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NUS-145

TOPICAL REPORT

RADIOBIOLOGICAL CONSIDERATIONS IN
REENTRY OF NUCLEAR REACTOR FRAGMENTS

MASTER

For

Space Nuclear Propulsion Office
Atomic Energy Commission
National Aeronautics and Space Administration

Contract SNPC-6

GROUP 1

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Morton I. Goldman
Albert W. DeAgazio
Luis F. Garcia
John S. Lee

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AEC RESEARCH AND DEVELOPMENT REPORT

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This document contains Confidential-Restricted Data relating to civilian applications of atomic energy.

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By

Morton I. Goldman
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Nuclear Utility Services, Inc.

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For The Atomic Energy Commission

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TABLE OF CONTENTS

	<u>Page</u>
TABLE OF CONTENTS	i
LIST OF FIGURES AND TABLES	iii
I. INTRODUCTION	1
II. DOSE ROUTES	3
A. External Exposure	3
B. Internal Exposure	4
III. PHYSICAL DOSIMETRY	5
A. External Dose	5
1. Gamma Dose	5
2. Beta Dose	9
B. Internal Dose	14
1. General	14
2. Range Average Dose Models	20
3. Lung Dose Model	25
4. Gastrointestinal Tract Dose Model	27
IV. RADIOBIOLOGICAL CONSIDERATIONS	29
A. External Radiation Exposure	30
B. Internal Dose Considerations	31
1. Inhalation	31
a. NAS - NRC Report	32
b. Experimental Findings	34
2. Ingestion	38
a. General	38
b. Experimental Findings	39
3. Summary	43
V. PARTICLE AND POPULATION DISTRIBUTIONS	46

TABLE OF CONTENTS (cont'd)

	<u>Page</u>
A. Particle Distributions	46
B. Population Distributions	48
VI. DISCUSSION	51
VII. RECOMMENDATIONS	53
BIBLIOGRAPHY	54

LIST OF FIGURES

<u>Figure No.</u>	<u>Title</u>	<u>Page</u>
1	Allowable Initial Activity on Ground/Rad	10
2	Relationships of Preliminary Dose Evaluations to Core Inventory	45
3	World Population Density with MA-9 Trajectory Overlay	50

LIST OF TABLES

<u>Table No.</u>	<u>Title</u>	<u>Page</u>
I	Fraction of Beta Activity Escaping Particle	12
II	Beta Dose Rates	12
III	Dose Rate in Tissue Around 1- μ c Particle	15
IV	Average Dose Rate for 1- μ c Particle	15
V	Limiting Values for Lung Dose Model	26
VI	Limiting Values for GI Tract Model	28
VII	Limiting Values Applicable to Range Average Dose of 100 rads to Lung and GI Tract (LLI)	44
VIII	Particle Size Distribution	47

I. INTRODUCTION

The program for operation and testing of nuclear rocket vehicles carries with it the implicit probability of failures which may create radiation hazards. The destruction of a nuclear stage following operation (whether normal or abnormal) will create fragments with appreciable fission product, or other radioactive, inventories which may be dispersed over great or small areas of the earth depending upon the circumstances. Aside from the problem of hazards in the area of the launch site, a major concern in the safety evaluation of nuclear rocket operations must be with the damage that could accrue from the essentially uncontrolled radiation exposures to fragments impacting on the earth's surface.

In order to provide a reliable engine capable of sustained, efficient performance, reactor core materials must be able to withstand high temperatures without damage. This factor implies certain limits on ablative behavior of core material during reentry, and produces a hazard significantly different from that of world-wide weapons fallout with which this problem has been compared. (1) In the rocket case, relatively large particles can be expected to impact almost anywhere along the trajectory band of the rocket prior to destruction; in the weapon test case large particles settle from the cloud within a relatively short distance of the burst, and only the widely-dispersed very small particles constitute a significant "off-site" or world-wide problem. The major function of a post-operational destruct (or other safety) system employed in nuclear rocket engines must be to produce a velocity and/or size distribution of fragments which, upon returning to the earth's surface, will not cause significant injuries to persons in the vicinity of such particles.

In order to provide a basis for destruct system criteria, the effects accruing from exposure to single (or a few) particles has been examined. Models for dose estimation have been chosen, and the relationship between dose, particle activity and particle size determined on the basis of these models. At this point it is well to indicate that the resolution of both dose models and dose values, particularly for the internal exposure case, is extremely difficult, since presently accepted models and doses are predicated on reasonable uniform exposure of the affected organ both in space and time. For the case where inhalation or ingestion of a single particle occurs, the dosimetry and the radiobiology are so diffuse as to be practically nonexistent. These are discussed in considerable detail in Sections III and IV.

An additional consideration which is necessary from a practical point of view is the probability of interaction between particles and people. If the probability is vanishingly small, considerably less concern is warranted than if the probability is small but significant. Unfortunately, this value is not amenable to precise calculation at this time. Nevertheless, rough estimates can be made; and these indicate that probabilities of interaction are large enough to warrant careful consideration of the resultant potential hazards. The basis for this estimate is presented in Section V.

The establishment of permissible doses under such circumstances is clearly not the function of NUS, but rather of the government agencies involved. Some discussion of dose values and their implications is included in Section VI, together with an indication of areas of technical ignorance which are in need of research to define more precisely present dosimetric and radiobiological concepts.

II. DOSE ROUTES

In common with most other releases of radioactive material to the environment, reactor debris can cause radiation exposure either from outside the body (external) or from inside the body (internal) following ingestion or inhalation. The destruction of a nuclear stage at high altitudes will produce fragments of radioactive debris ranging from sub-micron particles to about a centimeter in size. These fragments will settle at differing velocities and will be transported horizontally to differing extents by the wind. There will be, however, a significant probability of both external and internal exposure from these fragments.

A. EXTERNAL EXPOSURE

Section V of this report discusses the fragment density on the earth's surface and concludes that for external irradiation, it is reasonable to expect that no more than one particle will affect any one individual.

Both beta and gamma doses may be delivered by fragments deposited on the ground, on roof or building surfaces, on clothing, etc. The external gamma dose from such particles is believed to be more significant than the beta dose, since beta radiation is not highly penetrating. Most areas of the body would be protected to a great extent by any clothing worn. On the other hand, gamma radiation is more highly penetrating, requiring two to ten inches of ordinary building material to reduce the dose by a factor of 2. However, the fallout of reentering particles may occur in areas of the world where little clothing is customarily worn; consequently, the beta dose from single particles is considered.

The fragment size and fission product age affect the delivered beta dose rates since self-absorption of the beta particles within the fragment is strongly dependent upon the effective beta energy (which decreases with age) and the particle diameter (which would also be smaller for slower returning particles). This does not play as important a role in gamma dose rates from reentered fragments, however.

Section III of this report discusses the physical dosimetry of beta and gamma radiation from single particles external to the body. The results are expressed in initial particle activities on the ground as a function of time to reach the ground for a given dose. These results can then be related to mission characteristics by considering ablation and diffusion losses during reentry, as well as the acceptability of dose values under these circumstances.

Beta values for fragments deposited on the skin are not considered; they may, however, be inferred from the internal dose considerations.

B. INTERNAL EXPOSURE

As mentioned above, particle sizes to be expected will probably range downward from about a centimeter to a micron and below. The larger sizes are obviously excluded from consideration as internal radiation sources, but below some "critical" size (perhaps about 10μ) inhalation and retention in the lung is probable; between 10 and about 50μ , inhaled particles are transferred to the throat and thence to the gastrointestinal tract; below 0.5 - 1 mm in size, an uncertain probability of ingestion exists with certain foods (fresh leafy vegetables, fruits, and so forth).

The behavior of the fission products contained in these particles has considerable bearing on the resultant biological effects. If individual fission products can be extracted by body fluids with which the particles are contacted, these elements will be translocated to parts of the body in a selective fashion. Thus, for example, extracted radioiodine would constitute a source of exposure for the thyroid; strontium and barium for bone; and so forth. If, on the other hand, the radioactive material is retained in the particle matrix, exposure is largely restricted to the lung for inhaled particles and the GI tract for ingested particles. Some additional exposure to adjacent organs may also occur from insoluble particles, but the major doses are restricted to those indicated above.

The following section describes the models used for estimating exposure and some of the uncertainties involved. Of particular note is the apparent significance of the GI tract exposure, and the relative paucity of information on both dosimetry and biological consequences of given radiation exposures from single inhaled or ingested particles.

III. PHYSICAL DOSIMETRY

In situations as nebulous as those considered here, the models employed for dose estimation will always be subject to question. External dose estimates involve a knowledge of the source characteristics (decay rate and energy spectrum), distance (and intervening attenuation) between source and receptor, and the duration of exposure. All of these are highly uncertain at present, and the latter two may never be resolved. Internal dose estimates require knowledge of the pattern of behavior of particulates in the body, and this knowledge is not available at present. Nevertheless, some models are necessary if a beginning is to be made on estimation of doses from sources both external and internal to the body.

A. EXTERNAL DOSE

The model for external dose requires consideration of both dose rate and total dose delivered in order to properly evaluate the over-all effect. A dose delivered in a few hours will have a significantly greater effect than the same dose delivered over months to years. Since reactor fragments may have widely variant decay periods prior to their return to the earth's surface, the model employed must recognize the dose protraction period and its effect.

For example, for those cases in which delay times are short prior to return, initial decay rates will be rapid, and the fragment will deliver a substantial portion of its decay energy within the first day following return to earth. For those fragments of the core which have a long residence time in space prior to return to the earth, the resulting rate of decay on return will be much more gradual. To handle the widely variant dose delivery effects and obtain a consistent biological effect is somewhat difficult.

1. Gamma Dose

Report No. 29 of the NCRP ⁽²⁾ suggests the concept of predicting the biological consequences of exposure to radiation during a national emergency by correcting the accumulated whole body dose by biological recovery that will occur. It is presumed that this equivalent residual dose (ERD) will exhibit the same symptoms of radiation injury as a brief dose of the same size. This concept has been applied to determining limiting activities of radioactive

debris produced by the destruction of a nuclear rocket stage. The ERD concept assumes that:

- a. Ten per cent of the injury is not reparable.
- b. The remaining injury is repaired at 2.5% per day.
- c. Recovery does not start until four days after the start of exposure and is continuous thereafter.

Thus, the ERD dose may be expressed analytically as follows:

$$\text{ERD} = D_0 \left[0.1 + 0.9 (0.975)^{t-4} \right] + \dot{D} \int_4^t \left[0.01 + 0.9 (0.975)^t \right] dt$$

where:

D_0 = dose for the first four days, rads

$\dot{D}$ = dose rate after the fourth day, rads/day

t = exposure time, days

The assumptions used to calculate the limiting activity for a given dose are as follows:

- a. The receptor is exposed to a single particle indefinitely.
- b. The receptor is one meter from the particle.
- c. The gamma activity after reactor shutdown follows the Way-Wigner relationship.
- d. There is, on the average, one γ emission per two nuclear disintegrations.
- e. The average gamma energy is 0.7 Mev.

Consider an isotropic point source, where R_γ is defined as the γ emission rate expressed in β curies. Then the dose in the first four days is given by:

$$D_o = G \int_0^4 R_Y dt$$

From the Way-Wigner relationship

$$R_Y = \alpha (t + \tau)^{-1.2}$$

Where t is the time from particle impact on the ground and, τ , the time from reactor shutdown to the ground impact.

Then

$$D_o = G \int_0^4 \alpha (t + \tau)^{-1.2} dt = -5 G \alpha \left[(\tau + 4)^{-0.2} - \tau^{-0.2} \right]$$

At $t = 0$

$$\alpha = \frac{R_Y(\tau)}{\tau^{-1.2}}$$

Where $R_Y(\tau)$ is the activity at the time the particle reaches the ground.

Then:

$$D_o = -5 G R_Y(\tau) \left[\frac{(\tau + 4)^{-0.2} - \tau^{-0.2}}{\tau^{-1.2}} \right]$$

At 1 meter from a source

$$G = \frac{3.7 \times 10^{10} \text{ dis}}{\text{curie-sec}} \cdot \frac{1 \text{ } \gamma \text{ photon}}{2 \text{ dis}} \cdot \frac{1.4 \times 10^{-6} \text{ rad-cm}^2\text{-sec}}{\text{hr-} \gamma \text{ photon}} \cdot \frac{1}{4\pi (10^2)^2 \text{ cm}^2}$$

$$G = 0.206 \frac{\text{rad}}{\text{hr-curie}} \quad \text{or} \quad 4.95 \frac{\text{rad}}{\text{day-curie}}$$

Thus:

$$D_o = -24.7 R_Y(\tau) \left[\frac{(\tau + 4)^{-0.2} - \tau^{-0.2}}{\tau^{-1.2}} \right]$$

The dose rate may be expressed as:

$$\dot{D} = G R_Y = G \alpha (t + \tau)^{-1.2}$$

But when $t = 0$

$$\alpha = \frac{R_Y(\tau)}{\tau^{-1.2}}$$

Then

$$\dot{D} = 4.95 R_Y(\tau) \left[\frac{t + \tau}{\tau} \right]^{-1.2}$$

Substituting the expressions for D_0 and $\dot{D}$ into the expression for ERD dose

$$\begin{aligned} \text{ERD} = & -24.7 R_Y(\tau) \left[\frac{(\tau + 4)^{-0.2} - \tau^{-0.2}}{\tau^{-1.2}} \right] \left[0.1 + 0.9 (0.975)^{t-4} \right] + \\ & 4.95 R_Y(\tau) \int_4^t \left[\frac{t + \tau}{\tau} \right]^{-1.2} \left[0.1 + 0.9 (0.975)^t \right] dt \end{aligned}$$

Now, for an infinite exposure ($t = \infty$)

$$\begin{aligned} \text{ERD} = & -2.47 R_Y(\tau) \left[\frac{(\tau + 4)^{-0.2} - \tau^{-0.2}}{\tau^{-1.2}} \right] + 4.95 \frac{R_Y(\tau)}{\tau^{-1.2}} \cdot \\ & \left[0.5 (\tau + 4)^{-0.2} + 0.9 \int_4^{\infty} (t + \tau)^{-1.2} (0.975)^t dt \right] \end{aligned}$$

The resulting expression can be solved by graphical methods for various values of τ .

If the expression is solved for very small values of τ , the resulting activity will be excessive, i.e. lethal doses can be obtained before repair begins. To avoid this limitation an additional criterion for the accumulated dose in the first four days must be superimposed. The criterion established requires the dose accumulated in the first four days not to exceed the ERD accumulated over an infinite

duration of exposure.

Values for the limiting initial activity, i.e, the activity of the particle at the time it reaches the ground, for gamma doses of 1 rad (ERD) at one meter are shown in Figure 1 as a function of the time to reach the ground.

2. Beta Dose

The method used to calculate beta doses is that presented in Hine and Brownell (3) and is based on the observation that out to a distance equal to one relaxation length, the dose from a point source obeys an inverse square relationship. Beyond this distance, the dose is exponentially attenuated. This can be expressed as a two part function where ν is the absorption coefficient in cm^2/gm , ρ is the density in gm/cm^3 , x is the absorber thickness, and $J(x)$ the point source dose function:

$$J(x) = \frac{k}{(\nu\rho x)^2} \quad x < \frac{1}{\nu\rho}$$

$$J(x) = \frac{k}{\nu\rho x} e^{1-\nu\rho x} \quad x \geq \frac{1}{\nu\rho}$$

Combining the two relationships yields:

$$J(x) = \frac{k}{(\nu\rho x)^2} \left[1 - \nu\rho x e^{1-\nu\rho x} \right] + \nu\rho x e^{1-\nu\rho x}$$

$$\text{Where} \quad \left[1 - \nu\rho x e^{1-\nu\rho x} \right] \equiv 0 \quad \text{at } x \geq \frac{1}{\nu\rho}$$

Experimental measurements have shown the dose function can be well represented over a wide energy range by incorporating an energy dependent constant as follows:

FIG. 1

ALLOWABLE INITIAL ACTIVITY ON GROUND/RAD

$$J(x) = \frac{k}{(\nu \rho x)^2} \left\{ c \left[1 - \frac{\nu \rho x}{c} e^{1 - \frac{\nu \rho x}{c}} \right] + \nu \rho x e^{1 - \nu \rho x} \right\}$$

However, for high energy β particles, i.e., above 1.5 Mev maximum energy, c is approximately unity.

The constant, k , is selected such that the total integrated energy absorption in a very large sphere is equal to the average beta energy. Thus,

$$k = 1.28 \times 10^{-9} \rho^2 \nu^3 \bar{E} \alpha \quad \text{rad/disintegration}$$

$$\alpha = \left[3c^2 - (c^2 - 1)e \right]^{-1}$$

The absorption coefficient in tissue has been found to be:

$$\nu = \frac{18.6}{(E_0 - 0.036)^{1.37}} \left(2 - \frac{\bar{E}_\beta}{E_\beta^*} \right) \text{ cm}^2/\text{gm}$$

Where:

E_0 is the maximum beta energy in Mev

$\bar{E}_\beta$ is the average beta energy

E_β^* is the hypothetical average beta energy for a hypothetical allowed spectrum.

The quantity $\left(2 - \frac{\bar{E}_\beta}{E_\beta^*} \right)$ is unity for allowed spectra and is assumed to be unity for fission product mixtures.

The self absorption is estimated from Figures 23 and 25 of reference (3). These figures give 1) the ratio of the dose at the center of a spherical source to the dose inside a large distribution with the same radioactivity concentration and 2) the ratio of the average dose in a spherical source to the dose at the center. Multiplying these two values gives the fraction of the infinite dose absorbed within the spherical particle itself.

Table I shows the fraction of beta activity escaping various size particles with a density of 2 gm/cm³.

TABLE I
FRACTION OF BETA ACTIVITY ESCAPING PARTICLE

<u>Size Particle (μ)</u>	<u>Fission-Product Mixture Age</u>		
	<u>≤ 1 hr</u>	<u>10 hr</u>	<u>≥ 100 hr</u>
10000	0.32	0.19	0.06
6000	0.48	0.31	0.11
2000	0.79	0.63	0.31
1000	0.89	0.81	0.53
600	0.93	0.89	0.67
200	1.00	0.96	0.89
100 and smaller	1.00	1.00	1.00

The change in self absorption of beta activity is due to the changing beta energy of the fission product mixture with time. (4)

Table II shows the results obtained by solving the point source dose function, $J(x)$, and multiplying by the fraction of activity escaping the particle.

TABLE II
BETA DOSE RATES

<u>Size Particle (μ)</u>	<u>Beta dose rate at 1 meter $\left[\frac{\text{rad}}{\text{hr-curie}} \right]$</u>		
	<u>≤ 1 hr</u>	<u>10 hr</u>	<u>≥ 100 hr</u>
10000	7.40	5.35	0.33
6000	11.4	8.75	0.60
2000	18.7	17.8	1.68
1000	21.1	22.8	2.88
600	22.0	25.0	3.64
200	23.7	27.1	4.83
100 and smaller	23.7	28.2	5.42

The comparable gamma dose rate from particles is not significantly affected by particle size or change in energy with fission product age; as calculated previously, it is

$$0.206 \frac{\text{rad}}{\text{curie-hr}} \quad \text{at 1 meter.}$$

The beta dose rate varies from approximately 1.6 to 140 times the gamma dose rate depending upon the size of the particle and the time for the particle to reach the ground while the total beta dose will vary from 1.6 to 43 times the gamma dose.

B. INTERNAL DOSE

1. General

The established limits for internal exposure to radioactive material are based on the assumption that the material (or at least its radiation absorption) is uniformly distributed throughout the organ of reference. For specific cases, e.g. those concerning bone-seekers, in which biological data indicate a heterogeneous distribution in the organ, compensation is made by the use of a "relative damage factor," (n), or "distribution factor," (DF). But the particle problem differs from these cases principally because the radiation levels to which the surrounding tissues are exposed depends very strongly on the distance from the particle.

For radiation having a finite range in tissue (e.g. beta particles), the dose rate over a very short distance varies non-linearly from very high levels at or near contact with the particle to a zero value at a distance about equal to the range of the particles, as shown in Table 3. Since most of the energy released from a beta-emitting particle embedded in tissue is absorbed within distances which are a fraction of the maximum range of the particles, a major problem involves selecting the volume of tissue over which the dose (energy absorbed per unit mass) is to be averaged. As indicated in Table 4, the dose value obtained when this is a microvolume immediately surrounding the particle is drastically different from that obtained when the volume is taken to be that of the entire organ. It is of fundamental importance that the "type" of dose reported is the one which is biologically significant and can be correlated with biological effects which are observed when the entire volume of the organ (or organism) is exposed uniformly.

Results from some experiments indicate that it may be valid to average the dose over the entire organ; results from other experiments indicate a contradictory conclusion.⁽⁵⁾ Present data is not conclusive enough to be able to take a definitive stand at this time. In this evaluation of internal beta-emitters, a more or less middle stand is taken between the contact dose and the dose which would be produced by assuming uniform distribution of the radioactive material in the entire organ of reference.

duration of exposure.

Values for the limiting initial activity, i.e, the activity of the particle at the time it reaches the ground, for gamma doses of 1 rad (ERD) at one meter are shown in Figure 1 as a function of the time to reach the ground.

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$$J(x) = \frac{k}{(\nu\rho x)^2} \quad x < \frac{1}{\nu\rho}$$

$$J(x) = \frac{k}{\nu\rho x} e^{1-\nu\rho x} \quad x \geq \frac{1}{\nu\rho}$$

Combining the two relationships yields:

$$J(x) = \frac{k}{(\nu\rho x)^2} \left[1 - \nu\rho x e^{1-\nu\rho x} \right] + \nu\rho x e^{1-\nu\rho x}$$

$$\text{Where} \quad \left[1 - \nu\rho x e^{1-\nu\rho x} \right] \equiv 0 \quad \text{at } x \geq \frac{1}{\nu\rho}$$

Experimental measurements have shown the dose function can be well represented over a wide energy range by incorporating an energy dependent constant as follows:

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The constant, k , is selected such that the total integrated energy absorption in a very large sphere is equal to the average beta energy. Thus,

$$k = 1.28 \times 10^{-9} \rho^2 \nu^3 \bar{E} \alpha \quad \text{rad/disintegration}$$

$$\alpha = \left[3 c^2 - (c^2 - 1) e \right]^{-1}$$

The absorption coefficient in tissue has been found to be:

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Where:

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100 and smaller	1.00	1.00	1.00

The change in self absorption of beta activity is due to the changing beta energy of the fission product mixture with time. (4)

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200	23.7	27.1	4.83
100 and smaller	23.7	28.2	5.42

The comparable gamma dose rate from particles is not significantly affected by particle size or change in energy with fission product age; as calculated previously, it is

$$0.206 \frac{\text{rad}}{\text{curie-hr}} \quad \text{at 1 meter.}$$

The beta dose rate varies from approximately 1.6 to 140 times the gamma dose rate depending upon the size of the particle and the time for the particle to reach the ground while the total beta dose will vary from 1.6 to 43 times the gamma dose.

B. INTERNAL DOSE

1. General

The established limits for internal exposure to radioactive material are based on the assumption that the material (or at least its radiation absorption) is uniformly distributed throughout the organ of reference. For specific cases, e.g. those concerning bone-seekers, in which biological data indicate a heterogeneous distribution in the organ, compensation is made by the use of a "relative damage factor," (n), or "distribution factor," (DF). But the particle problem differs from these cases principally because the radiation levels to which the surrounding tissues are exposed depends very strongly on the distance from the particle.

For radiation having a finite range in tissue (e.g. beta particles), the dose rate over a very short distance varies non-linearly from very high levels at or near contact with the particle to a zero value at a distance about equal to the range of the particles, as shown in Table 3. Since most of the energy released from a beta-emitting particle embedded in tissue is absorbed within distances which are a fraction of the maximum range of the particles, a major problem involves selecting the volume of tissue over which the dose (energy absorbed per unit mass) is to be averaged. As indicated in Table 4, the dose value obtained when this is a microvolume immediately surrounding the particle is drastically different from that obtained when the volume is taken to be that of the entire organ. It is of fundamental importance that the "type" of dose reported is the one which is biologically significant and can be correlated with biological effects which are observed when the entire volume of the organ (or organism) is exposed uniformly.

Results from some experiments indicate that it may be valid to average the dose over the entire organ; results from other experiments indicate a contradictory conclusion.⁽⁵⁾ Present data is not conclusive enough to be able to take a definitive stand at this time. In this evaluation of internal beta-emitters, a more or less middle stand is taken between the contact dose and the dose which would be produced by assuming uniform distribution of the radioactive material in the entire organ of reference.

TABLE III
DOSE RATE IN TISSUE AROUND 1 - μ c PARTICLE⁽⁵⁾

Distance (μ m)	Dose Rate (Rads/Hour)			
	C-14	Sr-90(a)	P-32	Y-90
10	2,000,000	530,000	380,000	270,000
100	1,500	5,000	3,700	2,700
200	40	1,110	930	680
400	0.03	200	230	160
600		60	100	70
1,000		10	30	26
2,000		0.03	7	7
4,000			0.7	1
6,000			0.1	0.3
8,000			0.01	-
10,000				0.02

(a) Anomalous Spectral Distribution; does not include Y-90 daughter.

Note: The calculated beta dose rate at a distance of 10 μ m from a one-microcurie FP point source embedded in tissue is 621,000 rads/hour.

TABLE IV
AVERAGE DOSE RATE FOR 1 - μ c PARTICLE⁽⁵⁾

Radius of Sphere (cm)	Volume of Tissue (cm ³)	Average Dose Rate (Rads/hr/ μ c)			
		C-14	Sr-90	P-32	Y-90
0.01	4.2x10 ⁻⁶	2.1x10 ⁴	1.5x10 ⁴	9.4x10 ³	9.0x10 ³
0.1	0.0042	25	87	110	79
0.62	1.0	0.096	0.043	1.4	1.7
1.0	4.2	0.023	0.10	0.36	0.48

As such, unless it is specifically indicated otherwise, dose values for this study are those obtained by averaging the absorbed energy over a spherical volume of tissue of radius equal to the range of the most energetic beta particles.

Before proceeding with the development of the model used for evaluation of the internal particle problem, it may be of interest to review briefly, recent changes in terminology and definitions applied to radiation dosimetry and particularly applicable to the internal dose problem.

The International Commission on Radiological Protection (ICRP) and the International Commission on Radiological Units and Measurements (ICRU) have agreed upon the following formulation.^(6,7)

- a. For protection purposes it is useful to define a quantity which will be termed the "dose equivalent," DE.
- b. (DE) is defined as the product of absorbed dose, D, Quality Factor, (QF), Dose Distribution Factor, DF, and other necessary modifying factors.
$$(DE) = D(QF) (DF)$$
- c. The unit of dose equivalent is the "rem." The dose equivalent is numerically equal to the dose in rads multiplied by the appropriate modifying factors.

Although this statement does not cover a number of theoretical aspects (in particular the physical dimensions of some of the quantities) it fulfills the immediate requirement for an unequivocal specification of a scale that may be used for numerical expression in radiation protection.

It was recommended that the term "RBE" (Relative Biological Effectiveness) be used in radiobiology only and that "Quality Factor," (QF), be the name used for the linear-energy-transfer-dependent factor by which absorbed doses are to be multiplied to obtain for purposes of radiation protection a quantity that expresses on a common scale for all ionizing radiations the irradiation incurred by exposed persons. The term "Distribution Factor," DF, is to be used to express the modification of biological effect due to non-uniform distribution of internally deposited isotopes.

Following normal practice, the term QF and the term RBE, used without further qualification, imply the biological effectiveness of high LET relative to low LET radiation. (LET refers to linear energy transfer of charged particles in a medium and is the quotient of dE_L by dl , where dE_L is the average energy locally imparted to the medium by a charged particle of specified energy in traversing a distance of dl). Many factors, apart from LET, affect the RBE of two radiations:

- a. the nature of the biological effect considered;
- b. the nature and condition of the biological material (including physiological state, temperature, oxygen, concentration, and so forth);
- c. the magnitude of the biological effect and thus the dose level;
- d. the distribution of dose in time (dose-rate, fractionation, and so forth);
- e. the dose distribution within the irradiated material. (By "dose distribution" is meant the distribution of absorbed dose on a scale large enough for the statistical variation of energy increments to be negligible. This "macroscopic" dose distribution is to be distinguished from the microscopic distribution of energy absorption events.

Since, for emergency exposures, both dose and dose-rate will be considerably higher than in normal protection work, it is possible that RBE values different from the QF values should be used, which would introduce various problems related to the individual's exposure record, including that of the meaning of the "rem." However, in the absence of relevant mammalian data it is reasonable to assume that single doses of low LET radiation of 25r or less may be treated as equal in effect to low dose rate irradiation. It is therefore recommended that when "accidental high exposures" or "emergency exposures" of 25 rems or less have to be taken into account in an individual's exposure status, such exposures be evaluated in the same manner as in normal protection work. Concerning exposures at high doses and high dose-rates which may be encountered in radiation accidents of various sorts, any estimate of radiation dose made under high dose and dose-rate conditions must take account of the fact that the accepted QF values will not apply, even if the body were uniformly irradiated, which it normally is

not; the dose distribution is generally extremely non-uniform and so introduces further serious complications.⁽⁷⁾

The RBE Committee has made the following statement:⁽⁷⁾

"Another example, where mean dose is almost certainly not the correct parameter to take is that of radioactive particles in the lung. Unfortunately, there are as yet no experimental data to indicate the appropriate parameter. Also the dose and dose-rate near the particles will almost certainly exceed those appropriate to QF. An estimate of the dose equivalent merely by the product of QF and mean dose may be greatly in error, but further experimental advance is needed before a better estimate can be made."

This same comment is applicable to the particle problem in general; biologically meaningful dosimetry for radioactive point sources has not as yet been established. As was indicated above, there are various ways in which the dose delivered by point sources embedded in tissue may be described physically. Three such "types" of dose are:

- a. Contact Dose-- computed from the calculated surface dose-rate of the source. Alternately, for a very small source treated as being dimensionless, the equivalent would be the dose at a specified distance from the point source. Such a distance would be arbitrarily selected, or else determined by the known (or assumed) distance from the source to the target cells of interest.
- b. Range Average Dose--computed from the average energy absorbed per gram of tissue in a sphere of radius equal to the maximum range of the beta particles in tissue. For sources of dimensions large enough so that they are treated as discrete rather than point sources, the tissue volume of interest is that of a concentric spherical shell of a radial thickness equal to the maximum range of the beta particles.
- c. Organ Average Dose--computed from the average energy absorbed per gram of tissue in the entire organ concerned. For point sources, this is the same as assuming the radioactive material is uniformly distributed throughout the organ, or that there is uniform absorption of energy throughout the organ.

These concepts for particle dosimetry have been used by experimenters;⁽⁸⁾ however, the ICRP and ICRU have not designated these nor any equivalent concepts for particle dosimetry. The conceptual difficulties have been recognized, and it is expected that, as

soon as data considered sufficient is available, the ICRP and the ICRU will designate means for biologically meaningful dosimetry for radioactive point sources and will establish applicable limiting dose values (equivalent to the currently used maximum permissible dose) for the various types of exposures involving particles. Until such action comes about, the particle problem must be handled with caution in order to avoid conclusions which may be overly conservative or overly non-conservative. Obviously, either extreme is to be avoided.

There are many factors which must be taken into account in the complete evaluation of the biological hazard presented by the inhalation or ingestion of a single radioactive particle. It appears, however, that primary consideration must be given to the extreme variation of dose-rate (and dose) over relatively short distances from the particle. This fact, perhaps more than any of the many others, gives rise to the difficulty in formulating biologically significant concepts of dosimetry for the particle problem. As indicated in the RBE Committee statement quoted above, the dose and dose-rate near the particle will almost certainly exceed those appropriate to QF (Quality Factor); and estimation of the biologically significant dose as merely the product of QF and mean absorbed dose may be greatly in error. Because of the unique characteristics of the particle problem, it is not at all improbable that very different dosimetric concepts may have to be devised for this problem. Such concepts undoubtedly would have to take into account such factors as: type of radiation (alpha, beta, etc.); range of the particles in the tissue of interest; magnitude of the dose and dose-rate and their variation throughout the particle range--specifically, the dose and dose-rate at the target cells of interest; volume of tissue involved, etc.; it is conceivable that new dosimetric concepts for the particle problem may be stated in terms of range-average dose (in rads) multiplied by an appropriate modifying factor, P , which would be a combined function of QF and DF. For incremental ranges of particle energy, the qualifying factor P would probably be given numerical values applicable to incremental ranges of total activity (A , μc) contained in the particle. This would supposedly take care of such factors as dose, dose-rate, and tissue volume involved. Another qualifying factor, which may be designated as S , might also be used to define the effective shielding afforded by radiation-insensitive material intervening between the particle and the target cells of interest. In view of the fact that most of the dose resulting from a particle is delivered within a distance which is a small fraction

of the particle range, the possibility of there being such effective shielding is exceedingly important. Still another qualifying factor, which may be designated as M, could take into account motion of the particle in a particular organ due to physiological characteristics. There could be other modifying factors. As such, the biologically significant dose (which could be expressed in rem units) would then be given as:

$$\text{Dose Equivalent (DE)} = (\text{RAD}) \times P \times S \times M \times$$

At present, however, none of the modifying factors are known; so that numerically one can specify the dose only in terms of the range-average dose (RAD). All pertinent information known and assumed must be specified with the numerical value of the range-average dose in order for the information to be meaningful. At a time when dosimetric concepts and limits applicable to the particle problem have been established, it may be possible to re-evaluate such information and convert the data to biologically significant numerical values.

2. Range Average Dose Models

There are two basic models which may be used to express range-average dose in the particle case:

a. Stationary

The particle is assumed to remain in a fixed position during its entire retention time in the organ concerned.

b. Mobile

The particle is considered to move within the organ in accordance with known, or assumed, physiological characteristics.

The stationary model was the one used in this report. Reference to this has been made by the assumption that the particle remains "in situ" during its retention time in the organ concerned.

In general, if the particle is considered to be in motion while in a given organ, the resultant range-average dose (in rads) will be proportionately and considerably smaller than that resulting from the particle remaining stationary. This is due to the much larger volume of tissue involved in the "mobile" model. Comparison of the biological significance of the two doses simply on the basis of the numerical values of the range-average doses (in rads) would not

be a valid one. Intuitively, and assuming that all the cells concerned are equally radiosensitive, it seems that a comparison of the two models will have to be based on the number of cells which are killed and, perhaps more importantly, on the number of surviving, dosed cells and the absorbed dose delivered to these cells. Depending on the activity as well as rate of movement of the particle in the "mobile" model, it is possible that the contact, rather than the range-average, dose may be the more logical "type" of dose to use for evaluation of the resultant exposure.

The mobile model has been used by another investigator in a theoretical study⁽⁹⁾ of the significance of the particle problem in the inhalation case. Under certain conditions, however, radiation damage to cells (such as the ciliary cells in the respiratory system) which affect the motion of the particle may be such as to preclude motion of the particle. This very situation has been inferred from the results of biological experiments⁽¹⁰⁾ in which, contrary to expectation, large-size inhaled particles were retained in the lungs of exposed animals.

With the above discussion in mind, the following assumptions were used to evaluate internal doses from inhaled and ingested insoluble particles:

- a. The fission product (FP) activity is contained and uniformly distributed in a particle of material whose density is 2.0 grams/cm³.
- b. The characteristics of the fission product activity are:
 - (1) One beta particle and 0.5 gamma photon are emitted per disintegration.
 - (2) Energy of gammas = 0.7 Mev (average).
 - (3) Energy of betas = 0.8 Mev (maximum); 0.3 Mev (average).
 - (4) Simple, allowed beta spectrum.
- c. The fission product particle is spherical.
- d. Any one individual is exposed to only one fission product particle.

According to reference⁽³⁾, the ratio of the beta dose from a very small spherical source to the beta dose from a point source is given by

$$\frac{D}{J(x)} = 1 + 0.1 (\mu b)^2 \quad b \leq 1/\mu \quad (1)$$

where b is the radius of the spherical source; μ is the apparent absorption coefficient; and the product μb is dimensionless.

Equation (1) may be used as follows to determine the maximum radius of a discrete spherical source which may be treated as a (hypothetical) point source to within one percent accuracy.

$$0.1 (\mu b)^2 = 0.01 \quad (2)$$

The value of μ for the FP betas was determined by using the following equation from reference (3):

$$\mu = \frac{18.6}{(E_0 - 0.036)^{1.37}} \left(\frac{2 - \bar{E}}{\bar{E}^*} \right) \text{ cm}^2/\text{gm of tissue} \quad (3)$$

where E_0 is the maximum energy in Mev, $\bar{E}$ is the actual average beta energy per disintegration in Mev; and $\bar{E}^*$ is the hypothetical average beta energy per disintegration for a hypothetical allowed spectrum with the same value of E_0 . Substituting for $E_0 = 0.8$ Mev and for $\bar{E} / \bar{E}^* = 1$,

$$\mu = 27 \text{ cm}^2/\text{gm}$$

or, for the particle material of $2 \text{ gm}/\text{cm}^3$ density,

$$\mu_m = 54 \text{ cm}^{-1}$$

Substituting this value in equation (2), gives

$$b = 58 \times 10^{-4} \text{ cm} = 58 \mu\text{m}$$

On this basis, particles of radius, b , equal to or less than $50 \mu\text{m}$ (diameter = $100 \mu\text{m}$) are treated in this study as point sources, thereby simplifying the calculations considerably.

Since the absorption and scattering of electrons does not change greatly with small changes in atomic number, ranges in aluminum (in gm/cm^2) apply with sufficient accuracy, for purposes of dosimetry, to tissue, water, plastics, and other materials of low atomic number. From reference (3)

$$\nu R_0 = 8.25 \quad (4)$$

where R_0 is the maximum beta range and the product νR_0 is dimensionless. For betas of maximum energy of 0.8 Mev in tissue (density = 1 gm/cm³), $\nu = 27 \text{ cm}^2/\text{gm} \times 1 \text{ gm/cm}^3 = 27 \text{ cm}^{-1}$. Substituting this value in equation (4), the value of the maximum range of the FP betas in tissue is

$$R_0 = 0.305 \text{ cm} = 3.05 \text{ mm of tissue}$$

Because of the shape of a beta spectrum, the scattering of the betas, and the inverse square attenuation, the dose from a localized source becomes very small at moderate fractions of the maximum beta range in relation to the dose near the source. It is of interest to indicate a measure of the distance from such a source in tissue, over which a biologically significant dose is distributed. According to reference (3), the distances, X_p , in tissue, within which P% of the dose from a point source is distributed, are given by:

$$X_{50} = 1.3/\nu = 0.48 \text{ mm}$$

$$X_{60} = 1.7/\nu = 0.63 \text{ mm}$$

$$X_{90} = 3.6/\nu = 1.33 \text{ mm}$$

$$X_{95} = 4.5/\nu = 1.67 \text{ mm}$$

The numerical values for X_p were obtained by substituting the value of ν ($= 27 \text{ cm}^{-1}$) for the FP betas in tissue. It is to be noted that these distances are significantly shorter than the maximum range, $R_0 = 3.05 \text{ mm}$. It is of interest that the value of X_{60} is the average distance of dose distribution from a beta point source of maximum energy between 0.5 and 1.5 Mev.

The distance within which is distributed a given fraction of the dose outside the source is in all cases smaller for sources of finite size than for a point source. If all dimensions of a small beta source are less than $1/\nu$, the distances of dose distribution outside the source are about the same as for a point source. Since $1/\nu$ for the FP particle is $1/54 \text{ cm} = 0.0185 \text{ cm} = 185 \mu\text{m}$, the values of X_p indicated above for a point source may be considered to hold for particles of diameter equal to or less than $100 \mu\text{m}$. Because of the finite range of betas, the actual distribution distances for a given per cent of the dose from a point source are less than X_p . However, values for X_p/R_0 which are

less than 0.5 are reasonably accurate. The larger values of the ratio are too large, by an amount which cannot at present be specified. ⁽³⁾

The range average dose resulting from a point source embedded in tissue for T days is given by the following equation:

$$D(\text{rads}) = A(\mu\text{c}) \times (3.7 \times 10^4) (\beta/\text{sec}-\mu\text{c}) \times \bar{E} (\text{Mev}/\beta) \times (86,400) \quad (5)$$

$$(\text{sec/day}) \times \frac{(1.6 \times 10^{-6}) (\text{ergs}/\text{Mev}) \times T (\text{days})}{(4/3) (\pi R_0^3) (\text{cm}^3) \times (1) (\text{gm}/\text{cm}^3) \times (100) (\text{erg}/\text{gm-rad})}$$

where $A(\mu\text{c})$ is the total activity contained in the point source, $\bar{E}$ is the average beta energy per disintegration, R_0 is the maximum beta range in tissue, and T is the retention time of the particle in situ. Only the beta radiation is considered in computing the range-average dose from the FP activity, since the amount of energy absorbed from the gamma radiation in such a small tissue volume will not be significant. In equation (5), no correction is made for decay of the activity, A, as it is assumed that the in-situ time, T, is small compared to the radioactive half-life of the FP activity.

Simplifying and substituting the values of $\bar{E}$ (=0.3 Mev) and R_0 (=0.305 cm) for the FP activity concerned, equation (5) becomes

$$D (\text{rads}) = 129 \times A (\mu\text{c}) \times T (\text{days})$$

or,

$$A (\mu\text{c}) = \frac{(7.75 \times 10^{-3}) \times D (\text{rads})}{T (\text{days})} \quad (6)$$

where $A(\mu\text{c})$ is the total FP activity in a point source which will produce a range average dose of D rads during a period of T days in situ.

Equation (6) may be used to calculate the FP activity in a point source which will produce a limiting, range average dose of D rads. For the FP particles concerned, it was stated above that spherical particles of diameters equal to or less than 100 μm could be treated as point sources. Hence, equation (6) may also be used for these particular, small particles. Likewise, equation (6) rewritten as indicated below in terms of A', activity

per unit volume ($\mu\text{C}/\text{cm}^3$) and A'' , activity per unit mass ($\mu\text{C}/\text{gm}$) is applicable to these small spherical particles. The density of the particle material is taken to be $2 \text{ grams}/\text{cm}^3$; and b is the radius of the particle.

$$A' (\mu\text{C}/\text{cm}^3) = \frac{A (\mu\text{C})}{(4/3) \pi b^3 (\text{cm})^3}$$

$$A' (\mu\text{C}/\text{cm}^3) = \frac{(1.85 \times 10^{-3}) \times D (\text{rads})}{b^3 (\text{cm}^3) \times T (\text{days})} \quad (7)$$

and,

$$A'' (\mu\text{C}/\text{gm}) = \frac{A' (\mu\text{C}/\text{cm}^3)}{2 (\text{gm}/\text{cm}^3)}$$

$$A'' (\mu\text{C}/\text{gm}) = \frac{(0.92 \times 10^{-3}) \times D (\text{rads})}{b^3 (\text{cm}^3) \times T (\text{days})} \quad (8)$$

For a given value of D and T , according to equations 6, 7, and 8, the value of $A (\mu\text{C})$ is independent of the radius, b , of the particles, while both $A' (\mu\text{C}/\text{cm}^3)$ and $A'' (\mu\text{C}/\text{gm})$ are inversely proportional to the cube of the particle radius, b . The sensitivity of A' and A'' to changes in the particle radius (or diameter) is readily apparent.

3. Lung Dose Model

a. Specific Assumptions

- (1) The maximum diameter of a respirable particle of material of density equal to $2 \text{ gm}/\text{cm}^3$ is $10 \mu\text{m}$.
- (2) The particle is insoluble in lung fluids.
- (3) The particle is retained in situ in the lung during the specified retention times, T .
- (4) The retention time, T , is short compared to the radioactive half-life of the FP activity.
- (5) The specified dose, D , refers to the range average dose.

b. Calculations

Because the diameter of the largest particle of concern in this lung case is $10 \mu\text{m}$ the particles may be treated as point

sources. Equations 6, 7, and 8 may then be used to compute the activity, A (μc), and the concentrations, A' ($\mu\text{c}/\text{cm}^3$) and A'' ($\mu\text{c}/\text{gm}$) for a given range-average dose, D , and retention time, T . For a given set of values for D and T , the value of A is the same for all particle sizes concerned; but the limiting (i.e. minimum) concentration is determined by the particle having the largest diameter, $10\ \mu\text{m}$ in this case.

A retention time of 4 days is assumed for the larger particles, based on the standard man model recommended by the International Commission on Radiological Protection⁽¹¹⁾, that half of the 25 per cent of the inhaled material which is deposited in the lung (lower respiratory passages) is eliminated from the lung and swallowed in the first 24 hours. A retention time of $120/0.693$ (≈ 173) days is also examined, based on the same ICRP respiratory characteristics model for standard man--specifically, that 12.5 per cent of inhaled insoluble particles are retained in the lungs with a half-life of 120 days. A much longer retention time of $4(365)/0.693$ (≈ 2110) days was also examined, based on experimental data which indicate that retention times of that order may be expected under certain conditions⁽¹²⁾. Such long retention times are more probable for the smaller particles than for the larger ones, which may be trapped in the nasal passages or upper respiratory tract and subsequently swallowed within a short time. For this reason, the concentrations for the longest retention time were computed only for particles of one micron.

Results of the computation using this model are tabulated below:

TABLE V
LIMITING VALUES FOR LUNG DOSE MODEL

Retention time in situ (days)	Particle Diameter (μm)	Total Activity/ Rad, A/D ($\mu\text{c}/$ rad)	Concentration/rad	
			A'/D ($\text{c}/\text{cm}^3\text{-rad}$)	A''/D ($\text{c}/\text{gm-rad}$)
4	10	1.9×10^{-3}	3.7	1.85
173	10	4.5×10^{-5}	0.085	0.042
2,110	1	3.7×10^{-6}	7.0	3.5

For particles of diameter smaller than $10\text{ }\mu\text{m}$ the concentration values, A' and A'' , may be readily obtained by multiplying the values indicated in the above table by the cube of the ratio of the diameters. For example, the concentration, A' or A'' , for a $1\text{-}\mu\text{m}$ particle is 10^3 times the corresponding values for a $10\text{ }\mu\text{m}$ particle.

4. Gastrointestinal Tract Dose Model

a. Specific Assumptions:

- (1) Based on the ICRP gastrointestinal (GI) tract model of the standard man⁽¹¹⁾ food remains in the stomach for 1 hour, in the small intestine for 4 hours, in the upper large intestine for 8 hours, and in the lower large intestine (LLI) for 18 hours.
- (2) The maximum diameter of an ingestable particle is 1mm ($1000\text{ }\mu\text{m}$).
- (3) The particle is insoluble, i.e. no absorption takes place during its passage through the GI tract.
- (4) The total retention time of the particle in the GI tract is small compared to the radioactive half-life of the FP activity.
- (5) The specified limiting dose, D , refers to the range-average dose.
- (6) The particle is in intimate contact with the intestinal wall.
- (7) The particle resides next to the wall for the residence time in the particular segment of the GI tract.

b. Calculations:

For reasons stated above, small FP particles of diameters up to $100\text{ }\mu\text{m}$ may be treated as point sources. Hence, for a specified set of values for D (range average dose) and T (retention time in situ), the activity, A (μC), calculated using equation (6), has the same value for all such point sources; and the limiting concentration values, A' ($\mu\text{C}/\text{cm}^3$) and A'' ($\mu\text{C}/\text{gm}$), may be computed using equations (7) and (8). Since,

according to the assumed model, a particle passing through the GI tract spends more time in the lower large intestine (LLI) than in the other components of the GI tract, the computations have been made taking the lower large intestine (LLI) as the critical organ. Results of these computations are shown below.

TABLE VI
LIMITING VALUES FOR GI TRACT MODEL

Retention Time, T, in situ (days)	Particle Diameter (μm)	Total Activity/ rad, A/D ($\mu\text{c}/$ rad)	Concentration/rad	
			A'/D (curies/ $\text{cm}^3\text{-rad}$)	A''/D (curies/ gm-rad)
18/24	10	10^{-2}	20	10
18/24	100	10^{-2}	0.02	0.01
18/24	1,000 (point)	10^{-2}	2×10^{-5}	10^{-5}
18/24	1,000 (discrete)	3.9×10^{-2}	7.4×10^{-5}	3.7×10^{-5}

The calculations for a 1,000 μm (=1mm) diameter particle were made because of a suggestion that it is possible that particles as large as these may be ingestable. Limiting values for the 1mm particle were computed treating the particle as both a point and a discrete source. Although it is not strictly valid to treat this larger size particle as a point source, the conservative estimates may be used for comparison purposes.

It should be re-emphasized at this point that the model treats the particle as stationary and contiguous to the intestinal wall. As described later, the dimensions of the intestine are sufficiently great, that shielding of the radiation by the intestinal contents can be accomplished by only a small layer of material. Similarly, small motions of the particle or intestinal wall will reduce the energy absorbed per unit volume by a considerable amount. In the absence of definitive experimental data on particle behavior in the intestine, a conservative model has been adopted.

IV. RADIOBIOLOGICAL CONSIDERATIONS

With the models for dose estimation presented in the previous section, it is possible to examine the limitation activity concentrations which would be imposed by the various dose limits for external and internal radiation exposures. Although the establishment of dose guides for this purpose is obviously the function of appropriate government agencies, some values are suggested, and some of the inadequacies and uncertainties in the bases for acceptable dose establishment are presented below.

The basic guide upon which the doses should be oriented is not itself easy to establish: ideally, even in accident cases, it would be desirable to maintain exposures of the general population to values which are consistent with the values employed for normal peacetime nuclear activities. An examination of the program for nuclear rocket propulsion indicates that while such guides may be easily met in normal operations, the stringent requirements imposed by such guides in the event of improbable but credible malfunctions would tend to impose such restrictions on the program that the use of such vehicles would be forever prohibited. Certainly conventional nuclear installations could not meet such restrictions in accident cases, and the imposition of these standards on power reactors would restrict them to the most desolate locations in the world.

It does not seem unreasonable on this basis to relax normal radiation standards when accident cases are considered; the precedent has been set for conventional nuclear reactors. The next point to be considered is the degree of relaxation permitted in the case of the nuclear rocket. For conventional nuclear installations, the dose standards are established with the clear recognition that the occurrence of an accident at the facility will be noted and the effects on the population determined and corrective action imposed. For the case of random reentry of rocket fragments, the possibility of taking effective control action is small; fragments may land in remote, inaccessible areas. The location of such areas may not be precisely defined, and therefore recovery actions, even if feasible from an accessibility point of view, might not be possible because of the lack of knowledge. It must be assumed therefore that protective actions, or recovery of fragments, will be an advantage which can rarely be claimed, and therefore should not be considered in establishing criteria.

A, EXTERNAL RADIATION EXPOSURE

In determining the possible biological damage that could occur as a result of gamma whole body exposure from a fragment, it is instructive to compare possible dose guides with dose values that produce certain recognizable effects. Report No. 29⁽²⁾ of the NCRP presents the following dose-effect relationship:

- up to 1r - Weekly dose that does not cause recognizable sickness
- 200r - Largest single dose that does not cause a disabling sickness in 90% of the cases
- 450r - Median lethal single dose
- 1000r - Accumulated dose that will not cause disabling sickness in majority of cases (provided the ERD dose does not exceed 200r)
- 2500r - Lifetime doses that have been tolerated

In addition to the above citation, which is directed at civil defense emergencies, recommendations have been made for emergency dose limits for radiation workers by NCRP. The group has stated⁽¹³⁾ that, "An accidental or emergency dose of 25 rems to the whole body or a major portion thereof, occurring only once in the lifetime of the person, need not be included in the determination of the radiation exposure status of that person."

The value of 25 rems has been given additional stature, if not legal standing, for non-occupational exposure from a reactor accident in the AEC's Reactor Site Criteria (10 CFR 100). In these criteria a whole body dose of 25 rems to nearby populations is indicated as a "guide" or reference value for evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence, and low risk of public exposure to radiation.

It would appear that external gamma doses equivalent to 25 rems delivered in short periods would provide a reasonable basis on which to evaluate the unlikely occurrence of external exposure to radiation from reentered fragments.

If the 25 rems of gamma radiation is used as a dose guide, the external beta dose delivered concomitantly would be expected to range from 40

to 1075 rads at one meter, as developed in Section III. Injury from external beta radiation is evidenced primarily by injury to the skin. The upper estimate of 1075 rads would be expected to cause some skin injury, perhaps equivalent to mild second degree thermal burns, to an unclothed individual. The overall effect on the individual would depend upon the amount and type of clothing worn, and hence the amount of skin surface so affected under the duration assumptions made. The lower estimate of 40 rads is about 35 per cent greater than the annual whole body skin dose permitted under AEC protection standards (10 CFR 20), and would probably not produce detectable damage to the skin.

On the basis of these external dose considerations, acceptable inventories in fragments on the ground can be obtained by multiplying appropriate ordinal values in Figure I by 25.

B. INTERNAL DOSE CONSIDERATIONS

At present, there are no established limits nor guides concerning internal exposure to discrete particles of radioactive material. There are many influencing factors involved in the evaluation of internal exposure cases. These factors involve physical, chemical, biological, and physiological data, which are presently too poorly known to make a definitive evaluation of the actual hazard of inhaling or ingesting discrete radioactive particles. The discussion that follows is intended to indicate what radiobiological factors are involved and to review relevant work in the field.

In general, the internal radiation hazards from fallout depends on two parameters: availability of the various fission products, and the biological effects produced when these fission products are deposited internally. Evaluation of the biological effects produced by internally deposited fallout can be expressed in terms of limiting pathological processes. At relatively high dose levels these are damage to bone marrow, bone, and to the gastrointestinal tract. At very low levels the limiting process is probably carcinogenic.⁽¹⁴⁾

1. Inhalation

The biological effects of inhaled radioactive particles has not been studied extensively. Most observations to date have been on the induction to tumors in lungs of rodents and most of these studies have been by intratracheal injection of radioactive particles or by implantation of various kinds of radioactive sources. Other effects which may occur in addition to or as a result of pulmonary pathology

are decreased survival times, hematological changes, increased susceptibility to diseases, and changes in pulmonary function.

Particles of 1 - 2 microns have a maximum percentage retention in the lower respiratory tract, and those of greater than 5 microns seldom reach the lower portions of the tract. Particle size is a most important factor in lung deposition. However, the physiology of respiration can influence dust deposition as profoundly as particle size.⁽¹⁵⁾ For truly insoluble materials, during the period soon after deposition the gastrointestinal tract can be expected to receive the greatest radiation dose due to swallowing of particles initially deposited in the nasal and oral passages and those removed from the respiratory tract by ciliary action.

a. NAS - NRC Report

A comprehensive report, including conclusions and recommendations, concerning the biological effects of inhaled radioactive particles has been made by the Subcommittee on Inhalation Hazards, Committee on Pathologic Effects of Atomic Radiation, National Academy of Sciences - National Research Council.⁽⁵⁾ Following are selected statements contained in this report:

(1) No definitive evaluation of the actual hazard of inhaling radioactive particles can be made because the many influencing factors are too poorly known.

(2) Lung exposures are often expressed as "mean doses" to the lung, a figure arrived at by calculating the exposure assuming uniform distribution of radioactive material throughout the lung. Uniform distribution of inhaled particles is not observed in practice. Since local concentration of particles results in non-uniform distribution of energy absorption, the dose delivered to limited volumes of lung tissue may vary by several orders of magnitude above and below the mean values. Thus the calculated mean dose to the lung should be used with caution in estimating biological effects.

(3) Present data indicate that the lung and respiratory tract do not differ markedly in radiosensitivity from other soft tissues.

(4) The evidence for induction of cancer by inhaled radioactive materials in experimental animals is convincing. There is no reason to think this cannot occur in man despite the lack of definitive evidence at present. Therefore, continued study of inhalation hazards is urgent, and the continuation of stringent environmental control measures is justified pending the completion of adequate studies.

(5) The influence of dose rate on lung tumor production and the damage leading to the production of tumors is not presently known, and the influence of dose rate as compared to total dose cannot be assessed. From the present data, it is not possible to state whether there is or is not a threshold dose for the induction of lung cancer in animals.

(6) Translocation of the radionuclides must be taken into account. Since most of the particulates that reach the pharynx by ciliary action are swallowed, it is necessary to consider radiation to the mucosa of the gastrointestinal tract. Also, recent findings on the accumulation of radioactive material in the tracheobronchial lymph nodes have pointed to their potential importance in evaluating inhalation hazards.

(7) Deposition and retention of inhaled radioactive particles are related to their physical and chemical properties, and the physiologic and anatomic characteristics of the host. For different radioactive particles, retention in the lung, translocation to other tissues, and excretion have not been successfully generalized by a single mathematical model.

(8) An additional unresolved point in pulmonary tumorigenesis is the effect of the spatial distribution of the radiation. It is not known whether differences in tumor production are due to the particular tissue in which the deposition occurs or to the localization and resulting strong irradiation of the tissue. The skin experiments using radioactive point sources as compared to flat plates indicate that, in the range where extremely large doses are given, with consequent killing of cells, the tumor production is considerably lessened for localized sources. This, however, sheds no light on the localization of smaller quantities of materials where the dose

rate is not adequate to definitely kill the cells within a given range of the radioactive material.

b. Experimental Findings

Experiments by Cember^(16,17) in which the lungs of rats were exposed to comparable doses of radiation by externally applied x-rays and by internally deposited radioactive barium sulfate particles showed that the x-rays had a very much greater biological effect (using death of the animal as a criterion) than point radiation sources from the dust particles. However, the absorbed doses, although physically approximately equal, were not biologically equivalent. It was inferred from the results of these experiments that the spatial concentration of absorbed energy around a point source of radiation did not necessarily represent a unique hazard, and that the volume of irradiated tissue, as well as the integral dose, was a significant parameter in the evaluation of the biological hazard to the lung as a consequence of the inhalation of radioactive particles.

In a later experiment, Cember⁽¹⁸⁾ exposed the lungs of rats to S-35 in the form of 1.45μ diameter BaSO_4 particles, administered weekly by intratracheal insufflations. Squamous metaplasia of the bronchial epithelium and bronchogenic squamous-cell carcinoma were found among the experimental animals, and it was concluded that this clearly demonstrated the carcinogenic properties of intrapulmonary radiation.

An experiment which may become a classic in the particle problem area was carried out by Cember and Watson,⁽⁸⁾ in which Strontium-90 glass beads were implanted in the lungs of rats by intrapleural injection. The radioactive material was uniformly distributed in the beads whose diameters were $0/32 \pm 0.11$ millimeters. The smallest bead contained 1.09 microcuries Sr-90, and the largest one, 59.3 microcuries; and the calculated surface dose rates from these beads were, respectively, 5.94×10^5 and 1.75×10^6 rads per day. The corresponding range-average dose rates were 77 and 1920 rads per day. Only one bead was implanted into the lung of each rat. Even though the absorbed dose to the lung was much greater than that delivered to rats by external x-radiation, no increase in death rate that could be attributable to radiation was observed. The experimenters conclude that this result is compatible with the hypothesis that the energy absorbed from a

single intrapulmonary radiation source is less lethal than the uniform absorption of the same amount of energy throughout the lung.

On the other hand, the distinguishing feature of the rats treated with the Sr-90 beads was the large number of primary malignant tumors found in the pulmonary tree--four squamous cell carcinomas, two lymphosarcomas, and one lymphoma. Of the seven malignant tumors, six were intimately associated with the radioactive beads. Since bronchogenic squamous cell carcinomata are not known to arise cryptogenically among rats, the experimenters attributed them to the radioactive beads. The malignancy frequency was 30 per cent and the carcinoma frequency was 17 per cent. The experimenters concluded that these frequencies are extremely high, when translated into cancer rates among humans.

Bronchogenic carcinoma was produced in rats that had been intratracheally injected with radiocerium in doses ranging from 5 to 50 microcuries of Ce-144 as 1μ -diameter $Ce^{144}F_3$ particulates. The smallest dose which produced a tumor was reported to be 2400 rads.⁽¹⁹⁾ In further studies,⁽²⁰⁾ doses as small as 0.5 microcuries of Ce-144 were used. The tumor frequency observed from these doses was 8/41. In this lowest dose level, the first death due to lung cancer, a lymphosarcoma, occurred 528 days after exposure. The radiation dose to the lungs in this case was 600 rads. In these experiments, dose refers to the organ average dose. In these later studies,⁽²⁰⁾ data seemed to imply a dose rate effect, whereby lower dose rates were more effective in producing lung cancer than were the higher dose rates. The experimenter comments, however, that close analysis of the data does not permit such an inference, since the dose values cited refer to the total absorbed doses from the date of exposure to the date of death, and the tumorigenic dose cannot be calculated because the exact time when the tumor began is not known.

Experiments were conducted in which Ru-106 coated pellets containing varying amounts (0.007 to 14 μ c) were inserted through a tracheotomy into the lungs of young rats. The induction time for tumors was generally inversely related to dosage; the incidence showed a definite relationship to pellet activity. The relationship between tumor production and the number of microcuries applied to the pellet was direct. Major interest was directed to the basal cells of the

bronchial epithelium where the malignant transition is presumed to occur. This cell layer is estimated to lie, on the average, about 100 microns from the surface of the pellet. Significant tumor incidence was observed at the lower dose levels; the tumor with the smallest dose occurred in an animal accumulating (from insertion of pellet to death) 1400 rads at 100 microns. It was pointed out that the pellet itself (without radioactivity) produces local injury and the tissue sensitivity could be changed, perhaps enhanced. On the other hand, the respiratory mucosa is subjected to many insults during life and it may well be that concern should lie with the sensitivity of modified rather than normal cells. If the average lung dose had been used instead of the basal cell dose, the computed dose in rads to the lungs would have been lower by a factor of approximately six. ⁽²¹⁾

Relatively insoluble particles may be removed from the lung by at least three natural routes: (1) Transport of the particles along the ciliated epithelium of the respiratory passages from where they enter the gastrointestinal tract and are eventually excreted in the feces; (2) Transport of solubilized fractions by blood to other tissues and/or excretion in urine; (3) Translocation to the bronchial lymph nodes where the retention time is long. Phagocytosis may play a role in removal of particles by all of these routes. Experiments with dogs inhaling $\text{Pu}^{239}\text{O}_2$ aerosol of 4.3μ mass median diameter (MMD) indicated that the Pu-239 was retained in the lungs with a half-life of about 300 days. However, accumulation of Pu-239 in the bronchial lymph nodes appeared to be continuous. Because of the high retention in the lungs and the continuous accumulation in the bronchial lymph nodes and other tissues, the total body retention half-time for Pu-239 was nearly 1,800 days. ⁽²²⁾

Beagle dogs were given single exposures to $\text{Pu}^{239}\text{O}_2$ aerosols. Deposition of $0.9 - 0.1\mu\text{c/g}$ of lung caused death in 31 dogs in 55 - 412 days after exposure. Average radiation dose to lungs was 4000 - 14,000 rads. Lymphopenia, polypnea, weight loss, and bradycardia developed prior to death. Gross and histopathologic tissue changes were limited to the lungs and associated lymph nodes, which contained 99 per cent of the plutonium content of the dog. One dog died 862 days following deposition of approximately $0.05\mu\text{c/g}$ of lung. Dogs exposed to lesser quantities of plutonium appeared normal 2 - 2 1/2 years after exposure except for lymphopenia. ⁽²³⁾ The findings of these studies are of particular importance in the evaluation of the FP particle problem concerned in this study, even though the FP particle is not an alpha-emitter.

Beagle dogs were given single exposures to one of seven $\text{Pu}^{239}\text{O}_2$ aerosols that differed in particle size from 0.2 to 7.6 μ mass median diameter (MMD). For aerosol of 4.3 μ MMD, the half-time retention of Pu-239 in lungs was about 300 days; however, during this period there was continuous accumulation of Pu-239 in bronchial lymph nodes, which resulted in about a 1500 - 1800 day half-time for total body retention of plutonium. Results of these studies emphasize the importance of the chemical form and the particle size of inhaled aerosols on retention, translocation, and excretion, and point out the problems to be encountered in estimating the body burden from excretion analyses. The data consistently indicated that inhalation of aerosols with a MMD of 5 - 7 μ is more hazardous than inhalation of aerosols with a MMD of 0.2 - 2 μ . Deposition in portions of the respiratory tract where clearance processes were not very effective, and subsequent retention in the lung, was greatest for the aerosols with the largest median diameters. Conversely, translocation of Pu from the lung to other tissues, excretion in urine, and excretion in feces were maximum when aerosols with small MMD's were inhaled. Solubility of particles is inversely related to size due to the larger surface area. It seems doubtful that solubility was responsible, however, for much of the translocation. A possibility is transport of intact particles through the lung parenchyma.

The most surprising result of this study was the relatively high rate of fecal excretion after inhalation of the aerosols with the small MMD's. It is generally believed that the larger particles have a greater probability of being cleared from the lungs and respiratory tract by ciliary processes. Perhaps the relatively high radiation dose from larger plutonium particles inhibited the process in some way. Little is known about the effects of radiation on the ciliated mucus secreting epithelium of the respiratory passages. Because of the noted accumulation of activity in the bronchial lymph nodes following inhalation of PuO_2 , it is possible that these lymph nodes rather than the lung should be considered the critical tissue. A decision on this will depend on findings of long-term toxicity studies now in progress.⁽¹⁰⁾

In summary, within a week after deposition of insoluble compounds, the lung can be expected to contain more than 90 per cent of the total body burden and will receive the largest radiation dose, although it may in time be exceeded by the

tracheo-bronchial lymph nodes. Also, translocation of small fractions of radionuclides may occur to more radiosensitive tissues or to tissues of greater concern because of genetic effects. In these cases doses smaller than those to the lung or tracheobronchial lymph nodes may be critical. If one can extrapolate from mice and dogs to man, these studies of the physiological behavior permit evaluation of the hazard and derivation of maximum permissible concentrations only on the basis of a maximum permissible radiation dose to the selected critical tissue. Knowledge of the biological effects are not yet adequate to evaluate the choice of the critical tissue although it was shown that radioactive particles in lung can cause malignant tumors and that lymphatic damage may possibly occur.⁽²⁴⁾

2. Ingestion

a. General

Morgan⁽²⁵⁾ has pointed out the importance of the GI tract in internal exposure cases. Of the 355 MPC values listed in the ICRP Handbook,⁽²⁶⁾ 71 per cent for ingestion and 41 per cent for inhalation refer to the GI tract as the critical organ. From physiological data furnished to the ICRP, it became evident that the GI tract receives the greatest absorbed dose from many of the radionuclides. The lower large intestine is usually the critical portion of the GI tract for three reasons:

- (1) Only radionuclides with a radioactive half-life greater than 1 hour were considered in the preparation of the ICRP Handbook.
- (2) The contaminated material remains in the lower large intestine 18 of the 31 hours that it is the GI tract.
- (3) The mass of material in which the radionuclide is diluted and to which the radiation dose is delivered is relatively small, i.e. 150 g in the lower large intestine, 250 g in the stomach, 1100 g in the small intestine, and 135 g in the upper large intestine.

The model employed for these determinations, however, considers continuous movement of a uniform dispersion of the contaminant in the ingesta; the present problem involves a presumed insoluble particle which is liable to reside immediately adjacent to the tract wall.

It is with the aim of evaluating the particle problem in realistic terms that a search has been made in the literature for experimental data which is directly or indirectly concerned with the particle problem. Unfortunately, there has not been very much radiobiological experimental work done involving single radioactive particles; and to make matters worse, contradictory results have been obtained. Nevertheless, some feeling for the biological hazards which may be associated with internal exposure to single radioactive particles may be inferred from experimental work which has been done thus far.

b. Experimental Findings

A classic experiment was carried out by Nold, Hayes, and Comar⁽²⁷⁾ in which internal radiation dose measurements were made in live dogs and goats which had been fed food uniformly contaminated with given amounts of radioactive Y-90. The dosimetric considerations and the calibration of the silver phosphate glass dosimeters used in this experiment are described in separate report.⁽²⁸⁾ The primary concern in this experiment was the estimation of radiation damage to the GI tract mucosa from ingestion of poorly absorbed radioactive materials. This experiment was considered of importance because it was recognized that one of the most immediate effects of radiation exposure is manifested by the GI tract.

The results of oral administration of Y-90 to dogs indicated that the large colon always received the highest dose. As such, the minute glass dosimeters were implanted in the mid - colon and terminal colon. It was also found that the Y-90 was not significantly absorbed from the GI tract. Practically all of the administered Y-90 was eliminated within 1 to 7 days after administration. A significant finding in this experiment was the twelve-fold average difference in dose between the diarrhetic and constipated groups of dogs. This emphasizes the importance of the physical state of bowel passages upon the dose delivered to the critical organ. Another finding of interest was swelling of the mucosa. The thickness of the mucosa and submucosa increased from 1.0 mm to approximately 1.7 mm on the third day following ingestion of the Y-90, but returned to normal about the 5th day. Approximately the same dose values resulted from the activity administered as a single dose or as a daily dose over a 5-day period.

An interesting factor noted was that the larger the dogs, the smaller was the dose received by the large colon. The dose to the terminal colon was 2430 rads for dogs of less than 16 pounds; 1015 rads for dogs of greater than 55 pounds. Concentration may account for this in part, since the 25 millicuries of Y-90 was mixed with less food for the smaller dogs. A state of diarrhea should have a noticeable effect on dose, since the resultant dose to the GI tract is directly proportional to the time during which exposure occurs and the concentration of radioactivity in the ingesta. In this experiment, diarrhea was found to decrease the dose to the colon by a factor of from 2 to 4. On the other hand, constipation increased the average dose to the colon by a factor of 3 to 7. Radioactive decay occurring during GI tract passage time is considered negligible, when the radioactive material has a half-life of several weeks or longer.

From the results of these experiments the experimenters concluded that the variation of delivered dose with different physiological states is of extreme importance. They also stated that because of the wide variations in dose which would occur as a result of diarrhea and constipation, the establishment of a specific value as the MPC appears to be of limited value unless qualifications are made for the commonly encountered deviations from the normal. A greater spread in doses would be expected in more marked deviations in these physiological states. (27)

For rats whose surgically exteriorized gastrointestinal tracts were exposed to 250 kvp X-radiation, mortality studies indicated the LD₅₀ was 1550 r and the mean survival time 7.1 days after irradiation. When the intestines were exposed in situ, the LD₅₀ was 1620 r and the mean survival time 5.7 days. Pathologic damage ranged from transitory mitotic inhibition after 900 r to severe focal mucosal ulceration after 1500 r and 1900 r. Residual damage was evident as long as 43 days after irradiation. The rats that received irradiation of the exteriorized intestine survived twice as long as rats receiving total-body exposure, although in both cases deaths are assumed to result from intestinal damage. Therefore, damage to the intestine must have been more severe after whole-body irradiation, or the survival time was markedly influenced by injury to other tissues. A pathologic finding of interest in

these experiments was the maintenance of an intact, although damaged surface epithelium on the mucosa. This is in contrast to the denudation of the epithelium usually reported from studies of the intestinal radiation syndrome. The preservation of the surface epithelium was consistent, apart from the focal areas of ulceration. Another unanticipated finding in this study was the persistence of residual damage in the form of isolated cystic crypts as late as 43 days after irradiation. ⁽²⁹⁾

Because of the possibility that the radiosensitive cells of the gastrointestinal tract might lie beyond the range of alpha particles originating from the contents of the tract, experiments ⁽³⁰⁾ were conducted to determine the effect on the tract from an orally administered alpha-emitter. Results of this study with plutonium indicated that death did not occur from a dose less than 88 mc/Kgm, and that all mortalities resulting from plutonium ingestion occurred within 24 hours. The time of death has been a generally accepted criterion of the intestinal radiation syndrome. The oral plutonium dose of 88 mc/Kgm, which resulted in 25 per cent mortality, is calculated to deliver at the surface of the contents of the small intestine a dose of approximately 650,000 rems. Since this very large dose resulted in none of the symptoms characteristic of the intestinal radiation syndrome, the investigators conclude that alpha-radiation originating within the contents of the intestine is largely ineffective in causing acute damage to the intestinal wall.

In contrast to the results with an alpha-emitter (Pu-239), similar studies involving the oral administration of Y-91 indicate that beta particles originating within the contents of the GI tract are approximately as effective as external X-irradiation in producing all of the symptoms of the intestinal radiation syndrome. ⁽³⁰⁾ Because of the characteristics of the FP particle problem, the results of this experiment with plutonium are considered to be most important. For radiations of low penetrating power, the dose to any particular cell in the intestinal wall will decrease with the depth of the cell below the mucosal surface. In the small intestine, the extreme folding of the intestinal epithelium may also result in surface cells at the base of the villi being effectively shielded from the dose at the surface of the contents. That such considerations may be of controlling importance is indicated by the

absence of acute radiation effects following intragastric administration of massive amounts of an alpha-emitter (PU-239). However, intragastric administration of Y-91 (a beta-emitter) produces radiation effects which are entirely consistent with calculations of intestinal tract dose. Rats fed 8 mc Y-91 died after 5 - 6 days with symptoms typical of intestinal radiation injury. The calculated dose to the small intestine from the ingested 8 mc Y-91 is about 3000 r, well above the LD₅₀ for intestinal radiation death. Following feeding of 4 mc Y-91, rats died after about 10 days with symptoms typical of total body irradiation. The calculated maximum dose to the small intestine in this case is about 1400 r, which is below the LD₅₀ for intestinal radiation death. For the 4 mc case, the calculated dose to the large intestine is about 5000 r.⁽³¹⁾

In the case of evaluating acutely toxic effects of ingested radionuclides, the possibility that the radiation itself may affect passage time must not be overlooked. Gastric emptying time is significantly prolonged by total body x-radiation in doses as low as 75 r. The effect of irradiation of the stomach by contained radionuclides has not been studied, but may well have a significant effect on the retention of the radionuclides in the stomach, and on the subsequent rate of passage through the intestinal tract. Because of anatomical and physiological differences, rat data cannot be extrapolated directly to man. The rat received a relatively much larger dose to the stomach and small intestine as compared with the large intestine. This is due to a lower ratio of contents between small and large intestine in the rat as compared to man and to the relatively shorter retention time of radionuclides in the large intestine of the rat as compared to man.⁽³¹⁾

The LD₅₀ for orally administered Y-91 in the rat is about 17 mc/kg, and the average survival time is 8.4 days. The calculated radiation dose from this amount of Y-91 is: small intestine--1150 rads; ascending colon--2800 rads; and descending colon--4700 rads. Pathologic changes due to Y-91 were primarily present in the large intestine owing to longer retention of intestinal contents in that segment. Damage was qualitatively similar to that observed previously after X-irradiation. When the Y-91 dose was increased to the extent that the calculated beta dose to the small intestine was of a magnitude similar to the amount of X-radiation that causes intestinal damage, the small intestine did exhibit marked pathologic changes. The absence of significant damage after oral administration of 155 mc/kg of plutonium clearly indicates the lack of effectiveness of unabsorbed alpha-emitters in causing gastrointestinal radiation injury. The calculated dose at the surface of the mucosa of the small intestine for this amount of Pu-239 is approximately

100,000 rads. The range of the alpha particles is apparently not sufficient to penetrate to the radiosensitive crypt cells. Quantities as high as 230 mc/kg of the alpha-emitter Pu-239 were orally administered to rats without causing death. A series of rats sacrificed at 3, 6, and 9 days showed only superficial epithelial damage at 3 days. No other evidence of injury was noted.⁽³²⁾

In a study⁽³³⁾ concerned with the long-term effects of a chronically administered, poorly absorbed, beta-emitter, mature female rats were maintained for 60 days on drinking water containing Y-90. A high-level group of 50 animals ingested an average of 410 μ c/day; a low-level group of 50 animals ingested 90 μ c/day. Nine animals from the high-level group died during the exposure period, and all animals in this group experienced severe weight loss. The low-level group exhibited no gross symptoms during the exposure period which would distinguish them from the controls. The total accumulated beta dose to the large intestine was estimated to have been about 55,000 rads in the high-level group and about 12,000 rads in the low-level group. The median post-treatment survival times for the high- and the low-level groups were 350 days and 434 days, respectively; and 626 days for control animals.

The late effects of particular interest in this study included cancerous involvement of the oral cavity and reproductive organs. Numerous corneal ulcers were probably related, in part at least, to external contamination. A histopathologic study of the intestinal tract indicated virtually complete recovery of the early damage, and only one intestinal tumor was observed among the exposed animals. Any consideration of these results in conjunction with human exposure problems must take into account the differing geometries involved. For example, extra-intestinal exposure would be of reduced significance in the human. It is considered doubtful that very significant penetration of beta particles would occur, in the human, from the lumen of the intestine to the endometrium of the uterus.⁽³³⁾

3. Summary

As indicated above, it is not possible to establish scientifically valid dose limits for internal exposure from single insoluble particles. On the basis of the literature reviewed, however, it does seem reasonable to obtain an indication of the relative impact of

internal vs. external dose on particle inventories by choosing an arbitrary dose value for internal sources. This dose value is not intended to be suggested as an acceptable dose, but one which would probably not differ by more than an order of magnitude from whatever value is finally promulgated.

For this purpose, particle activity concentration values are presented in Table 7 for a range-average (local) dose of 100 rads to the lung and GI tract (LLI). These values, and the activity concentration in a 0.575 cm particle as a function of time of arrival on earth for an ERD of 25 rems of external gamma radiation after infinite exposure, are shown in Figure 2, together with the average concentrations in a reactor fragment for several values of integrated power (assuming Way-Wigner decay). It can be seen that ingested particle limits require concentrations lower than those which can be expected at short times after shut-down. Even if a range-average dose of 1000 rads to the GI tract (LLI) is taken as the "acceptable" guide value, delay times of several weeks to several months might be required, assuming the validity of the model used.

TABLE VII

LIMITING VALUES APPLICABLE TO RANGE AVERAGE DOSE
OF 100 RADS TO LUNG AND GI TRACT (LLI)

Retention Time, T, In Situ (days)	Particle Dia. (μm)	Total Activity, A, (microcuries)	Concentration	
			A' (curies/ cm ³)	A'' (curies/ gm)
<u>LUNG</u>				
4	10	0.19	370	185
173	10	4.5x10 ⁻³	8.5	4.2
2,110	1	3.7x10 ⁻⁴	700	350
<u>GI Tract (LLI)</u>				
18/24	10	1.0	2,000	1,000
"	100	1.0	2	1
"	1,000(point)	1.0	2x10 ⁻³	1x10 ⁻³
"	1,000(discrete)	3.9	7.4x10 ⁻³	3.7x10 ⁻³

Note: For a given set of values of T (retention time) and D (range-average dose), the value of A is the same for all particles treated as point sources. The following relationship, applicable to such sources, may be used to compute readily the corresponding values of A' and A'' for particles of diameter, d, different from those given above: A_1 (or A_1') = A_2 (or A_2') $\times (d_2/d_1)^3$. This relationship, however, is not valid for particles treated as discrete sources.

FIG. 2
RELATIONSHIPS OF PRELIMINARY
DOSE EVALUATIONS TO CORE INVENTORY

V. PARTICLE AND POPULATION DISTRIBUTIONS

One of the more significant factors determining the biological significance of reentered reactor fragments is their distribution on the earth's surface. There is obviously a considerable difference in significance for those particles which impact in ocean, desert or other uninhabited areas and those particles which may be deposited in rather densely populated urban, suburban or agricultural areas of the world. Estimates of significance require consideration of both particle and population distributions. Although the trajectories for fragments created in the destruct of a nuclear rocket are not yet available, approximations can be made which indicate the relative degree of significance associated with such fragment reentry. Similarly, estimates of population distributions and land usage can provide a basis for approximation of the interactions of particles with people.

A. PARTICLE DISTRIBUTIONS

Size and spatial distribution data on fragments of a reactor core destroyed in space are not available. Although particle size distribution data from explosive destruct tests should exist, it had not been made available at the time of preparation of this report. For estimates of particle size distribution, therefore, the relationship presented in NUS-122⁽¹⁾ for 100 lbs. of high explosive was used, and the results are shown in Table 8.

For the purposes of this treatment it was assumed that particles less than 10μ in diameter spread gradually over a range of 30° in latitude about a great circle (roughly paralleling the weapons fallout pattern); between $10 - 50\mu$ particle, over a 20° range; and $>50\mu$ particles, over a 10° range. Also indicated in Table 8 is the average density per square mile of earth surface, based on the assumed dispersion in latitude indicated above. It can be seen that the variations in assumptions of size distribution of lateral spread do not greatly affect the density per square mile within the limits used here. For the heavier particles, a reduction in deposition length from 360° to 90° , and from 10° to 5° in latitude would result in an increase of about an order of magnitude to the range of $10 - 100$ particles per square mile.

For respirable particles ($1 - 10\mu$) an estimate of inhalation can be made by assuming that (eventually) the particles are uniformly distributed over an altitude range of 20 miles. Under this assumption and using a projected surface density of 6×10^6 particles per square mile, the average concentration in air is about 7×10^{-5} particles/ m^3 . With an

TABLE VIII
PARTICLE SIZE DISTRIBUTION

<u>d, μ</u>	<u>Volume%</u>	<u>log mean d, μ</u>	<u>No. of mean particles</u>	
			Total ^a	mi ⁻²
Case I - d _{max} = 7 mm				
1-10	1.4	3.2	3.9x10 ¹⁴	7.7x10 ^{6b}
10-50	3.1	22.5	2.5x10 ¹¹	7500 ^c
50-100	2.8	70.8	7.1x10 ¹⁰	4100 ^d
100-500	13.8	225	1.1x10 ¹⁰	640
500-1000	11.9	708	3.0x10 ⁸	17
1000-7000	66.6	2650	3.2x10 ⁷	1.9
Case II - d _{max} = 19 mm				
1-10	0.9	3.2	2.5x10 ¹⁴	5x10 ^{6b}
10-50	2.0	22.5	1.6x10 ¹¹	4650 ^c
50-100	1.9	70.8	4.8x10 ¹⁰	2800 ^d
100-500	9.0	225	7.2x10 ⁹	420
500-1000	7.9	708	2.0x10 ⁸	12
1000-5000	33.6	2250	2.7x10 ⁷	1.6
5000-19000	44.5	9750	4.4x10 ⁵	0.026

Notes:

(a) Total volume of loaded graphite equals 4.75x10⁵cm³

(b) Assumes 30° spread in latitude (5.11x10⁷mi²)

(c) Assumes 20° spread in latitude (3.41x10⁷mi²)

(d) Assumes 10° spread in latitude (1.72x10⁷mi²) for d>50 μ

average breathing rate of $20 \text{ m}^3/\text{day}$, the probability of inhaling a particle is about 1.4×10^{-3} per day, or about 1 in 2 years. For particles in the $10 - 50\mu$ size range, which may be inhaled and subsequently swallowed, a similar analysis would indicate a concentration in air of about 7×10^{-8} particles/ m^3 , and a corresponding inhalation probability of about 1.4×10^{-6} per day.

The direct ingestion route is much less amenable to the type of exercise performed above. This route involves questions of the portion of land areas covered by fresh fruit or vegetable crops, the yield per unit area, the processing employed, and the delay between deposition and ingestion. For an indication of the scope of the problem, it is assumed that 0.1 per cent of the total land area is devoted to crops eaten raw, that crop coverage is 50%, and that one square mile of land provides fresh fruits and vegetables for 1000 people. (U. S. value approximates 400 persons per square mile of harvested cropland; total harvested cropland is about 14% of the total land area). Ingestable particles are considered to range from 50μ to 500μ and perhaps to 1mm in diameter.

Particle deposition density in this size range approximates 4000 per square mile, of which half is on the vegetation and half is deposited directly on the ground. If the vegetation serves 1000 people, then, assuming no further losses in processing (washing, peeling, etc.), about 2 particles might be ingested per person.

On the basis of these crude assumptions, the use for dose estimation purposes of one particle per person does not seem unreasonable. The total number of people so affected, however, is dependent upon the distribution of population and the land usage along the trajectories of interest.

B. POPULATION DISTRIBUTIONS

Gross population density distribution over the world is presented in Figure 3. The range of density shown is necessarily limited, and it should be noted that regions of much higher density exist in urban and suburban areas. For example, the density along the Nile River Valley near Cairo is reported⁽³⁵⁾ to be about 6500 per square mile, and in the Yangtze River Valley of mainland China, about 3500 per square mile. Urban areas may contain more than 15,000 per square mile (Manhattan holds about 85,000 per square mile). The average population over the regions of the world within the limits of 35° North and South latitudes has been estimated from population data:⁽³⁵⁾

for Asia, it approximates 190 per square mile; for Australia, 3.4 per square mile; for the United States, 50 per square mile; for Central and South America, about 30 per square mile; and for Africa (exclusive of Algeria and Tunisia), approximately 25 per square mile. An examination of typical trajectories for an orbiting vehicle (or fragments) has been made by using the track followed by the MA-9 flight. Although differences between this flight and those projected for an operational nuclear vehicle are recognized, the relative proportions of overland and over-water travel can be examined on the basis of these data. In the flight of MA-9 the maximum overland travel encompassed some 50 per cent of the path of Orbit 11, and a minimum percentage of overland travel of about 7.3 per cent on Orbit 19. In the case of Orbit 11, the trajectory passes over sparsely inhabited areas in both South America and Northern Africa, but also passes directly over such congested areas as Recife in Brazil and Cairo and the Nile Delta in Egypt as well as the densely populated regions of Southeast China, including Canton and Hong Kong. The only major city over which Orbit 19 passes is Monterrey, Mexico, with a population of some 600,000 individuals. These orbits are also plotted in Figure 3. An index of the total population over which a particular orbit passes is provided by the product of the orbit length projection in any area and its density. For the two orbits mentioned above, the index number is approximately 3×10^5 for Orbit 11 and approximately 2×10^4 for Orbit 19.

If, as assumed in the previous section, 0.1 per cent of the land area is used for growing fruits and vegetables eaten raw, then Orbit 11 ($\pm 5^\circ$ band) would intersect about 8.6×10^3 square miles of such land. Again, on the basis of previous assumptions, the number of people affected by this dose route would approximate 8.6 million for this orbit. For Orbit 19 the number would be about 1.25 million. In order to improve this estimate a detailed examination of land use practice for each of the over-flight areas would be required, the practicality of which is open to question. It would appear, on the basis of these estimates, that the number of people affected could be very large, and, therefore, the problem is worthy of additional evaluation of its physical, biological, and radiological aspects.

FIG. 3
WORLD POPULATION DENSITY
WITH MA-9 TRAJECTORY
OVERLAY

(Life Pictorial Atlas, Life and Rand McNally, 1961)

VI. DISCUSSION

It has been shown that the problems associated with reentered fragments of nuclear rocket reactors are real. Apparent particle distributions are not so diffuse as not to produce credible exposure situations, although refinement in the analysis is certainly required. It would also appear that problems generated by external exposure to radiation from fragments are not difficult to resolve. Acute whole body gamma dose guides have been considered by several groups, and a value of 25 rems for once-in-a-lifetime accidental or emergency exposure seems to enjoy a reasonable consensus. As shown in Figure 2, a spherical particle with dimensions less than about 1/4" would satisfy such a whole body gamma dose criterion of 25 rems at any time following reentry. However, major unresolved problems appear to reside in the internal dose area, particularly with reference to lung and GI tract exposures. The concurrent external beta dose to the skin from a particle which delivers a whole body gamma dose of 25 rems (ERD) will vary from 40 to 1075 rads depending upon the size and age of the particle. The beta dose to portions of the body protected by clothing may be substantially reduced due to the limited penetrating ability of beta radiation. Additionally, the range of beta radiation in air decreases rapidly from about 13 meters just after fission to about 3 meters 100 hours after fission. Thus, the degree of skin injury from beta radiation will be determined to a great extent by the type and amount of clothing worn and the mobility of the individual.

In the internal exposure case, particularly where discrete, insoluble particles are concerned, not only is there an absence of dose guidance, but there is also a relative vacuum with respect to the physical dosimetry, that is the tissue volume over which such doses should be computed, or averaged. Although it is recognized that uniform distribution of inhaled particles is not observed in practice, lung exposures are expressed as "mean organ doses," calculated by assuming uniform distribution of the radionuclides throughout the lung. Similarly, the ICRP model for calculating the dose to the walls of the GI tract from ingested radionuclides assumes uniform distribution of the radionuclides in the contents of the GI tract; thus, the dose to the walls of the GI tract is one-half of the computed absorbed dose in the contents of the GI tract.

The particle problem for the inhalation case has been reviewed by a special committee of the National Academy of Sciences - National Research Council, with the purpose of applying the findings to an evaluation of the potential hazards of airborne radionuclides. The report⁽⁵⁾ of this study is a comprehensive review of data available at this time. This report does not contain recommendations of a

specific concept for the computation of physical dose and dose-rates for the particle problem. However, one of the conclusions in this report states that, because of the non-uniform distribution of energy absorption and the associated very-high dose rates at or near contact with the particle, the calculated "mean dose" to the lung should be used with caution in the estimation of biological effects. For the particle ingestion case, no equivalent study-report is presently available.

Two basic questions appear to require answers in the GI tract case: first, the question of dosimetry--that is, the magnitude of the dose received by the biologically important segment of the wall from insoluble beta emitting particles passing through the GI tract; and second, the biological significance of a relatively high dose to a very small segment of the wall. Both questions appear to require experimental work to provide answers. The apparent significance of these doses in terms of restrictions on reactor operations of destruct system requirements suggests that carefully planned experimental work is worthy of support.

Additionally, for exposures from sources such as those treated in this study, dose guide values applicable to a specified type of "acceptable risk" will be required from cognizant government agencies. Different sets of dose guides are possible, depending on the type and degree of "acceptable risk" (genetic and somatic effects, ranging from non-observable to disabling or lethal) used as the basis for the necessary criteria. As such, dose guides may have a very wide range of values. Undoubtedly, criteria used to establish such dose guide values will include political as well as technical considerations.

VII. RECOMMENDATIONS

The following recommendations seem warranted on the basis of the examination reported above:

1. Studies are required to provide better information on size distribution of reactor fragments following destruct and reentry, and on the spatial and temporal distribution of such particles on the earth's surface.
2. A statement of policy with respect to "acceptable risk" from accidental exposures of this sort must evolve from the cognizant Government agencies. This policy will be required not only to evaluate nuclear rocket operations, but to set in large measure the scope of required radiobiological investigations by indicating the kinds of effects to be sought.
3. Research is required on the potential source material with respect to both dosimetric and radiochemical nature. Dosimetric measurements on irradiated fuel material, heated to simulate operation and reentry conditions, are important to refined evaluation of beta doses, as affected by the distribution of fission products remaining in the fuel material following such treatment. Similarly the determination of solubility of the residual fission products in lung and intestinal fluids remains substantially unknown at present, and may lead to consideration of other body organs as "critical."
4. Research on the extent and significance of internal exposure from single (or few) insoluble particles is required. This research should be conducted on large animals, i.e. those with intestinal dimensions large in comparison to the range of the beta radiation of interest. Lung data would also be useful, although there is comparatively more data on this presently available.

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