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SUMMARY REPORT

GAMMA SPECTRAL DATA FOR SHIELDING AND HEATING CALCULATIONS

by

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ABSTRACT

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Spectra of gamma rays following neutron absorption and inelastic-scattering events in H, Be, C, O, Al, Cr; Fe, Ni, Zr, natural W and the W isotopes, U^{235} and U^{238} are tabulated. The gamma thermal capture spectra for Cd and Sm are presented. A detailed study of the prompt and delayed fission gammas (both intensities and time variations) is also given. Descriptions are given of the sources of information and the calculations performed. In addition, an evaluation of the reliability of the data is given.

Authors

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1. SUMMARY

The United Nuclear Corporation is presently modifying its UNC-SAM Monte Carlo radiation transport system of programs* for use in the tungsten nuclear rocket program. The modified system and its main tracking routine will be called ATHENA (Attenuation, Tracking, and Heating for NASA). In association with this program, gamma heating studies in the core are being performed for which gamma spectral data are necessary. Therefore, a detailed study was undertaken to determine the gamma spectra and intensities following neutron absorption and inelastic-scattering events in the elements and nuclides of interest – H, Be, C, O, Al, Cr, Fe, Ni, Zr, natural W and the separate W isotopes, U^{235}, and U^{238}. The capture spectra for the strongly capturing elements Cd and Sm are given. Finally, a detailed study of the prompt and delayed fission gammas was made providing a detailed description of the post-fission sources from 0 to 10 hr.

The thermal capture gamma spectra are generally well-known. The energy and intensity of the discrete spectra are represented by their "best" experimental values. Continuous gamma spectra are represented by a sufficient number of discrete gamma energies to represent the spectrum adequately. Wherever discrete gamma lines of relatively high intensity are imposed on a continuum, the gammas of energy close to the discrete line have been lumped together with the discrete gamma. This procedure was also followed for the inelastic gamma spectra.

*For a description of the UNC-SAM code see Reference 1.

It was assumed that the capture spectrum is independent of the neutron energy at which the capture occurs. This assumption is generally adequate for most problems. The assumption that no gammas follow the charged-particle reactions (except for oxygen) will tend to cause an underestimate of the total heating.

The gamma spectra following inelastic scattering, for neutron energies below ~4 Mev, are believed to be adequate, since they are based on experimental level-excitation cross sections. In the intermediate neutron energy range (4 to 8 Mev) the inelastic gamma spectra, based in part on statistical theory, are not uniformly reliable. For neutron energies above 8 Mev, the spectra are based on statistical theory which includes (n,2n) and (n,3n) processes. As the neutron energy increases above 8 Mev, the validity of the parameters used in the theory becomes increasingly questionable. In general, the shape of the spectra is given more reliably than the absolute values. For the problems in which it is planned to use the data presented in this report the inadequacy is not important because neutrons having energies > 8 Mev constitute only 1/2 of 1% of the fission source.[2]

Inasmuch as the tungsten being used in the reactor may be enriched in one or more isotopes, additional calculations were made to provide capture gamma spectra for the isotopes W^{182}, W^{183}, W^{184}, and W^{186}.

2. GAMMA SPECTRAL DATA

2.1 ABSORPTION AND INELASTIC GAMMA SPECTRA

2.1.1 General Comments on the Absorption Spectra

This section describes a study made of the gamma spectra following absorption and inelastic-scatter events in elements of interest in the tungsten nuclear rocket program. The data are to be used in the modified United Nuclear Monte Carlo three-dimensional transport code to determine, among other quantities, gamma heating in and near the core.

The relevant data are presented in Tables 1 through 26, 32, and 33. Generally, there are two tables per element (each element being identified in the programs by a 5-digit integer). One table gives the absorption, and the other gives the inelastic spectrum. Each table consists of a matrix specifying, for a given neutron energy bin, the number of gammas produced per event (absorption or inelastic) for each of several discrete gamma energies.* The neutron energies given represent the upper energy of the bin. The spectra are assumed to be constant within each neutron energy group.

The absorption spectrum for each element (except for Cd and Sm) includes both capture and charged-particle events, the spectrum of each being weighted by its

*For several of the elements the gamma energies were combined so as to condense the tables; the complete, more detailed spectra are available in punched-card format for use in the ATHENA system.

average cross section in the given neutron energy group. If P_c and P_{cp} are the capture and charged-particle number spectra (number of photons produced) with corresponding average cross sections σ_c and σ_{cp}, then the absorption number spectrum P_a for that neutron energy bin E_n is

$$P_a(E_n, E_\gamma) = \frac{\sigma_c(E_n)P_c(E_\gamma) + \sigma_{cp}(E_n)P_{cp}(E_n, E_\gamma)}{\sigma_c(E_n) + \sigma_{cp}(E_n)} \qquad (1)$$

For most of the elements (the exception being oxygen) it was assumed that $P_{cp}=0$, i.e., no gammas are created following a charged-particle reaction. In general, charged-particle reactions produce low-energy (<1 Mev) gammas. Thus they can be neglected in shielding problems where the desired quantity is the gamma-ray dose at the outside of the shield. For problems in which one desires the total local gamma heating, neglect of gammas from charged-particle reactions will underestimate the total heating. A saving factor is that, in general, charged-particle reactions are important only at high (several Mev) neutron energies. Hence for a fission source where neutrons with E > 6 Mev represent only 2.5% of the source, the fraction becoming 0.5% for E > 8 Mev, the underestimate of the heating will be small. Moreover, at high neutron energies, in addition to the charged-particle reactions, inelastic-scattering events (with $\sigma_{inel} \gtrsim \sigma_{cp}$ generally) yield gamma rays, the latter being of higher energies than those from the charged-particle reactions.

Eq. 1 implies that the capture spectrum is independent of the neutron energy at which the capture occurs. This is not generally valid for resonance regions in which different resonances may excite different levels. However, for a continuous neutron energy distribution the composite thermal spectrum is adequate. The possible variation of the capture spectrum with neutron energy, and its consequences for any given element, could be further investigated. The thermal capture spectrum represents high-energy (several Mev) captures even less adequately than in the resonance region. This does not represent any difficulty, however, as the capture cross section is generally very small in the Mev range. Moreover, for a fission spectrum, there are relatively few high-energy neutrons.

2.1.2 General Comments on the Inelastic Spectra

The inelastic gamma spectrum for each element includes the (n,2n) and (n,3n) processes in addition to the purely inelastic (n,n′γ) reaction. For neutron energies below ~4 Mev, the values tabulated are based on experimental level-excitation cross sections. In some cases these were supplemented by Hauser-Feshbach calculations. In the intermediate neutron energy range (4 to 8 Mev) the spectra are based on statistical-model calculations, supplemented by some meager experimental data. The data of Perkin[3] for Al, Fe, Ni, and W, though complete, are of dubious quality. He occasionally gives production cross sections for gammas of energies which are not physically possible for the particular element and neutron energy considered. Perkin also normalizes the cross section separately at each neutron energy for which data are presented. This introduces spurious structure in the gamma spectra, which we have attempted to remove.

As the neutron energy increases beyond 8 Mev, there is an increasing paucity of experimental data (except for some data at 14 Mev). To compound the difficulty, the validity of the parameters used in the statistical theory [which includes (n,2n) and (n,3n) processes] becomes increasingly questionable as the neutron energy extends further away from the energy range at which experimental data exist. In general, the spectral shapes are given more reliably than the absolute values. However, since the fission spectrum has very few neutrons above 8 Mev, inaccuracies in the associated gamma spectra are not too important.

2.2 DISCUSSION OF THE REFERENCES USED AND CALCULATIONS PERFORMED IN PREPARING THE SPECTRAL INFORMATION FOR EACH ELEMENT

2.2.1 Hydrogen

The gamma spectrum following a neutron interaction with hydrogen is a model of simplicity. There are no inelastic scatterings, and an absorption at any energy produces a single 2.23-Mev gamma.[4] The gamma spectrum is shown in Table 1.

2.2.2 Beryllium

The gamma spectrum following a neutron absorption in beryllium is based mainly on the work of Draper and Bostrom.[5,6] The spectrum is given in Table 2. There is no $(n,n'\gamma)$ reaction in Be at the neutron energies of interest ($\leq$18.0 Mev). However, the (n,2n) reaction does lead to the creation of a 2.43-Mev gamma. The number of such gammas created per "inelastic" event as a function of neutron energy is given in Table 3. The data are based on the work of A. Krumbein.[7]

2.2.3 Carbon

E. Troubetzkoy and H. Goldstein[4] give the spectrum of gamma rays following neutron capture in carbon. The spectrum was assumed constant for incident neutron energies below 223 ev, above which the (n,γ) cross section becomes zero. Above 223 ev no gammas are produced per neutron absorption as the gammas following the charged-particle reactions are neglected. The spectrum is shown in Table 4.

The gamma spectrum following inelastic scattering in carbon was based on statistical theory for neutron energies above 10 Mev. For $E_n \leq 10$ Mev, experimental level-excitation cross sections were used. It was assumed that the 7.66-Mev level does not decay by gamma emission.[8] The data are presented in Table 5.

2.2.4 Oxygen

There are no neutron capture events in oxygen. The absorption spectrum given in Table 6 represents the production of the 3.5-Mev gamma following the (n,α) reaction. The gamma spectrum following inelastic scattering was based on the experimental level-excitation cross sections measured by the Rice group and quoted in BNL-325.[9] These were supplemented by statistical model calculations. The details are given in Reference 10. The gamma spectrum, as a function of neutron energy, is given in Table 7.

2.2.5 Aluminum

The capture spectrum was taken from J. E. Draper and C. O. Bostrom[6] and the compilation of E. Troubetzkoy and H. Goldstein.[4] The gamma spectrum emitted per inelastic-scattering event in aluminum was deduced as follows. The work of Towle and Gilboy[11] and the compilations found in References 12 and 13 were supplemented by Hauser-Feshbach calculations to determine the excitation cross sections for the six lowest levels for incident neutron energies below 4.0 Mev. These agreed remarkably well (to within 15%) with the calculations of M. Leimdörfer (personal communication).

For neutron energies above 4 Mev, statistical-model calculations were used. These agreed well with the discrete-level data of Reference 11 and the Hauser-Feshbach calculations near 4 Mev. The calculations were supplemented by the experimental data of Perkin[3] for neutron energies to 8.5 Mev and by the data of Thompson and Engesser at $E_n = 14$ Mev (Reference 14).

The capture and inelastic gamma spectra are tabulated in Tables 8 and 9.

2.2.6 Chromium

The spectral intensities of gamma rays resulting from neutron capture in chromium were taken from the compilation of Troubetzkoy and Goldstein.[4] The range of gamma energies from 0 to 9.72 Mev has been divided into seven groups. Integrated intensities are given for each group. Since the charged-particle reactions [which far outweigh the (n,γ) reaction for $E_n \geq 2.5$ Mev] are assumed to be accompanied by negligible gamma emission, the absorption gamma spectrum equals the capture spectrum for neutron energies up to 2.5 Mev. Thereafter it drops sharply to zero.

The gamma spectra following inelastic scattering in chromium were based on the level-excitation cross sections of Van Patter,[15] for incident neutron energies below 3 Mev. For incident neutron energies above 3 Mev the gamma emission was calculated from statistical theory.

The capture and inelastic gamma spectra following neutron interactions in chromium are given in Tables 10 and 11.

2.2.7 Iron

The absorption gamma spectrum for iron was taken to be the capture spectrum, as given by Troubetzkoy and Goldstein,[4] for incident neutron energies below 4.5 Mev. For $E_n > 4.5$ Mev the very small (n,γ) cross section is negligible compared with the (n,p) and (n,α) reactions. It is assumed that the charged-particle reactions produce no gammas.

The production of gamma rays by inelastic neutron scattering from iron, for incident neutron energies below 4 Mev, were taken from the experimental data of Montague and Paul.[16] For neutron energies above 4 Mev, the data of Perkin[3] which extend to $E_n = 8.5$ Mev, and the data of Caldwell[17] for $E_n = 14$ Mev were used.

The capture and inelastic gamma spectra are tabulated in Tables 12 and 13.

2.2.8 Nickel

The absorption gamma spectrum in nickel was assumed to be equal to the capture spectrum, which was taken from the compilation of Troubetzkoy and Goldstein.[4] The spectrum of gamma rays following inelastic scattering, for neutron energies below 4 Mev, was based mainly on the discrete-level excitation cross sections of Broder et al.,[18] supplemented by the work of Day[19] and Cranberg and Levin.[20] For neutron energies between 4 and 8.5 Mev the data of Perkin[3] were used. Above 8.5 Mev, the spectra were calculated by statistical theory with parameters adjusted to fit Perkin's data at $E_n = 8.5$ Mev.

The spectra are tabulated in Tables 14 and 15.

2.2.9 Zirconium

The capture gamma spectrum is taken from the compilation of Troubetzkoy and Goldstein[4] for gamma energies above 3 Mev. For $E_\gamma < 3$ Mev the capture spectrum for zirconium was assumed to approximate that for molybdenum (which has a similar spectrum above $E_\gamma = 3$ Mev). The values were taken from Reference 4. The gamma spectrum following inelastic scattering events in zirconium was obtained from the data of M. Fleishman,[21] using the total inelastic scattering cross sections of J. Ray (UNC Phys./Math Memo No. 1679, Dec. 1960). The spectra are tabulated in Tables 16a and 16b.

2.2.10 Cadmium

The capture gamma spectrum given in Table 17 is based on the compilation of Troubetzkoy and Goldstein[4] and on the work of Smither.[22] Nothing is presented in this report on the inelastic gamma spectrum as cadmium will be present in relatively low concentrations for its effect as a strongly capturing medium.

2.2.11 Samarium

The capture spectrum given in Table 18 is from Troubetzkoy and Goldstein[4] and Groshev.[23] No inelastic gamma spectrum is given.

2.2.12 Tungsten

A. Natural Tungsten

The gamma spectrum following thermal neutron capture is based on the experimental data found in References 24 to 27. The spectrum, assumed to be independent of neutron energy, is given in Table 19.

For low neutron incident energies (E_n < 2 Mev), the spectrum of gamma rays from inelastic scattering is based mainly on the discrete level-excitation cross sections of Smith.[28] At higher neutron energies the data from Perkin[3] were supplemented by statistical-model calculations. The values obtained are presented in Table 20.

B. The Tungsten Isotopes – W^{182}, W^{183}, W^{184}, and W^{186}

Additional calculations have been made to provide capture gamma spectra for the separated tungsten isotopes W^{182}, W^{183}, W^{184}, and W^{186}. These spectra were obtained from that for natural tungsten, as described in Troubetzkoy and Goldstein[4] in conjunction with data on binding energies and on certain gamma lines assignable to particular isotopes, as given by Treado and Chagnon.[29]

Briefly, the procedure was as follows. The capture gammas for natural tungsten are represented by a 12-group spectrum ($n_i E_i$), i = 1,2,..., 12; i.e., per absorption in natural tungsten there are emitted n_1 gammas of energy E_1, etc. A "background" spectrum was constructed, assumed common to all of the isotopes, by subtracting (with proper weighting) contributions attributable to particular isotopes. This background is

$$n_i^{(B)} = n_i - \sum_{\substack{j=182,183\\184,186}} a_j n_{j,i} \tag{2}$$

where $n_{j,i}$ is the number of gammas emitted, in the energy range containing E_i, by isotope j, and a_j is the abundance of isotope j in natural tungsten, multiplied by its thermal capture cross section and divided by the sum of the products of abundances and cross sections, i.e., a_j represents the probability that a given capture in natural tungsten is a capture in the isotope j.

Following this, the background spectrum was renormalized separately for each isotope j, yielding the constants c_j, so as to set the total gamma emission per capture in isotope j equal to the binding energy U_j:

$$C_j \sum_i n_i^{(B)} E_i + \sum_i n_{j,i} E'_{j,i} = U_j \tag{3}$$

where $E'_{j,i}$ is the actual energy of a line emitted by isotope j in the range represented by E_i, and U_j = 6.10, 7.48, 5.86, 5.34 for j = 182, 183, 184, 186, as reported by Treado and Chagnon. Finally, using the C_j found from Eq. 3, a 12-group capture gamma spectrum was computed for each isotope, following the formula

$$N_{j,i} = C_j n_i^{(B)} + \sum_k n_{j,k} \left(\frac{E'_{j,k}}{E_i}\right), \tag{4}$$

the summation being over lines k in the i^{th} energy bin of the spectrum. The last factor ensures the energy balance satisfied by the spectra

$$\sum_{i=1}^{12} N_{j,i} E_i = U_j \tag{5}$$

The capture spectra are presented in Tables 21 through 24.

2.2.13 Uranium-235 and Uranium-238 – Nonfission Gammas

In the proposed Monte Carlo calculations, the fission-related gammas will be generated from a prescribed power pattern and operating history, without regard to the histories of individual neutrons after fission. On the other hand, neutron absorptions encountered during the neutron Monte Carlo are recorded on an interaction tape which is later processed by the GASP program to generate a source for the secondary gamma problem. Since no distinction is made in this recording between nonfission and fission captures, the element-dependent gamma-production input to GASP should define the nonfission gammas produced per absorption (i.e., per capture-or-fission), since the fission gammas are treated in a separate (primary gamma) calculation.

These nonfission gammas per absorption were computed as follows. Let $n_{i,j}$ be the number of gammas of energy i produced as a result of the nonfission capture of a neutron in energy bin j. Then

$$N_{i,j} = \frac{\sigma_\gamma(E_j)}{\sigma_\gamma(E_j) + \sigma_f(E_j)} \cdot n_{i,j}$$

(where σ_γ and σ_f are the capture and fission cross sections) represents the number of nonfission gammas of energy i produced per fission or nonfission capture of a neutron in energy bin j. In the calculations for U^{235} the $n_{i,j}$ were assumed to be independent of j (the neutron energy). The capture spectrum (for U^{235}) was taken to be the same as that of the prompt fission gammas (to be discussed later). Their intensity corresponds to a total of 6.429 Mev/capture (see References 30, 31). To generate the $N_{i,j}$ for U^{235} the following average cross section values were used (data from Reference 8).

AVERAGE CROSS SECTIONS – $\sigma_\gamma/(\sigma_\gamma+\sigma_f)$ vs ENERGY FOR U^{235}

Neutron Bin	Lower Energy Limit, Mev	$\overline{\sigma_\gamma/(\sigma_\gamma+\sigma_f)}$
1	3.7 (-8)	0.15
2	1.0 (-7)	0.17
3	1.0 (-6)	0.25
4	4.0 (-6)	0.60
5	7.0 (-6)	0.30
6	1.0 (-5)	0.34
7	1.5 (-2)	0.28
8	1.0 (-1)	0.15
9	1.0 (+0)	0.035
10	4.0 (+0)	0.003
11	10.0 (+0)	0.0005
	1.81 (+1)	

For U^{235} the $N_{i,j}$ (number of nonfission gammas per absorption) are given in Table 25.

The thermal capture gamma spectrum, the $n_{i,j}$ for U^{238} is based on the work of Campion (see References 32 and 8). The spectrum was assumed to be independent of neutron energy; the total emission is 6.37 Mev/capture. Table 26, giving the number of nonfission gammas per absorption in U^{238}, is based on the following average cross section values (data from Reference 8).

AVERAGE CROSS SECTIONS – $\sigma_\gamma/(\sigma_\gamma + \sigma_f)$ vs ENERGY FOR U^{238}

Neutron Bin	Lower Energy Limit, Mev	$\overline{\sigma_\gamma/(\sigma_\gamma+\sigma_f)}$
1	3.7 (-8)	1.00
2	1.0 (+0)	0.75
3	1.3 (+0)	0.35
4	1.5 (+0)	0.10
5	1.8 (+0)	0.08
6	2.3 (+0)	0.04
7	4.0 (+0)	0.01
8	7.0 (+0)	0.001
	2.0 (+1)	

A. Prompt and Delayed Gammas from Fission in U^{235}

Presented below are the spectra, intensities, and time variations which have been compiled and calculated. This is followed by some discussion of the sources of information, and recommendations for future work.

B. Results

Recommended Values for the Gamma Release Rates and Integrals Following Fission of U^{235}

1. The total gamma energy release rate and integral, 0 to 10 seconds, is shown in Table 27 and Fig. 1.
2. The spectrum of the prompt (0 to 5.0×10^{-8} second) radiation is shown in Table 28.
3. The integral data (gammas released during first 1.0 second after fission) indicated in the last column of Table 27 are summarized in Table 29.

   At present the spectrum is assumed to be constant over the entire 1.0 second. Table 28 could be modified to represent the range 0 to 1.0 second by multiplying the entries for Mev/fission and photons/fission by 8.470/7.394 = 1.146.
4. Photon release rates, in 12 groups (1.0 second to 10 hours) are shown in Table 30.
5. Integrated gamma output, Mev/fission, in 12 energy groups and three time bands from 1.0 second to infinity is shown in Table 31.

C. Discussion: Sources of Data – Calculations

0 to 5×10^{-8} Second

The principal source of data on the spectrum and time dependence of the "prompt" gammas (0 to 5×10^{-8} second) is Maienschein et al., References 33 and 34.

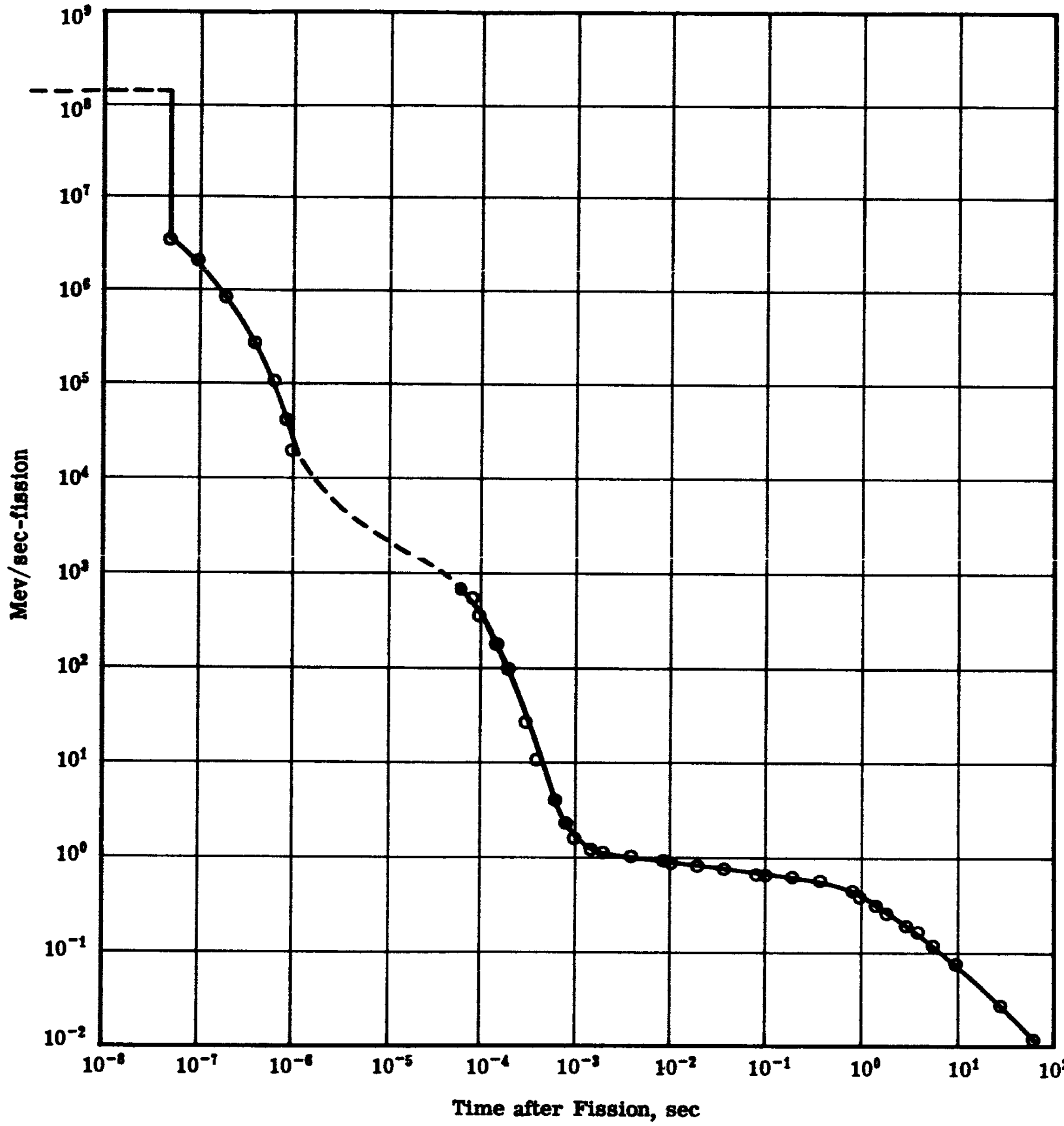

Fig. 1 — Gamma Energy Release Rate as Function of Time after Fission of U^{235}

Fig. 4.1.1 of Reference 32 was integrated to provide gamma sources (for the first 5×10^{-8} second following fission) in each of the 12 gamma groups which are used in the tabulation of delayed gamma intensities, plus one group including the 5.5 to 7.5 Mev contributions. The energy range below 0.3 Mev, not reported in References 33 and 34, has been augmented by 0.24 Mev/fission, following Skliarevskii[35] and Roos.[36] This led to a value of 7.39 Mev over the range of 0.02 to 7.5 Mev, 0 to 5×10^{-8} second.

5×10^{-8} to 1.0×10^{-6} Second

Here (and over the whole first second) the spectrum is assumed to be the same as that for the very prompt radiation. The time dependence from 5×10^{-8} to 10^{-6} second was estimated by combining the four intensity-vs-time curves of Fig. 4.2.2, Reference 34, each weighted by the average energy of the pertinent gammas. Small extrapolations of the 0.70-Mev and 1.30-Mev curves of that figure permitted a calculation of a plausible shape of the intensity-vs-time curve for the energy region of 0.15 to 1.42 Mev. This shape was assumed to describe the time variation of the entire gamma source in this time interval.

For the normalization of this portion of the curve, use was made of Maienschein's experimental result that, from 5×10^{-8} to 10^{-6}, about 5.7% as many counts were observed over a fairly wide range (0.16 to 1.93 Mev) as were observed in the first 5×10^{-8} second for the same energy range. Hence the total energy emission in this time range is taken to be $0.057 \times 7.39 = 0.421$ Mev.

1.0×10^{-6} to 6.0×10^{-5} Second

This range was filled in by graphical interpolation between the earlier and later times. The possible error cannot be too large as the integrated energy release over this range is only about 0.1 Mev.

6.0×10^{-5} to 1.0 Second

The shape of the curve of total energy release rate vs time in this time range was taken to be the same as that given for $E_\gamma \geq 0.51$ Mev in Reference 37, Fig. 9 and Table 1. The normalization from photons/fission-sec to Mev/fission-sec was made by matching the U^{235} portion of that reference at 1.0 second to the absolute intensity (in Mev/fission-sec) implied by the table of intensities which we have given for times $\geq$ 1.0 second.[38,39] This normalization appears well-founded since the ratio of Walton's intensities to Zigman and Mackin's is very nearly constant over the range from 1 to 4 seconds.

1.0 Second to 3.64×10^4 Seconds (10.14 Hours)

The data up to 1.74×10^4 seconds are reproduced from the work of Zigman and Mackin[38] as reported in Watson.[39] The first nine columns (covering 0.02 to 4.0 Mev) were extrapolated graphically from 1.74×10^4 to 3.64×10^4 seconds. The last three columns (covering 4.0 to 5.5 Mev) were extrapolated from 2.75×10^3 or 4.03×10^3 seconds.

3.64×10^4 Seconds to Infinity

The last time bands considered, extending the energy-release computations to 10^8 seconds and infinity, were treated by assuming that the exponents, c, in power fits to the various energy-rate curves, $W' = at^{-c}$ Mev/sec, were the same for $t > 3.64 \times 10^4$ seconds as in the range $1.74 \times 10^4 \leq t \leq 3.64 \times 10^4$ seconds.

2.2.14 Uranium-235 Energy Distribution of Gamma Rays Following Neutron-Producing Reactions

The gamma-ray spectrum following inelastic-scattering events in U^{235} was derived, for low incident neutron energies, from Hauser-Feshbach calculations, taking into account the known discrete levels.[40] At higher energies, statistical model calculations were performed. The spectra are given in Table 32.

2.2.15 Uranium-238 Energy Distribution of Gamma Rays Following Neutron-Producing Reactions

The gamma-ray spectrum following inelastic scattering for $E_n < 1$ Mev was derived from level-excitation measurements.[41-44] These measurements were extended by Hauser-Feshbach calculations, taking into account competition from fission and capture processes. The neutron penetrabilities were calculated using the nonlocal potential of Perey and Buck.[45] For $E_n > 1$ Mev the gamma spectra are based on statistical theory, including (n,n′), (n,2n), and (n,3n) processes. The spectra are presented in Table 33.

TABLE 1 — HYDROGEN – NUMBER OF GAMMA RAYS EMITTED PER ABSORPTION

	E_γ, Mev
Energy	2.23
0.03 ev	1.0
18.02 Mev	1.0

TABLE 2 — BERYLLIUM – NUMBER OF GAMMA RAYS EMITTED PER ABSORPTION

	E_γ, Mev					
E, Mev	.8550	2.5900	3.3650	3.4410	5.9560	6.8070
2,00000E 01	0	-0	-0	-0	-0	-0
1,00000E-03	,2400	,2100	,2800	,1100	,0200	,6500
1,00000E-09	,2400	,2100	,2800	,1100	,0200	,6500

TABLE 3 — BERYLLIUM – NUMBER OF GAMMA RAYS EMITTED PER NEUTRON-PRODUCING REACTION

	E_γ, Mev
E, Mev	2.43
1.80200E 01	.2610
1.63028E 01	.2900
1.47514E 01	.3200
1.33476E 01	.3600
1.20774E 01	.4000
1.09281E 01	.4450
9.88815E 00	.5100
8.94717E 00	.5700
8.09573E 00	.6450
7.32532E 00	.7100
6.62823E 00	.7750
5.99747E 00	.8140
5.42673E 00	.8370
4.91031E 00	.8380
4.44303E 00	.8200
4.02022E 00	.7600
3.63765E 00	.6300
3.29148E 00	.3000
2.97825E 00	.1200
2.69484E 00	0
1.00000E-10	0

TABLE 4 — CARBON – NUMBER OF GAMMA RAYS EMITTED PER CAPTURE

E_γ, Mev		
1.27	3.68	4.95
.3000	.3000	.7000

TABLE 5 — CARBON – NUMBER OF GAMMA RAYS EMITTED PER NEUTRON-PRODUCING REACTION

	E_γ, Mev											
E, Mev	0.25	0.75	1.25	1.75	2.25	2.75	3.25	3.75	4.438	5.25	6.00	9.00
1.80200E 01	.0000	.0000	.0001	.0003	.0006	.0010	.0015	.0020	.0677	.0099	.0196	.2704
1.71000E 01	.0000	.0000	.0001	.0003	.0005	.0008	.0012	.0016	.1432	.0074	.0152	.2089
1.63000E 01	.0000	.0000	.0001	.0002	.0004	.0006	.0009	.0012	.2140	.0053	.0120	.1584
1.55000E 01	.0000	.0000	.0001	.0002	.0003	.0004	.0006	.0008	.3397	.0033	.0083	.1044
1.47500E 01	.0000	.0000	.0000	.0001	.0002	.0003	.0004	.0004	.4254	.0022	.0057	.0668
1.40000E 01	.0000	.0000	.0000	.0001	.0001	.0001	.0002	.0002	.4971	.0013	.0041	.0386
1.33000E 01	.0000	.0000	.0000	.0000	.0001	.0001	.0001	.0001	.5778	.0005	.0030	.0193
1.27000E 01	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0001	.6555	.0001	.0019	.0085
1.21000E 01	0	.0000	.0000	.0000	.0000	.0000	.0000	.0001	.7032	.0000	.0008	.0035
1.15000E 01	0	.0000	.0000	.0000	0	.0000	.0000	.0000	.7811	.0000	.0002	.0009
1.09000E 01	0	0	0	0	0	.0000	.0000	0	.8447	0	.0000	.0001
1.04000E 01	0	0	0	0	0	0	0	0	.8859	0	0	.0000
9.89000E 00	0	0	0	0	0	0	0	0	.9170	0	0	.0000
9.41000E 00	0	0	0	0	0	0	0	0	.9458	0	0	0
8.95000E 00	0	0	0	0	0	0	0	0	.9690	0	0	0
8.65000E 00	0	0	0	0	0	0	0	0	.9836	0	0	0
8.51000E 00	0	0	0	0	0	0	0	0	.9902	0	0	0
8.35000E 00	0	0	0	0	0	0	0	0	.9939	0	0	0
8.20000E 00	0	0	0	0	0	0	0	0	.9974	0	0	0
8.05000E 00	0	0	0	0	0	0	0	0	.9979	0	0	0
7.90000E 00	0	0	0	0	0	0	0	0	.9983	0	0	0
7.83000E 00	0	0	0	0	0	0	0	0	1.0000	0	0	0
4.90000E 00	0	0	0	0	0	0	0	0	1.0000	0	0	0
4.82000E 00	0	0	0	0	0	0	0	0	0	0	0	0

TABLE 6 — OXYGEN – NUMBER OF GAMMA RAYS EMITTED PER ABSORPTION

E, Mev	E_γ, Mev 3.5
2,00000E 01	.4230
1,71000E 01	.4330
1,63000E 01	.4340
1,55000E 01	.4310
1,47500E 01	.4220
1,40000E 01	.4070
1,33000E 01	.3930
1,27000E 01	.3990
1,21000E 01	.3560
1,15000E 01	.3670
1,09000E 01	.4620
1,04000E 01	.4970
9,89000E 00	.5000
9,41000E 00	.5000
8,95000E 00	.5000
8,51000E 00	.4500
8,10000E 00	.4000
7,70000E 00	.1000
7,33000E 00	0
1,00000E-10	0

TABLE 7 — OXYGEN – NUMBER OF GAMMA

E, Mev	.75	1.25	1.75	2.25	2.75	3.25	3.75	4.25	4
1,81000E 01	,0099	,0010	,0023	,0060	,0093	,0143	,0164	,0178	.
1,71000E 01	,0101	,0010	,0023	,0060	,0099	,0143	,0161	,0173	.
1,63000E 01	,0104	,0010	,0023	,0059	,0091	,0141	,0158	,0166	.
1,55000E 01	,0106	,0010	,0022	,0058	,0089	,0138	,0152	,0156	.
1,47500E 01	,0107	,0009	,0020	,0057	,0086	,0135	,0145	,0146	.
1,40000E 01	,0112	,0009	,0019	,0055	,0083	,0130	,0136	,0134	.
1,33000E 01	,0116	,0008	,0016	,0052	,0078	,0124	,0128	,0125	.
1,27000E 01	,0120	,0006	,0014	,0049	,0073	,0120	,0123	,0116	.
1,21000E 01	,0122	,0005	,0010	,0045	,0069	,0117	,0116	,0099	.
1,15000E 01	,0118	,0003	,0007	,0042	,0067	,0112	,0098	,0065	.
1,09000E 01	,0113	,0001	,0005	,0041	,0061	,0094	,0062	,0031	.
1,04000E 01	,0109	,0000	,0004	,0037	,0048	,0064	,0033	,0008	
9,89000E 00	,0083	0	,0003	,0027	,0023	,0034	,0008	0	
9,41000E 00	,0093	0	,0002	,0008	,0006	,0006	0	0	
9,34000E 00	,0077	0	0	0	0	0	0	0	
9,20000E 00	,0062	0	0	0	0	0	0	0	
9,10000E 00	,0074	0	0	0	0	0	0	0	
8,95000E 00	,0064	0	0	0	0	0	0	0	
8,84000E 00	,0067	0	0	0	0	0	0	0	
8,70000E 00	,0065	0	0	0	0	0	0	0	
8,51000E 00	,0073	0	0	0	0	0	0	0	
8,35000E 00	,0060	0	0	0	0	0	0	0	
8,10000E 00	,0070	0	0	0	0	0	0	0	
7,87000E 00	,0046	0	0	0	0	0	0	0	
7,70000E 00	,0049	0	0	0	0	0	0	0	
7,40000E 00	,0006	0	0	0	0	0	0	0	
7,33000E 00	0	0	0	0	0	0	0	0	
6,30000E 00	0	0	0	0	0	0	0	0	
6,00000E 00	0	0	0	0	0	0	0	0	
1,00000E-10	0	0	0	0	0	0	0	0	

#1

RAYS EMITTED PER NEUTRON-PRODUCING REACTION

	E_γ, Mev										
.75	5.25	5.75	6.25	6.75	7.25	7.75	9.0	11.0	13.0	15.0	17.0
185	.0184	.0177	.1548	.1112	.1029	.0112	.2225	.2640	.1429	.0660	.0191
176	.0172	.0161	.1559	.1115	.1031	.0089	.2227	.2672	.1411	.0561	.0065
165	.0158	.0145	.1577	.1127	.1044	.0074	.2233	.2732	.1361	.0422	.0006
153	.0142	.0127	.1585	.1131	.1035	.0052	.2264	.2797	.1260	.0230	0
139	.0127	.0112	.1645	.1163	.0997	.0023	.2344	.2841	.1089	.0073	0
126	.0114	.0095	.1724	.1227	.0968	.0004	.2469	.2841	.0794	0	0
115	.0096	.0067	.1803	.1299	.0941	0	.2608	.2756	.0436	0	0
098	.0068	.0034	.1927	.1401	.0917	0	.2729	.2555	.0162	0	0
068	.0033	.0012	.2045	.1480	.0889	0	.2829	.2135	.0004	0	0
032	.0007	0	.2487	.1630	.0844	0	.3042	.1550	0	0	0
007	0	0	.3120	.1824	.0766	0	.3142	.0834	0	0	0
0	0	0	.4356	.1864	.0669	0	.2687	.0220	0	0	0
0	0	0	.6976	.1306	.0380	0	.1414	0	0	0	0
0	0	0	.7738	.1462	.0421	0	.0424	0	0	0	0
0	0	0	.8421	.1242	.0361	0	0	0	0	0	0
0	0	0	.8705	.0986	.0279	0	0	0	0	0	0
0	0	0	.8490	.1166	.0345	0	0	0	0	0	0
0	0	0	.8697	.1019	.0284	0	0	0	0	0	0
0	0	0	.8638	.1051	.0310	0	0	0	0	0	0
0	0	0	.8670	.1039	.0290	0	0	0	0	0	0
0	0	0	.8523	.1145	.0332	0	0	0	0	0	0
0	0	0	.8788	.0932	.0280	0	0	0	0	0	0
0	0	0	.8573	.1087	.0340	0	0	0	0	0	0
0	0	0	.9058	.0737	.0205	0	0	0	0	0	0
0	0	0	.9001	.0774	.0225	0	0	0	0	0	0
0	0	0	.9870	.0130	0	0	0	0	0	0	0
0	0	0	1.0000	0	0	0	0	0	0	0	0
0	0	0	1.0000	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0

#2

TABLE 8 — ALUMINUM – NUMBER OF GAMMA RAYS EMITTED PER CAPTURE

E, Mev	
.20	2.5000
1.5	.9500
1.78	1.0000
2.5	.7000
4.0	.8000
6.0	.2000
7.73	.2500

TABLE 9 — ALUMINUM – NUMBER OF GAMMA R

E, Mev	.75	1.25	1.75	2.25	2.75	3.25	3
1.80200E 01	.1000	.2100	.3100	.1600	.1100	.1100	.1
1.40000E 01	.1200	.2200	.3200	.1700	.1000	.1000	.1
1.09000E 01	.1300	.2500	.3100	.1800	.1000	.1000	.0
8.51000E 00	.1600	.2800	.3000	.2000	.1300	.1200	.0
6.63000E 00	.2000	.3100	.2800	.2700	.1400	.1100	.0
5.16000E 00	.2100	.3500	.2400	.3700	.1300	.0500	.0
4.02000E 00	.2200	.4000	.1800	.3800	.0700	.0100	
3.13000E 00	.1700	.4700	.0400	.3300	0	0	
2.44000E 00	.2900	.7100	0	0	0	0	
1.90000E 00	.3100	.6900	0	0	0	0	
1.48000E 00	.3600	.6400	0	0	0	0	
1.15000E 00	1.0000	0	0	0	0	0	
8.97000E-01	0	0	0	0	0	0	
1.00000E-03	0	0	0	0	0	0	

#1

AYS EMITTED PER NEUTRON-PRODUCING REACTION

E_γ, Mev								
.75	4.25	4.75	5.25	5.75	6.25	6.75	7.5	8.5
100	.0900	.1100	.1000	.0900	.0900	.0800	.1500	.4400
000	.0900	.1000	.0900	.0800	.0700	.0600	.1200	.2300
900	.0900	.0900	.0800	.0700	.0600	.0400	.0600	.0600
700	.0900	.0800	.0600	.0400	.0300	.0100	0	0
400	.0600	.0500	.0200	.0050	0	0	0	0
100	.0100	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0

#2

TABLE 10 — CHROMIUM – NUMBER OF GAMMA RAYS EMITTED PER CAPTURE

E_γ, Mev						
0.5	1.5	2.5	4.0	6.0	8.0	9.716
.8500	.4100	.2100	.1200	.2300	.3900	.0640

TABLE 11 — CHROMIUM – NUMBER OF GAMMA RAYS EMITTED PER NEUTRON-PRODUCING REACTION

E, Mev	Eγ, Mev 0.25	0.75	1.25	1.75	2.25	2.75	3.25	3.75	4.75	5.5	8.0	10.5
1.80200E 01	.0053	.4361	.6066	.3192	.3293	.3747	.4004	.3829	.6471	.4494	.4777	.0517
1.71000E 01	.0052	.4350	.6030	.3123	.3190	.3615	.3852	.3668	.6155	.4227	.4374	.0441
1.63000E 01	.0051	.4339	.5996	.3057	.3092	.3490	.3710	.3518	.5867	.3990	.4037	.0386
1.55000E 01	.0050	.4326	.5957	.2984	.2983	.3355	.3557	.3358	.5566	.3750	.3714	.0343
1.47500E 01	.0049	.4313	.5916	.2907	.2872	.3216	.3403	.3200	.5273	.3524	.3427	.0312
1.40000E 01	.0047	.4298	.5871	.2821	.2750	.3066	.3237	.3033	.4970	.3296	.3157	.0290
1.33000E 01	.0046	.4282	.5823	.2733	.2625	.2914	.3072	.2867	.4677	.3085	.2925	.0277
1.27000E 01	.0044	.4267	.5776	.2650	.2507	.2774	.2922	.2719	.4419	.2904	.2747	.0265
1.21000E 01	.0043	.4250	.5726	.2559	.2388	.2623	.2762	.2565	.4156	.2725	.2597	.0248
1.15000E 01	.0041	.4231	.5669	.2457	.2241	.2460	.2592	.2401	.3887	.2547	.2488	.0215
1.09000E 01	.0039	.4209	.5605	.2345	.2089	.2286	.2413	.2231	.3613	.2375	.2427	.0162
1.04000E 01	.0037	.4187	.5546	.2243	.1952	.2130	.2255	.2085	.3380	.2243	.2404	.0104
9.89000E 00	.0034	.4164	.5480	.2129	.1802	.1962	.2087	.1930	.3145	.2132	.2382	.0050
9.41000E 00	.0032	.4140	.5412	.2013	.1651	.1795	.1922	.1782	.2936	.2067	.2318	.0015
8.95000E 00	.0029	.4122	.5349	.1896	.1498	.1628	.1761	.1643	.2764	.2057	.2187	0
8.51000E 00	.0027	.4097	.5275	.1773	.1345	.1466	.1609	.1519	.2647	.2100	.1959	0
8.10000E 00	.0024	.4077	.5204	.1656	.1202	.1318	.1478	.1423	.2605	.2174	.1656	0
7.70000E 00	.0021	.4057	.5132	.1541	.1066	.1183	.1369	.1361	.2636	.2214	.1256	0
7.33000E 00	.0019	.4040	.5065	.1437	.0948	.1075	.1296	.1341	.2721	.2171	.0858	0
6.97000E 00	.0016	.4028	.5009	.1344	.0846	.0994	.1261	.1360	.2814	.1971	.0493	0
6.63000E 00	.0014	.4023	.4971	.1270	.0769	.0947	.1265	.1412	.2849	.1620	.0239	0
6.30000E 00	.0012	.4024	.4963	.1217	.0715	.0932	.1303	.1478	.2757	.1178	.0071	0
6.00000E 00	.0010	.4032	.4987	.1190	.0686	.0939	.1353	.1499	.2492	.0755	0	0
5.70000E 00	.0008	.4049	.5053	.1186	.0673	.0961	.1396	.1472	.2081	.0396	0	0
5.43000E 00	.0007	.4072	.5155	.1203	.0678	.0977	.1390	.1383	.1592	.0175	0	0
5.16000E 00	.0005	.4108	.5303	.1238	.0669	.0968	.1327	.1208	.1107	.0029	0	0
4.91000E 00	.0004	.4151	.5474	.1279	.0660	.0919	.1222	.0963	.0679	0	0	0
4.67000E 00	.0003	.4202	.5659	.1320	.0633	.0835	.1053	.0713	.0369	0	0	0
4.44000E 00	.0002	.4262	.5840	.1349	.0585	.0724	.0839	.0489	.0172	0	0	0
4.23000E 00	.0002	.4324	.5995	.1359	.0518	.0588	.0639	.0329	.0051	0	0	0
4.02000E 00	.0001	.4393	.6122	.1346	.0441	.0427	.0450	.0195	.0000	0	0	0
3.82000E 00	.0001	.4467	.6205	.1307	.0358	.0278	.0308	.0079	0	0	0	0
3.64000E 00	.0000	.4540	.6236	.1250	.0274	.0161	.0208	.0016	0	0	0	0
3.46000E 00	.0000	.4621	.6217	.1169	.0192	.0069	.0116	0	0	0	0	0
3.29000E 00	.0000	.4706	.6145	.1067	.0128	.0082	.0046	0	0	0	0	0

TABLE 11 — CHROMIUM (CONTINUED)

	E_γ, Mev											
E, Mev	0.25	0.75	1.25	1.75	2.25	2.75	3.25	3.75	4.75	5.5	8.0	10.5
3.13000E 00	0	.1213	1.0400	.1532	.0091	0	0	0	0	0	0	0
2.97000E 00	0	.1274	1.0192	.0143	.0161	0	0	0	0	0	0	0
2.83000E 00	0	.0682	.9896	.0091	.0147	0	0	0	0	0	0	0
2.69000E 00	0	.1016	.9516	.0102	.0120	0	0	0	0	0	0	0
2.56000E 00	0	.1784	.9364	.0105	.0107	0	0	0	0	0	0	0
2.44000E 00	0	.2023	.9249	.0111	.0051	0	0	0	0	0	0	0
2.32000E 00	0	.1521	.9215	.0118	0	0	0	0	0	0	0	0
2.21000E 00	0	.1165	.9266	.0115	0	0	0	0	0	0	0	0
2.10000E 00	0	.1162	.9186	.0094	0	0	0	0	0	0	0	0
2.00000E 00	0	.1209	.9139	0	0	0	0	0	0	0	0	0
1.90000E 00	0	.1253	.8996	0	0	0	0	0	0	0	0	0
1.81000E 00	0	.1120	.8880	0	0	0	0	0	0	0	0	0
1.72000E 00	0	.1207	.8792	0	0	0	0	0	0	0	0	0
1.63000E 00	0	.1239	.8762	0	0	0	0	0	0	0	0	0
1.55000E 00	0	.1530	.8469	0	0	0	0	0	0	0	0	0
1.48000E 00	0	.3156	.6844	0	0	0	0	0	0	0	0	0
1.41000E 00	0	.5248	.4752	0	0	0	0	0	0	0	0	0
1.34000E 00	0	.5327	.4673	0	0	0	0	0	0	0	0	0
1.27000E 00	0	.6538	.3461	0	0	0	0	0	0	0	0	0
1.21000E 00	0	.7291	.2708	0	0	0	0	0	0	0	0	0
1.15000E 00	0	.7412	.2588	0	0	0	0	0	0	0	0	0
1.09600E 00	0	.8000	.2000	0	0	0	0	0	0	0	0	0
1.04200E 00	0	1.0000	0	0	0	0	0	0	0	0	0	0
9.91000E-01	0	1.0000	0	0	0	0	0	0	0	0	0	0
9.43000E-01	0	1.0000	0	0	0	0	0	0	0	0	0	0
8.97000E-01	0	1.0000	0	0	0	0	0	0	0	0	0	0
8.53000E-01	0	1.0000	0	0	0	0	0	0	0	0	0	0
8.12000E-01	0	1.0000	0	0	0	0	0	0	0	0	0	0
7.72000E-01	0	1.0000	0	0	0	0	0	0	0	0	0	0
7.34000E-01	0	1.0000	0	0	0	0	0	0	0	0	0	0
6.99000E-01	0	1.0000	0	0	0	0	0	0	0	0	0	0
6.66000E-01	0	1.0000	0	0	0	0	0	0	0	0	0	0
6.32000E-01	0	1.0000	0	0	0	0	0	0	0	0	0	0
6.01000E-01	0	1.0000	0	0	0	0	0	0	0	0	0	0
5.72000E-01	0	0	0	0	0	0	0	0	0	0	0	0

TABLE 12 — IRON – NUMBER OF GAMMA RAYS EMITTED PER CAPTURE

E_γ, Mev	
.38	.7500
1.6	.6000
2.6	.2700
3.7	.2300
6.0	.2500
7.63	.3800
9.3	.0210

TABLE 13 — IRON – NUMBER OF GAMMA RAYS EMITTED PER NEUTRON-PRODUCING REACTION

	E_γ, Mev									
E, Mev	.85	1.5	2.5	3.5	4.5	5.5	6.5	7.5	8.5	9.5
1.81000E 01	.5000	1.1150	.3200	.1360	.0970	.0250	.0030	0	0	0
1.71000E 01	.6400	1.1100	.3300	.1500	.1040	.0400	.0100	.0060	.0020	0
1.63000E 01	.7500	1.1050	.3450	.1670	.1120	.0550	.0270	.0150	.0070	.0010
1.55000E 01	.8800	1.1000	.3600	.1850	.1180	.0700	.0550	.0300	.0140	.0030
1.47500E 01	1.0000	1.0900	.3800	.2000	.1240	.0820	.0740	.0500	.0200	.0050
1.40000E 01	1.1000	1.0800	.3950	.2190	.1300	.0930	.0790	.0530	.0260	.0070
1.33000E 01	1.1800	1.0600	.4050	.2350	.1330	.1100	.0790	.0540	.0270	.0090
1.27000E 01	1.2500	1.0400	.4200	.2500	.1350	.1220	.0790	.0550	.0280	.0096
1.21000E 01	1.3000	1.0200	.4350	.2650	.1360	.1270	.0780	.0550	.0280	.0098
1.15000E 01	1.3500	1.0100	.4500	.2800	.1400	.1320	.0750	.0540	.0280	.0098
1.09000E 01	1.3900	1.0000	.4700	.2950	.1410	.1360	.0730	.0510	.0270	.0097
1.04000E 01	1.4100	.9900	.4800	.3020	.1420	.1380	.0700	.0470	.0250	.0093
9.89000E 00	1.4200	.9700	.4950	.3100	.1410	.1400	.0660	.0430	.0230	.0080
9.41000E 00	1.4300	.9500	.5050	.3180	.1400	.1400	.0610	.0390	.0200	0
8.95000E 00	1.4400	.9300	.5200	.3250	.1380	.1380	.0540	.0320	.0160	0
8.51000E 00	1.4400	.9050	.5300	.3280	.1330	.1350	.0480	.0260	0	0
8.10000E 00	1.4300	.8800	.5400	.3310	.1260	.1200	.0410	.0190	0	0
7.70000E 00	1.4200	.8550	.5500	.3300	.1170	.1050	.0320	.0060	0	0
7.33000E 00	1.4100	.8300	.5600	.3280	.1060	.0900	.0210	0	0	0
6.97000E 00	1.4000	.8050	.5700	.3230	.0950	.0750	.0100	0	0	0
6.63000E 00	1.3700	.7800	.5700	.3170	.0850	.0600	.0030	0	0	0
6.30000E 00	1.3300	.7500	.5700	.3080	.0730	.0450	0	0	0	0
6.00000E 00	1.3000	.7200	.5650	.2970	.0640	.0300	0	0	0	0
5.70000E 00	1.2700	.6900	.5600	.2830	.0540	.0180	0	0	0	0
5.43000E 00	1.2400	.6650	.5500	.2700	.0450	.0070	0	0	0	0
5.16000E 00	1.1900	.6400	.5350	.2530	.0350	0	0	0	0	0
4.91000E 00	1.1500	.6100	.5150	.2380	.0250	0	0	0	0	0

TABLE 13 — IRON (CONTINUED)

E, Mev	E_γ, Mev .85	1.5	2.5	3.5	4.5	5.5	6.5	7.5	8.5	9.5
4,67000E 00	1,1100	,5800	,4950	,2200	,0150	0	0	0	0	0
4,44000E 00	1,0600	,5500	,4700	,2000	,0050	0	0	0	0	0
4,23000E 00	1,0300	,5200	,4400	,1800	0	0	0	0	0	0
4,02000E 00	1,0100	,4900	,4000	,1550	0	0	0	0	0	0
3,82000E 00	.9800	,4600	,3600	,1250	0	0	0	0	0	0
3,64000E 00	.9600	,4300	,3200	,1000	0	0	0	0	0	0
3,46000E 00	.9400	,3900	,2600	,0560	0	0	0	0	0	0
3,29000E 00	.9200	,3400	,2000	0	0	0	0	0	0	0
3,13000E 00	.9250	,3000	,1000	0	0	0	0	0	0	0
2,97000E 00	.9250	,2600	,0300	0	0	0	0	0	0	0
2,83000E 00	.9300	,2200	0	0	0	0	0	0	0	0
2,69000E 00	.9400	,1700	0	0	0	0	0	0	0	0
2,56000E 00	.9000	,1000	0	0	0	0	0	0	0	0
2,44000E 00	.9140	,0860	0	0	0	0	0	0	0	0
2,32000E 00	.9240	,0760	0	0	0	0	0	0	0	0
2,21000E 00	.9290	,0710	0	0	0	0	0	0	0	0
2,10000E 00	.9380	,0620	0	0	0	0	0	0	0	0
2,00000E 00	.9460	,0540	0	0	0	0	0	0	0	0
1,90000E 00	.9550	,0450	0	0	0	0	0	0	0	0
1,81000E 00	.9640	,0360	0	0	0	0	0	0	0	0
1,72000E 00	.9730	,0270	0	0	0	0	0	0	0	0
1,63000E 00	.9800	,0200	0	0	0	0	0	0	0	0
1,55000E 00	.9900	,0100	0	0	0	0	0	0	0	0
1,41000E 00	1,0000	0	0	0	0	0	0	0	0	0
8,50000E-01	1,0000	0	0	0	0	0	0	0	0	0
8,40000E-01	0	0	0	0	0	0	0	0	0	0
1,00000E-10	0	0	0	0	0	0	0	0	0	0

TABLE 14 — NICKEL – NUMBER OF GAMMA RAYS EMITTED PER ABSORPTION

E, Mev	E_γ, Mev 0.50	1.5	2.5	4.0	5.820	6.580	6.839	7.528	7.817	8.532	9.0
1.80170E 01	.0030	.0013	.0008	.0008	.0002	.0002	.0007	.0002	.0003	.0005	.0012
1.40320E 01	.0023	.0010	.0006	.0006	.0002	.0001	.0006	.0001	.0002	.0004	.0009
1.09280E 01	.0022	.0010	.0006	.0006	.0002	.0001	.0005	.0001	.0002	.0003	.0008
8.51080E 00	.0024	.0011	.0006	.0006	.0002	.0001	.0006	.0001	.0002	.0004	.0009
6.62820E 00	.0031	.0014	.0008	.0008	.0003	.0002	.0008	.0002	.0003	.0005	.0012
5.16210E 00	.0050	.0020	.0012	.0012	.0004	.0002	.0011	.0003	.0004	.0005	.0018
4.02020E 00	.0124	.0056	.0032	.0032	.0010	.0007	.0030	.0007	.0011	.0020	.0049
3.13100E 00	.0183	.0081	.0047	.0047	.0015	.0010	.0045	.0011	.0016	.0029	.0071
2.43840E 00	.0526	.0234	.0135	.0135	.0043	.0028	.0128	.0031	.0046	.0085	.0205
1.89900E 00	.1730	.0764	.0442	.0442	.0140	.0093	.0420	.0102	.0152	.0279	.0675
1.47900E 00	.4570	.2030	.1170	.1170	.0370	.0246	.1110	.0268	.0402	.0735	.1780
1.15180E 00	.8400	.3730	.2120	.2120	.0680	.0452	.2020	.0493	.0738	.1350	.3270
8.97030E-01	.9000	.4000	.2300	.2300	.0730	.0485	.2185	.0528	.0792	.1450	.3510

TABLE 15 — NICKEL – NUMBER OF GAMMA RAYS EMITTED PER NEUTRON-PRODUCING REACTION

	E_γ, Mev											
E, Mev	0.5	1.0	1.5	2.0	2.5	3.0	3.5	4.5	6.0	7.5	9.0	10.5
1,80170E 01	,8356	,3170	,9690	,3660	,5860	,4500	,3730	,7760	,3630	,1306	,0388	,0143
1,40320E 01	,8327	,3030	,9380	,3160	,5020	,3740	,2940	,5630	,2346	,0797	,0232	,0035
1,09280E 01	,8290	,2840	,8960	,2530	,4420	,2900	,2130	,3970	,1756	,0352	,0033	0
8,51080E 00	,8245	,2620	,8510	,1940	,3830	,2430	,1810	,2670	,0532	,0042	0	0
6,62820E 00	,8220	,2510	1,0000	,2870	,2200	,1430	,0875	,0992	,0050	0	0	0
5,16210E 00	,0223	,2520	,7350	,1050	,0735	,0525	,0314	,0226	0	0	0	0
4,02020E 00	,8210	,0230	1,3400	,0100	,0100	,0150	,0220	0	0	0	0	0
3,13100E 00	0	,1518	1,0170	,0261	0	0	0	0	0	0	0	0
2,43840E 00	0	0	1,0000	0	0	0	0	0	0	0	0	0
1,89900E 00	0	0	1,0000	0	0	0	0	0	0	0	0	0
1,47900E 00	0	0	1,0000	0	0	0	0	0	0	0	0	0

TABLE 16 — ZIRCONIUM – NUMBER OF GAMMA RAYS EMITTED PER CAPTURE AND PER NEUTRON-PRODUCING REACTION

(16a) Number of Gamma Rays Emitted per Capture

E_γ, Mev				
0.75	1.5	3.5	6.0	7.5
1.3	0.2	1.13	0.35	0.04

(16b) Number of Gamma Rays Emitted per Neutron-Producing Reaction

	E_γ, Mev					
E, Mev	1.0	2.0	3.0	4.0	5.0	6.0
18.0	.39	.32	.25	.19	.10	.03
10.9	.50	.38	.31	.31	.19	.10
8.51	.53	.38	.32	.32	.21	.09
6.63	.58	.40	.30	.23	.12	
5.16	.67	.42	.29	.12	0	
4.02	.80	.44	.15	0		
3.13	1.0	.50	0			
2.44	1.0	.30				
1.90	1.0	0				
1.48	1.0					
1.15	1.0					
0.90	0					

TABLE 17 — CADMIUM – NUMBER OF GAMMA RAYS EMITTED PER CAPTURE

E, MeV	E_γ, Mev							
	.5000	.5580	.6510	1.4000	2.5000	3.5000	5.5000	8.0000
2.00000E 01	.3500	.8800	.1900	.9200	.9600	.7300	.1700	.0100
1.00000E-09	.3500	.8800	.1900	.9200	.9600	.7300	.1700	.0100

TABLE 18 — SAMARIUM – NUMBER OF GAMMA RAYS EMITTED PER CAPTURE

E, Mev	E_γ, Mev							
	.3340	.4390	.6500	1.3000	2.2000	3.5000	5.7000	7.2000
2.00000E 01	.8200	.5400	.6500	1.5000	1.1000	.4500	.0500	.0100
1.00000E-09	.8200	.5400	.6500	1.5000	1.1000	.4500	.0500	.0100

TABLE 19 — NATURAL TUNGSTEN – NUMBER OF GAMMA RAYS EMITTED PER CAPTURE

E_{γ}, Mev											
0.25	0.75	1.25	1.75	2.25	2.75	3.25	3.80	4.60	5.25	5.60	6.60
.6000	.6530	.4280	.3950	.3300	.2600	.2300	.1980	.1000	.0600	.0420	.0720

TABLE 20 — NATURAL TUNGSTEN – NUMBER OF GAMMA RAYS EMITTED PER NEUTRON-PRODUCING REACTION

E, Mev	E_γ, Mev 0.25	1.00	1.75	2.25	2.75	3.25	3.75	4.25	4.75	5.25	5.75
1.80000E 01	.4500	2.6000	.4100	.1800	.0900	.0500	.0300	.0200	.0100	0	0
1.40000E 01	.2400	2.2000	.4500	.2200	.1000	.0600	.0400	.0200	.0200	.0200	.0100
1.09000E 01	.2000	2.1000	.6000	.3000	.2000	.0800	.0700	.0400	.0300	.0300	.0200
8.51000E 00	.2400	2.1000	.8400	.3600	.2600	.1400	.1200	.0800	.0500	.0400	.0200
6.63000E 00	.3200	2.2000	1.0000	.4000	.2000	.1800	.1400	.0700	.0400	.0200	0
5.16000E 00	.3600	2.1000	.9500	.3800	.0800	.0800	.0400	0	0	0	0
4.02000E 00	.3800	1.9000	.8600	.2300	0	0	0	0	0	0	0
3.13000E 00	.3800	1.7000	.6900	.0800	0	0	0	0	0	0	0
2.44000E 00	.4200	1.4000	.4600	0	0	0	0	0	0	0	0
1.90000E 00	.4200	1.0000	.1500	0	0	0	0	0	0	0	0
1.48000E 00	.6500	.4000	0	0	0	0	0	0	0	0	0
1.15000E 00	.8500	.2000	0	0	0	0	0	0	0	0	0
9.00000E-01	1.0000	.1000	0	0	0	0	0	0	0	0	0
7.00000E-01	1.0000	0	0	0	0	0	0	0	0	0	0
1.20000E-01	0	0	0	0	0	0	0	0	0	0	0

TABLE 21 — TUNGSTEN-182 – NUMBER OF GAMMA RAYS EMITTED PER CAPTURE

E_{γ}, Mev											
.2500	.7500	1.2500	1.7500	2.2500	2.7500	3.2500	3.8000	4.6000	5.2500	5.6000	6.6000
.5000	.6530	.4280	.3950	.3300	.2600	.2100	.1840	.0615	.0633	.0102	.1100

TABLE 22 – TUNGSTEN-183 – NUMBER OF GAMMA RAYS EMITTED PER CAPTURE

E_{γ}, Mev											
.2500	.7500	1.2500	1.7500	2.2500	2.7500	3.2500	3.8000	4.6000	5.2500	5.6000	6.6000
.8300	.9040	.5920	.5470	.4570	.3600	.2910	.2250	.0585	.0604	0	.0745

TABLE 23 – TUNGSTEN-184 – NUMBER OF GAMMA RAYS EMITTED PER CAPTURE

E_γ, Mev											
.2500	.7500	1.2500	1.7500	2.2500	2.7500	3.2500	3.8000	4.6000	5.2500	5.6000	6.6000
.5030	.6560	.4300	.3970	.3320	.2610	.2620	.1640	.0628	.0439	.0306	.0541

TABLE 24 — TUNGSTEN-186 – NUMBER OF GAMMA RAYS EMITTED PER CAPTURE

E_γ, Mev											
.2500	.7500	1.2500	1.7500	2.2500	2.7500	3.2500	3.8000	4.6000	5.2500	5.6000	6.6000
.4630	.5040	.3300	.3050	.2550	.2010	.1620	.1720	.1220	.0530	.0650	.0415

TABLE 25 — URANIUM-235 – NUMBER OF NONFISSION GAMMA RAYS EMITTED PER ABSORPTION

E, Mev	E_γ, Mev .0894	.6000	1.1020	1.5590	1.9900	2.3920	2.7930	3.2400	3.7420	4.2430	4.7430	5.2440	6.4200
1.82000E 01	.0034	.0014	.0005	.0003	.0001	.0001	.0001	.0000	.0000	.0000	.0000	.0000	.0000
1.00000E 01	.0207	.0085	.0032	.0015	.0009	.0006	.0003	.0002	.0002	.0001	.0000	.0000	.0000
4.00000E 00	.2420	.0990	.0368	.0176	.0101	.0067	.0041	.0029	.0018	.0011	.0006	.0004	.0005
1.00000E 00	1.0300	.4240	.1580	.0762	.0432	.0286	.0174	.0125	.0076	.0048	.0024	.0015	.0021
1.00000E-01	1.9300	.7920	.2940	.1420	.0806	.0535	.0325	.0233	.0142	.0089	.0046	.0028	.0040
1.50000E-02	2.[illegible]500	.9620	.3570	.1730	.0979	.0649	.0394	.0283	.0172	.0108	.0055	.0034	.0048
1.00000E-05	2.[illegible]700	.8490	.3150	.1520	.0864	.0573	.0348	.0250	.0152	.0095	.0049	.0030	.0043
7.00000E-06	4.1400	1.7000	.6300	.3050	.1730	.1150	.0696	.0499	.0304	.0190	.0098	.0061	.0085
4.00000E-06	1.7200	.7[illegible]60	.2620	.1270	.0720	.0478	.0290	.0208	.0127	.0079	.0041	.0025	.0036
1.00000E-06	1.1700	.4810	.1780	.0864	.0490	.0325	.0197	.0141	.0086	.0054	.0028	.0017	.0024
1.00000E-07	1.[illegible]300	.4240	.1580	.0762	.0432	.0286	.0174	.0125	.0076	.0048	.0024	.0015	.0021
3.70000E-08	0	-0	-0	-0	-0	-0	0	-0	-0	-0	-0	-0	0

TABLE 26 — URANIUM-238 – NUMBER OF NONFISSION GAMMA RAYS EMITTED PER ABSORPTION

E, Mev	E_γ, Mev .2500	.7500	1.2500	1.7500	2.2500	2.7500	3.2500	3.7500	4.5000
2.00000E 01	.0046	.0016	.0007	.0005	.0003	.0004	.0000	.0000	.0001
7.00000E 00	.0465	.0160	.0068	.0054	.0031	.0036	.0002	.0004	.0007
4.00000E 00	.1900	.1640	.0270	.0220	.0123	.0144	.0008	.0016	.0028
2.30000E 00	.3720	.1280	.0546	.0432	.0246	.0288	.0015	.0032	.0056
1.80000E 00	.4650	.1600	.0683	.0540	.0307	.0360	.0019	.0040	.0070
1.50000E 00	1.6300	.5600	.2400	.1900	.1070	.1260	.0067	.0140	.0245
1.30000E 00	3.5000	1.2[illegible]00	.5120	.4050	.2300	.2700	.0140	.0300	.0530
1.00000E 00	4.6500	1.6[illegible]00	.6830	.5400	.3070	.3600	.0190	.0400	.0700
3.70000E-08	0	-0	-0	-0	-0	-0	0	-0	-0

TABLE 27 — TOTAL GAMMA ENERGY RELEASE RATE AND INTEGRAL, 0 TO 10 SECONDS

t, sec	Mev/sec-fission	Integral, Mev/fission
0-5.0(-8)	1.48(+8)(average)	7.394
5.0(-8)	3.51(+6)(instantaneous)	
1.0(-7)	2.15(+6)(instantaneous)	
2.0(-7)	8.93(+5)(instantaneous)	
4.0(-7)	2.72(+5)	0.421
6.0(-7)	1.02(+5)	
8.0(-7)	4.26(+4)	
1.0(-6)	1.95(+4)	0.102
6.0(-5)	6.20(+2)	
8.0(-5)	5.14(+2)	
1.0(-4)	3.60(+2)	
1.5(-4)	1.77(+2)	
2.0(-4)	8.99(+1)	
3.0(-4)	2.66(+1)	0.0478
4.0(-4)	1.15(+1)	
6.0(-4)	3.94(+0)	
8.0(-4)	2.20(+0)	
1.0(-3)	1.58(+0)	
1.5(-3)	1.21(+0)	
2.0(-3)	1.12(+0)	
4.0(-3)	9.91(-1)	
8.0(-3)	8.68(-1)	
1.0(-2)	8.06(-1)	0.0716
2.0(-2)	7.60(-1)	
4.0(-2)	7.06(-1)	
8.0(-2)	6.66(-1)	
1.0(-1)	6.50(-1)	
2.0(-1)	6.05(-1)	0.433
4.0(-1)	5.11(-1)	
8.0(-1)	4.12(-1)	
1.0(+0)	3.78(-1)	
1.5(+0)	2.98(-1)	
2.0(+0)	2.47(-1)	
3.0(+0)	1.89(-1)	1.307
4.0(+0)	1.54(-1)	
5.5(+0)	1.21(-1)	
1.0(+1)	7.22(-2)	

TABLE 28 — SPECTRUM OF THE PROMPT (0 TO 5 × 10^{-8} SECOND) GAMMAS FROM U^{235} FISSION

Energy, Mev	Mev/fission, 0 to 5×10^{-8} sec	$\bar{E} = \sqrt{E_1 E_2}$	Photons/fission, at $\bar{E}$
0.02-0.4	0.710	0.08944	7.94
0.4 -0.9	1.950	0.6000	3.25
0.9 -1.35	1.327	1.102	1.204
1.35-1.8	0.912	1.559	0.585
1.8 -2.2	0.659	1.990	0.331
2.2 -2.6	0.526	2.392	0.220
2.6 -3.0	0.372	2.793	0.133
3.0 -3.5	0.310	3.240	0.0957
3.5 -4.0	0.218	3.742	0.0583
4.0 -4.5	0.155	4.243	0.0365
4.5 -5.0	0.089	4.743	0.0188
5.0 -5.5	0.061	5.244	0.0116
5.5 -7.5	0.105	6.423	0.0163
Total	7.394		

TABLE 29 — GAMMA ENERGY RELEASE, 0 TO 1 SECOND AFTER FISSION

Time Band, sec	Mev/fission
0-5.0 × 10^{-8}	7.394
5.0(-8)-1.0(-6)	0.421
1.0(-6)-6.0(-5)	0.102
6.0(-5)-1.0(-3)	0.0478
1.0(-3)-0.1	0.0716
0.1-1.0	0.4332
Total	8.470

TABLE 30 — PHOTONS PER SECOND PER FISSION AS

Seconds	.020 .400 .089	.400 .900 .600	.900 1.350 1.102	1.350 1.800 1.559	1.800 2.200 1.990	2. 2. 2.
1.0000E 00	1.600E-01	1.200E-01	7.200E-02	3.400E-02	2.500E-02	8.5
1.5000E 00	1.300E-01	9.600E-02	5.300E-02	2.700E-02	1.800E-02	6.2
2.0000E 00	1.000E-01	8.100E-02	4.100E-02	2.300E-02	1.400E-02	7.6
3.0000E 00	7.300E-02	6.100E-02	2.900E-02	1.700E-02	1.100E-02	6.7
4.0000E 00	5.500E-02	5.000E-02	2.300E-02	1.400E-02	8.500E-03	5.8
6.0000E 00	3.800E-02	3.700E-02	1.600E-02	1.000E-02	6.000E-03	4.6
9.0000E 00	2.400E-02	2.600E-02	1.100E-02	7.100E-03	4.200E-03	3.2
1.3000E 01	1.700E-02	1.900E-02	7.600E-03	5.200E-03	3.000E-03	2.4
1.9000E 01	1.100E-02	1.300E-02	5.300E-03	3.700E-03	2.200E-03	1.7
2.8000E 01	8.800E-03	8.600E-03	3.800E-03	2.600E-03	1.500E-03	1.1
4.1000E 01	5.000E-03	5.900E-03	2.700E-03	1.800E-03	1.000E-03	8.0
6.0000E 01	3.300E-03	3.900E-03	1.900E-03	1.200E-03	6.700E-04	5.2
8.8000E 01	2.100E-03	2.600E-03	1.300E-03	8.500E-04	4.400E-04	3.5
1.2900E 02	1.300E-03	1.700E-03	9.600E-04	5.700E-04	2.900E-04	2.3
1.8900E 02	8.500E-04	1.200E-03	6.900E-04	3.800E-04	1.900E-04	1.5
2.7700E 02	5.600E-04	7.700E-04	4.800E-04	2.500E-04	1.300E-04	1.0
4.0600E 02	3.800E-04	5.200E-04	3.400E-04	1.700E-04	8.000E-05	7.0
5.9500E 02	2.900E-04	3.400E-04	2.300E-04	1.100E-04	5.200E-05	4.2
8.7000E 02	2.300E-04	2.300E-04	1.500E-04	7.000E-05	3.400E-05	2.8
1.2800E 03	1.700E-04	1.500E-04	9.200E-05	4.600E-05	2.200E-05	1.7
1.8700E 03	1.200E-04	1.000E-04	5.800E-05	3.100E-05	1.300E-05	1.1
2.7500E 03	8.300E-05	6.600E-05	3.000E-05	1.800E-05	8.200E-06	6.8
4.0300E 03	4.900E-05	4.500E-05	1.700E-05	1.300E-05	5.000E-06	4.0
5.9000E 03	3.000E-05	2.900E-05	6.900E-06	7.600E-06	3.200E-06	2.5
8.6400E 03	1.700E-05	1.900E-05	4.700E-06	4.900E-06	2.100E-06	1.4
1.2700E 04	1.100E-05	1.200E-05	2.400E-06	2.700E-06	1.300E-06	6.8
1.7400E 04	7.800E-06	8.800E-06	1.600E-06	1.700E-06	9.600E-07	4.2
3.6500E 04	3.100E-06	3.800E-06	5.600E-07	5.600E-07	4.200E-07	1.3

*These data, as well as the spectra and intensities for earlier times (10^{-16}
generator program written for the ATHENA system.

#1

. FUNCTION OF ENERGY GROUP AND TIME AFTER FISSION*

E1						
E2						
EG, Mev						
.200	2.600	3.000	3.500	4.000	4.500	5.000
.600	3.000	3.500	4.000	4.500	5.000	5.500
.392	2.793	3.240	3.742	4.243	4.743	5.244
00E-03	7.500E-03	6.300E-03	4.600E-03	3.700E-03	2.100E-03	9.100E-04
00E-03	6.500E-03	5.500E-03	4.000E-03	2.700E-03	1.500E-03	6.000E-04
00E-03	5.800E-03	4.800E-03	3.600E-03	2.100E-03	1.200E-03	5.600E-04
00E-03	4.800E-03	3.900E-03	2.900E-03	1.500E-03	8.700E-04	4.200E-04
00E-03	4.000E-03	3.400E-03	2.400E-03	1.200E-03	6.800E-04	3.400E-04
00E-03	3.200E-03	2.600E-03	1.800E-03	8.700E-04	4.800E-04	2.400E-04
00E-03	2.300E-03	1.800E-03	1.300E-03	6.100E-04	3.400E-04	1.700E-04
00E-03	1.700E-03	1.300E-03	9.900E-04	4.400E-04	2.400E-04	1.200E-04
00E-03	1.200E-03	9.400E-04	7.000E-04	3.200E-04	1.700E-04	8.400E-05
00E-03	8.200E-04	6.100E-04	4.600E-04	2.100E-04	1.100E-04	5.400E-05
00E-04	5.800E-04	4.200E-04	3.100E-04	1.500E-04	7.500E-05	3.600E-05
00E-04	3.700E-04	2.700E-04	2.000E-04	1.000E-04	4.700E-05	2.300E-05
00E-04	2.400E-04	1.700E-04	1.200E-04	6.700E-05	2.900E-05	1.400E-05
00E-04	1.500E-04	1.000E-04	7.600E-05	4.300E-05	1.800E-05	8.300E-06
00E-04	9.000E-05	6.000E-05	4.500E-05	2.500E-05	1.000E-05	4.400E-06
00E-04	5.300E-05	3.500E-05	2.700E-05	1.500E-05	5.200E-06	2.300E-06
00E-05	3.300E-05	2.100E-05	1.600E-05	6.200E-06	2.800E-06	1.200E-06
00E-05	1.800E-05	1.200E-05	8.700E-06	3.600E-06	1.300E-06	4.600E-07
00E-05	1.200E-05	6.800E-06	5.000E-06	1.700E-06	5.400E-07	1.800E-07
00E-05	6.000E-06	3.700E-06	2.700E-06	6.200E-07	2.000E-07	4.900E-08
00E-05	3.400E-06	2.000E-06	1.400E-06	2.100E-07	6.400E-08	1.100E-08
00E-06	1.700E-06	9.800E-07	7.100E-07	4.500E-08	1.300E-08	1.000E-09
00E-06	8.300E-07	4.700E-07	3.400E-07	6.000E-09	2.000E-09	1.100E-10
00E-06	3.400E-07	1.900E-07	1.400E-07	7.500E-10	2.400E-10	1.200E-11
00E-06	1.300E-07	7.800E-08	5.700E-08	6.000E-11	2.200E-11	1.400E-12
00E-07	4.800E-08	2.800E-08	2.100E-08	7.500E-12	2.000E-12	1.100E-13
00E-07	2.500E-08	1.400E-08	1.000E-08	1.300E-12	3.300E-13	2.200E-14
00E-07	4.500E-09	2.200E-09	1.600E-09	1.500E-14	3.600E-15	3.100E-16

o 1.0 second), are included as internal data in the VANGEN source-

#2

TABLE 31 — INTEGRATED GAMMA OUTPUT, MEV/FISSION, IN 12 GROUPS AND 3 TIME INTERVALS FROM 1.0 SECOND TO INFINITY

		Time Interval, sec			
Group	$\overline{E}$, Mev	1.0 to 3.64×10^4	3.64×10^4 to 1.0×10^8	1.0×10^8 to ∞	Total, 1.0 to ∞
1	0.089	1.77(−1)	3.42(−2)	5.90(−3)	0.217
2	0.600	1.25(+0)	4.04(−1)	2.38(−1)	1.89
3	1.102	1.07(+0)	5.15(−2)	2.09(−2)	1.12
4	1.559	9.29(−1)	6.24(−2)	1.25(−3)	0.993
5	1.990	6.31(−1)	1.53(−1)	1.01(−1)	0.885
6	2.392	5.53(−1)	1.96(−2)	2.20(−4)	0.573
7	2.793	3.61(−1)	3.50(−4)	0.	0.361
8	3.240	3.05(−1)	1.70(−4)	0.	0.305
9	3.742	2.59(−1)	1.50(−4)	0.	0.259
10	4.243	1.41(−1)	0.0	0.	0.141
11	4.743	7.71(−2)	0.0	0.	0.0771
12	5.244	4.01(−2)	0.0	0.	0.0401
Total*		5.79	0.73	0.35	6.87
Total†					8.47
Total Fission Gammas					15.34

*0.02-5.5 Mev.
†0.02-7.5 Mev, 0-1.0 sec.

TABLE 32 — URANIUM-235 – NUMBER OF GAMMA RAYS EMITTED PER NEUTRON-PRODUCING REACTION

E, Mev	E_γ, Mev: 0.25	0.75	1.25	1.75	2.25	2.75	3.25	3.75	4.50	5.50	6.50
1.80174E 01	.9777	1.4130	.6998	.2072	.0547	.0167	.0080	.0050	.0045	.0010	.0001
1.71390E 01	.9099	1.1772	.5261	.1386	.0392	.0191	.0143	.0106	.0099	.0022	.0001
1.63030E 01	.8267	.9197	.3810	.1058	.0453	.0338	.0287	.0217	.0205	.0044	.0003
1.55080E 01	.7387	.6766	.2887	.1128	.0728	.0624	.0545	.0416	.0389	.0082	.0005
1.47510E 01	.6687	.4982	.2649	.1568	.1204	.1071	.0940	.0721	.0673	.0138	.0007
1.40320E 01	.6372	.4178	.3025	.2219	.1779	.1613	.1435	.1104	.1024	.0196	.0008
1.33480E 01	.6429	.4189	.3558	.2684	.2202	.2064	.1871	.1446	.1320	.0223	.0007
1.26970E 01	.6556	.4217	.3433	.2537	.2185	.2178	.2033	.1572	.1368	.0174	.0000
1.20770E 01	.6515	.3773	.2754	.1986	.1879	.2049	.1960	.1477	.1123	.0070	0
1.14880E 01	.6444	.3341	.2161	.1545	.1655	.1932	.1812	.1243	.0682	.0007	0
1.09280E 01	.6396	.3050	.1782	.1290	.1516	.1769	.1500	.0815	.0221	0	0
1.03950E 01	.6394	.2935	.1623	.1163	.1345	.1416	.0945	.0292	.0014	0	0
9.88820E 00	.6440	.2976	.1598	.1029	.1009	.0838	.0321	.0020	.0000	0	0
9.45090E 00	.6557	.3127	.1590	.0821	.0568	.0278	.0028	.0003	.0001	0	0
8.94720E 00	.7202	.3364	.1577	.0634	.0260	.0062	.0015	.0004	.0001	0	0
8.51080E 00	.7543	.3771	.1677	.0617	.0219	.0080	.0028	.0008	.0002	0	0
8.09570E 00	.8115	.4558	.2094	.0865	.0367	.0143	.0048	.0014	.0004	0	0
7.70090E 00	.9043	.6043	.3086	.1436	.0628	.0243	.0083	.0024	.0007	0	0
7.32530E 00	1.0489	.8638	.4879	.2340	.1019	.0393	.0132	.0038	.0011	0	0
6.96810E 00	1.2584	1.2635	.7452	.3562	.1542	.0591	.0197	.0056	.0015	0	0
6.62820E 00	1.5204	1.7482	1.0326	.4888	.2092	.0791	.0260	.0072	.0020	0	0
6.30500E 00	1.7222	2.0681	1.2059	.5591	.2332	.0854	.0270	.0072	.0019	0	0
5.99750E 00	1.7646	2.1070	1.2024	.5389	.2153	.0747	.0220	.0054	.0011	0	0
5.70500E 00	1.7602	2.0787	1.1578	.4988	.1895	.0617	.0167	.0036	.0007	0	0
5.42670E 00	1.7555	2.0493	1.1127	.4599	.1655	.0502	.0124	.0024	.0003	0	0
5.16210E 00	1.7504	2.0185	1.0668	.4217	.1431	.0401	.0089	.0015	.0001	0	0
4.91030E 00	1.7450	1.9865	1.0207	.3849	.1225	.0315	.0062	.0008	.0001	0	0
4.67080E 00	1.7393	1.9532	.9742	.3494	.1038	.0243	.0041	.0005	.0000	0	0
4.44300E 00	1.7331	1.9184	.9274	.3152	.0869	.0183	.0027	.0003	.0000	0	0
4.22630E 00	1.7266	1.8825	.8807	.2828	.0719	.0134	.0017	.0001	.0000	0	0
4.02020E 00	1.7196	1.8448	.8338	.2519	.0585	.0096	.0010	.0001	0	0	0
3.82420E 00	1.7121	1.8061	.7873	.2228	.0469	.0066	.0007	.0001	0	0	0
3.63760E 00	1.7041	1.7654	.7407	.1954	.0370	.0046	.0005	.0000	0	0	0
3.46020E 00	1.6955	1.7236	.6948	.1701	.0286	.0032	.0004	0	0	0	0
3.29150E 00	1.6865	1.6804	.6496	.1469	.0219	.0023	.0002	0	0	0	0
3.13100E 00	1.6769	1.6360	.6052	.1256	.0166	.0018	.0004	0	0	0	0
2.97830E 00	1.6666	1.5902	.5619	.1065	.0128	.0014	0	0	0	0	0
2.83300E 00	1.6556	1.5427	.5194	.0896	.0101	.0009	0	0	0	0	0
2.69480E 00	1.6439	1.4943	.4783	.0750	.0086	.0004	0	0	0	0	0
2.56340E 00	1.6314	1.4443	.4384	.0628	.0073	.0000	0	0	0	0	0

TABLE 32 — URANIUM-235 (CONTINUED)

E, Mev	E_γ, Mev: 0.25	0.75	1.25	1.75	2.25	2.75	3.25	3.75	4.50	5.50	6.50
2.43840E 00	1.6182	1.3934	.4003	.0534	.0059	0	0	0	0	0	0
2.31950E 00	1.6042	1.3417	.3641	.0468	.0042	0	0	0	0	0	0
2.20630E 00	1.5892	1.2884	.3301	.0424	.0022	0	0	0	0	0	0
2.09870E 00	1.5736	1.2346	.2994	.0391	.0006	0	0	0	0	0	0
1.99640E 00	1.5567	1.1789	.2724	.0351	0	0	0	0	0	0	0
1.89900E 00	1.5396	1.1229	.2505	.0300	0	0	0	0	0	0	0
1.80640E 00	1.5203	1.0647	.2343	.0228	0	0	0	0	0	0	0
1.71830E 00	1.5004	1.0072	.2237	.0145	0	0	0	0	0	0	0
1.63450E 00	1.4785	.9503	.2154	.0063	0	0	0	0	0	0	0
1.55480E 00	1.4569	.8964	.2058	.0010	0	0	0	0	0	0	0
1.47900E 00	1.4337	.8468	.1916	0	0	0	0	0	0	0	0
1.40680E 00	1.4087	.8041	.1735	0	0	0	0	0	0	0	0
1.33820E 00	1.3828	.7713	.1505	0	0	0	0	0	0	0	0
1.27300E 00	1.3586	.7509	.1217	0	0	0	0	0	0	0	0
1.21090E 00	1.3309	.7375	.0845	0	0	0	0	0	0	0	0
1.15180E 00	1.3076	.7285	.0491	0	0	0	0	0	0	0	0
1.09560E 00	1.2835	.7141	.0218	0	0	0	0	0	0	0	0
1.04220E 00	1.2622	.6887	.0049	0	0	0	0	0	0	0	0
9.91370E-01	1.2456	.6503	0	0	0	0	0	0	0	0	0
9.43020E-01	1.2346	.6044	0	0	0	0	0	0	0	0	0
8.97030E-01	1.2306	.5559	0	0	0	0	0	0	0	0	0
8.53280E-01	1.2322	.4946	0	0	0	0	0	0	0	0	0
8.11670E-01	1.2390	.4311	0	0	0	0	0	0	0	0	0
7.72080E-01	1.2540	.3662	0	0	0	0	0	0	0	0	0
7.34430E-01	1.2665	.2909	0	0	0	0	0	0	0	0	0
6.98610E-01	1.2808	.2268	0	0	0	0	0	0	0	0	0
6.64540E-01	1.2833	.1596	0	0	0	0	0	0	0	0	0
6.32130E-01	1.2800	.1031	0	0	0	0	0	0	0	0	0
6.01300E-01	1.2682	.0595	0	0	0	0	0	0	0	0	0
5.71970E-01	1.2448	.0326	0	0	0	0	0	0	0	0	0
5.44080E-01	1.2115	.0116	0	0	0	0	0	0	0	0	0
5.17540E-01	1.1736	.0021	0	0	0	0	0	0	0	0	0
4.92300E-01	1.1336	0	0	0	0	0	0	0	0	0	0
4.68290E-01	1.0973	0	0	0	0	0	0	0	0	0	0
4.45450E-01	1.0671	0	0	0	0	0	0	0	0	0	0
4.23730E-01	1.0448	0	0	0	0	0	0	0	0	0	0
4.03060E-01	1.0230	0	0	0	0	0	0	0	0	0	0
3.83410E-01	1.0103	0	0	0	0	0	0	0	0	0	0
3.64710E-01	1.0040	0	0	0	0	0	0	0	0	0	0
3.46920E-01	1.0000	0	0	0	0	0	0	0	0	0	0
1.27000E-02	0	0	0	0	0	0	0	0	0	0	0

TABLE 33 — URANIUM-238 – NUMBER OF GAMMA RAYS EMITTED PER NEUTRON-PRODUCING REACTION

E, Mev	E_γ, Mev: 0.25	0.75	1.25	1.75	2.25	2.75	3.25	3.75	4.50	5.50	6.50
1.80200E 01	.7385	.2957	.1635	.0955	.0584	.0424	.0414	.0372	.0355	.0074	.0007
1.71000E 01	.7096	.2854	.1476	.0773	.0393	.0220	.0178	.0149	.0224	.0142	.0015
1.63000E 01	.7334	.2688	.1273	.0603	.0281	.0163	.0162	.0191	.0411	.0279	.0029
1.55000E 01	.7265	.2480	.1123	.1014	.0311	.0247	.0295	.0366	.0780	.0516	.0050
1.47500E 01	.7178	.2404	.1198	.0721	.0485	.0422	.0513	.0636	.1352	.0842	.0075
1.40000E 01	.7138	.2563	.1562	.1047	.0732	.0656	.0809	.0999	.2075	.1182	.0089
1.33000E 01	.7243	.2980	.1919	.1325	.0914	.0854	.1071	.1312	.2601	.1251	.0067
1.27000E 01	.7391	.3154	.1948	.1276	.0938	.0920	.1164	.1396	.2529	.0879	.0020
1.21000E 01	.7411	.3076	.1818	.1180	.0893	.0885	.1091	.1239	.1834	.0295	.0000
1.15000E 01	.7389	.3032	.1776	.1154	.0847	.0765	.0863	.0865	.0835	.0023	.0001
1.09000E 01	.7404	.3041	.1783	.1122	.0731	.0535	.0494	.0374	.0154	.0007	.0001
1.04000E 01	.7417	.3041	.1745	.1022	.0568	.0311	.0204	.0091	.0022	.0015	.0003
9.89000E 00	.7438	.3001	.1625	.1535	.0377	.0136	.0049	.0017	.0033	.0029	.0006
9.41000E 00	.7478	.2905	.1433	.0631	.0216	.0053	.0024	.0028	.0065	.0056	.0011
8.95000E 00	.7542	.2734	.1180	.0422	.0114	.0041	.0043	.0054	.0126	.0109	.0021
8.51000E 00	.7630	.2465	.0926	.0279	.0102	.0072	.0080	.0102	.0237	.0203	.0038
8.10000E 00	.7776	.2125	.0705	.0274	.0167	.0128	.0144	.0183	.0425	.0363	.0066
7.70000E 00	.8009	.1767	.0693	.0426	.0289	.0222	.0252	.0320	.0742	.0629	.0110
7.33000E 00	.8376	.1577	.0959	.0686	.0465	.0361	.0412	.0525	.1217	.1021	.0168
6.97000E 00	.9016	.1764	.1446	.1037	.0704	.0553	.0640	.0817	.1896	.1567	.0231
6.63000E 00	.9993	.2295	.1961	.1393	.0949	.0763	.0903	.1157	.2650	.2161	.0251
6.30000E 00	1.0975	.2643	.2182	.1516	.1041	.0883	.1096	.1416	.3284	.2477	.0122
6.00000E 00	1.1226	.2585	.2028	.1367	.0963	.0897	.1169	.1518	.3480	.2147	0
5.70000E 00	1.1207	.2467	.1848	.1221	.0908	.0931	.1235	.1599	.3555	.1398	0
5.43000E 00	1.1284	.2394	.1736	.1149	.0915	.0979	.1297	.1653	.3513	.0628	0
5.16000E 00	1.1369	.2357	.1686	.1142	.0959	.1027	.1334	.1644	.3057	.0081	0
4.91000E 00	1.1476	.2356	.1691	.1184	.1009	.1046	.1309	.1533	.2157	0	0
4.67000E 00	1.1602	.2384	.1739	.1249	.1047	.1021	.1201	.1362	.1235	0	0
4.44000E 00	1.1744	.2434	.1811	.1314	.1063	.0947	.1028	.1142	.0503	0	0
4.23000E 00	1.1867	.2489	.1880	.1361	.1049	.0838	.0842	.0820	.0111	0	0
4.02000E 00	1.1984	.2546	.1942	.1389	.1009	.0704	.0651	.0416	.0000	0	0
3.82000E 00	1.2083	.2596	.1985	.1394	.0950	.0578	.0473	.0123	0	0	0
3.64000E 00	1.2163	.2634	.2011	.1380	.0883	.0479	.0294	.0016	0	0	0
3.46000E 00	1.2237	.2667	.2023	.1351	.0806	.0386	.0131	0	0	0	0
3.29000E 00	1.2304	.2693	.2024	.1310	.0731	.0297	.0039	0	0	0	0
3.13000E 00	1.2370	.2716	.2016	.1260	.0652	.0208	.0006	0	0	0	0
2.97000E 00	1.2443	.2736	.2000	.1197	.0565	.0120	0	0	0	0	0
2.83000E 00	1.2514	.2754	.1979	.1133	.0477	.0058	0	0	0	0	0
2.69000E 00	1.2595	.2771	.1949	.1055	.0375	.0017	0	0	0	0	0
2.56000E 00	1.2680	.2786	.1912	.0970	.0269	.0001	0	0	0	0	0
2.44000E 00	1.2769	.2796	.1868	.0879	.0170	0	0	0	0	0	0

TABLE 33 — URANIUM-238 (CONTINUED)

E, Mev	E_γ, Mev 0.25	0.75	1.25	1.75	2.25	2.75	3.25	3.75	4.50	5.50	6.50
2,32000E 00	1,2870	,2804	,1812	,0770	,0090	0	0	0	0	0	0
2,21000E 00	1,2976	,2806	,1748	,0651	,0037	0	0	0	0	0	0
2,10000E 00	1,3099	,2803	,1672	,0522	,0017	0	0	0	0	0	0
2,00000E 00	1,3226	,2793	,1583	,0380	0	0	0	0	0	0	0
1,90000E 00	1,3371	,2775	,1475	,0251	0	0	0	0	0	0	0
1,81000E 00	1,3520	,2749	,1354	,0153	0	0	0	0	0	0	0
1,72000E 00	1,3691	,2711	,1209	,0077	0	0	0	0	0	0	0
1,63000E 00	1,3891	,2657	,1037	,0025	0	0	0	0	0	0	0
1,55000E 00	1,4099	,2593	,0874	,0015	0	0	0	0	0	0	0
1,48000E 00	1,4299	,2518	,0694	0	0	0	0	0	0	0	0
1,41000E 00	,9497	,3290	,0527	0	0	0	0	0	0	0	0
1,34000E 00	1,0355	,3137	,0377	0	0	0	0	0	0	0	0
1,27000E 00	1,0858	,2913	,0250	0	0	0	0	0	0	0	0
1,21000E 00	1,1536	,2778	,0150	0	0	0	0	0	0	0	0
1,15000E 00	1,1754	,2562	,0091	0	0	0	0	0	0	0	0
1,09600E 00	1,1907	,2345	0	0	0	0	0	0	0	0	0
1,04200E 00	1,1938	,2183	0	0	0	0	0	0	0	0	0
9,91000E-01	1,1960	,2007	0	0	0	0	0	0	0	0	0
9,43000E-01	1,1970	,1778	0	0	0	0	0	0	0	0	0
8,97000E-01	1,1980	,1545	0	0	0	0	0	0	0	0	0
8,53000E-01	1,2224	,1157	0	0	0	0	0	0	0	0	0
8,12000E-01	1,2342	,0760	0	0	0	0	0	0	0	0	0
7,72000E-01	1,2454	,0469	0	0	0	0	0	0	0	0	0
7,34000E-01	1,2565	,0220	0	0	0	0	0	0	0	0	0
6,99000E-01	1,2619	0	0	0	0	0	0	0	0	0	0
6,66000E-01	1,2328	0	0	0	0	0	0	0	0	0	0
6,32000E-01	1,2079	0	0	0	0	0	0	0	0	0	0
6,01000E-01	1,1978	0	0	0	0	0	0	0	0	0	0
5,72000E-01	1,6890	0	0	0	0	0	0	0	0	0	0
5,44000E-01	1,1527	0	0	0	0	0	0	0	0	0	0
5,18000E-01	1,1329	0	0	0	0	0	0	0	0	0	0
4,92000E-01	1,1980	0	0	0	0	0	0	0	0	0	0
4,68000E-01	1,1012	0	0	0	0	0	0	0	0	0	0
4,45000E-01	1,0784	0	0	0	0	0	0	0	0	0	0
4,24000E-01	1,0420	0	0	0	0	0	0	0	0	0	0
4,03000E-01	1,0042	0	0	0	0	0	0	0	0	0	0
3,83000E-01	1,0004	0	0	0	0	0	0	0	0	0	0
3,65000E-01	1,0000	0	0	0	0	0	0	0	0	0	0

3. CONCLUSIONS

The tables presented in this report represent an adequate representation of the gamma spectra following neutron absorptions and inelastic-scattering events from the elements of interest in the tungsten nuclear rocket program. These include H, Be, C, O, Al, Cr, Fe, Ni, Zr, natural W and four W isotopes, U^{235} and U^{238}.

The thermal capture gamma spectra are generally well-known. The assumption that the capture spectrum is independent of the neutron energy should be further investigated. The errors introduced in the gamma sources by neglecting gammas from charged particles, though probably small, should be investigated. Indeed, the charged particles themselves will, in general, deposit several Mev of energy locally at the point of interaction. Neglecting the latter contribution might be more important than neglecting the gammas produced in the charged-particle reactions.

The gamma spectra following inelastic scattering, for incident neutron energies below ~4 Mev, are reasonably accurate as they are based on experimental level-excitation cross sections. The resultant gamma spectra become less reliable in proportion to the extent to which the experimental data are supplemented by Hauser-Feshbach calculations.

In the intermediate neutron energy range (4 to 8 Mev) the inelastic gamma spectra are not uniformly reliable. There is a real need for good experimental data in this energy range, for almost all the elements considered in this report. The only relatively complete data, those of Perkin,[3] are of poor quality. The inelastic gamma spectra tabulated for this energy range are generally based (in varying degree) on statistical theory. The parameters used in the theoretical model are at times questionable, and the unreliability increases as the incident neutron energy increases.

The lack of good experimental data becomes more acute when one considers high (>8 Mev) incident neutron energies. There are practically no data except perhaps at $E_n = 14$ Mev. For $E_n > 8$ Mev the tabulated spectra are almost exclusively derived from statistical theory. In general, the spectral shapes are given more reliably than the absolute magnitudes. Fortunately the problems to be run in this program use a fission source, which has only one-half of one per cent of the source above 8 Mev.

As tungsten is an element of major importance in the program, it was decided to obtain capture spectra for several tungsten isotopes. The spectra are based on those for natural tungsten, modified for each isotope according to its binding energy and the gamma transitions appropriate to the particular isotope. It remains outside the scope of this report to obtain the inelastic gamma spectra for the various tungsten isotopes.

The gamma-ray spectra following neutron interactions with uranium-235 and uranium-238 were separated into two parts. The gammas associated with non-fission capture events were treated in the same manner as those of the other elements.

A very complete study was made of the fission gammas – both prompt and delayed (from 0 to 10 hr). These data are believed to be accurate to within the 15% uncertainty claimed for the prompt radiation, and as such should be adequate for most practical applications. However, some uncertainties remain which might be investigated. One stems from the assumption of a constant gamma spectrum for the first second, which may not be entirely correct. For example, the average prompt photon energy implied by Table 28 is $\overline{E}$ = 0.53 Mev/photon, whereas Table 30 implies that $\overline{E}$ varies from 0.85 Mev at 1 second to about 1.0 Mev over the range from 10 seconds to about 5 minutes, after which it decreases to 0.6 Mev at 10 hr.

Also, the extrapolations performed to extend Table 30 from about 5 hr to 10 hr, while introducing relatively small contributions to the total energy release, could be important in computing heating rates for long times after shutdown. Hence these extrapolations should be examined more carefully if problems in these time regions are contemplated.

In general, the accuracy of both the nonfission and the fission gamma spectra are believed to be adequate for the problems to be considered in the tungsten nuclear rocket program.

4. REFERENCES

1. Eisenman, B. and Nakache, F. R.: UNC-SAM: A FORTRAN Monte Carlo System for the Evaluation of Neutron or Gamma-Ray Transport in Three-Dimensional Geometry, UNC-5093 (Aug. 1964).
2. Goldstein, H.: "Fundamental Aspects of Reactor Shielding," Addison-Wesley Publishing Co., Inc., Reading, Mass., 1959.
3. Perkin, J. L.: Nuclear Phys., 60:561 (1964).
4. Troubetzkoy, E. and Goldstein, H.: A Compilation of Information on Gamma-Ray Spectra Resulting from Thermal-Neutron Capture, ORNL-2904 (1960).
5. CINDA – An Index to the Literature on Microscopic Neutron Data, published by Columbia University and ENEA Neutron Data Compilation Centre, May 1965.
6. Draper, J. E. and Bostrom, C. O.: Nuclear Phys., 47:108 (1963).
7. Krumbein, A. D.: Neutron Cross Sections for Beryllium, Vol. B, UNC-5014 (May 1962).
8. Troubetzkoy, E. S. et al.: Neutron Cross Sections, UNC-5099 (Dec. 1964).
9. Hughes, D. J. and Schwartz, R. B.: Neutron Cross Sections, 2nd Ed., BNL-325 (July 1958). Also Hughes, D. J. et. al.: Neutron Cross Sections, Supplement No. 1, BNL-325 (Jan. 1960).
10. Ray, J. H. et al.: Neutron Cross Sections, UNC-5139 (Nov. 1965).
11. Towle, J. H. and Gilboy, W. B.: Nuclear Phys, 39:300 (Dec. 1962).
12. Troubetzkoy, E. S.: Fast Neutron Cross Sections of Fe, Si, Al, and O, Vol. C, NDA 2111-3 (Nov. 1959).
13. Tralli, N. et al.: Neutron Cross Sections, UNC-5002 (Jan. 1962).
14. Thompson, W. E. and Engesser, F. C.: USNRDL-TR-861 (June 1965).
15. Van Patter, D. M. et al.: Phys Rev., 128:1246 (1962).
16. Montague, J. H. and Paul, E. B.: Nuclear Phys., 30:93 (1962).

17. Caldwell, R. L., Mills, W. R., Jr., and Hickman, J. B., Jr.: Nuclear Sci, and Eng., 8:173 (1960).

18. Broder, D. L., et al.: Atomnaia Energia, 16:103 (1964).

19. Heinrich, F. and Tanner, F.: Helv. Phys. Acta, 36(3):298 (1963).

20. Buckingham, B. R. S. et al.: AWRE 0-28/60 (1961).

21. Fleishman, M. R.: UNUCOR-634 (Mar. 1963).

22. Smither, R. K.: Phys. Rev., 124:183 (1961).

23. Groshev, L. V. et al.: Nuclear Phys., 43:669 (1963).

24. Groshev, L. V. et al.: "Atlas of γ-Ray Spectra from Radiative Capture of Thermal Neutrons," Atomisdat, Moscow, 1958.

25. Hamermesh, B.: Phys. Rev., 80:415 (1950).

26. Kinsey, B. B. and Bartholemew, G. A.: Can. J. Phys. 31:1051 (1953).

27. Kubitschek, H. E. and Dankoff, S. M.: Phys. Rev., 76:531 (1949).

28. Smith, A.: Reports to the AEC Nuclear Cross Section Advisory Group, p. 12, WASH-1031 (Feb. 1961).

29. Treado, P. A. and Chagnon, P. R.: Nuclear Phys., 34(3):623 (June 1962).

30. Peele, R. W. et al., in Pile Neutron Research Physics, p. 273, IAEA, Vienna (1962).

31. Bertini, H. W. et al.: Basic γ Data for ART Heat Deposition Calculations, ORNL-2113 (Oct. 1956).

32. Campion, P. S.: Can. J. Phys., 37:377 (1959).

33. Peele, R. W., Maienschein, F. C., and Love, T. A.: The Energy Spectrum of Prompt Gamma Rays Accompanying the Thermal Fission of U^{235}, p. 45, ORNL-2609 (Feb. 15, 1958).

34. Maienschein, F. C. et al.: Characteristics of Fission Product Gamma Rays Emitted between 5×10^{-8} and 10^{-6} Second after Thermal Fission of U^{235}, p. 47, ORNL-2609 (1958).

35. Skliarevskii, V. V. et al.: Investigation of U^{235} Fission Gamma Rays in the Energy Region up to 250 kev, J. Exp. Theor. Phys., 5:2 (Sept. 1957).

36. Roos, M.: Sources of Gamma Radiation in a Reactor Core, Nuclear Energy, Part B (Reactor Technology)I, 98 (1959).

37. Walton, R. B. et al.: Delayed Gamma Rays for Photofission of U^{238}, U^{235}, and Th^{232}, Phys. Rev. 134(4B):B824 (May 25, 1964).

38. Zigman, P. and Mackin, J.: Early Time Decay of Fission Product Mixtures – II; Gamma-Energy Release and Ionization Rates Following Thermal Neutron Fission of U^{235}, Health Phys. 5:79.

39. Watson, J. E., Jr.: Analysis of Calculated and Measured Fission Product Activities, BRL Report No. 1239 (Feb. 1964).

40. Landolt-Börnstein Tables, New Series, Vol. 1, "Energy Levels of Nuclei," Springer-Verlag, Berlin, 1961.

41. Cranberg, L.: Neutron Scattering by U^{235}, Pu^{239}, and U^{238}, LA-2177 (1959).

42. Cranberg, L and Levin, J. S.: Phys. Rev., 109:2063 (1958).

43. Batchelor, R. and Towle, J. H.: Proc.Phys. Soc. (London), 73:193 (1959).

44. Smith, A. B., Private Communication to H. Goldstein (1960).

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