GAMMA HEATING CALCULATIONS
IN GRAPHITE MODERATED SYSTEMS

G. E. Hanson
G. J. Busselman

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by

G. E. Hanson

G. J. Busselman

Battelle-Northwest

Battelle Memorial Institute

Richland, Washington

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ABSTRACT

Calculations were carried out to determine the amount of fission product gamma ray energy absorbed in various regions of a tubular fuel element and the surrounding graphite of a unit cell. Two different methods were used, involving two separate computer codes; a point-kernel shielding code and a transport code. Several approximations of the annular fuel element were used in the shielding code. Results from the two methods are compared and discussed, and the best approximation using the simpler shielding code is determined.

INTRODUCTION

A water-cooled tubular fuel element (see Appendix C, Figure 1) has been investigated to determine the amounts of fission product gamma ray energy which are absorbed in the various regions of the element and the surrounding graphite of a unit cell. These calculations were performed as part of an investigation of the effect of a complete and instantaneous loss of water. It is assumed that such an accident would cause the immediate shutdown of the reactor; thus, only the fission products already residing in the fuel are assumed to contribute gamma rays to the system. The calculations were carried out using two different computer codes, and the results are presented and compared in the following sections. Computational methods, graphs, and tables of photon cross sections are given in the appendices.

FIRST APPROACH

The computer code ISOSHL⁽¹⁾, a shielding code using point-kernel integration, was used to calculate the γ attenuation in cylindrical geometry in the various materials. RIBD (Radioactive Isotope Buildup and Decay), a subroutine of ISOSHL, was used to provide the spectrum of γ -rays from the fission product inventory supplying the source. Use of the code was somewhat complicated by the fact that ISOSHL requires the innermost region to contain the source, which is not true in the case of the tubular fuel investigated, which surrounds the lithium aluminate target.

Three different calculational approaches were considered: 1) a solid uranium fuel element; 2) Case 1 with the effect of the inner source region removed by subtraction; and 3) a smeared rod of uranium, target, and cladding.

The second case was a combination of two different rods. The solid uranium rod of case 1 was divided into two regions of uranium, so that the source (reduced proportionately) was only in the inner region, which extended to the actual inner fuel radius. When the results of the two-region rod were subtracted from those of case 1, the remainders gave the answers one would obtain from a ring of fission products in uranium surrounding a non-fissioning inner core of uranium. This is still not quite the actual physical situation, since the lithium aluminate target in the center of the actual element is much less dense than uranium and therefore would attenuate fewer γ -rays.

Results of the calculations for these three cases are given in Table I. All of the percentages quoted are at zero time after shutdown. Results calculated for later times remain almost constant, so that the percent of heat escaping the fuel does not change significantly with time, up to 12 minutes from shutdown.

Contributions to the cell under investigation from surrounding cells were obtained by calculating the percentage of the total γ heat at distances from the

original cell equal to the first, second, and third rings of surrounding fuel elements. This percentage when multiplied by the fractional solid angle intercepted by the number of elements in that particular ring was added to the percentage of the total γ heat absorbed within the fuel, target, and cladding of the original cell. This process assumes an even γ distribution over the entire core. The last ring of cells used was considered to be a solid ring of fuel at that radius.

TABLE I

Case	<u>A</u> % of γ -Heat Escaping from Fuel, Target, and Cladding	<u>B</u> γ -Heat from Surrounding Lattice ab- sorbed in Fuel, Target, and Cladding	<u>C</u> Net Heat Absorbed in Process Tube & Graphite	<u>D</u> Net Heat Absorbed in Fuel, Target, & Cladding (100%-C)
1. Solid U-rod	21.90%	3.68%	18.22%	81.78%
2. Solid rod minus 2- region rod	39.07%	6.57%	32.50%	67.50%
3. Target, fuel, & clad smeared throughout rod	58.82%	9.88%	48.94%	51.06%

SECOND APPROACH

The computer code, ANISN,⁽²⁾ was used to continue the gamma-heating study. ANISN is a one-dimensional, S_n transport code, which allows general anisotropic scattering and shell sources. For these reasons it was chosen to continue the study which was started with ISOSHL. The range of values obtained with ISOSHL varied rather widely with the approximations used, and it was believed that the more flexible and sophisticated ANISN could help to resolve these variations.

Five gamma ray energy groups were used for the calculations, and the source was computed from the RIBD subroutine of the ISOSHL program. Two different cases were

run; one with a white boundary and an albedo of 1.0, and the other with a vacuum boundary. The latter case was run to determine what percent of the heat produced escapes from the cell, and where the corresponding incoming heat from the surrounding lattice is absorbed, since the net leakage in an infinite array would be zero.

The results of the two cases are tabulated below:

TABLE II
Percent of Total γ -ray Energy Absorbed in Each Region

<u>Region</u>	<u>Vacuum Boundary</u>	<u>White Boundary</u>
Target	1.3027	1.3724
Al Clad	.7275	.7657
Zr Clad	.6425	.6748
Air	.0008	.0009
Zr Clad	.6446	.6757
Fuel	60.9637	64.0710
Zr Clad	1.1444	1.3005
Air	.0007	.0008
Zr process tube	6.2595	8.3047
Inner graphite	6.6382	13.1952
Outer graphite	2.6110	9.6393
Cell total	80.9356	100.0000
Leakage	19.0644	- - - -

Thus, the vacuum boundary shows 65.43% of the heat is absorbed inside of the process tube, 15.51% in the process tube and graphite, and 19.06% leaks out of the cell. The white boundary with albedo = 1.0 shows 68.86% of the heat is absorbed inside of the process tube, and 31.14% in the process tube and graphite. Therefore, most of the heat from the surrounding lattice is absorbed in the graphite and process tube and very little ($\sim 3\%$ of the total heat produced) which escapes the fuel is returned to it.

COMPARISON AND SUMMARY

After the results from the ISOSHL D calculations were obtained, it was first thought that a reasonably accurate answer would probably lie somewhere between the answers of cases 2 and 3 (Table I). It was felt that case 2 attenuated too many gammas in the target area because a non-fissioning uranium core still effectively remained in the center after the subtraction, instead of the much less dense target and cladding. Case 3, on the other hand, would allow too many gammas to escape because the density of the smear was much less than that of the fuel. The limit to the flexibility of ISOSHL D was reached at this point, and ANISN was used to attempt to resolve the difference between cases 2 and 3.

However, the answers obtained from ANISN agreed quite closely with those of case 2. From Table II one can see that the amount of heat absorbed within the inner fuel boundary (i.e., in the actual low density target and cladding) is of relatively little importance when compared with the absorption in the fuel ring itself. One might conclude from the agreement of the two results with targets of such different densities that the total amount of heat absorbed in the fuel, target, and cladding would remain fairly constant no matter what type of material is used for the inner target region. This is apparently the case because photons emitted toward the target must re-enter the dense fuel ring if they are not absorbed within the target. Thus, the probability of a photon escaping to the outer region of the cell (process tube and graphite) when it is originally emitted radially inward must be very small.

A summary of the results of the two cases showing their close agreement is presented in the following table.

TABLE III

<u>Case</u>	<u>% of Heat Absorbed in Fuel, Target, & Cladding</u>	<u>% of Heat Absorbed in Process Tube & Graphite</u>
ISOSHL D Case 2 (Solid rod minus 2-region rod)	67.50%	32.50%
ANISN (White boundary)	68.86%	31.14%

A detailed study of the propagation of uncertainties in these numbers would be profitable in that one would obtain a better indication of their agreement and reliability.

REFERENCES

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2. W. W. Engle, Jr., A User's Manual for ANISN - A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering, K-1693, United Carbide Nuclear Division, Oak Ridge, Tennessee; March 30, 1967.

APPENDIX AConversion of ISOSHL D Results to Fractional Heat Absorbed

The output of the ISHSHLD code consists of source values (photons/sec) for 16 energy groups, the average energy for each group, and the energy flux (Mev/cm²-sec) for each group at some specified radius. The total γ -decay heat per centimeter of length of the fuel is given by the following equation:

$$\frac{\sum_{i=1}^{16} S_i \left(\frac{\text{photons}}{\text{sec}} \right) \bar{E}_i \left(\frac{\text{Mev}}{\text{photon}} \right) (1.603 \times 10^{-13} \frac{\text{Watt-sec}}{\text{Mev}})}{\text{Length of fuel (cm)}} = \frac{\text{Watts}}{\text{cm}}$$

The amount of heat observed at any particular radius (per centimeter of length of the fuel) can be found from the following:

$$\phi \left(\frac{\text{Mev}}{\text{cm}^2\text{-sec}} \right) 2\pi R(\text{cm}) (1.603 \times 10^{-13} \frac{\text{Watt-sec}}{\text{Mev}}) = \frac{\text{Watts}}{\text{cm}}$$

Thus, the percent of the γ heat which escapes to the point in question is given by the ratio of the heat observed at that point to the total γ -decay heat. This ratio is then determined independent of the power level. The percent of the γ -heat which is absorbed between the center of the element and the point is consequently given by one minus that ratio.

APPENDIX BDetails of ANISN Input and Results

Data for the photon source for ANISN was supplied by two other codes; ISOSHL and Program S.⁽¹⁾ The photon source used was assumed to be the product of a spatial distribution and a photon energy distribution.⁽²⁾ The spatial distribution was obtained from the fission density terms supplied by an analysis of the fuel. with Program S.⁽³⁾ The photon energy spectrum was calculated using the energy production rate from the RIBD fission product inventory of ISOSHL normalized to $5.96 \frac{\text{Mev}}{\text{fission-sec}}$, the value quoted for the rate, per fission, at which energy is carried off by gamma rays from the fission products of U^{235} .⁽⁴⁾ Five energy groups were chosen to match the five groups of the cross section set (see Appendix D). ISOSHL's 16 energy groups were collapsed to this group structure:

<u>Group</u>	<u>Energy Range (Mev)</u>
1	$3 \rightarrow \infty$
2	$2 \rightarrow 3$
3	$1 \rightarrow 2$
4	$.115 \rightarrow 1$
5	$0 \rightarrow .115$

With the source thus obtained and the energy absorption cross sections from the cross section set, the energy deposition is given by the absorption rates calculated by ANISN for each region of the cell. In terms of the absolute amounts of energy deposited, no answer was obtained, due to several normalizations of the source terms. However, the relative amounts of heat deposited in each region are easily obtained, since ANISN also prints the total absorption for the entire system.

The energy "flux" shapes for both the white and vacuum boundaries are shown in Appendix C, Figure 2. The rise in the flux in the inner air regions is interpreted as the geometrical buildup by the code due to the shell source.

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APPENDIX C

<u>Region</u>	<u>Description</u>	<u>Outer Radius (cm)</u>
1	Target	1.365
2	Al Clad	1.619
3	Zr Clad	1.695
4	Air	2.181
5	Zr Clad	2.244
6	Fuel	2.955
7	Zr Clad	3.044
8	Air	3.442
9	Zr Process Tube	4.135
10	Inner Graphite	8.598
11	Outer Graphite	12.160

FIGURE 1
Tubular Fuel Element Equivalent Cylindrical Cell

APPENDIX C

FIGURE 2
ANISN Energy "Flux" Shapes

APPENDIX DPhoton Cross Sections

The following are photon energy deposition and isotropic and anisotropic energy transfer cross sections (in barns/atom) for the elements in the tubular fuel assembly. ⁽¹⁾

URANIUM

<u>Group</u>	<u>Sigma Scatter</u>	<u>Sigma Absorption</u>
1	5.5586 + 00	1.1293 + 01
2	6.4816 + 00	1.0967 + 01
3	8.7620 + 00	1.4176 + 01
4	1.8416 + 01	1.6303 + 02
5	4.2823 + 01	1.8502 + 04

Transfer Matrix - Zeroth Order

<u>To From</u>	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>
1	8.0263-01	1.4322+00	1.0268+00	2.2970+00	0.
2	0.	1.4772+00	2.4764+00	2.5280+00	0.
3	0.	0.	3.5975+00	5.1645+00	0.
4	0.	0.	0.	1.7233+00	1.1828+00
5	0.	0.	0.	0.	4.2823+01

Transfer Matrix - First Order

1	7.9232-01	1.3535+00	8.2268-01	1.1746-01	0.
2	0.	1.4409+00	2.1579+00	3.1426-01	0.
3	0.	0.	3.3593+00	2.0164+00	0.
4	0.	0.	0.	8.0943+00	-2.3693-01
5	0.	0.	0.	0.	5.8288+00

D-2

OXYGEN

<u>Group</u>	<u>Sigma Scatter</u>	<u>Sigma Absorption</u>
1	3.5396 - 01	5.2708 - 01
2	4.6844 - 01	5.8682 - 01
3	7.1306 - 01	6.8166 - 01
4	1.6014 + 00	7.6082 - 01
5	3.7238 + 00	1.9218 + 01

Transfer Matrix - Zeroth Order

<u>To From</u>	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>
1	6.9794-02	1.2454-01	8.9288-02	7.0338-02	0.
2	0.	1.2846-01	2.1534-01	1.2465-01	0.
3	0.	0.	3.1282-01	4.0023-01	0.
4	0.	0.	0.	1.4986+00	1.0285-01
5	0.	0.	0.	0.	3.7238+00

Transfer Matrix - First Order

1	6.8898-02	1.1770-01	7.1538-02	1.0214-02	0.
2	0.	1.2530-01	1.8764-01	2.7327-02	0.
3	0.	0.	2.9211-01	1.7534-01	0.
4	0.	0.	0.	7.0386-01	-2.0602-02
5	0.	0.	0.	0.	5.0686-01

ZIRCONIUM

<u>Group</u>	<u>Sigma Scatter</u>	<u>Sigma Absorption</u>
1	2.0163 + 00	3.2719 + 00
2	2.5152 + 00	3.2478 + 00
3	3.6104 + 00	3.5699 + 00
4	8.0070 + 00	9.3864 + 00
5	1.8619 + 01	2.7700 + 03

Transfer Matrix - Zeroth Order

<u>To From</u>	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>
1	3.4897-01	6.2268-01	4.4644-01	5.9817-01	0.
2	0.	6.4228-01	1.0767+00	7.9625-01	0.
3	0.	0.	1.5641+00	2.0463+00	0.
4	0.	0.	0.	7.4928+00	5.1424-01
5	0.	0.	0.	0.	1.8619+01

Transfer Matrix - First Order

1	3.4449-01	5.8848-01	3.5769-01	5.1068-02	0.
2	0.	6.2648-01	9.3820-01	1.3664-01	0.
3	0.	0.	1.4606+00	8.7668-01	0.
4	0.	0.	0.	3.5193+00	-1.0301-01
5	0.	0.	0.	0.	2.5343+00

D-4

LITHIUM

<u>Group</u>	<u>Sigma Scatter</u>	<u>Sigma Absorption</u>
1	1.2985 - 01	1.9061 - 01
2	1.7364 - 01	2.1703 - 01
3	2.6687 - 01	2.5525 - 01
4	6.0053 - 01	2.8429 - 01
5	1.3964 + 00	3.5519 - 01

Transfer Matrix - Zeroth Order

<u>To From</u>	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>
1	2.6173-02	4.6701-02	3.3483-02	2.3489-02	0.
2	0.	4.8171-02	8.0751-02	4.4716-02	0.
3	0.	0.	1.1731-01	1.4956-01	0.
4	0.	0.	0.	5.6196-01	3.8568-02
5	0.	0.	0.	0.	1.3964+00

Transfer Matrix - First Order

1	2.5837-02	4.4136-02	2.6827-02	3.8301-03	0.
2	0.	4.6986-02	7.0365-02	1.0248-02	0.
3	0.	0.	1.0954-01	6.5751-02	0.
4	0.	0.	0.	2.6395-01	-7.7259-03
5	0.	0.	0.	0.	1.9007-01

D-5

CARBON

<u>Group</u>	<u>Sigma Scatter</u>	<u>Sigma Absorption</u>
1	2.6316 - 01	3.8967 - 01
2	3.4971 - 01	4.3769 - 01
3	5.3437 - 01	5.1095 - 01
4	1.2011 + 00	5.6898 - 01
5	2.7928 + 00	5.2725 + 00

Transfer Matrix - Zeroth Order

<u>To From</u>	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>
1	5.2345-02	9.3402-02	6.6966-02	5.0443-02	0.
2	0.	9.6342-02	1.6150-01	9.1866-02	0.
3	0.	0.	2.3462-01	2.9975-01	0.
4	0.	0.	0.	1.1239+00	7.7136-02
5	0.	0.	0.	0.	2.7928+00

Transfer Matrix - First Order

1	5.1673-02	8.8272-02	5.3653-02	7.6602-03	0.
2	0.	9.3972-02	1.4073-01	2.0495-02	0.
3	0.	0.	2.1908-01	1.3150-01	0.
4	0.	0.	0.	5.2789-01	-1.5452-02
5	0.	0.	0.	0.	3.8014-01

D-6

NITROGEN

<u>Group</u>	<u>Sigma Scatter</u>	<u>Sigma Absorption</u>
1	3.0836 - 01	4.5791 - 01
2	4.0894 - 01	5.1206 - 01
3	6.2368 - 01	5.9628 - 01
4	1.4012 + 00	6.6432 - 01
5	3.2583 + 00	1.0566 + 01

Transfer Matrix - Zeroth Order

<u>To From</u>	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>
1	6.1069-02	1.0897-01	7.8127-02	6.0198-02	0.
2	0.	1.1240-01	1.8842-01	1.0812-01	0.
3	0.	0.	2.7372-01	3.4996-01	0.
4	0.	0.	0.	1.3112+00	8.9992-02
5	0.	0.	0.	0.	3.2583+00

Transfer Matrix - First Order

1	6.0285-02	1.0298-01	6.2595-02	8.9369-03	0.
2	0.	1.0963-01	1.6418-01	2.3911-02	0.
3	0.	0.	2.5560-01	1.5342-01	0.
4	0.	0.	0.	6.1587-01	-1.8027-02
5	0.	0.	0.	0.	4.4350-01

ALUMINUM

<u>Group</u>	<u>Sigma Scatter</u>	<u>Sigma Absorption</u>
1	5.8770 - 01	8.8705 - 01
2	7.7000 - 01	9.6673 - 01
3	1.1610 + 00	1.1093 + 00
4	2.6023 + 00	1.2619 + 00
5	6.0511 + 00	1.5486 + 02

Transfer Matrix - Zeroth Order

<u>To From</u>	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>
1	1.1341-01	2.0237-01	1.4509-01	1.2682-01	0.
2	0.	2.0874-01	3.4992-01	2.1134-01	0.
3	0.	0.	5.0834-01	6.5267-01	0.
4	0.	0.	0.	2.4352+00	1.6713-01
5	0.	0.	0.	0.	6.0511+00

Transfer Matrix - First Order

1	1.1196-01	1.9126-01	1.1625-01	1.6597-02	0.
2	0.	2.0361-01	3.0491-01	4.4407-02	0.
3	0.	0.	4.7468-01	2.8492-01	0.
4	0.	0.	0.	1.1438+00	-3.3479-02
5	0.	0.	0.	0.	8.2364-01

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1. M. S. Ferry, Photon Cross Sections, APEX 613, General Electric, Cincinnati, Ohio, August 1961.

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