore, this source of neutron energy is neglected here. Similar considerations lead to the

neglect of the (γ, n)- reaction in deuterium as neutron source: the average cross-section for this process amounts only to about 2.2 mb for the energy range of interest and only about a quarter of the whole γ -sources has energies higher than the threshold energy for this reaction (2.23 Mev).

Consequently, it remains the fast fission effect.

Taking the average v-values for U-238 and Th-232:

U-238 : 2.90
Th-232: 2.60

the relation (1) provides the energy, carried by one neutron which has been released by fast fission. After multiplying these values by the corresponding v, the following kinetic neutron energies per fast fission event, $E_{n,\,f,\,k}$ are obtained:

k = Th-232: 5.1 Mev
k = U-238: 5.8 Mev

The mean kinetic neutron energy E_n generated per fission in the reactor results now from the suitable combination of these individual reactions. The correct weighting factors for the $E_{n,\,k}$ are the probabilities that the corresponding reaction occurs. Considering a mixture of thermally fissionable materials, the ratio of processes in the individual isotopes is determined by the effective macroscopic fission cross-sections Σ_{fi}. The fast fission ratio δ gives the ratio of fast fissions in fertile isotopes to fissions in thermally fissionable isotopes.

Hence, the general formula for E_n may be written as:

$$\bar{E}_n = \frac{\left(\sum_k E_{n,f,k} \cdot \delta_k\right) \sum_k \Sigma_{fi,k} + \sum_k \Sigma_{fi,k} \cdot E_{n,k}}{\sum_k \Sigma_{fi,k} \,(1+\delta)}$$

2.2. Gamma-ray energy and spectra

The γ-ray sources in a reactor are more numerous than the ones of neutrons. In the present report they may be grouped as follows:

(a) Prompt fission γ-rays, including the short-life γ-radiation emitted in a time interval till 10^{-6} sec. after the fission event.

(b) Fission-product γ-rays, including the intermediate life and delayed γ-radiation available in a reactor (this means, emitted till about 5.10^{7} sec. after the fission event).

(c) γ-radiation due to (n,γ)- processes in the fuel.

(d) γ-radiation due to (n,γ)- processes in the non-fuel reactor constituents.

(e) Inelastic scattering γ-rays.

(f) γ-rays by annihilation of positrons.

(g) γ-rays by Bremsstrahlung.

The collection of the individual yields and spectra of the sources has been mainly based on the techniques and data reported in (Ref. 6). There, an extensive survey about experimental and theoretical results is given. But partly these data have been updated by means of results of (Ref. 7) and (Ref. 8). In the following, some details concerning the individual γ-sources are reported.

(a, b) The prompt fission and fission product γ-rays depend on the fissioned isotope and the correct way to evaluate the γ-radiation for the reactor would be to apply a

similar averaging prescription as for the liberated kinetic neutron energy. But for lack of suitable information, only the γ -rays by fissioning of the U-235 nucleus have been tabled here. It is hoped that the error, introduced thereby in the entire energy balance for the γ -sources of the reactor, will not be significant. In a certain measure, this assumption is supported by (Ref. 5), pp. 25-27, where it is mentioned that, for the prompt fission γ-ray spectrum shape of U-233 and Pu-239, no observable differences from that of U-235 has been found by means of experiments. Recent experimental results seem to prove that the same can not be said for the fission product γ -rays. However, the differences in spectrum as well as in source strength for the two isotopes Pu-239 and U-235, which sustain mainly the fission process in the here-considered reactor, are not remarkable.

Under these circumstances, the γ -sources of fission in U-235 are taken as γ-sources for any fission process in the reactor. They are listed in Table 1.

TABLE 1

γ-spectra by the fission process

Upper energy limit [Mev]	Prompt fission γ -rays $\left[\frac{\text{Mev}}{\text{fission}}\right]$	Fission product γ -rays $\left[\frac{\text{Mev}}{\text{fission}}\right]$
0.5	1.37	0.72
1.0	1.80	1.35
1.5	1.43	1.54
2.0	0.95	1.29
3.0	1.33	1.41
4.0	0.58	0.57
6.0	0.39	0.22
8.0	0.05	
TOTAL	7.90	7.10

(c) After having captured a neutron, the compound nucleus is brought to an excited state. The excess energy is often emitted by means of γ- quanta. Per capture of a thermal neutron in U-235, 6.42 $\frac{\text{Mev}}{\text{capture}}$ γ-energy are released. By multiplication with the capture processes per fission, one receives the yield of γ-energy.

If $\Sigma_{c,\,235}$ means the macroscopic effective capture cross-section in U-235 and $\bar{\Sigma}_f$ the macroscopic effective fission cross-section of the fuel, this yield amounts to:

$$E_\gamma = \frac{\bar{\Sigma}_{c,\,235}}{\bar{\Sigma}_f} \cdot 6.42 \left[\frac{\text{Mev}}{\text{fission}}\right]$$

Similarly, the thermal capture in U-238 may be treated. If $\Sigma_{c,\,238}$ means the macroscopic effective capture cross-section for thermal neutrons in U-238, this yield amounts to:

$$E_\gamma = \frac{\Sigma_{c,\,238}}{\Sigma_f} \cdot 4.70 \left[\frac{\text{Mev}}{\text{fission}}\right]$$

The quantity 4.70 gives the released γ-energy in Mev per absorption event. In addition to the thermal absorption in the fuel, one has still to investigate the resonance absorption. The number of neutrons captured in the resonance region per fission may be derived by:

$$\nu \cdot \varepsilon \cdot (1-p) \cdot e^{-B^2\tau}$$

where ν, ε, p and $e^{-B^2\tau}$ mean the number of created neutrons per fission, the fast fission factor, the resonance escape probability and the leakage of fast neutrons from the reactor core, respectively. The yield of this process amounts approximately to:

$$E_\gamma = \nu \cdot \varepsilon \cdot (1-p) \cdot e^{-B^2\tau} \cdot 4.7 \left[\frac{\text{Mev}}{\text{fission}}\right]$$

At equilibrium, every capture in U-238 is followed by one disintegration of a U-239 nucleus and a Np-239 nucleus. Thus one has a simple relation to the capture processes in U-238. The energy released by the disintegration per capture in U-238 amounts to 0.4 Mev.

(d) The number of parasitic absorptions per fission in the reactor lattice cell may be derived by:

$$\frac{\nu}{\eta} \cdot \frac{1-f}{f}$$

A subdivision of this capture rate to the different nuclei within the cell has to be done in proportion to the product of macroscopic absorption cross-section and mean flux in the corresponding material. The emitted γ-energy and its spectral distribution for capture processes is tabled for a lot of kinds of nuclei in (Ref. 5).

(e) During the slowing-down process, fast neutrons suffer inelastic scattering and lose thus kinetic energy. The target nucleus is excited to some level above ground state and, within a very short time after the scattering event, the excess energy is lost by emission of γ-quanta.

It is supposed that the total neutron energy transferred to the target nucleus arises as γ-energy.

For U-238, the γ-spectrum as function of the energy of the colliding neutron is given in (Ref. 6).

(f, g) Each time a positron is annihilated, two 0.5 Mev γ-quanta are created. Positron emitters are only rarely to be found in the reactor. Therefore, the main source of positrons is represented by the pair

decay of γ-radiation. Because of its small range, it may be assumed that the recombination process will take place at the location of the creation of the positron. Consequently, by this process γ-energy is not carried away. There is only an internal change of the spectrum distribution.

Bremsstrahlung is generated by deceleration of charged particles. Since the β-radiation generated in the reactor is not sufficiently energetic, this process does not contribute remarkably to the γ-source of the reactor. Moreover, the Bremsstrahlung has a rather weak spectrum and is therefore absorbed with great probability in the neighborhood of its place of generation.

2.3. Beta-ray energy and spectrum

During the decay of the fission products of U-235, about 7.6 $\frac{\text{Mev}}{\text{fission}}$ in form of kinetic energy of β-particles is released. Since no corresponding information for the other fissionable isotopes have been found in the literature, this term is taken to be representative for all fissions which happen in the reactor. The spectrum has been taken from (Ref. 9) and is reproduced in Table 2.

TABLE 2

β-spectrum by fissioning in U-235

Energy range [Mev]	Average energy [Mev]	β-source, Q_β $\left[\frac{\text{Mev}}{\text{fission}}\right]$
0.1 - 0.4	0.25	0.46
0.4 - 0.9	0.65	1.82
0.9 - 1.35	1.13	2.10
1.35 - 1.8	1.58	0.93
1.8 - 2.2	2.00	1.47
2.2 - 2.6	2.40	0.18
> 2.6	3.20	0.64
	TOTAL	7.60

2.4. Kinetic energy of fission fragments

Although the computation of the deposition of the fission fragment energy does not represent difficulties, its absolute value may be tabled here for the individual isotopes. The average fission fragment energy, $\bar{E}_{ff}$, per fission in the reactor may then be calculated in the same way as $\bar{E}_n$, specified in chapter 2.1.

The fission fragments lose their kinetic energy by ionizing and exciting the atoms along their paths as well as by atomic collisions. Their maximum range in uranium metal and aluminium amounts to $6.7.10^{-4}$ cm and $1.4.10^{-3}$ cm, respectively. Therefore, one may conclude that all the fission fragment energy is dissipated to the fuel.

Taking the values of (Ref. 10), p. 14, and normalizing to the well-known fission fragment energy of a U-235 nucleus, splitted by a thermal neutron, one obtains Table 3.

TABLE 3

Fission fragment energy for different isotopes

Fissile nucleus	E_{ff} $\left[\frac{\text{Mev}}{\text{fission}}\right]$
U-233	162.8
U-235	166.0
U-238	167.9*
Pu-239	173.4

* Assuming an average energy of the incident neutron of 2.5 Mev. The value varies strongly with the energy of the incident neutron.

3. ENERGY DEPOSITION BY NEUTRONS

3.1. General remarks

In subsection 2.1., the formula for the kinetic energy of neutrons $\overline{E}_n$ created by an average fission event in the reactor has been specified. Now one has to investigate in which way this energy is transferred to the individual reactor constituents and by which calculation models the transfer mechanisms may be adquetely described.

In passing through matter neutrons can interact with atoms by the following processes:

Elastic scattering
Inelastic scattering
Capture
Fission

Only in the first of these processes, the loss of kinetic neutron energy agrees with the recoil energy of the interacting nucleus. Referring to the decomposition of the organic liquid by dense ionisation, it is consequently only this process which must be taken into account. In the framework of the present report, it is assumed that the neutron energy, lost by inelastic scattering, appears completely as γ-radiation. Captures and thermal fissions happen at low energies. Neutrons involved in these processes carry only a small amount of energy, which is without importance for the balance of the kinetic neutron energy as well as for the polymerisation effect in the organic coolant and thus it is neglected here. Also the fast fission effect influences only slightly the neutron energy balance, because the greatest deal of fissions occurs at thermal energies. It may be taken into account diminishing the energy generated per fission in the reactor for an amount E_F,

$$E_F = C \cdot \frac{\delta}{1 + \delta}$$

where δ is the fast fission ratio of the lattice cell and c is the mean energy in Mev of the neutron which causes the fast fission process.

Finally, there is a further event which reduces the kinetic neutron energy available in a reactor, namely the leakage of fast neutrons from the core. The method to evaluate it shall be described in the subsequent section.

3.2. Determination of the neutron energy leakage from a finite core

An obvious technique to derive the energy leakage by neutrons from the core would be the solution of the multi-group diffusion equation. The term $D_k B^2 \phi_k$ would, after normalization for the flux, represent the fraction of neutrons of the k^{th} energy group, escaping from the core if D, B^2 and ϕ mean the diffusion coefficient, the buckling and the neutron flux. Unfortunately, the buckling method is rather unsuitable for the fast neutron calculations. But just these neutrons transport the most energy out of the core. Therefore, one must look for a more appropriate calculation procedure.

Supposing a uniform spatial fission source and a sufficient smallness of the energy interval for an energy group of neutrons, the neutron current from the core j_c (E) can be better approximated by the aid of the theory of collision probabilities:

$$j_c(E) = \frac{\nu N(E)}{4 \bar{\Sigma}_c(E)}$$

N (E), ν and $\bar{\Sigma}_c$ (E) are the fraction of neutrons emitted per energy interval by the fission process, the number of neutrons created per fission and the removal cross-section of the homogenized core, respectively. All thses data may be procured by the code GAM (Ref. 12). Multiplying j_c (E) by the ratio of surface to volume of the reactor core, A_c/V_c,

and by the energy E [Mev] related to the corresponding neutron group and integrating about all possible energies, one gets the leakage energy E_L from the core

$$E_L \left[\frac{\text{Mev}}{\text{fission}}\right] = \frac{\nu A_c}{4 V_c} \int_0^\infty \frac{N(E)\, E\, dE}{\Sigma_c(E)}$$

The integral in the formula for E_L may be replaced by a sum of energy groups without losing accuracy, if a sufficiently great number of groups is chosen.

3.3. Energy transfer by scattering

The remaining energy, which is deposited within the reactor core by scattering processes, amounts to E_S

$$E_s = \bar{E}_n - E_F - E_L$$

Now, E_s has to be distributed correctly to the different materials of the core. A mean value of energy deposition for the core may be derived, restricting the consideration to a single lattice cell. If a space-dependent deposition rate for the kinetic neutron energy is needed, one could obtain it by weighting with the fast neutron flux distribution in the core.

The transfer of neutron energy to the i^{th} material in the lattice cell is proportioned to the number of collisions in this material and to the mean energy loss per collision.

For the number of collisions R_i, the formula holds

$$R_i \left[\frac{\text{collisions}}{\text{sec}}\right] = V_i \int_{E_l}^{\bar{E}_u} N_i \phi_i(E) \left[\sigma_i^{el}(E) + \sigma_i^{inel}(E)\right] dE \,,$$

where V_i [cm^3], $N_i \left[\frac{\text{nuclei}}{cm^3}\right]$, $\phi_i(E) \left[\frac{\text{neutrons}}{cm^2\, sec}\right]$, $\sigma_i^{el}(E)$ [cm^2] and $\sigma_i^{inel}(E)$ [cm^2]

mean the volume of the i^{th} material in the lattice cell, the atomic density, the mean neutron flux as function of the energy, the microscopic elastic scattering cross-section and the microscopic inelastic scattering cross-section in the i^{th} material, respectively. The integration covers all energies which are interesting for the energy transfer. E_ℓ means the lower integration limit, E_u the upper one.

One gets the energy deposition to the material i, E_{si}, multiplying the term under the integral in the formula for R_i by the energy loss per collision $\Delta E_i(E)$,

$$E_{si} = V_i \int_{E_\ell}^{E_u} N_i \phi_i(E) \left[\sigma_i^{el}(E)\, \Delta E_i^{el}(E) + \sigma_i^{inel}(E)\, \Delta E_i^{inel}(E)\right] dE$$

If ΔE is inserted with the dimension [Mev] also E_{si} will get this dimension.

It stands to reason that it is impossible to evaluate the integral for E_{si} exactly. But it is possible to construct for $\phi(E)$, $\sigma(E)$ and $\Delta E(E)$ step functions. Then the integral can be replaced by a sum of terms of the various energy groups :

$$E_{si} = \sum_j V_i N_i \phi_{i,j} \left[\sigma_{i,j}^{el} \Delta E_{i,j} + \sigma_{i,j}^{inel} \Delta E_{i,j}\right]$$

As mentioned above, index i refers to the material; the new index j refers to the energy group. It runs from 1 to the number of groups employed.

3.4. Determination of the cross-sections as function of the energy

The sum-formula for the deposited neutron energy of the preceding sub-section may be rewritten as

$$E_{si} = V_i N_i \sum_k \sum_j \phi_{k,i} \left(\sigma_{k\to j,i}^{el} \Delta E_{k\to j} + \sigma_{k\to j,i}^{inel} \Delta E_{k\to j}\right)$$

k and j are running indexes for the number of employed energy groups, m.

$$1 \leq k_{ij} \leq m \quad , \quad k < j$$

$\sigma_{k \to j,i}$ is the elastic or inelastic transfer cross-section from group k to group j for the material i.

$\Delta E_{k \to j,i}$ is the difference in energy between group k and j. $\phi_{k,i}$ is the neutron flux in arbitrary units within the energy group k and in the material i.

The choice of number of groups will depend on the grade of variation of the parameters applied in the formula. In any case, it seems advisable to treat the high-energy spectrum as accurate as possible, taking there a rather fine subdivision of the groups, because the energy loss per elastic collision is directly proportional to the energy at which the scattering event occurs, what results in a high weight for these groups in the sum of the whole dissipated energy. A quantitative estimate of the distribution of the deposited energy as function of the energy at which the collisions happen is cited in (Ref.11). It was found there that about 99% of the deposited energy were transferred by the neutron groups above 5 kev. From this fact one can conclude that the choice of the lower energy limit will hardly influence the result and that the difficulties which could arise by the resonance absorption will be of no account in connection with the problem of the energy deposition.

The upper energy boundary is determined by the fission spectrum. In practice one may choose 10 Mev for it, because the fraction of neutrons with higher energies is a negligible quantity.

The transfer cross-sections appearing in the formula for the calculation of E_{si} can be obtained executing a GAM computation (Ref.12) for the moderator zone of the lattice cell and a central homogenized zone, containing all the other cell constituents.

The ΔE values result automatically from the group division.

3.5. Spatial distribution of the fast spectrum in the lattice cell

Although the subdivision of the lattice cell, described above, is somewhat schematic, it is believed that the employment of the so-derived cross-sections will not lead to appreciable errors in the E_{si}'s. But there is a geometrical problem which can influence considerably the energy deposition to the organic liquid: fissions happen only in that region of the lattice cell where fuel exists. In the case in investigation, the fuel is surrounded by the organic liquid. Consequently, the likelihood for the first collisions of neutrons with the organic liquid will be greater than the probability that the first collisions will occur in the heavy water. And since the first collisions are the most efficient events referring to the energy transfer - particularly in the case of hydrogen - one would obtain an underestimate of the energy deposited to the organic liquid. Hence one can not use the spectrum distribution calculated by GAM. Rather, one has to investigate this space-dependent effect by a code as the Winfrith-DSN code (Ref.13), taking the same subdivision of the lattice cell as for the GAM-calculation. Inserting the $\phi_{k,\,i}$ obtained by the DSN code in the formula for E_{si} of sub-section 3.4., the fractions of the available neutron energy are received, which are dissipated to the i materials in the cell. These E_{si}, finally, have to be multiplied by the normalization factor N,

$$N = \frac{E_s}{\sum_i E_{si}}$$

in order to find the energy rates in Mev per fission deposited to the individual materials.

4. ENERGY DEPOSITION BY GAMMA-RAYS

4.1. Survey about energy transfer and calculation model

The energy transfer of γ-rays to matter occurs mainly by the following three processes:

Photoelectric effect
Compton effet
Pair production

The photoelectric effect results in the conversion of nearly the whole γ-energy to kinetic energy of an electron. It is predominant for low-energy γ-rays and for heavy atoms and decreases rapidly with increasing energy.

The Compton effect is characterized by the fact that only a fraction of the energy of the photon is deposited to the atom. This imparted energy appears as recoil energy of an orbitral electron.

Pair formation is released only by photons of energies higher than 1.02 Mev. The probability for this process is increased with the photon energy and the atomic weight of the nuclei. The absorbed γ-energy is converted to mass in form of one electron and one positron. The excess energy of the γ-quantum over 1.02 Mev appears as kinetic energy of the particles.

The electrons and positrons generated by these processes in turn lose their energy by inelastic collisions with atoms. The atoms are ionized or excited. In the last case, the atoms return to their ground state, emitting quanta of heat and light. As mentioned in section 2.2., a small part of the energy of electrons and positrons may be again re-converted to γ-energy by annihilation of positrons and by Bremsstrahlung.

Since the migration range of electrons and positrons is rather small (see β-ray energy deposition!), it seems to be justifiable to assume that energy lost by γ-rays in Compton scattering and photoelectric process is converted into heat directly at the point of collision.

Similarly as for the evaluation of the neutron energy deposition, the starting-point of the considerations regarding the dissipation of γ-energy in the reactor core will be the known total γ-ray source and the spectra derived by the methods discussed in subsection 2.2. Also the general course of the calculation is the same as for neutrons.

The finite dimensions of the reactor are taken into account by reducing the available source-strength for the energy leakage. Then the reactor is treated as an infinite one, where in each lattice cell the like processes occur. Therefore, the investigation may be restricted to a single lattice cell. A γ-ray photon often suffers as many as five to ten scattering processes before its eventual absorption and the initial straight penetration goes over into a diffusion process. This fact justifies in a certain measure to investigate the γ-energy deposition taking a flat γ-flux distribution in a lattice cell and the application of a homogenized cell model during the evaluation of the dissipation rates to the individual constituents of the lattice cell. Therefore, in a first step the corresponding formalism is developed.

Corrections for the homogeneous model must be derived in the case that there is a remarkable self-shielding effect for γ-rays in a heterogeneous lattice cell, where the main fraction of the γ-sources is located in that part of the cell which contains the fuel. But this effect is negligible, if the fuel zone is not too thick. However, a method to determine this heterogeneity factor is described in one of the following sub-sections.

4.2 Gamma-energy leakage from the reactor core

As for neutrons, the finiteness of the reactor shall be taken into account by an appropriate decrease of the gamma-ray source strength Ψ_j for the j^{th} energy group. The Ψ_j's result from the technique discussed in section 2.2. They are introduced in an approximative formula for the gamma - energy leakage from the core, which agrees in its construction with the one for the neutron leakage and has been taken from (Ref. 11) :

$$\Psi_j^* \left[\frac{\text{Mev}}{\text{fission}}\right] = \frac{A_C}{4 V_C} \frac{\Psi_j}{\mu_{h,j}} \left[\frac{A_{1,j}}{1+\alpha_{1,j}} + \frac{A_{2,j}}{1+\alpha_{2,j}}\right]$$

A_C and V_C are, as in the case of the neutron leakage, the surface and volume of the core, respectively. $A_{1,j}$, $A_{2,j}$, $\alpha_{1,j}$ and $\alpha_{2,j}$ are energy buildup constants for gamma-rays and $\mu_{h,j}\left[\frac{1}{cm}\right]$ is the total linear attenuation coefficient. All the constants are that of the homogenized core. The γ-sources are taken to be uniformly distributed throughout the core.

4.3. Calculation model for γ-energy deposition in an homogenous Core

Once the γ-sources corrected for the leakage, one will receive averaged dissipation rates of γ-energy, treating the reactor core as an infinite one. This corresponds to the assumption that all gamma-quanta produced by the fission processes must also be absorbed by the individual constituents of the reactor. For an homogeneous medium, the γ-energy flux is given by

$$\phi_{h,j}\left[\frac{\text{Mev} \times \text{cm}}{\text{fission}}\right] = B_j \frac{\Psi_j^*}{\mu_{h,j}} .$$

The subscripts h and j denote the homogeneous model and the j^{th} energy group, respectively. B_j is the corresponding energy absorption buildup factor; $\Psi_j^*\left[\frac{\text{Mev}}{\text{fission}}\right]$ means the for leakage reduced gamma-ray source strength and $\mu_{h,j}\left[\frac{1}{cm}\right]$ the

total linear attenuation coefficient.

Equating the generated energy per volume unit with the absorbed energy

$$\sum_j \psi_j^* = \sum_j \mu^e_{h,j}\, \phi_{h,j} \equiv \sum_j B_j \frac{\mu^e_{h,j}}{\mu_{h,j}}\, \psi_j^*$$

one obtains the buildup factor:

$$B_j = \frac{\mu_{h,j}}{\mu^e_{h,j}}$$

$\mu^e_{h,j}$ $\left[\frac{1}{cm}\right]$ is the energy absorption coefficient.

Thus the energy flux in an homogeneous medium is reduced to

$$\phi_{h,j} \left[\frac{\text{Mev x cm}}{\text{fission}}\right] = \frac{\psi_j^*}{\mu^e_{h,j}}$$

The fraction of gamma-energy absorbed in the i^{th} material amount then to

$$E_i \left[\frac{\text{Mev}}{\text{fission}}\right] = \sum_j f_i\, \mu^e_{i,j}\, \phi_{h,j} \equiv \sum_j f_i\, \mu^e_{i,j} \frac{\psi_j^*}{\mu^e_{h,j}}$$

where f_i is the volume fraction of the i^{th} material in the lattice cell. Employing the energy absorption mass attenuation coefficients for every energy group j and for each material i, $\left(\mu^e/\rho\right)_{i,j}$, the equation above may be rewritten:

$$E_i \left[\frac{\text{Mev}}{\text{fission}}\right] = \sum_j a_i \left(\mu^e/\rho\right)_{i,j} \frac{\psi_j^*}{\left(\mu^e/\rho\right)_{h,j}}$$

As already said, h refers to the homogeneous model. a_i denotes the weight fraction of the i^{th} material in the lattice cell.

It may be noticed that the somewhat strange unit for the γ-energy flux $\phi_{h_{ij}}$ is caused by the normalization procedure. Normalizing the power of the reactor to one fission per cm3 per sec., the unit of ϕ is converted to $\left[\frac{\text{Mev}}{\text{cm2 sec}}\right]$.

4.4. Corrections for the heterogeneity of the lattice cell

The decision whether or not one must execute a calculation employing a heterogeneous technique, which is more troublesome and time-consuming, depends mainly on the self - shielding of the fuel zone and, hence, on its geometrical dimension.

Nearly all the γ-rays are generated in the fuel zone of the considered reactor type. The energy generated and absorbed in this zone may be evaluated with reasonable accuracy by considering only the first absorbing collision and assuming a uniform source strength distribution in the fuel.

$$E^{(1)} = \sum_j \varphi_j^* \times \bar{r}_{c_{ij}} \cdot (r \times \mu_j^e)$$

P_c is the collision probability defined in (Ref. 14), $r\,[cm]$ is the radius of the cylindrical source and μ^e is the corresponding energy absorption coefficient. j refers, as above, to the energy group.

If $E^{(1)}$ exceeds a certain fraction of the γ-energy absorbed within the fuel zone and derived by the homogeneous technique discussed in the preceding subsection, one can conclude that the homogeneous model is not appropriate for this case.

Then one has in addition to the fraction of energy, which arises in the zone and which is also absorbed here, to add a fraction of γ-energy which is dissipated to this zone

coming from neighbouring fuel elements. It may be estimated from the approximate formula given in (Ref. 11)

$$E^{(2)} = \sum_k \frac{\tau}{\pi(a_k+\tau)} \sum_j \left[Ki_2(y_{j,k}) - Ki_2(y_{j,k}+\alpha_j)\right] \psi_j^*$$

where the subscript k refers to all sources of γ-energy within the core. In practice, one can of course neglect sources which are sufficiently far from the considered fuel zone.

The functions Ki are tabulated in (Ref. 15) and a [cm] means the shortest distance between the surfaces of the both interacting source zones. For the arguments α and y, the following relations hold:

$$\alpha = \frac{\pi\tau}{2}\mu_{sz}$$

$$y = a\mu_m + \tau\mu_{sz}$$

where $\mu_{sz}\left[\frac{1}{cm}\right]$ and $\mu_m\left[\frac{1}{cm}\right]$ represent the total linear attenuation coefficients of the source zone and of the moderator, respectively.

In the present case, as source zone a homogeneous zone of the lattice cell, containing all materials situated inside the calandria tube, was defined. The outer zone consists consequently only of moderator. The homogenization of the central source zone for γ-radiation is justified because the rather thin pins of the investigated 18-rod fuel cluster do not show appreciable self-shielding.

The self-shielding factor SF for the central source zone of the lattice cell may be defined quite generally as

$$SF = \frac{\sum_m \sum_j f_m \mu^e_{m,j} \bar{\phi}_{central,j}}{\sum_m \sum_j f'_m \mu^e_{m,j} \phi_{h,j}} = \frac{E^{(1)} + E^{(2)}}{\sum_m \sum_j \mu^e_{m,j} \phi_{h,j}}$$

The subscript m shall refer to the materials within the central zone. $\bar{\phi}_{central}$ is the mean γ-flux in the central zone and f_m, f'_m are the volume fractions of the corresponding materials within the source zone and the whole lattice cell.

Hence, the energy dissipated really to the material m in the source zone is given by the product of SF and the rate coming out by means of the model which treats the lattice cell as an homogeneous mixture of its constituents and a flat γ-ray flux distribution :

$$E_m = SF \sum_j a_m \left(\mu^e/\rho\right)_{m,j} \frac{\psi_j^*}{\left(\mu^e/\rho\right)_{h,j}}$$

Automatically, this relation leads to a correction for the energy absorbed in the outer zone because the sum of the total absorbed gamma-energy must be constant, namely, $\sum_j \psi_j^*$.

5. ENERGY DEPOSITION BY BETA PARTICLES

Under normal conditions, the evaluation of the beta energy deposition for power reactors does not represent any problem because the fuel is surrounded by a canning which absorbs the electrons very locally. Only for high-energetic β-rays and for small sheath thicknesses of the fuel pins a fraction of beta particles, created directly by the fission process in the fuel, could enter in the coolant and then it would be absorbed here. It seems advisable to check for every reactor type anew whether a remarkable decomposition of the organic coolant is caused by that radiation source.

The beta-energy generated by one fission together with the spectrum distribution for 7 energy groups and the mean energy per group are tabulated in sub-section 2.3. The formulae for the evaluation of the escaping β-particles from a source are derived in (Ref. 9) for a source slab which extends itself in one direction to infinite.

In order to stay on the safe side by employing the formulae from (Ref. 9) for the energy absorption by the organic liquid, E_β, in the case of cylindrical fuel pins (e.g., in order to overestimate the absorption rate there!), the following expressions are suggested:

$$E_\beta \left[\frac{\text{Mev}}{\text{fission}}\right] = \begin{cases} 0 & \text{for } R < D \\ \frac{Q_\beta}{2RT}\left[(R-D)(R-3D) - 2D^2 \ln\frac{D}{R}\right] & \text{for } D < R < D+T \\ \frac{Q_\beta}{2RT}\left[T(4R-3T-6D) + 2(T+D)^2 \ln\frac{T+D}{R} - 2D^2 \ln\frac{D}{R}\right] & \text{for } D+T < R \end{cases}$$

$Q_\beta \left[\frac{\text{Mev}}{\text{fission}}\right]$, T [mm], D [mm], R [mm] mean the uniform β-ray source in the fuel pin, the radius of the pin, the effective thickness of the canning and the range of beta particles, respectively. Inserting the energy of the particles in units of Mev, the range may be calculated by

$$R\ [\text{mm}] = \begin{cases} 1.795\ \frac{E^n}{N \times Z} & \text{for} \quad 0.01 \leq E \leq 3.0 \\ \frac{1.535\ E - 0.308}{N \times Z} & \text{for} \quad 1.0 \leq E \leq 20 \end{cases}$$

where $N\left[\frac{10^{24}}{\text{cm}3}\right]$ and Z are the atomic density and the atom number of the absorbing material.

Finally, for the effective canning thickness, the relation holds

$$D = D_a \ \frac{(N \times Z)\ \text{canning}}{(N^X Z)\ \text{fuel}}$$

with D_a equal to the actual cladding thickness.

6. NUMERICAL RESULTS FOR THE ENERGY DEPOSITION RATES WITHIN THE 250 MWe-ORGEL PROTOTYPE*

6.1. Description of the reactor

The calculations have been executed for the un - irradiated initial core of the 250 MWe-ORGEL prototype consisting of:

- 18-rod clusters as fuel elements
- Uranium carbide, enriched in U-235 (1.8 relative) as fuel
- HB-40 ($C_{18}H_{22}$) as coolant
- SAP (aluminum) as cladding
- Zircalloy-2 as central guide tube of the fuel element, pressure tube and calandria tube
- D_2O as moderator

Moreover, the following terms have been used during the course of the computation:

a) Geometrical reactor data

Radius of the reactor core	[cm]	200
Height of the core	[cm]	360

b) Geometrical data of a lattice cell

Radius of a pin in the fuel cluster, T	[cm]	0.915
Thickness of the cladding, D_a	[cm]	0.0915
Outer radius of the calandria tube, r	[cm]	6.55
Lattice cell radius	[cm]	13.65

c) Weight-fractions in the lattice cell

Uranium carbide	0.469
HB-40	0.019
D_2O	0.364

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* designed by the industrial group GAAA-INTERATOM-MONTECATINI EDISON in the frame of the ORGEL prototype contest.

SAP	0.027
Zircalloy-2	0.121

D) Weight-fractions within the fuel zone of the lattice cell

Uranium carbide	0.737
HB-40	0.030
SAP	0.042
Zircalloy-2	0.191

e) Densities of the materials situated in the reactor in [g/cm3]

Uranium carbide	13.5
HB-40	0.8484
SAP	2.7
Zircalloy-2	6.55
D_2O	1.084

f) Neutronic data of the cell

Fast fission ratio, δ	0.076
Fast fission factor, ε	1.0414
Resonance escape probability, p	0.8454
Thermal fission factor, η	1.5316
Thermal utilization factor, f	0.9253
Slowing down area, τ	100.1 cm2

$$\frac{\Sigma_{c,\,238}}{\Sigma_f} = 0.393$$

$$\frac{\Sigma_{c,\,235}}{\Sigma_f} = 0.209$$

6.2. Used basic data

For the evaluation of the neutron energy deposition, 15 fast neutron groups have been used. The energy boundaries have been chosen as follows:

Group number:	1	2	3	4
Lower energy boundary:	4.72 Mev	2.87	1.35	0.821

5	6	7	8
0.498	0.235	0.111	64.7 kev

9	10	11	12
31.8	11.7	5.53	2.61

13	14	15
1.23	0.58	0.17

The γ-energy leakage was for lack of other data calculated, using the build-up constants for water. Against this, the total linear attenuation coefficients have been averaged for the homogeneous core. All computations referring to the γ-energy dissipations were carried out by means of 8 energy groups.

Energy group	Upper energy boundary [Mev]	A_1	A_2	α_1	α_2
1	0.5	25	-24	-0.142	-0.070
2	1.0	12	-11	-0.095	0.016
3	1.5	9.2	-8.2	-0.081	0.051
4	2.0	6.4	-5.4	-0.067	0.086
5	3.0	5.2	-4.2	-0.059	0.010
6	4.0	4.0	-3.0	-0.050	0.012
7	6.0	14	-13	-0.008	0.019
8	8.0	12	-11	-0.005	0.020

Total mass attenuation coefficients μ/ρ for γ-rays

Material / Energy group	Mean energy [Mev]	Al	H	C	U	Fe	O
1	0.25	0.110	0.225	0.113	0.950	0.119	0.114
2	0.75	0.0700	0.145	0.0730	0.103	0.0686	0.0730
3	1.25	0.0557	0.115	0.0577	0.0653	0.0540	0.0577
4	1.75	0.0466	0.0953	0.0481	0.0516	0.0454	0.0481
5	2.5	0.0388	0.0776	0.0393	0.0458	0.0384	0.0395
6	3.5	0.0328	0.0632	0.0315	0.0440	0.0342	0.0330
7	5.0	0.0278	0.0506	0.0271	0.0446	0.0314	0.0278
8	7.0	0.0238	0.0408	0.0229	0.0466	0.0300	0.0238

The homogenization prescription for μ/ρ is

$$\left(\frac{\mu}{\rho}\right)_h = \sum_i a_i \left(\frac{\mu}{\rho}\right)_i$$

$$\rho_h = \sum_i \frac{\rho_i f_i}{\sum_i f_i}$$

f_i means the volume fraction of the i^{th} material in the lattice cell. For reasons of lack of the data for deuterium and zirconium, the corresponding values of hydrogen and iron have been employed.

Energy absorption mass attenuation coefficients μ^e/ρ for γ-rays

Material / Energy group	Mean energy [Mev]	Al	H	C	U	Fe	O
1	0.25	0.0277	0.0558	0.0279	0.565	0.0376	0.0281
2	0.75	0.0280	0.0579	0.0291	0.0680	0.0277	0.0291
3	1.25	0.0259	0.0533	0.0268	0.0412	0.0252	0.0269
4	1.75	0.0240	0.0488	0.0247	0.0335	0.0237	0.0248
5	2.5	0.0220	0.0432	0.0222	0.0325	0.0226	0.0224
6	3.5	0.0205	0.0376	0.0199	0.0340	0.0223	0.0202
7	5.0	0.0193	0.0319	0.0176	0.0372	0.0227	0.0183
8	7.0	0.0185	0.0270	0.0159	0.0417	0.0236	0.0168

6.3. Summary of the results

An average fission occurring in the considered core releases the following energy yields:

Kinetic energy of fission fragments $\overline{E}_{ff} = 166.2 \left[\frac{\text{Mev}}{\text{fission}}\right]$

Kinetic energy of neutrons $\overline{E}_{n} = 4.87 \left[\frac{\text{Mev}}{\text{fission}}\right]$

Gamma-ray energy and spectrum $\overline{E}_{\gamma} = 21.07 \left[\frac{\text{Mev}}{\text{fission}}\right]$

Group	Energy range [Mev]	$\nu_j \left[\frac{\text{Mev}}{\text{fission}}\right]$
1	0 - 0.5	3.33
2	0.5 - 1.0	4.18
3	1.0 - 1.5	3.86
4	1.5 - 2.0	3.16
5	2.0 - 3.0	4.76
6	3.0 - 4.0	1.11
7	4.0 - 6.0	0.51
8	6.0 - 8.0	0.16

Beta-ray energy $E_{\beta} = 7.60 \left[\frac{\text{Mev}}{\text{fission}}\right]$

The sum of these individual yields of the released energy amounts to 200.5 $\left[\frac{\text{Mev}}{\text{fission}}\right]$. About 0.21 $\left[\frac{\text{Mev}}{\text{fission}}\right]$ of the liberated neutron energy leaks out from the core and 0.14 $\left[\frac{\text{Mev}}{\text{fission}}\right]$ are consumed by the fast fission effect. Hence, the neutron energy available in the reactor and transferred by scattering events results as $E_S = 4.52 \left[\frac{\text{Mev}}{\text{fission}}\right]$.

This remaining quantity E_S is dissipated in the following way to the reactor constituents:

to material	Neutron energy dissipated by elastic scattering $\left[\frac{\text{Mev}}{\text{fission}}\right]$	by inelastic scattering $\left[\frac{\text{Mev}}{\text{fission}}\right]$
UC - fuel	0.14	0.65
SAP-cladding	0.03	0.07
HB40-coolant	0.72	0.01
Zr2-tubes	0.02	0.19
D_2O-moderator	2.68	0.01
Total $\left[\frac{\text{Mev}}{\text{fission}}\right]$	3.59	0.93

Only the energy deposited by the elastic scattering process is converted to thermal energy, whereas the inelastic scattering represents a conversion of kinetic neutron energy to γ-energy. Since mostly the inelastic scattering takes place in uranium, the application of the γ-spectrum from the processes in uranium for all inelastic scatterings seems to be justified.

Increasing the η_j's given above for the 0.93 $\left[\frac{\text{Mev}}{\text{fission}}\right]$ converted from the kinetic neutron energy and reducing these terms for the leakage of γ-energy from the core (leakage = 1.30 $\left[\frac{\text{Mev}}{\text{fission}}\right]$), the η_j^*'s are obtained:

Group	Energy range [Mev]	η_j^* $\left[\frac{\text{Mev}}{\text{fission}}\right]$
1	0 - 0.5	3.27
2	0.5 - 1.0	3.96
3	1.0 - 1.5	4.11
4	1.5 - 2.0	3.20
5	2.0 - 3.0	4.49
6	3.0 - 4.0	1.05
7	4.0 - 6.0	0.47
8	6.0 - 8.0	0.15
	TOTAL $\left[\frac{\text{Mev}}{\text{fission}}\right]$	20.70

After evaluating the gamma dissipation rates by the homogeneous model, one receives the following distribution:

Gamma-energy dissipated (homogeneous model)	
To material	$\left[\frac{\text{Mev}}{\text{fission}}\right]$
UC-fuel	12.42
SAP-cladding	0.39
HB40-coolant	0.28
Zr2-tubes	1.77
D_2O-moderator	5.84

In calculating the self-shielding effect, the fraction $E^{(2)}$ has been derived by two different procedures, taking once for μ_m the μ of D_2O and the other time the μ of the homogenized core. Because the reality lies surely between these two extremes, SF has been finally determined by a mean value of $E^{(2)}$. Then SF is equal to 1.08.

Correcting the above-given dissipation rates for γ-energy, one obtains:

Gamma-energy dissipated (corrected for heterogeneity	
To material	$\left[\frac{\text{Mev}}{\text{fission}}\right]$
UC-fuel	13.46
SAP-cladding	0.41
HB40-coolant	0.30
Zr2-tubes	1.92
D_2O-moderator	4.61

Nearly all the β -particles generated to the fuel are also absorbed there and in the cladding material. Only 0.08 $\left[\frac{\text{Mev}}{\text{fission}}\right]$ escape from the fuel and are absorbed in the organic coolant.

In the subsequent table, the essential results which characterize the energy balance of the 250 MWel-ORGEL prototype, are once more tabulated. E_t means the total energy.

Form of released energy per fission	$\overline{E}_{ff}$	$\overline{E}_n$	$\overline{E}_\gamma$	$\overline{E}_\beta$	$\overline{E}_t$
Quantity of released energy per fission [Mev]	166.2	4.87	21.07	7.60	199.74
Quantity of energy per fission, available in the reactor [Mev]	166.2	3.59	20.70	7.60	198.09
Leakage from the core [Mev]	0.	0.21	1.30	0.	1.51
Energy absorbed in fuel and canning $\left[\frac{\text{Mev}}{\text{fission}}\right]$	166.2	0.17	13.87	7.52	187.76
Energy absorbed in organic coolant $\left[\frac{\text{Mev}}{\text{fission}}\right]$	0.	0.72	0.30	0.08	1.10
Energy absorbed in tubes $\left[\frac{\text{Mev}}{\text{fission}}\right]$	0.	0.02	1.92	0.	1.94
Energy absorbed in moderator $\left[\frac{\text{Mev}}{\text{fission}}\right]$	0.	2.68	4.61	0.	7.29

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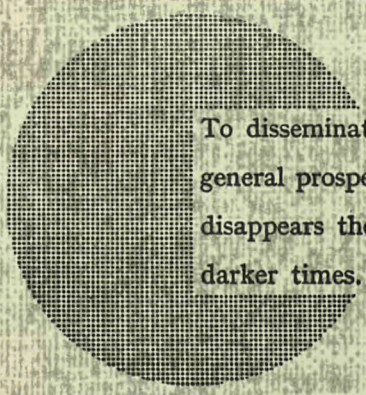

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Alfred Nobel

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