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VOY-C1-TR8

15 DECEMBER 1966

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SPACECRAFT RADIATION MAPPING
STUDY FOR THE RTG-STUDY
VOYAGER TASK C

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UNDER JPL CONTRACT No. 951112
MODIFICATION No. 3

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SECTION 1

INTRODUCTION

The purpose of this study is to provide an estimate of the radiation fields that exist in and around the Voyager Spacecraft as a result of utilizing plutonium-238 (Pu-238) fueled Radioisotope Thermoelectric Generators (RTG's) as the primary electrical power sources. The feasibility of utilizing these RTG's as primary power sources for the Voyager Spacecraft is presently under study as part of the current Voyager Phase 1A, Task C effort (Reference 1). The radiation mapping study discussed here is part of that feasibility study.

Such radiation maps are required in order to carry out meaningful assessments of the possible effects the radiations emanating from the RTG's will have on the Voyager Bus Science Payload and Support Subsystems. The radiation sensitivity of these subsystems have been the subject of previous studies under the present Phase 1A, Task C effort (References 2 and 3). These previous studies have established radiation sensitivity thresholds for the spacecraft subsystems and components independent of what the actual radiation environment might be at the subsystem and component locations.

Preliminary indications of the seriousness of the effects on any particular subsystem have previously been made utilizing a very simplified description of the radiation fields (Reference 4). These descriptions consisted of a definition of the type, the intensity, and the energy distribution of the nuclear radiation emitted by the curium-244 and plutonium-238 fueled RTG's. The RTG's were treated as point isotropic sources in an infinite void medium, thus, yielding a somewhat conservative estimate of the field intensity since any absorption of the radiation by the structure and equipment was neglected.

The results of these preliminary studies, however, have indicated the feasibility of using PU-238 fueled RTG's from the viewpoint of the deleterious effects of the radiation on the various subsystems comprising the Voyager Spacecraft. Also, the results of these studies showed that curium-244 would probably be useable only in conjunction with rather massive neutron shields. For this reason curium-244 has not been investigated further in the present work.

The present study goes well beyond these preliminary estimates of the radiation field and provides a much more realistic estimate of the radiation environment that exists at the various subsystem and component locations. The geometry of the Spacecraft Bus is considered along with the proper mass distributions. Allowance is also made for a number of interactions between the radiation and the vehicle structure and equipment. However, only those radiations emanating directly from the RTG sources have been considered here. No secondary radiations arising from the interaction of the source radiation with the spacecraft materials have been determined in the present study.

The radiations of concern, of course, are the high energy neutrons and gammas emitted by the Pu-238 fuel, which are certainly capable of producing various types and quantities of secondary radiation. However, it is felt that these secondary radiations will only be of possible concern to a small number (if any) of the science payloads. Therefore, any secondary radiation arising from the source radiation in the vicinity of the affected instrument will be determined as part of the instrument sensitivity study (Reference 3).

The mapping study has been carried out utilizing a fast neutron and gamma radiation shielding computer code known as QAD-PR, which was developed by the Los Alamos Scientific Laboratory (Reference 5). It is felt that the use of this code to perform the present mapping studies represented a reasonable compromise between the complexity of the shielding calculations and the overall accuracies required for the sensitivity studies. A brief discussion of this code is given in Appendix A. The spacecraft configuration used for the present study utilizes eight 75-watt (e) RTG's which are symmetrically mounted on the ring of the twelve-sided Voyager Bus. All eight units are fueled with Pu-238. A detailed description of this configuration is given in Section 3.

A summary of the results is given in Section 2. This includes the calculated values of the neutron and gamma dose rates, and the yearly integrated dose values at selected points of concern in and around the spacecraft. In addition, isodose plots for regions external to the central body of the vehicle are also given. This is followed by a discussion of the spacecraft and RTG configurations along with the assumptions and approximations utilized in sectionalizing the spacecraft for convenient computer analysis. Also included is a discussion of the estimated accuracies of the calculations.

SECTION 2

SUMMARY

The configuration of the Spacecraft Bus utilized in the present study is shown in Figure 2-1. A sectionalized model of this configuration is shown in Figure 2-2. The sectionalized model was used in performing the computer calculation. Figure 2-1 is a transparent overlay for Figure 2-2 so that a comparison of the vehicle configuration with the sectionalized model can be easily made. It is seen that the retropropulsion engine, equipment bay, harnessing, and the other more massive sections along with the RTG units are reasonably well represented. Since the retropropulsion engine represents a very significant fraction of the total Spacecraft Bus weight, it has a reasonably strong influence on the radiation field. This is particularly so since the engine fuel has a high hydrogen content. Thus, it must be reasonably well represented in the sectionalized model. On the other hand, the very light structures, such as the high gain antenna and support struts, have very little effect on the radiation field and, therefore, have been neglected. This sectionalization is further discussed below. The essential results of the mapping calculations are shown in Tables 2-1 and 2-2. Table 2-1 shows the dose rates for both the gammas and neutrons at various selected points in and around the Spacecraft Bus. The location of these selected points at which the dose is calculated (commonly called receiver points) is also given in Table 2-1 in terms of the cylindrical coordinates "r" and "Z". This coordinate system is defined in Figure 2-2. Since symmetry around the central vehicle axis is assumed, no specification of the azimuthal angle is required. Figure 2-2 also shows the location of the points listed in Table 2-1. Similarly, Table 2-2 shows the total integrated dose of neutrons and gammas after a one-year time period for the same receiver points.

In addition to the selected receiver points listed in Tables 2-1 and 2-2, the dose at many other receiver points has been determined in order to construct a more generalized radiation map. These results are contained in Figure 2-3 which shows isodose contours around the annular region containing the RTG sources. The isodose lines are not extended into the vehicle structure since they tend to take on rather complex shapes.

It is very difficult to assess the accuracy of these calculations without direct experimental results. However, some degree of confidence in the analytical techniques and nuclear cross section data utilized in the calculations can be obtained from previous experience gained with the use of these calculational techniques on other programs where experimental results were available. Such comparisons have been made utilizing experimental data from the SNAP-27 program. Here it was found that the experimental data and the results of calculations similar to those developed here were in agreement to about 5 percent. While this excellent agreement may be somewhat fortuitous, it does lend some confidence to the results.

The present results appear to adequately satisfy the requirements for the radiation effects assessment. However, further studies may be required which treat specific areas of the vehicle geometry more realistically in addition to a more detailed analysis of the energy distributions and the intensities of the nuclear particles. This may be particularly important around certain scientific instruments. A number of codes presently exist which can perform such calculations to a high degree of accuracy. These would include two-dimensional solutions of the Boltzmann Transport Equation by Carlson's discrete ordinate method, such as codes TDC or 2DXY, or complex geometry Monte Carlo treatments, such as codes 05R or FMC-G and FMC-N.

Figure 2-1. Voyager Spacecraft Bus Configuration

Figure 2-2. Vehicle Schematic for Voyager Bus Radiation Mapping (Pu-238 Fuel)

Figure 2-3. Isodose Contours

Table 2-1. Dose Rate Values for Voyager Bus, Pu-238 RTG Radiation Map

Receiver					Receiver				
No.	Coordinates		γ	η	No.	Coordinates		γ	η
	R (cm)	Z (cm)				R (cm)	Z (cm)		
			$\frac{\text{Rad}}{\text{hr}}$	$\frac{\text{Neutrons}}{\text{cm}^2 \cdot \text{sec}}$				$\frac{\text{Rad}}{\text{hr}}$	$\frac{\text{Neutrons}}{\text{cm}^2 \cdot \text{sec}}$
1	0	-500	1.4×10^{-3}	9.3×10^1	22	65	240	1.87×10^{-2}	9.65×10^2
2	0	0	2.72	5.75	23	126	253	2.91	1.54×10^3
3	55	0	4.87	2.75×10^2	24	330	282	2.28	1.59
4	230	0	5.15	3.46×10^2	25	192	295	5.84	3.32
5	55	73	5.39	3.12	26	255	330	3.03	2.17
6	126	73	8.31	5.15	27	295	350	1.81	1.2
7	55	121	5.79	3.54	28	0	333	2.2×10^{-4}	1.8
8	126	121	1.06×10^{-2}	6.30	29	65	333	1.47×10^{-2}	7.4×10^2
9	0	147	6.0×10^{-5}	5.30×10^{-1}	30	150	380	2.19	1.2×10^3
10	65	147	6.82×10^{-3}	4.05×10^2	31	330	368	1.22	7.85×10^2
11	111	155	8.73×10^{-3}	5.50	32	0	500	1.15	6.45×10^2
12	126	155	1.17×10^{-2}	6.85	33	237.6	282	4.58	4.5×10^3
13	104	185.5	1.30	7.80	34	305	212	1.66	1.08
14	111	185.5	1.25	7.70	35	305	355	1.61	1.05
15	118.5	185.5	1.31	8.05	36	162	355	2.68	1.53
16	126	185.5	1.59	9.20	37	0	282	2.7×10^{-4}	2.23
17	136	185.5	1.86	1.06×10^3	38	130	282	3.09×10^{-2}	1.62×10^3
18	230	185.5	1.91	1.32	39	330	282	1.65×10^{-2}	1.04×10^3
19	111	216	2.2	1.22	40	430	282	6.83×10^{-3}	4.48×10^2
20	126	216	2.4	1.32	41	1000	282	8.8×10^{-4}	6×10^1
21	0	240	2.25×10^{-4}	2.9×10^{-3}	42	230	1282	8.2×10^{-4}	5.6×10^1

Table 2-2. Total Yearly Dose Values for Voyager Bus, Pu-238 RTG Radiation Map

Receiver			Total Yearly Dose		Receiver			Total Yearly Dose	
No	Coordinates		γ Rad	η <u>Neutrons</u> cm ²	No.	Coordinates		γ Rad	η <u>Neutrons</u> cm ²
	R (cm)	Z (cm)				R (cm)	Z (cm)		
1	0	-500	1.2×10^1	2.92×10^9	22	65	240	1.65×10^2	3.03×10^{10}
2	0	0	2.39	1.8×10^9	23	126	253	2.56	4.84
3	55	0	4.27	8.65×10^9	24	330	282	2.0	5.0
4	230	0	4.52	1.09×10^{10}	25	192	295	5.13	1.04×10^{11}
5	55	73	4.73	9.80×10^9	26	255	330	2.67	6.8×10^{10}
6	126	73	7.30	1.62×10^{10}	27	295	350	1.59	3.76×10^{10}
7	55	121	5.08	1.11	28	0	333	1.9	5.65×10^7
8	126	121	9.3	1.98	29	65	333	1.29×10^2	2.32×10^{10}
9	0	147	5.3×10^{-1}	1.67×10^7	30	150	380	1.93	3.77
10	65	147	6.0×10^1	1.27×10^{10}	31	330	368	1.07	2.46
11	111	155	7.7×10^1	1.73	32	0	500	1.01	2.02
12	126	155	1.03×10^2	2.15	33	237.6	282	4.02	1.41×10^{11}
13	104	185.5	1.15	2.45	34	305	212	1.46	3.4×10^{10}
14	111	185.5	1.10	2.42	35	305	355	1.42	3.3
15	118.5	185.5	1.15	2.53	36	162	355	2.36	4.8
16	126	185.5	1.40	2.89	37	0	282	2.38	7.0×10^7
17	136	185.5	1.64	3.33	38	130	282	2.72×10^2	5.1×10^{10}
18	230	185.5	1.68	4.15	39	330	282	1.45×10^2	3.26
19	111	216	1.94	3.83	40	430	282	6×10^1	1.4
20	126	216	2.11	4.15	41	1000	282	7.75	1.88×10^9
21	0	240	2.0×10^1	5.9×10^7	42	230	1282	7.20	1.75×10^9

SECTION 3

DISCUSSION

3.1 RTG-SPACECRAFT CONFIGURATION

The RTG Spacecraft configuration utilized in the present mapping study is shown in Figure 2-1. This outline drawing indicates the overall layout of the spacecraft and the assumed positions of the RTG's. There are eight Pu-238 fueled, 75-watt (e) RTG units symmetrically placed around the spacecraft. The following sections briefly describe the characteristics of the RTG sources and the geometric model of the spacecraft that has been utilized in the mapping study.

3.1.1 RTG RADIATION SOURCES

The characteristic radiation resulting from the natural decay of Pu-238, which includes spontaneous fission, decay of the fission product nuclei, and the neutron-producing reactions of decay alpha particles with light element nuclei, has been previously described (References 2 and 4) and is here summarized in Tables 3-1 and 3-2. The total thermal power of the eight RTG units was taken as 16,040 watts (th).

Although previous studies have assumed a thin right cylindrical annular fuel region in each RTG, the present work assumes an RTG with a cermet type fuel. Because of its excellent helium retention characteristic, a cermet type fuel does not require a large central cavity in the fuel to contain the helium. Therefore, the fuel cylinder for this study is assumed to be much more dense than that in the earlier studies. This, in turn, makes the self-shielding of the neutrons and gamma photons within the source much more effective and, thus, reduces the net neutron and gamma intensity leaving the source capsules.

Also, since the Pu-238 is fissionable by fast neutrons, the neutron population from (α, n) reactions and from spontaneous fission neutrons can be increased when such neutrons cause further fissions to take place. Based upon calculations for similar systems, the overall neutron multiplication is estimated to be 1.2, including interactions between adjacent RTG's. This multiplication factor, however, is not considered in the present calculations.

Table 3-1. Gamma Photon Yield From Pu-238 Fuel Form

Characteristic Energy (Mev)	Energy Band (Mev)	Yield (Photon-sec ⁻¹ -watt ⁻¹)
0.0435	0 -0.07	1.85×10^7
0.0998	0.07-0.13	1.01×10^7
0.153	0.13-0.18	1.71×10^6
0.203	0.18-0.50	9.07×10^3
0.776	0.50-0.80	4.34×10^5
0.810	0.80-0.90	1.81×10^5
1.0	0.90-1.3	1.90×10^4
1.5	1.3 -1.9	1.37×10^4
2.3	1.9 -2.7	1.13×10^4
3.0	2.7 -4.0	6.60×10^3
5.5	4.0 -7.0	5.23×10^3

Table 3-2. Neutron Yield From Pu-238 Fuel Form

Energy Group Limits (Mev)	Neutron Yield (Neutrons-sec ⁻¹ -watt ⁻¹)
0.0-1.0	1130
1.0-3.0	2100
3.0-5.0	690
5.0-8.0	230
8.0-13.0	20
<p>Note: The above yields are for fuel relatively uncontaminated by light elements; the (α, n) reactions with such elements constitute largest source of neutrons.</p>	

For the present study, the RTG capsule sizes are as shown in Figure 2-1. The influence of the various types of thermal radiator fins presently under consideration on the radiation field has been neglected in the present study.

3.1.2 GEOMETRIC MODEL

The geometric model of the Spacecraft utilized in the present study is shown in Figure 2-2. Such sectionalization is required in order to simplify the task of specifying the spacecraft geometry required in the computer calculations. Utilizing the transparent overlay of Figure 2-1, it can be seen that the sectionalized model closely approximates certain key regions of the spacecraft such as the retropropulsion engine, midcourse/orbit adjust engines, equipment bays, etc. The retropropulsion engine, in particular, is of considerable importance due to its large mass and the fact that its fuel contains a considerable amount of hydrogen which has strong effects on the neutron flux. The mass distribution of certain key subsystems is listed in Table 3-3.

Table 3-3. Voyager Mass Distribution

Item	Weight (Pounds)
Capsule	5000
Retro Engine	9513
MC/OA Engines	2187
A/C Gas System	247
Bus Str. and Equipment	1347
Ant. and Scan	242

Utilizing the mass distribution and the materials in each of the regions defined in Figure 2-2, the average density and composition for each region was determined. The exceptions to this "homogenizing" of the regions are the RTG fuel sources and the cladding around them. In order to better simulate the proper amount of self-shielding, the fuel cladding material was left at full density and the fuel itself at 45 degrees of full density as far as determining

the self-shielding for any particular source point was concerned. In this way the effective self-shielding of the discrete sources was preserved even though the source integrations were carried out assuming all eight RTG units consisted of a diluted source material "smeared out" over the volume of a right cylindrical, annular source region symmetrical around the spacecraft. The source region so defined was divided into 200 subdivisions as depicted in Figure 3-1.

Figure 3-1. Simulated RTG Source

In addition to the above considerations, the RTG sources were also assumed to be aligned vertically rather than at the actual 30-degree inclined position. This is depicted in Figure 2-2. Such an assumption lead to considerable simplification in specifying the source boundaries required for computer input data.

3.2 SHIELDING CALCULATIONS

The shielding calculations carried out under the present study have been performed utilizing a computer program designated as QAD-PR. A brief discussion of this program and a

description of the nuclear cross section and buildup factor data required for program inputs are given here. A somewhat more general discussion of this program is given in Appendix A.

3.2.1 PROGRAM QAD-PR

The QAD-PR computer code is a program used to compute fast neutron and gamma ray penetration through shielding assemblies and materials. It performs point kernel calculations for complex geometries including the use of buildup factors for gamma radiation and removal data based on NDA moments fit data and Albert-Welton kernel attenuation for the neutron radiation.

The source volume is generally subdivided into a number of lesser volumes which are sufficiently small to permit the asymptotic approximation. Thus, compared with the distance between the volume element and a receiver point (where the dose is desired), the source volume can be represented by a point isotropic source. For the present work, the RTG units were represented by an annular ring source. This annular source was subdivided into 200 source volumes. It is felt that this many source points was sufficient to satisfy the above asymptotic approximation solution with reasonable accuracy. The accuracy of the results are very dependent upon the accuracy of the nuclear cross section data which make up the radiation attenuation coefficients. A brief discussion of this cross section data utilized in the present work is given below.

3.2.2 NUCLEAR INTERACTIONS

All gamma photon interactions with materials encountered in its path have been represented by the usual total mass attenuation coefficients. In the energy range of interest, from 0.04 Mev to 7.0 Mev, most reactions will be Compton scattering. However, where appropriate, pair production and photoelectric effects are included. Curves of the total attenuation coefficients used in this study are shown in Figure 3-2. These coefficients are based upon the cross section data of Reference 6.

Figure 3-2. Total Mass Attenuation Coefficient (Without Coherent Scattering)

Scattering of photons into the beam from surrounding matter is treated by the usual buildup approximations. Buildup factors are based on the least squares fit of a second order polynomial to experimental data (Reference 7). The dose buildup curve for aluminum is shown in Figure 3-3.

Figure 3-3. Dose Buildup Factors for Aluminum

The removal of source neutrons from the beam is treated by the semi-empirical removal cross section. This is the cross section which best characterizes the effective attenuation of Uranium 235 fission neutrons in an infinite medium of a given material compared with the attenuation by water in a similar geometry.

The effect of scattering neutrons into the beam is approximated by using the so-called "Albert-Welton kernel". Here the measured, uncollided neutron flux, ϕ_u , from a Uranium 235 fission plate in an infinite pool of light water is fit by least squares to an expression such as (Reference 8):

$$\phi_u = \frac{P_o Y \alpha_1}{4\pi} \left(\frac{H^{\alpha_2}}{R^2} \right) \exp \left[-\alpha_3 H^{\alpha_4} + \Sigma_o H + \sum_i (\rho_i \Sigma_i t) \right]$$

Where:

- P_o = isotopic power level, watts
- Y = neutron yield from the primary source, neut/sec/watt
- Σ_o = removal cross section of oxygen in water, cm^2/gm
- H = relaxation lengths of material compared with that of water
- ρ_i = mass density of shield material, i , gm/cm^3
- Σ_i = removal cross section of shield, i , cm^2/gm
- t = ray thickness through shield layer, i , cm
- $\alpha_1, \alpha_2, \alpha_3, \alpha_4$ = constants

The energy of the uncollided flux is biased toward the higher energies because down-scattering to a lower energy group is not considered in this model. However, absorptions are handled separately for each energy group and the uncollided flux can thus see a significantly altered gamma spectrum with respect to a reference in or near the source. The neutron spectrum will not be altered because of the single removal cross section which applies uniformly over all neutron energies.

3.2.3 VALIDATION OF QAD-PR AND CROSS SECTION DATA

The validity of utilizing a particular analytical tool lies with the accuracy of the results obtained. For the present work it is difficult to determine the accuracy of the results without direct comparison with experimental results. However, the QAD-PR program has been previously used, along with much of the above cross section data, to map the radiation field around a single PU-238 fueled capsule for the SNAP-27 Program. The calculated values agreed to within approximately 5 percent of the measured dose rates. While such good agreement may be fortuitous, it nevertheless serves to validate both the program and the cross section data and to lend credence to the source definition.

SECTION 4
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APPENDIX A

QAD-PR COMPUTER CODE

The QAD-PR code, written at the Los Alamos Scientific Laboratory, is a program used to compute fast neutron and gamma ray penetration through shielding assemblies and materials. The original version of QAD-PR, known as QAD, has since undergone revision to its present form which is described in this Appendix. The computations are based on point kernel ray tracing techniques which include the use of polynomial buildup factors for gamma radiation; and for neutron penetration, removal based on NDA moments fit data and Albert-Welton kernel attenuation are used.

The code accommodates up to 20 gamma ray source group energies and 10 fast neutron energy source groups with a lower edge of 0.33 Mev as per the moments fit data for a fission source. The source may be specified in either rectangular, cylindrical or spherical geometry. Zones (or space regions) are formed by surface boundaries which intersect to result in the desired volume enclosures. These boundaries are in general curved surfaces or planes which are described by the input information. The material composition of each zone is specified which enables the number of mean free paths for the neutron groups to be computed from a source point to a receiver point (detector); with this, application of the material gamma attenuation, geometrical attenuation and integration over the source volume result in computed radiation intensities at the receiver point. Any number of receiver points may be prescribed per problem.

Computed radiation intensities in the form of a variety of printout information is available from the program computations. For gamma radiation, the code prints out the direct beam energy flux and mean buildup factors for each energy group; also the dose rates and energy absorption rates for four additional materials both for direct beam penetration (without buildup) and penetration including buildup may be obtained as printout information upon supplying the appropriate conversion factors as input. The units for all the printout intensities are determined by the input. The mean buildup factors for each energy group are also listed which are the ratios of the gamma intensity including buildup factors to the direct beam intensities.

The neutron printout information is the differential number spectrum, the number flux and energy flux. The neutron dose rate and heating rates in four other materials may also be obtained by supplying the appropriate input information. Provisions are also made in the program for computing an alternate neutron dose rate based on Albert-Welton and neutron removal attenuation; again, the form of the output information is determined by the input constants.