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TOTAL ENERGY DEPOSITION RATES
IN A GRAPHITE MODERATOR
WITH TWO DIFFERENT FUELS

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TOTAL ENERGY DEPOSITION RATES IN A GRAPHITE MODERATOR
WITH TWO DIFFERENT FUELS

by

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INTRODUCTION AND SUMMARY

The problem of the heating of the moderator due to neutrons and gamma rays is of particular interest to engineers working with graphite reactors. The growth or shrinkage of the graphite depends upon its temperature, and thus on the power level of the reactor. Therefore, it is of great use to the materials engineer to be able to predict for various fuel assemblies the heat absorbed as a function of position in the graphite. This report contains the results of graphite heating calculations for two different unit cells and the methods employed in the calculations.

Briefly, the calculations were made in the following manner: spatial and energy sources were separately computed for the gamma rays and then combined as input sources for the gamma energy deposition calculations. The neutron energy deposition was determined using neutron fluxes from previous gamma source calculations.

The two fuel assemblies investigated both consist of water-cooled tubular uranium fuel elements surrounded by a 7-1/2" square block of graphite. They will be referred to in the report as unit cells A and B. Dimensions and nuclei densities for A and B are given in Appendix 1.

The results of the study are shown graphically on Pages 10 through 12. The interval heating rates for neutrons and gammas are quite dependent upon the enrichment and thickness of the fuel. Normalized to the same power level, the graphite in cell B, with highly enriched fuel, absorbs 33% more energy per second than cell A, with natural uranium fuel. Both the total graphite heating rate and the spatial dependance of the heating rates were of particular interest in these calculations. The methods presented could also be extended to give

energy deposition rates in any non-fuel region for similar or more complex geometries.

THEORY

The determination of spatial heating rates in the graphite moderator requires two separate calculations: one for gamma ray heating and one for neutron heating. The theoretical basis for each of these calculations will be presented in this section.

Gamma heating calculations make use of the concept of an "energy flux," defined as:

$$\phi(E) = E\psi(E) \text{ MeV}\cdot\text{cm}^{-2}\cdot\text{sec}^{-1}\cdot\text{MeV}^{-1}, \quad (1)$$

where, $\psi(E)$ is the photon number flux; and a corresponding "energy source density," given by:

$$S(E) = E\Delta(E) \text{ MeV}\cdot\text{cm}^{-3}\cdot\text{sec}^{-1}\cdot\text{MeV}^{-1} \quad (2)$$

where, $\Delta(E)$ is the photon number source.

With these definitions, the transport equation for the energy flux in steady state becomes:

$$\begin{aligned} \vec{\Omega} \cdot \vec{\nabla} \phi(E, \vec{r}, \vec{\Omega}) + \Sigma_t(E) \phi(E, \vec{r}, \vec{\Omega}) \\ = \int d\vec{\Omega}' \int (E/E') \Sigma_{KN}(E' \rightarrow E; \vec{\Omega}' \rightarrow \vec{\Omega}) \phi(E', \vec{r}, \vec{\Omega}') dE' \\ + S(E, \vec{r}, \vec{\Omega}) \quad (1) \end{aligned} \quad (3)$$

where, Σ_{KN} is the Klein-Nishina expression for the Compton scattering. This equation can be solved by a standard transport code, provided the scattering

cross sections all contain the ratio E/E' given in Equation (3), and provided the sources contain an energy term. If one then obtains reaction rates as a function of radius for the macroscopic gamma absorption cross sections (Σ_i^{abs}), the gamma heating rates (H_γ) can be read directly from the output, since:

$$H_\gamma(r) (\text{MeV} \cdot \text{cm}^{-3} \cdot \text{sec}^{-1}) = \sum_i^{\text{All Groups}} \phi_i(r) (\text{MeV} \cdot \text{cm}^{-2} \cdot \text{sec}^{-1}) \Sigma_i^{abs} (\text{cm}^{-1}) \quad (4)$$

The neutron heating rates (H_N) can be obtained with the same neutron fluxes used to generate the spatial gamma sources. These fluxes appear in the following equations for neutron heating rates:⁽²⁾

$$H_N(r) = \int_0^{E_{\max}} \overline{\Delta E} (E, r) \Sigma_s(E, r) \phi(E, r) dE, \quad (5)$$

where, $\overline{\Delta E}$ is the average energy loss per scattering collision for neutrons of energy E at point r , and Σ_s is the total macroscopic scattering cross section. For elastic scattering by an element of atomic weight A , for neutrons of energy E :

$$\overline{\Delta E} = E \left[\frac{2A}{(A+1)^2} \right] (1 - \overline{\cos \psi}) \quad (6)$$

where, $\overline{\cos \psi}$ is the average cosine of the scattering angle in the center-of-mass system. However, for isotropic, elastic scattering $\overline{\cos \psi} = 0$; thus we have:

$$\overline{\Delta E} = E \left[\frac{2A}{(A+1)^2} \right]. \quad (7)$$

ORIGINS OF SPATIAL GAMMA SOURCE

The spatial distribution of gamma rays in the unit cell was determined from n- γ and the fission reaction rates using a one-dimensional transport code, DTF-IV.⁽³⁾ Cross sections for the code were supplied by Battelle Revised THERMOS⁽⁴⁾ for the thermal group and by HRG⁽⁵⁾ for the five epi-thermal groups.

The first calculation done for both cells A and B was one to obtain an infinite multiplication factor, k_{∞} , which was then compared to experimental values. When good agreement was reached, the fluxes generated were used to calculate spatially dependent reaction rates in each region of the cell using the n- γ cross sections of the various isotopes. Next, more reaction rates were generated, this time in the fuel region only using the fission cross sections of ^{235}U and ^{238}U .

Thus, reaction rates were generated for all points in the unit cell for both radioactive capture and fission reactions. Also obtained as a by-product of the fission activation was the total fission rate (per cm) in the fuel, which serves as the normalizing factor for the rest of the calculations on a given cell.

ORIGINS OF GAMMA ENERGY SOURCE

The gamma energy release associated with a given neutron capture or fission reaction was the next thing which had to be determined. These terms were then combined with the spatial sources to provide gamma energy sources.

The amount of gamma energy released per neutron captured was obtained from Greenborg⁽⁶⁾ for seven gamma energy groups given on the following page:

TABLE I

Gamma Group Structure

<u>Group No.</u>	<u>Energy Bounds (MeV)</u>
1	10 - 9
2	9 - 7
3	7 - 5
4	5 - 3
5	3 - 2
6	2 - 1
7	1 - .01

These 7 groups were subsequently chosen for use in the gamma heating calculation, since the gamma group structure is independent of the neutron energy groups used for the spatial source calculations.

To obtain the amount of gamma energy released per fission in each group, both prompt fission and fission product gammas were considered. The value used for the total gamma energy released per fission for prompt gammas was 7.827 MeV and that used for the total gamma energy released per fission for fission product gammas was 7.219 MeV.⁽⁷⁾

These values were then divided among the 7 energy groups. For fission product gammas, the division was accomplished with the help of the computer code ISOSHL⁽⁸⁾, which lists the gamma energy production rate for 16 fine energy groups. These rates, when combined in our 7-group structure, were compared to the total energy production rate, and the resulting ratios determined what part of the 7.219 MeV/Fission fell within each group. For the prompt fission gammas, a standard table⁽⁷⁾ was used, collapsed to our 7-groups.

ANISN GAMMA ENERGY DEPOSITION CALCULATIONS

The computer code, ANISN,⁽⁹⁾ a one-dimensional transport code, was used for the final gamma heating calculations. ANISN was chosen over DTF-IV for its more useful output in energy deposition calculations, and because the gamma cross section generating code, MUG, punched cross sections in the ANISN format.

MUG, a revision of PROGRAM GAMMA,⁽¹⁰⁾ supplied the energy-flux cross sections as defined in the Theory Section. The transfers all contained the E/E' ratio, and the absorption cross sections used in the energy deposition reaction rates had the form for group i , of:

$$\sigma_{a_i} = \sigma_{pe_i} + \left(\frac{E_i - 1.02}{E_i} \right) \sigma_{pp_i} + \sigma_{tot. C_i} - \sum_j \frac{E_j}{E_i} \sigma_{i \rightarrow j}, \quad E_i > 1.02 \text{ MeV} \quad (8)$$

where, σ_{pe} = photoelectric absorption cross section;
 σ_{pp} = pair production absorption cross section;
 $\sigma_{tot. C_i}$ = total Compton cross section;
 $\sigma_{i \rightarrow j}$ = Compton scatter cross section from group i to group j ;
 E_i = average energy of group i .

If one notes that

$$\sigma_{tot. C_i} = \sum_j \sigma_{i \rightarrow j},$$

then Equation (8) becomes:

$$\sigma_{a_i} = \sigma_{pe_i} + \frac{E_i - 1.02}{E_i} \sigma_{pp_i} + \sum_j \sigma_{i \rightarrow j} \left(\frac{E_i - E_j}{E_i} \right). \quad (9)$$

The spectral and spatial gamma sources discussed previously were combined in the preparation of the ANISN distributed source input. Reaction rates were multiplied by the gamma energy released for each of the 7 energy groups and for each space point in the cell. The sources thus had the required units of $\text{MeV}/\text{cm}^3\text{-sec}$, noting that the MeV term in the denominator of Equation (2) has been removed by the energy integration in the group-wise approximation to the transport equation.

Thus, the ANISN calculation (a P-1 calculation was done due to the large forward scattering of gammas) provided reaction rates by space point of the macroscopic value of the absorption cross section of Equation (9). However, these turn out to be just the energy deposition rates (Equation (4)) when it is remembered that the flux has the units of $\text{MeV}/\text{cm}^3\text{-sec}$, the "energy flux." It is of some interest to list and compare the gamma heating rates by region for the two cells. This is done in Appendix 2. The dependence of the gamma energy deposition on the enrichment and thickness of the fuel region is quite evident.

NEUTRON ENERGY DEPOSITION CALCULATIONS

Referring to Equation (7) of the Theory Section, for carbon:

$$\overline{\Delta E} = 0.142 E. \quad (10)$$

However, a group-wise approximation must be made, so that, for group i,

$$\overline{\Delta E}_i = 0.142 \overline{E}_i, \quad (11)$$

where,

$$\overline{E}_i = \frac{\int E \phi(E) dE}{\int \phi(E) dE} \quad (12)$$

and the integrals are over group i . The computer code HRG, used to generate epi-thermal cross sections for the neutron calculation, was used to provide the graphite flux spectrum. HRG calculates the flux for 68 fine lethargy groups, each with a lethargy width of 0.25. Thus, since $\phi(E)dE = \phi(u)du$, the equation for $\overline{E}_i$ can be fairly well approximated by:

$$\overline{E}_i \approx \frac{0.25 \sum_{j \text{ in } i} \overline{E}_j \phi_j}{0.25 \sum_{j \text{ in } i} \phi_j} = \frac{\sum_{j \text{ in } i} \overline{E}_j \phi_j}{\sum_{j \text{ in } i} \phi_j} \quad (13)$$

where,

ϕ_j = flux in fine group j ;

$\overline{E}_j$ = average E for fine group $j \approx \frac{E_{\text{upper}} + E_{\text{lower}}}{2}$; and the

summations are over all fine groups j within the broad group i .

With $\overline{E}_i$, and hence $\overline{\Delta E}_i$, thus calculated, Equation (5) may be approximated group-wise by:

$$H_N(r) \approx \sum_i \overline{\Delta E}_i \Sigma_{s_i} \phi(r)_i, \quad (14)$$

where,

Σ_{s_i} = total macroscopic scattering cross section for carbon for group i;

$\phi(r)_i$ = broad group flux from DTF for each mesh point, r, in the graphite;

and the summation is over the 6 broad groups used in the neutron flux calculations in DTF.

Thus, neutron energy deposition rates were determined in the same units (MeV/cm³-sec), and normalized to the same value (the # Fissions/cm-sec from DTF-IV), as the gamma energy deposition rates.

RESULTS

To obtain some idea of the relative contribution of neutrons and gammas to the total interval heating rates, Figures (1) and (2) were constructed. They show the relative interval deposition rates as a function of radius.

Figure (1) shows the values for unit cell A, a natural uranium, water-cooled fuel element in a 7-1/2" square block of graphite in a helium atmosphere. The total graphite heat deposition rates (MeV/cm-sec) found by summing the products of the heating rates times the interval volumes (actually volume per unit length) show that approximately 62% of the heating is due to gammas and 38% to neutron slowing down.

Figure (2) shows the results for unit cell B, a highly-enriched, water-cooled fuel element in the same graphite block as cell A. Thus, the difference in cells A and B is the enrichment and thickness of the fuel, and corresponding changes in the thickness of the coolant channels. From Figure (2), cell B shows

FIGURE 1
Unit Cell A
Relative Interval Heating Rates

FIGURE 2
Unit Cell B
Relative Interval Heating Rates

FIGURE 3
Total Relative Interval Heating Rates Normalized to 1.0
Fission/sec-cm

a higher contribution from gammas than cell A, due in large part to the increased probability of gammas escaping from the less dense, aluminum-alloyed fuel to the outer graphite region, and a lower contribution from neutrons, due to more energy loss in the thicker water regions. The total graphite deposition rates for cell B show that approximately 80% of the heating is due to gammas and 20% to neutrons.

Finally, the neutron and gamma interval heat deposition rates were added together, and the resulting total interval deposition rates were then normalized to the same value, $1 \frac{\text{Fission}}{\text{cm-sec}}$; thus, the two cells can be compared on a normalized basis. Figure(3) shows a comparison of the total interval deposition rates. The total graphite heat deposition rates for the sum of neutron and gamma contributions indicate that the graphite in cell B absorbs 33% more energy per second than in cell A, for the same power level. Table II below gives the values for gamma, neutron, and total energy deposition in the graphite:

TABLE II

Comparison of Graphite Heating Rates

	<u>Gamma</u> <u>(MeV/Fission)</u>	<u>Neutron</u> <u>(MeV/Fission)</u>	<u>Total</u> <u>(MeV/Fission)</u>	<u>% of Total Energy</u> <u>Released per Fission</u>
Cell A	6.08	3.75	9.83	4.92
Cell B	10.44	2.60	13.04	6.76

The total energy released per fission was determined by taking the oft-quoted value of 200 MeV/Fission, subtracting the energy carried off by neutrinos ~ 10 MeV/Fission, and adding the energy gained due to n- γ reactions. For cell A this value was ~ 200 MeV/Fission, and for cell B the value was ~ 193 MeV/Fission.

A word is in order about the rising of the gamma interval deposition rates in Figures (1) and (2), and thus in the total rates in Figure (3). Although the capture gamma source produced in the graphite is increasing toward the cell edge due to the rising thermal neutron flux, the calculated effect does not appear to be due entirely to any physical gamma sources present in the graphite, and may be a quirk in the white boundary condition of ANISN. The correction for this in the total heating rates would be less than 1% and was not included.

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APPENDIX I

Cell Dimensions and Nuclei Densities

Unit Cell A:

<u>Region</u>	<u>Thickness (cm)</u>	<u>Isotopes</u>	<u>Nuclei Density ($\frac{\text{nuclei}}{\text{barn-cm}}$)</u>
Coolant	.53340	H	6.62750 -2
		O	3.31375 -2
Clad	.14351	Al	6.02500 -2
Fuel	1.14046	²³⁵ U	3.44500 -4
		²³⁸ U	4.75054 -2
Clad	.12319	Al	6.02500 -2
Coolant	.24892	H	6.62750 -2
		O	3.31375 -2
Process Tube	.10541	Zr	4.23736 -2
Graphite	8.45292 (Cylindricized)	C (Helium atmosphere)	8.13375 -2

Unit Cell B:

Coolant	.84607	H	6.62750 -2
		O	3.31375 -2
Clad	.15240	Al	6.02500 -2
Fuel	.61519	Al	5.07750 -2
		²³⁵ U	5.42660 -4
		²³⁸ U	4.03294 -5
Clad	.15240	Al	6.02500 -2
Coolant	.42342	H	6.62750 -2
		O	3.31375 -2
Process Tube	.10541	Zr	4.23736 -2
Graphite	8.45292 (Cylindricized)	C (Helium atmosphere)	8.13375 -2

APPENDIX II

Percent of Gamma Energy Deposited by Region

The following table gives the percentages of the total gamma source energy which are deposited per second in each region for both unit cells:

<u>Region</u>	<u>Unit Cell A</u>	<u>Unit Cell B</u>
Coolant	0.17%	1.23%
Clad	0.31%	2.59%
Fuel	72.67%	22.26%
Clad	0.50%	3.37%
Coolant	0.35%	1.86%
Process Tube	1.96%	10.73%
Graphite	24.04%	57.96%

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