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**ROVER FLIGHT SAFETY PROGRAM PRELIMINARY REVIEW.
IV. SAFETY ANALYSIS REPORT:
Radiological Considerations In Nuclear Flight Safety**

by

**The Radiological Effects Working Group
of the Space Nuclear Propulsion Office**

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**U.S. NAVAL RADIOLOGICAL
DEFENSE LABORATORY**

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TECHNICAL MANAGEMENT OFFICE
P. E. Zigman, Head

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Organizations participating in the Radiological Effects Working Group of the Space Nuclear Propulsion Office:

U. S. Naval Radiological Defense Laboratory
Space Nuclear Propulsion Office
NUS Corporation
Lockheed Missile and Space Company
Battelle Northwest
Westinghouse Astronuclear Laboratory
Oak Ridge National Laboratory
Los Alamos Scientific Laboratory

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Eugene P. Cooper
Scientific Director

D. C. Campbell, CAPT USN
Commanding Officer and Director

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ABSTRACT

Accidents to space nuclear propulsion systems could lead to deposition of radioactive material on the surface of the earth and subsequently to interactions between man and the radioactive debris. This report describes preliminary models developed to evaluate possible safety problems resulting from the introduction of nuclear rocket debris into man's environment.

The physical and chemical characteristics of the debris are described as well as general exposure criteria and applicable biological effects information. The concept of probability of exposure as a relevant safety index is employed. Mathematical models are presented which describe the likelihood of different levels of total body gamma exposure experienced by a mobile receptor and the expected number of direct interactions between people and debris. For direct interactions, skin, lung, and gastrointestinal tract exposures are specifically considered. Beta and gamma dose calculational methods applicable to reactor debris problems have been developed as part of these studies and are described in detail.

CONTRIBUTORS

This volume of the "Rover Flight Safety Program Preliminary Review" was written by members of the Radiological Effects Working Group of the Space Nuclear Propulsion Office. The individuals participating in the development of each Section of the report are listed below. The final report was edited by P. Zigman, W. Ellett and A. DeAgazio. Most of the work described in this report was executed under a program developed by J. Mackin, W. Ellett and P. Zigman.

Section I - Purpose and Scope

P. Zigman and J. Mackin, USNRDL; G. Dix, SNPO.

Section II - Debris Definition

J. Mackin, E. Freiling and J. Lai, USNRDL;
J. Bridges, W. Brown and R. Trammell, WANL;
C. Lee, R. Hausman and P. Hutchinson, LMSC;
L. King, LASL

Section III - Approach to Dose Model Development

J. Mackin and P. Zigman, USNRDL;
M. Goldman, NUS;
R. Decker, SNPO

Section IV - Particle Exposure Models

D. Kochendorfer and J. Ulberg, USNRDL;
A. DeAgazio and M. Goldman, NUS;
R. Decker, SNPO; B. Fish, T. Clark, R. Boyett,
J. L. Thompson and W. Wilkie, ORNL;
W. Langham, LASL

Section V - Radiological Dose Models

W. Ellett, O. Hogan, J. Ulberg and J. Pestaner, USNRDL;
M. Berger, NBS; A. DeAgazio, NUS

Section VI - Complete Dose Models

D. Kochendorfer, J. Ulberg and W. Ellett, USNRDL;
A. DeAgazio and M. Goldman, NUS

Section VII - Radiation Exposure Criteria and Radiopathology

J. Teresi, USNRDL; M. Goldman, NUS;
B. Fish, ORNL; W. Bair and R. Thompson, BNW

Section VIII - Conclusions and Recommendations

W. Ellett and P. Zigman, USNRDL; A. DeAgazio and
M. Goldman, NUS

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GLOSSARY

$A.F.(m,E_i),$	Absorbed Fraction of gamma energy E_i for organ mass m
$[A^*(t)_j]_I,$	Activity in μci for a particle of size j at time of inhalation t
$B,$	Breathing rate. - meter cubed per day
$a^*,$	Nasal breathing area. - centimeters squared
$a^{**},$	Mouth breathing area. - centimeters squared
$a(H),$	Area of horizontal surfaces. - centimeters squared
$a(t)_j,$	Total ground area shielded from size j particles. - centimeters squared
$a(V)_k,$	Vertical cross sectional area of the cylinder representing body part k . - centimeter squared
$B(E_i),$	Buildup factor for gamma energy band E_i
$D^*(I)_j,$	Beta dose to an organ from isotope I in a particle size j . - rad
$D_j(d)$	Gamma exposure at a distance d from a source of size j . - roentgens
$D_j(0)_\gamma,$	Average gamma dose to an organ from size j debris. - rad
$D(s)_{j,i},$	Average exposure to a mobile receptor in the i th activity class from a source of size j located a distance S for the center of motion. - roentgens
$D(s)_{j,i,g},$	Reduced exposure due to local shielding appropriate for cultural group g . - roentgens

$D(u.p.)_{j,i}$,	Average gamma exposure to group i from a uniform planar distribution of size j particles. - roentgens
D_{∞}/A_t	Organ dose per μ ci intake. - m rad
d,	Distance from a source to a point one meter above the base of a receptor. - meters
E ,	Average energy of electrons escaping debris. - Mev
E_i ,	Gamma energy in the ith energy band. - Mev
E_{max} ,	End point energy of Fission Product Beta Spectrum. - Mev
E_0 ,	Initial electron energy. - Mev
F_j^* ,	Fraction of size j particle passing through the nose.
$F(av)_j$,	Fraction of particles deposited in alveolar ducts and sacs.
$F(l)_j$,	Fraction of particles not deposited in upper respiratory tract.
$F(p)_{j,c}$,	Contamination reduction fraction by food processing for size j debris. on crop c
$f(I)$,	Fraction of time inhalation occurs.
$f(H)$,	Fraction of horizontal body surface exposed.
$f(N)$,	Fraction of inspired volume occurring via nose.
$f(M)$,	Fraction of inspired volume occurring via mouth.
$f(P)_i$,	Fraction of the total population in group i.
$f(V)_k$,	Fraction of vertical area exposed on body part k.
f_i ,	Fraction of time a given standard deviation of motion σ_i applied to a member of the ith activity class.
f_j^* ,	Fraction of particles of size j inhaled via the nose.

f_j^{**} ,	Fraction of particles of size j inhaled via the mouth.
f_o ,	Fraction of time spent outdoors.
$f(*)$,	Fraction of activity reaching critical organ via inhalation.
$f(**)$,	Fraction of activity reaching critical organ via mouth.
$f(r)_{j,c}$,	Fraction of size j particles retained on foliage crop. c
g ,	Grams.
g_o ,	Gravitational acceleration. - centimeter per square second
j ,	Particle size class.
$K(E_i)$,	Exposure conversion constant for gamma energy band E_i . - roentgens meters squared per Mev
M_j ,	Total number of particles of size j .
m ,	Mass of critical organ. - grams
$\bar{m}_j$	Average number of size j particles per unit area. - number per square meter
N_o ,	Total number of people exposed to M_j particles.
$\bar{M}$,	Average population density. - number per square meter
$\bar{M}_i$,	Average population density for group i . - number per square meter
$\frac{N_i}{N_o} (D \geq D(s))_{j,i}$,	Fraction of the population receiving an exposure of at least $D(s)_{j,i}$
P_i ,	Probability a mobile receptor in group i is within a given annular region about his center of motion.
$[P.E.]_{j,i}$,	Average population exposure to individuals in group i from single particles of size j . - man roentgens
$\sum [P.E.]_p$,	Total population exposure from single particles. - man roentgens

$\Sigma[P.E.]$,	Total population exposure. - man roentgens
$P(M)_j$,	Probability that M_j particles will strike and stick to an individual.
$P(M_j t)$,	Probability that M_j particles stick on the skin for a time t .
$P(M)_j(\ell)$,	Probability of inhaling M_j particles of size j .
$P(M_j)(g)$,	Probability that M_j particles reach the gastrointestinal tract.
$P'(g)$,	Probability that a particle is retained t hours in the gastrointestinal tract.
$P(t)_j$,	Probability that a particle of size j will stick to skin for a time t .
$Q_j(E_i)$	Gamma energy emitted in energy band E_i from a particle of size j . - Mev
Q_γ ,	Time integrated gamma energy emission. - Mev
Q_β ,	Time integrated beta energy emission. - Mev
r ,	Distance between a source and the base of a mobile receptor. - meters
S ,	Separation distance between a source and the center of motion of a mobile receptor. - meters
t ,	Time. - hours
T_e	Effective half life. - days
$V(T)_j$,	Terminal velocity of size j particle in still atmosphere. - meter per sec
$V(W)$,	Wind speed 1 meter above ground. - meter per sec
W ,	Air flow. - liters per minute
W_g ,	Exposure reduction factor to account for differences in local shielding to members of cultural group g .

$Y_c,$	Area planted in crop c needed to feed one person. - meters squared per person.
$Z_i,$	Average number of persons in group i within a distance S from a particle.
Γ_j^*	Impaction efficiency for the nasal area.
Γ_j^{**}	Impaction efficiency for the mouth area.
$\Gamma_{k,j},$	Impaction efficiency for body part k and particle size j.
γ_j	Particle diameter of size j debris. - microns
$\eta,$	Air viscosity. - grams per centimeter second
$\Lambda_j,$	Mean life on the skin of a size j particle. - hours
$\lambda,$	Excretion coefficient. - hours ⁻¹
$\mu_{ci},$	Microcuries - 3.7×10^4 disintegrations per second
$\mu(E_i)$	Linear absorption coefficient in air. - meter ⁻¹
$\xi_j,$	Sticking efficiency of size j particle
$\rho,$	Specific gravity of debris particles
$\rho_0,$	Air density. - grams per centimeter cubed
$\rho_i,$	Standard deviation of motion for a member of the ith activity class. - meter
$\bar{\Phi}_j,$	Average number of particles retained in the alveoli.
$\bar{\Phi}^*,$	Average number of particles retained in the alveoli via the nose.
$\bar{\Phi}^{**},$	Average number of particles retained in the alveoli via the mouth.
$\bar{\chi}_j,$	Mean time integrated suspended particle concentration - particle sec per meter cubed.

$\bar{\Psi}_j$,	Mean number of size j particles striking and sticking to an individual.
$\bar{\Psi}^*$,	Mean number of particles inhaled via nose.
$\bar{\Psi}^{**}$,	Mean number of particles inhaled via mouth.
$\bar{\Psi}_j''$,	Mean number of particles of size j impacting in breathing zone.
$\bar{\Psi}(g)_j$,	Total average number of size j particles.
$\bar{\Psi}(H)_j$,	Mean number of size j particles ingested per person.
$\bar{\Psi}'(I)_j$,	Mean number of particles inhaled per person.
$\bar{\Psi}(O)_j$,	Mean number of particles of size j retained by a clothed individual.
$\bar{\Psi}(s)_j$,	Mean number of particles swallowed following inhalation.
Ω_k ,	Diameter of body part k. - centimeters

SECTION I

PURPOSE AND SCOPE

A. INTRODUCTION

The United States, through the AEC-NASA Space Nuclear Propulsion Office (SNPO), is developing nuclear rockets for space use. It is expected that normal operations of such systems will terminate with the radioactive, nuclear engine in either a long-lived earth orbit or in permanent space residence. Successful missions conducted as presently conceived will not at any time present a radiological problem. If, however, an accident occurs which results in thrust cessation following some portion of nuclear powered flight, and before attainment of a long lived orbit or earth escape velocity, some fraction of the radioactivity developed during reaction operation will eventually return to the earth's surface.

If, following an accident, the intact reactor or other radioactive debris is deposited on the earth's surface, this material could under certain circumstances interact with man in such fashion as to pose a radiological problem. The magnitude and nature of this safety problem would depend upon pre-accident flight and reactor operations, the accident itself, applied countermeasures, and the circumstances of debris-man interaction.

In the event of a pre-orbit accident, missions incorporating sub-orbital reactor start-up could return a substantial portion of the generated radioactivity. In this case applied countermeasures could markedly reduce any potential radiological safety problem. Aborted missions which involve start-up of the reactor in orbit would return a much smaller fraction of the contained radioactivity because of the longer delay before earth deposition. Again, certain flight modes and countermeasures might be utilized to minimize potential radiological problems and increase total safety in use of the nuclear propulsion system.

In order to reduce any potential radiological problem from a given mission it is necessary to examine alternate flight and reactor

operation modes for the mission and to inspect the desirability of applying any particular countermeasure. These evaluations rest ultimately on analyses and comparisons of population-debris interactions for the alternate actions possible. In turn, the analyses and comparisons depend on appropriate interaction models and on required input data. Pertinent evaluation methods and, as necessary, associated input data are being developed by the Space Nuclear Propulsion Office's Radiological Effects Program. These models and methods will be used by other SNPO investigators to consider the radiological safety of any particular mission. This document describes the present status of the methods and data developed under the auspices of the Radiological Effects Program.

B. STATEMENT OF THE PROBLEM

Accidents to space nuclear propulsion systems could lead to deposition of radioactive debris on the earth's surface and subsequent interaction of the debris with man. The magnitude of the safety problem which would result from such interactions would depend on a number of factors. These include the extent and nature of both direct and indirect interactions; the size distribution and physical-chemical characteristics and behavior of the debris; and the radioactivity content (amount and kind) of the material. It is necessary that many factors be examined, individually and in concert, to evaluate radiological safety following in-flight accidents to any nuclear propulsion system. To carry out such examinations it is necessary to develop a variety of experimental data and to formulate analytical models and techniques to represent the extent of human-debris interaction and to depict radiation dose distributions.

C. BACKGROUND

Computer codes under development by SNPO contractors ^(1,2) are being employed to calculate the space-time history of nuclear propulsion reactor debris formed under different mission/accident/applied counter-conditions. These codes permit definition of terrestrial radiation sources and debris deposition patterns, but are not designed to allow detailed calculations of man-debris interactions.

The Technical Management Office of the U. S. Naval Radiological Defense Laboratory (USNRDL) was designated to develop a technology for

computations of the many interactions between debris from nuclear propulsion reactors and man. This technology, expressed in terms of computer codes, when combined with the space-time debris history codes will provide a means to evaluate and judge the radiological safety in the use of space nuclear propulsion systems. Development of the interaction assessment methodology has been carried out through design and implementation of a Radiological Effects Program with the following objectives and scope:

To provide the means to assess the radiological importance of nuclear radiation from injection of contaminated nuclear propulsion reactor fuel and system materials into man's environment. The Program is concerned with all radiological safety aspects in the use of any nuclear propulsion system. Situations of debris injection include passive (no countermeasure) and destructive reactor re-entry following scheduled flight or application of countermeasure devices. Program endeavors are directed to the development of radiological safety assessment methods and to the acquisition of necessary man-debris interaction data, not to the safety inspection of any given mission.

It is planned that the methods and information developed via the Radiological Effects Program will be used by other components of the total SNPO program in examination and evaluation of the radiological safety of specific missions.

D. APPROACH

To achieve the stated objectives it is necessary that the Radiological Effects Program address itself to determining the extent to which contaminated debris would interact or become associated with man and following interaction, estimation of the manner and extent to which absorbed ionizing radiation would be distributed in man. To fulfill these functions the Radiological Effects Program is subdivided into twelve major sections. Figure I-1 depicts the twelve major components (Projects) and the objectives of each, in terms of information flow. The diagram also indicates the involvements of participating organizations.

Figure I-1 SNPO Radiological Effects Program Plan

As will be shown in subsequent sections of the present document, analytical model development has evolved from the project definition and information flow scheme described in Figure I-1. Although shown schematically as a segment of the Radiological Effects Program, responsibility for dose evaluation, Project 9, rests ultimately with other components of the SNPO Flight Safety Program.

Following Program formulation a Radiological Effects Working Group (REWG) was established. Execution of the Radiological Effects Program has been accomplished in large part via the Working Group, chaired by the USNRDL Technical Management Office, and composed of both direct Program participants and a number of expert contributors and consultants.

E. PLAN OF THIS DOCUMENT

The analytical methods and supporting experimental information produced by, and under development in, the Radiological Effects Program for application to future assessments of the radiological safety of nuclear flight operations are described in this document and in a classified volume containing appendices.⁽³⁾ Following the present Section, the manner of presentation is organized as follows:

Section II - This portion of the report considers the nature of the radioactive debris that could be returned, under certain circumstances, to the earth. Debris return following either passive re-entry or after intentional explosive destruct are considered.

Section III - This section discusses the basis for selection of the methods used in dose model development.

Section IV - This section presents the models developed to calculate the probability that exposed persons and debris will interact in a postulated manner.

Section V - This portion of the report describes the radiological dose models used to calculate either the gamma ray exposure or the beta and gamma dose from contact in a given exposure situation.

Section VI - This section indicates how the models developed in sections IV and V are combined to evaluate potential exposures associated with any given debris deposition pattern. A summary of the sensitivity analysis performed for each model is included.

Section VII - This section describes presently developed criteria used in the evaluation of radiation safety and presents a summary of some of the studies of pathological effects from exposure to reactor debris.

Section VIII - This section discusses the adequacy of the evolved dose models, relevancy of existing criteria and adequacy of the available information on biological effects as well as recommendations for further study.

A classified appendix containing Confidential Restricted Data is in preparation. This Appendix will be published as a separate USNRDL Technical Report entitled "Appendix to Rover Flight Safety Program Preliminary Review, Vol. IV - Safety Analysis Report - Radiological Considerations in Nuclear Flight Safety".

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SECTION II

DEBRIS DEFINITION

A. INTRODUCTION

The present emphasis in the nuclear rocket program has been placed on the graphite core, liquid hydrogen-cooled reactor developed from the LASL KIWI concept. The very basis for selection of graphite as the core structural material (its excellent high temperature properties and ablation resistance), precludes the complete disposal of re-entering material by burnup in the upper atmosphere. Since a certain amount of core debris may return to earth in the event of an accident, it becomes necessary to examine the nature and distribution of this debris as it affects the nature and extent of interactions with people.

The details of the nuclear stage and engine are presented elsewhere.^(1,2) The reference fuel consists of small uranium dicarbide (UC_2) spheres coated with a layer of pyrolytic carbon. These beads are distributed throughout a graphite matrix and the mixture fabricated into fuel elements. Each element is pierced the entire length by coolant channels which are coated with niobium carbide to prevent erosion by the coolant.

In the mission safety studies of nuclear rocket flight operations reported elsewhere in this series^(1,2), two potential re-entry modes are considered: these are passive re-entry of an initially intact nuclear rocket reactor, and re-entry of debris resulting from the explosive destruction of the reactor. These two re-entry modes will produce different kinds and degrees of radiological interactions due to differences in the debris size and earth surface distribution of radioactive materials. It is the purpose of this report to present models which can be used to assess the radiological interactions for these potential re-entry modes. Since the nature of the debris is a factor in determining the applicability of the postulated interaction models, it is the intent of this section to characterize the re-entered material to the extent necessary to justify the models presented. Among the debris characteristics of major importance are the size, activity content, spatial distribution and the time of arrival of material on the ground.

The debris size largely determines the radioactive source strength since, as a first approximation, the larger the size the greater the activity content. The size of the fragment also determines the routes through which individuals may become exposed. Particles up to about 10 microns are primarily responsible for lung doses whereas particles up to 1000 microns can be responsible for delivering doses to the gastrointestinal tract, and in the case of direct external contact with an individual, a skin dose. Large particles, those much above 1000 microns can neither be inhaled nor readily ingested, nor will they stick to the body for an appreciable length of time. These large fragments however, ranging up to an entire fuel element in size, could under certain circumstances contain sufficient activity to deliver significant external whole body gamma doses at substantial distances. Knowledge of chemical and physical properties of the debris as well as its post-deposition behavior will further contribute to predicting its interaction with humans. These factors are discussed in succeeding sections of this report.

Debris deposition density is important in any radiological safety evaluation since this quantity, coupled with population density in the affected area, determines the extent of interaction with humans. Of major significance is the geographic location of the debris; the difference in probable interaction resulting from deposition in the ocean or lightly populated areas as opposed to deposition in an area of high population density is obvious. In addition, the amount of radioactivity and associated decay energy spectra must also be known. To determine this, the reactor operating history and decay after reactor shutdown, as well as selective fission product losses before earth impact must be known. Therefore, studies of the trajectories involved, post-operational and re-entry phenomena and subsequent transport mechanisms are necessary. These factors are treated in other volumes of this series^(1,2) for specific mission profiles.

In the following sections, debris is generally characterized for each of the potential re-entry modes, passive and explosive destruct.

B. PASSIVE RE-ENTRY DEBRIS

In the case where no countermeasure is employed, the disassembly and dispersal of the nuclear engine is accomplished solely by the natural forces acting upon it. This is referred to as the passive re-entry or passive disassembly mode, and is described in Volume II⁽¹⁾ of this series.

1. Debris Size

In the case of passive disassembly of an intact reactor, fuel elements will be ejected from the engine at some time following mission abort. These may be integral elements, or portions of elements resulting from breakage during the ejection process. During re-entry or impact upon the earth's surface these elements may be further degraded. The most probable fragment size has not been determined. However, it appears unlikely that much size reduction will occur during ejection from the reactor or during re-entry other than cross-sectional fracturing which will produce short sections with a cross-sectional dimension about equal to the original fuel element diameter. Studies currently being performed for the passive re-entry mode⁽¹⁾ assume that the re-entering debris is in the form of integral fuel elements. As new information on fragment size for this re-entry mode becomes available, the presently developed dose model can be easily modified to use this information.

2. Debris Distribution

The distribution of debris from passive re-entry of an intact reactor will depend to a large extent upon the operational history of the engine since the fission product decay heat generated prior to engine disassembly may cause loss of structural integrity. The tumbling characteristics of the reactor following failure will also play a role in the distribution of material on the earth's surface.

The precise distribution of debris will depend upon the mission, system design and the mode and time of failure. For near-orbit disassembly, the area of coverage is highly variable extending from about one thousand square kilometers to more than one million square kilometers. The latter figure represents a 30 km wide band about the earth. For a low altitude disassembly, the area of coverage would be much smaller, probably on the order of a several square kilometers.

3. Debris Location

Using existing computer programs⁽¹⁾, impact location of engine debris can be predicted provided orbital lifetime is less than ten revolutions. Thus, a failure following a suborbit or low orbit start would result in a predictable debris impact location. For debris with longer orbital lifetimes, impact location cannot be predicted. Typical flight paths for the first orbit about the earth would extend from 40° North latitude to 40° South latitude. All areas lying under the flight path would be potentially affected by debris impact in the event of an accident. Since each succeeding orbital trace is displaced from its

predecessor, the location of the potentially affected area is changed until a band between about 40°N latitude and 40°S latitude is encompassed.

4. Character of Debris

Prior to re-entry, but after some engine operating time, it is expected that some of the graphite matrix will have sublimed. The pyrocoated bead structure will also be partially degraded due to post-operational heating⁽³⁾, which would allow diffusion of the fuel material and fission products through the graphite matrix. The amount of material diffused increases with increasing fuel temperature and time at elevated temperature. Thus, although re-entry and ablation do not significantly change the physical or chemical properties of the fuel element⁽⁴⁾, the re-entered material may have an altered fission product composition due to the post-operational heating. Such variations in fission product composition can be treated by the dose models developed in this report.

C. DESTRUCT RE-ENTRY DEBRIS

Explosive destruct tests to date^(5,6) have produced a complete spectrum of particle sizes ranging from approximately one fuel element diameter to sub-micron particles. Thus, debris interaction and radiation dose distribution models have been developed to consider interactions resulting from debris ranging in size from sub-micron particles to fragments with dimensions of the same order as intact fuel elements.

1. Debris Size

Various experimental investigations of particle size following the explosive destruct of reactor fuel elements have been published^(5,6,7). In one case a full scale model of a NERVA core containing beaded fuel elements has been destroyed in a test carefully simulating proposed destruct systems.^(5,6) Though very important for the final analysis of any selected mission, the results of these experiments are not necessary for understanding the models presented in this report. The models developed are applicable to any debris size distribution evolved from either experimental data or theoretical considerations. To facilitate the handling of the particle size information in the physical interaction and radiological dose models particle sizes have been divided into twelve ranges according to their diameter as shown in Table II-1. The assumption is made that all particles in a given size range behave physically and biologically in essentially the same fashion as the geometric mean for that group.

TABLE II-1

DEBRIS SIZE CLASSES

<u>Class</u>	<u>Size Range Diam. (Microns)</u>	<u>Log Mean Diam. (Microns)</u>
12	0 - 10	3.16
11	10 - 20	14.11
10	20 - 50	31.62
9	50 - 100	70.71
8	100 - 200	141.10
7	200 - 500	316.20
6	500 - 1000	707.10
5	1000 - 2000	1,411.00
4	2000 - 5000	3,162.00
3	5000 - 10,000	7,071.00
2	10,000 - 20,000	14,110.00
1	20,000 - 50,000	31,620.00

2. Geographical Debris Distribution

The area over which fragments would be distributed will be strongly influenced by the velocities imparted by the explosion. Velocity increments in the pitch plane will affect space residence time whereas lateral increments serve to alter the angle between the orbital plane and the equator, creating a crossrange spread on the ground.

To calculate locations of ground impact points for particles, an engine destruct analyzer computer program has been developed by the Lockheed Missile and Space Co. (LMSC).⁽²⁾ The results of analysis using

this program include prediction of the location of reactor fragments and the particle number density on the ground. This information is required input for utilizing the dose models developed in this report. Readers are referred to reference⁽²⁾ for a discussion of the assumptions and methods used in the Lockheed analysis.

3. Debris Location

As in the passive re-entry case, impact location of debris can be predicted if the orbital lifetime is less than ten revolutions. For mission failure times and destruct velocity increments which result in longer orbital lifetimes impact location is indeterminate.

4. Character of Debris

In the explosive destruct case, it is assumed that the core is destroyed before operational heating effects occur. The only changes in debris character will be those due to the re-entry heating. The graphite-uranium mixture will undergo some ablation during the re-entry phase, as discussed in reference⁽³⁾, but no chemical or physical changes (other than size) in the character of the debris would be expected. Radioactive metallic materials from the core and engine structure would also be expected to undergo some ablation during the re-entry phase⁽⁴⁾, but with no significant effect on possible radiological interactions other than reduction in mass.

An additional characteristic of significance for small-size debris is the rate at which contained fission products can be eluted or dissolved in water or body fluids. If the uranium carbide beads retain their pyrocarbon coating the fuel material is very resistant to leaching of the contained fission products. Solubility tests reported by J. R. Lai* on beads irradiated to provide tracer levels of activity indicate that immersions of up to one week in distilled water, 0.1 HCl, isotonic saline or simulated gastric juice removes no detectable activity. These intact coated beads contained approximately 2×10^{11} fission product atoms. If leaching occurred in these experiments it would have been observed if greater than one part in 10^4 . Similar experiments with nonactivated beads have been carried out at Los Alamos Scientific Laboratory (LASL) where no change in the weight of beads held for 24 hours in monkey gastric juices have been observed.⁽⁸⁾ However, some mass loss has been observed in uncoated beads and it is possible that some fission product could be removed from beads damaged by the destruct process. Measurements of fission product loss from uncoated and broken coated beads are now in progress at the U.S. Naval Radiological Defense Laboratory.

*"Radionuclide Solubility from Simulated Nerva Debris" by J. R. Lai, USNRDL Technical Report (In preparation).

D. SUMMARY

Characteristics of re-entered debris which must be considered in evaluating radiological interaction probabilities and magnitudes have been identified in the preceding discussion for the passive and explosive destruct re-entry cases. Numerical values for these characteristics have not been given in most instances; such values will vary with the particular mission and failure mode considered and, consequently, are generated for each specific analysis as indicated in Volumes I and II of this series.

The development of complete radiological interaction models, however, does not require specific numerical values, but merely a recognition of the range over which parameter variation can reasonably be expected. In the sections that follow, typical debris parameters such as activity, areal distributions, etc., will be used to indicate, for example, the sensitivity of the analytical models to the various assumptions made. It should be remembered, however, that these "source" values are not associated with particular flight profiles, but serve merely as a computational convenience.

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SECTION II

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SECTION III

APPROACH TO DOSE MODEL DEVELOPMENT

A. INTRODUCTION

The conventional safety analysis approach in the nuclear industry is to estimate the maximum doses assuming the worst possible conditions all exist simultaneously. In contradistinction, the degree of radiological interaction associated with flight operations of a nuclear rocket engine have been derived using models that provide realistic estimates of the doses that may be delivered under credible accident conditions.

This section presents a brief discussion of why such an approach is needed along with the general method used to develop adequate dose models. The actual dose models for various situations are presented in later sections of this report.

B. BASIC CONCEPT OF DOSE MODELS

All of the models developed for determining doses from nuclear rocket engine debris include two basic components: the first component describes the likelihood that a given set of exposure conditions will exist; the second component describes the radiological dose that will be delivered to a human for those exposure conditions. This breakdown is shown schematically in Figure III-1.

The exposure conditions for each possible receptor cannot be precisely predicted. A range of conditions will exist which depend upon the impact location, abundance and size of the debris, and the population density. Hence, the prediction of an interaction situation is given as the probability that a given exposure condition will occur. This probability is combined with the radiological dose for given exposure conditions and the results expressed as the distribution of dose over the affected population group. From such a population dose distribution, the mission safety can be assessed if the radiological effects are known.

Figure III-1 Dose Model Components as Presented in This Report

C. BASIS FOR PROBABILISTIC APPROACH

1. General Comments

Radiation exposures are usually determined by considering four factors: (1) the nature and intensity of the source; (2) the distance between source and receptor; (3) the presence of any shielding; and (4) the duration of the exposure. For a situation in which these factors can be identified within reasonable limits, they are sufficient in themselves. However, the estimation of the biological interactions which may result from nuclear rocket engine flight failures is complicated by the more or less random nature of the deposition of system debris in the biosphere. The debris may come to earth in an uninhabited desert or in a large city. In the first case, the probability of direct exposure of individuals to the debris will be vanishingly small whereas in the latter case the probability may be extremely high. Furthermore, the location of source and receptor with respect to each other, the presence or absence of significant shielding, and duration of exposure cannot be identified within reasonable limits.

Thus the mathematical description of a dose to a receptor from exposure to nuclear rocket system debris must include, in addition to the usual four factors listed above, a fifth factor which defines the probability that the receptor and the debris will interact in a certain manner.

As described in Section II of this report and in others^(1,2) the source characteristics are determined primarily by the history of the debris prior to earth impact. That is, the specific activity, debris size, and spatial distributions are established by the mission characteristics and accident situation assumed, and by re-entry behavior of the debris. For debris of specified characteristics, the radiation exposure for a given time period is amenable to fairly exact calculation.

When the individuals or population potentially exposed to these sources are considered, it becomes obvious that the specifications of exposure durations, source-receptor separation distances, and shielding are not amenable to simple selection except for "worst-cases". It is relatively simple to assume, for example, that an individual will remain at a distance of one meter from a large fuel fragment for the remainder of his lifetime; and this might, in fact, be possible in the case of an invalid who remains immobile. This case is certainly not representative, however, of the exposure history of the great majority of those potentially affected by such sources. In fact, it would be expected that there would exist within an exposed population group a distribution

of doses, ranging from none to that of the lifetime dose cited above. The magnitude of any individual dose would of course vary with the habits of the individual as well as the deposition density and activity of the debris.

Similarly, when considering ingestion of insoluble debris, it would be simplest to assume that each fragment of the appropriate size range is ingested, and that it remained at a given point in the gut section for the duration of its passage through the gastrointestinal tract. Actually, the average residence time of debris in the gut will vary widely between individuals as will the amount ingested. Thus, the internal doses from this route to an exposed population group can also be expected to follow a distribution function ranging from zero to some maximum.

Therefore, to more accurately describe the degree of interaction associated with re-entering debris, functions which describe the probabilities for interaction between the debris and the receptor in a given manner have been developed. These probability functions are used in combination with dose rate functions which describe the dose rate for a particular source and geometrical configuration, and a second probability function which describes the duration of exposure as a function of time. This treatment is applied to each of the exposure routes presented in the following sections.

2. Factors Considered

Debris returned to earth would consist of fuel fragments ranging in size from micron dimensions up to an entire fuel element. A number of routes exist by which humans may be exposed to ionizing radiation from these fragments. Potential primary exposure routes would include: external exposures to the whole body and to the skin from deposited debris; internal exposures to the lung due to inhalation of fine particulates; internal exposures to the gastrointestinal tract during the passage of ingested particles or particles inhaled and subsequently swallowed; and exposure to various body organs from dissolution and entry into the blood stream of radionuclides released by inhaled or ingested particulates. Routes for secondary exposures might result from radionuclides being released by debris deposited on the ground and reaching man through his water and food chains. An outline of the dose routes considered are shown in Figure III-2.

Obviously, the probability and extent of interaction will vary with the character of the debris, as well as with the nature of the interaction and the characteristics of the population group. For example, a very low probability would be expected for inhalation of

III-5

Figure III-2 Flow Diagram of Possible Routes of Exposure to Reactor Debris

particles greater than 20 microns, or for ingestion or retention on the skin of fragments with dimensions on the order of several millimeters (see Section IV). In the case of skin contact, the duration of exposure would also be affected by the particle size mass, smaller particles having a longer mean contact time than larger ones. Clearance of particles from the lung is also a highly variable process and its variation with particle size is not a simple or well understood process.

Finally, the probability and extent of interaction will vary with the number density and characteristics of the affected population group. The affect of the former is obvious; the characteristics of the people, however, play a role somewhat more difficult to evaluate in a quantitative fashion. The factors of apparent significance for external exposures, for example, would include the fraction of time spent indoors (under cover), the range of motion with respect to deposited debris, the nature and extent of clothing and (for skin contact exposure) the frequency of bathing and physical activity. Ingestion doses would be expected to vary with the character of local agricultural practices and diet.

The distribution of population over the earth's surface, the fraction of land devoted to agriculture, and those other cultural factors which are required for determining dose and dose probability for the band of the earth's surface between 40°N and 40°S latitude are given in another report.⁽³⁾ Given the surface distributions in space and time for the range of fragment sizes following their return to the earth, and given knowledge of the distribution of people on the earth's surface and their customs with respect to dress, bathing, housing, food, and agricultural practices, it should be possible to estimate the probability of human exposure to deposited debris.

Section IV of this report develops the methods for determining the expected degree of interaction between engine debris and people and Section V presents the radiological dose models for determining the dose delivered for the postulated exposure conditions.

REFERENCES

SECTION III

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SECTION IV

PARTICLE EXPOSURE MODELS

A. INTRODUCTION

1. General

The mathematical functions which describe the probabilities for interaction in a given manner between the debris and the receptor, and the duration of exposure are presented in this section. These probability functions are used with functions which describe the dose rate for a particular source and geometrical configuration. This treatment is applied to each potential exposure route.

2. Exposure Situations Considered

The debris from nuclear rocket flight failures will emit alpha, beta, and gamma radiation; the alpha arising from the enriched uranium in the fuel and the beta and gamma radiation primarily from the fission products and to a much lesser extent from induced activity in structural materials. As is usually the case, the biological significance of the alpha radiation arises only from deposition inside the body. However, the beta and gamma radiation can be significant in both external and internal exposures.

Gamma ray exposure is the primary consideration for those fragments of the reactor core which are approximately 1000 microns or larger in diameter. For a nominal power history prior to failure, fragments of this size or greater can deliver significant external whole body gamma doses. Although particles in the size range from 500 to 1000 microns are too large to inhale, there is a slight probability of ingestion. If so, both internal beta and gamma doses then become significant. As the particle size becomes smaller, the probability of ingesting and inhaling a particle becomes larger due to an increasing deposition density and also to the decreasing effectiveness of natural barriers which tend to prevent entry of foreign particles into the body. Internally, the dose from beta irradiation is considered to be more important than that from gamma since all the beta decay energy is expended in a very localized volume close to the source.

The specific routes of human exposure to debris are discussed below:

a. External Gamma

The external gamma dose model is described in Section IV-B and is calculated on the premise that the receptors do not always remain stationary but are free to move about in some manner. The whole body gamma dose is calculated for each particle size class along with the probability that an individual in a given population group receives a given dose from each particle size class. From the dose and exposure probability for each individual the dose distribution within the total population group is calculated; the total population exposure is also determined.

b. Skin

Skin exposure (Section IV-C) considers both the probability of a falling particle striking and sticking to bare skin and the probability of that particle sticking for a given time. For a selected particle size a radiological dose is calculated corresponding to the estimated retention time. The dose and probability models may be adjusted to allow for the fraction of various parts of the body covered by clothing. Exposure probabilities also consider the fraction of time individuals are outdoors. It is assumed that persons who are indoors are not subject to exposure by this route.

c. Lung

The exposure model for the lung presented in Section IV-D below includes the probability that a particle will reach the alveolar ducts and alveolar sacs. This model compensates for those particles that are inhaled but become trapped in the nasal passages and upper respiratory tract. These trapped particles do not reach the lung but are elevated due to the ciliary action of the epithelium of the respiratory tract and are eventually swallowed, contributing to the gastrointestinal dose discussed below.

d. Gastrointestinal Tract

The exposure of the gastrointestinal tract is presented in Section IV-E arises from particles that are ingested via contaminated food and from particles contributed by inhalation and subsequently swallowed.

The beta dose to the gastrointestinal tract wall from the ingested particles is calculated on an assumed particle position in the gut cross section and an expected residence time in a fixed position. The probability of exposure of a given magnitude is determined by considering the probability of ingesting a particle (including the contribution from the respiratory tract), and the probable retention time in the lower large intestine. For the larger particles, a whole body gamma dose from the traveling source is also calculated.

e. Metabolized Isotopes

It is demonstrated in Section IV-F that all nuclides with the exception of the radioiodines can probably be neglected when considering organ dose due to any soluble fraction of the fission product inventory. That analysis considers that the activity is completely soluble in the body immediately upon intake. This approach recognizes the present unavailability of complete solubility data on debris materials and indicates that such data for the radioiodine is most important.

3. Cultural Factors

As mentioned previously, the flight path of the nuclear rocket is such that debris from an abort could potentially impact in any area between 40°N and 40°S latitude. The behavioral and cultural characteristics of the peoples inhabiting this area vary greatly. Such factors as the type of housing and daily activities in addition to the more obvious population density differences could affect the external gamma dose. Internal doses will vary depending upon food intake and cooking habits while external skin contact doses will be affected by such factors as clothing, skin conditions and bathing frequency. Other factors such as these could be considered in evaluating the exposure probability and the models can be elaborated to any extent needed to evaluate a particular mission. The required cultural characteristics of various population groups have been estimated and are presented in another report.⁽¹⁾

B. EXTERNAL WHOLE-BODY EXPOSURE

1. Probability of Exposure

As noted earlier an external whole body gamma dose model has been developed for receptors who are mobile; i.e., not fixed in a given

location for the entire exposure period. Using this model the number of people receiving at least a given exposure and the overall population exposure for external gamma exposure from engine debris deposited in a random manner on the ground can be determined.

If the nuclear engine re-enters passively or if the deposited debris from an explosive destruct is spread over such a large area that the spacing between individual particles is very large, then it is probable that most individuals will be affected only by a single particle of any one size class. In the development given below the assumption is made that only one particle of any size class will affect a receptor at a time.

As the deposition density becomes large the mean separation distance between particles becomes small and the basic assumption used in the development of the mobile receptor and dose probability relationships cannot be considered to be applicable. However, for a high particle density a uniformly contaminated plane can be used to approximate the large numbers of discrete sources. The use of the mobile receptor model is limited to those cases where, at least 95% of the time, the receptor is in a region that contains on the average no more than one particle of a given size class. This limitation is arbitrarily selected but will not result in underestimates of the dose received by the receptor. A separate model for cases where a uniformly contaminated plane is a useful approximation is described in Section IV-B-1-c. The dose contribution from both models is included in the overall individual dose and population dose calculations.

It is possible to define the average gamma exposure the receptor will receive from a single particle if the manner in which he moves with respect to the source can be described. Of course, any analytical model employed to describe the mobility of a large segment of any population must be recognized as limited in precision and scope; such a model, however, is more realistic than assuming arbitrarily and permanently fixed receptors.

For any normal individual, it can be stated that some location exists about which that individual spends more time than at other locations at which large fractions of their time are spent. If the activity of individuals can be defined by such centers of motion and a distribution function about them, then the average exposure of that individual can be described for a given separation between the center of motion and the source. An exposure so calculated is an average in the sense that the receptor's various positions represent an adequate sampling of the distribution function during a time interval in which the source strength is essentially constant.

Three distribution functions describing the motion of receptors about a fixed location were considered: (1) a uniform probability out to a fixed spatial limit; (2) a linearly decreasing probability out to a fixed distance; and (3) a Gaussian distribution. The uniform probability distribution has been eliminated on the basis of personal experience and logic. A linearly decreasing probability distribution might be valid for semi-invalids and infants with a limited range of motion. The Gaussian distribution, however, allows for infrequent occurrences of very large distances and, hence does not impose any limit on distance to which the individual may move. Of course it is possible to select other distributions; however, it is shown in Appendix B that the type of function used to describe the receptor's motion has a small effect upon the calculated dose in comparison to the extent of motion permitted. For this report a two-dimensional Gaussian distribution has been selected as a reasonable approximation.

a. Exposure to a Mobile Receptor from a Single Particle

It is first assumed that the individual's location with reference to a fixed center of motion (C in Figure IV-1) is given by the independent variables x and z both of which are Gaussian distributed with a common standard deviation.

The probability density (probability per unit area) weighted for the fraction of time, f_i , the standard deviation of motion, σ_i , applies for an individual or group of similar individuals in the activity class i is given by

$$y_i(x,z) = \frac{f_i}{2 \pi \sigma_i^2} \exp - \left[\frac{x^2 + z^2}{2\sigma_i^2} \right] \quad (IV-1)$$

The probability that an individual is within a given area defined by the points (x_1, z_1) , (x_2, z_1) , (x_1, z_2) and (x_2, z_2) is given by

$$P = \int_{z_1}^{z_2} \int_{x_1}^{x_2} y(x,z) dx dz$$

To calculate the exposure it is convenient to use polar coordinates centered on a source Q, see Fig. IV-1.

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Figure IV-1 Mobile Receptor Geometry
Source at Q , Receptor at R , Center of Motion at C

From Fig. IV-1, if a source, Q, is located a distance, S from the center of motion, C, then

$$x^2 + z^2 = r^2 + S^2 - 2 rS \cos \theta$$

and it is noted that

$$\frac{x + S}{r} = \cos \theta \text{ and } \frac{z}{r} = \sin \theta$$

$$\text{or, } x = r \cos \theta - s \text{ and } z = r \sin \theta$$

then

$$J \left\{ \frac{x, z}{r, \theta} \right\} = \begin{vmatrix} \frac{\partial x}{\partial r} & \frac{\partial x}{\partial \theta} \\ \frac{\partial z}{\partial r} & \frac{\partial z}{\partial \theta} \end{vmatrix} = (\cos \theta)(r \cos \theta) - (\sin \theta)(-r \sin \theta) = r$$

Therefore, the probability that an individual is in an area defined by θ_1 , θ_2 , r_1 , and r_2 is given by

$$P = \int_{r_1}^{r_2} \int_{\theta_1}^{\theta_2} g(r, \theta) r d\theta dr \quad (\text{IV-2})$$

where

$$g(r, \theta)_i = \frac{f_i}{2 \pi \sigma_i^2} \exp \left[-\frac{1}{2\sigma_i^2} (S^2 + r^2 - 2 Sr \cos \theta) \right] \quad (\text{IV-3})$$

Thus the probability that an individual is found in the annular region between r_1 and r_2 is

$$P_i = \frac{f_i}{2 \pi \sigma_i^2} \int_{r_1}^{r_2} \int_0^{2\pi} r \exp \left[-\frac{1}{2\sigma_i^2} (S^2 + r^2 - 2Sr \cos \theta) \right] dr d\theta \quad (\text{IV-4})$$

The exposure from an isotropic point gamma source, $D_j(d)$ can be represented as a function of distance, r , and particle size, j , (see Section V-C-4). The average exposure $D(S)_{j,i}$ to a mobile receptor, for a given separation distance, S , is given by equation (IV-5) for an individual (or group of individuals) in activity class i for the fraction of time, f_i , the standard deviation of motion σ_i applies.

$$D(S)_{j,i} = \frac{f_i}{2 \pi \sigma_i^2} \int_0^{\infty} \int_0^{2\pi} D_j(d) r \exp \left[-\frac{1}{2\sigma_i^2} (S^2 + r^2 - 2Sr \cos \theta) \right] dr d\theta$$

(IV-5)

b. Average Population Exposure

The separation distance S in equation (IV-5) cannot be explicitly derived for an individual since the spatial distribution of the particles and receptors is assumed to be random. However since large numbers of particles and people are considered a minimum average exposures can be specified using the following parameters:

- M_j the total number of particles of size j
- $\bar{m}_j$ the average number of particles per unit area of size j (number per square meter)
- N_o total number of people in a given population exposed to M_j particles
- $\bar{n}$ average population density (number per square meter)
- $\bar{n}_i$ average population density for activity group i (number per square meter)

Since a large number of particles and people are being considered, the average population density can be used to obtain the average dose distribution between members of an exposed population group. If it is assumed that the occurrence of receptors has a Poisson distribution, the average number of people in activity group i who are at a distance $S > S_{\max}$ from a single particle or closer is

$$\bar{z}_i = \bar{n}_i \pi S^2$$

(IV-6)

For M_j particles, the total number of people in the i^{th} group who are at a distance S or closer to any particle of size j is

$$N_i = M_j \bar{z}_i = M_j \bar{n}_i \pi S^2 \quad (\text{IV-7})$$

The lowest possible exposure any of these individuals may get is that corresponding to a separation distance S . Thus, the fraction of the total population, $[N_i/N_o]$, receiving at least a given exposure $D \geq D(S)_{j,i}$ from particles of size j is

$$\left[\frac{N_i}{N_o} \right]_{D \geq D(S)_{j,i}} = \frac{M_j}{N_o} \bar{n}_i \pi S^2 \quad (\text{IV-8})$$

but since M_j and N_o are for the same area.

$$\frac{M_j}{N_o} = \frac{\bar{m}_j}{\bar{m}} \quad (\text{IV-9})$$

(IV-8) becomes

$$\left[\frac{N_i}{N_o} \right]_{D \geq D(S)_{j,i}} = \frac{\bar{n}_i \bar{m}_j \pi S^2}{\bar{m}} = f(P)_i \bar{m}_j \pi S^2 \text{ for } 0 \leq S \leq S_{\max} \quad (\text{IV-10})$$

where $f(P)_i$ is the fraction of the population in activity group i and $S_{\max}$ is an arbitrary upper limit set on S so that $D(S_{\max})_{j,i} \approx 0$.

It is obvious that $[N_i/N_o]$ cannot be greater than 1 for any values of $S < S_{\max}$, i.e. $\bar{m}_j \ll 1$. The uniformly contaminated plane model (Section IV-B-1-c) is appropriate to calculate the average exposure to the total population in cases where $\bar{m}_j$ is not a small number.

An expression for an average population exposure from particles of size j can be derived as follows: As before, the average number of people from the i^{th} group in an area of radius S about a single particle is given by the continuous function:

$$\bar{z}_i = \bar{n}_i \pi S^2 \quad (\text{IV-6})$$

The rate of change of this average with respect to S is:

$$\frac{d\bar{z}_i}{dS} = 2 \bar{n}_i \pi S \quad (\text{IV-11})$$

Thus the average population exposure for a single particle of size class j and population group i is

$$[\text{P.E.}]_{j,i} = \int_0^S \max \frac{dz_i}{dS} D(S)_{j,i} dS \quad (\text{IV-12})$$

$$[\text{P.E.}]_{j,i} = 2 \bar{n}_i \pi \int_0^S \max S D(S)_{j,i} dS$$

where $D(S)_{j,i}$ is given by equation (IV-5). For the entire population the total population exposure (man roentgens) from single particles is:

$$\Sigma[\text{P.E.}](P) = \sum_i \sum_j 2 \bar{n}_i \pi M_j \int_0^S \max D(S)_{j,i} dS \quad (\text{IV-13})$$

c. Exposure to Multiple Particles

A basic assumption used in the development of the probability relationships is that no one person will receive a significant dose from more than one particle except within the framework of several centers of motion for different diurnal periods. As the particle density becomes large the separation distance between particles will become smaller and this assumption is no longer useful. However, as the number of particles becomes large the uniformly contaminated plane source may be used to represent a large number of discrete point sources.

The uniformly contaminated plane dose model is used when the distance between particles becomes comparable with σ_i . As an arbitrary guide to the use of the uniformly contaminated plane model in place of equation (IV-5), the following criterion is applied:

The uniformly contaminated plane model is used if the particle density results in more than one particle, on the average, being in a region within which a mobile receptor spends 95% of his time.

That is equation (IV-5) is used when

$$2.45 \sigma_i \leq \left(\frac{1}{\bar{m}_j \pi} \right)^{\frac{1}{2}}$$

Table IV-1 shows related values of $\sigma_i - \bar{m}_j$. If $\bar{m}_j$ exceeds the indicated value for a given σ_i the uniformly contaminated plane model is used rather than the mobile receptor model.

TABLE IV-1

LIMITS FOR MOBILE RECEPTOR MODEL

σ_i (meters)	$\bar{m}_j \times 10^6$ (particles/km ²)
10	532
20	133
40	33
60	15
80	8
100	5

As an expression for the gamma exposure from a uniform planar distribution of size j particles, $D(u.p.)_{j,i}$, is developed in Section V-B-4. Except for variation in local shielding appropriate to each activity group, all members of the exposed population receives the same dose. Members of the ith activity group exposed to a uniform distribution of size j particles receive a population exposure equal to:

$$[P.E.](u.p.)_{j,i} = N_o f(p)_i D(u.p.)_{j,i} \quad (IV-15)$$

The total population exposure from a uniformly contaminated plane is:

$$\Sigma [P.E.](u.p.) = \sum_i \sum_j [P.E.](u.p.)_{j,i} \quad (IV-16)$$

This population exposure is added to that given by equation IV-13 for single particle exposures.

$$\Sigma[\text{P.E.}] = \Sigma[\text{P.E.}](P) + \Sigma[\text{P.E.}](u.p.) \quad (\text{IV-17})$$

2. Duration of External Exposures

Depending upon the time of failure and the specific mission, radioactive engine debris may arrive on the ground within time periods ranging from a few minutes to many years. For material which re-enters with little time delay, most of the contained activity (about 90%) will decay within a few days after earth impact and more than 95% within one year. On the other hand, if the re-entry is delayed one year then only long lived isotopes will be present and about 15% of the residual activity will decay in the second year. Since it is impossible to determine how long a piece of engine debris or a receptor will remain in a given area it is necessary to determine doses received for a wide range of possible exposure times. The selected exposure times are:

- a. up to four days after initial exposure
- b. up to thirteen weeks after initial exposure
- c. up to one year after initial exposure
- d. up to 30 years after initial exposure

By definition, the four day dose is considered to be an acute exposure. It is the period of time so designated by the National Committee on Radiation Protection and Measurements⁽²⁾ on the assumption that no biological repair takes place within 96 hours of exposure. The thirteen week and one year intervals are selected to correspond to those periods used by the United States regulatory agencies for dose determinations. The 30 year period is selected since this is specified to be the genetically important time span by the International Commission on Radiological Protection (ICRP) and the Federal Radiation Council (FRC).

3. Cultural Factors

The use of the mobile receptor dose model requires that certain information be known concerning the daily activities of each population group considered. Consequently, available data for countries between 40°N and 40°S latitude were reviewed and these countries classified into "culturally similar" groups.⁽¹⁾ The number of culturally similar groups for the specific information required are as follows:

a. seven groups for estimating the degree of motion and the fraction of the population engaged in a specific activity;

b. three housing models for determining the degree of protection afforded by housing and industrial structures; and

c. five clothing models to describe the weight and amount of clothing worn by the group.

Each country is considered separately to determine the fraction of the population in each age or occupation group and to determine the total population and population density.

The model for determining the dose to a receptor that moves about a fixed center allows for more than one such fixed center. For all individuals at least two centers are considered; one is for the period of time they are assumed to be sleeping and the other (or others) for the fractions of each day they are engaged in various activities. In addition the model allows for various reduction factors that arise due to shielding by structures or occupations such as fishing or mining.

Exposure reduction factors for nine culturally similar groups and eight activity categories have been estimated.⁽¹⁾ These reduction factors are applied directly to the expressions giving the gamma exposure. For example from equation IV-5 the exposure from a single particle to members of an activity group i living in a cultural group g is

$$D(S)_{j,i,g} = W_g D(S)_{j,i} \quad (IV-18)$$

C. EPIDERMAL CONTACT

1. Contact Probability

The evaluation of the degree of interaction between individuals and nuclear rocket debris via direct skin contact requires that cultural characteristics of the potentially affected population and the dynamics of wind-blown particles be defined. Since particle distributions can be expressed in terms of the deposition density on the ground it becomes desirable to relate the number of particles interacting with an individual to the number of particles deposited per unit ground area.

A simple method for obtaining the ratio of the number of interacting particles to the deposition density has been published and is adequate for the purposes of this work.⁽³⁾ The model assumes the man, in a standing position, is composed of an assemblage of cylindrical parts which represent two arms, two legs, two hands, a trunk, lower head and upper head. The model also attempts to consider the effects of local winds at ground level although it assumes that the local meteorology does not affect the terminal velocity of particles.

The total ground area, $a(t)_j$, shielded from falling particles of size j can be expressed by

$$a(t)_j = a(H) + \frac{V(W)}{V(T)_j} a(V)_k \quad (\text{cm}^2) \quad (\text{IV-19})$$

where

- $a(H)$ = Settling area of horizontal surfaces, (cm^2)
- $a(V)_k$ = Vertical cross sectional area of the cylindrical representing body part, k , (cm^2)
- $V(W)$ = Wind velocity 1 meter from ground, (m/sec)
- $V(T)_j$ = Terminal velocity of size j particle in still atmosphere (m/sec)

To obtain the mean number of particles striking and sticking to an individual, equation (IV-19) is multiplied by the ground particle deposition density, and the impaction and sticking efficiencies as explained below.

The impaction efficiency^(4,5) corrects for those particles which travel around the body with the air flow rather than intercepting the body (as they would if the particle traveled a straight line). The sticking efficiency allows for those particles which strike the body but bounce off. It is assumed that both the impaction efficiency and sticking efficiency are 100 percent for particles settling on horizontal surfaces. Equation (IV-20) gives the mean number of particles of size j striking and sticking to an individual

$$\bar{\Psi}_j = 10^{-4} \bar{m}_j \left[a(H) + \frac{V(W)}{V(T)_j} \xi_j \sum_{k=1}^{k=9} a(V)_k \Gamma_{k,j} \right] \quad (IV-20)$$

where

$\bar{\Psi}_j$ = Mean number of particles of size, j, striking and sticking to an individual.

$\Gamma_{k,j}$ = Impaction efficiency for body part, k, and particle size j

$\bar{m}_j$ = Mean particle density of deposition on ground for size j, (particle per meter squared)

ξ_j = Sticking efficiency, the fraction of impacted particles of size j that are retained on the skin.

To calculate $V(T)_j$ for particles $\geq 50\mu$ in diameter, the following equation⁽⁶⁾ derived from fallout data is used.

$$V(T)_j = \frac{1.325 \times 10^{-2} b \eta}{\rho_o} [\log(b \gamma_j \times 10^{-4} + 1.163)]^3 \quad \left(\frac{m}{sec}\right) \quad (IV-21)$$

where

b = $[2\rho_o g_o (\rho - \rho_o) / \eta^2]^{1/3}$, (cm⁻¹)

η = Air viscosity, (g/cm-sec)

ρ_o = Air density, (g/cm³)

g_o = Acceleration of gravity, (cm/sec²)

γ_j = Particle diameter of size j debris, (microns)

Stokes law is used for particles < 50 microns in diameter.

To obtain efficiency of impaction, Γ_{kj} , the empirical relationship by Landahl and Herrman⁽⁴⁾ is used:

$$\Gamma_{kj} = \frac{K^3 \alpha}{\beta + K^2 + K^3 \alpha} \frac{\gamma \times 10^{-4}}{\Omega_k} \quad (\text{IV-22})$$

where

$$K = \rho \gamma^2 \times 10^{-8} V_w / \Omega_k \text{ (g/cm-sec)}$$

$$\alpha, \beta = \text{Constants (see below).}$$

$$\Omega_k = \text{Diameter of body part } k, (\text{cm})$$

The largest cylinder tested experimentally by Landahl and Herrman in developing equation (IV-22) was 1.8 cm; the largest particle tested was about 30 microns in diameter. However, the theoretical work of Rantz as discussed by Humphrey⁽³⁾, treats data for larger cylinders and particle sizes. Using this model, a study was made using equation (IV-22) and the data from Rantz⁽⁵⁾ to determine the combination of values of α and β that would best reproduce the theoretical curves for impaction efficiency for the required range of diameters and particle density. The results of this study are given in Appendix C. The values for the constants α and β given below allow a fit to within a few percent over the entire range, and were selected for use in this study.

$$\alpha = 300, (\text{cm-sec/g})$$

$$\beta = 1.0 \times 10^{-7}, (\text{g}^2/\text{cm}^2\text{-sec}^2)$$

Results obtained using this model were compared with experimental data on deposition of wind-borne particles on human skin.⁽⁷⁾ The largest particle size used in this study had a mass median diameter of 6.5μ , and showed deposition efficiencies on hairy arms of .2% and 1% for wind speeds of 2 and 5 mph, respectively. Results on hairless surfaces showed an efficiency of 0.07% at 5 mph. As indicated in Appendix C, the analytical model used in this study predicts no impaction for this particle size and the wind speeds used. Additional investigation of this phenomenon is needed.

If impaction takes place, only a fraction of the striking particles will stick for a significant time period. Since experimental data on this factor is largely unavailable it is assumed that all particles smaller than 50 micron diameter will stick. For larger

particles it is assumed that the fraction of particles sticking ξ_j is inversely proportional to their diameter.⁽⁸⁾

$$\begin{aligned}\xi_j &= 1 \text{ if } 0 < \gamma_j < 50 \\ \xi_j &= \frac{50 \text{ Microns}}{\gamma_j \text{ (microns)}} \text{ if } \gamma_j > 50\end{aligned}\tag{IV-23}$$

If it is assumed that the particle deposition density varies according to a Poisson distribution (Section IV-B-1-b) then the probability that M_j particles will strike and stick to an individual is given by

$$P(M)_j \text{ skin} = \frac{(\bar{\Psi}_j)^{M_j}}{M_j!} \exp(-\bar{\Psi}_j)\tag{IV-24}$$

2. Duration of Exposure

Some experimental work has been done to determine how long a particle might remain on the skin. ORNL has reported work with wax and plastic spheres loaded with Zn-CdS (Ag).⁽⁹⁾ This work indicates that the surface conditions of the skin (mainly oiliness and perspiration), and the degree of physical activity of the individual are the most important parameters. Apparently the density of the particle is not as effective as these parameters in determining the retention time. The preliminary results obtained indicate that the number of particles remaining on the skin decreases exponentially with time.⁽⁹⁾ Thus the probability, $P(t)_j$, of a particle of size j remaining on the skin for at least time t is

$$P(t)_j = \exp(-t/\Lambda_j)\tag{IV-25}$$

where Λ_j is the mean life on the skin of particles of size j .

Over a range of particle sizes varying from 40 to 1000 microns retention time has been measured at ORNL as shown in Fig. IV-2.⁽⁹⁾ An analytical relationship fitting this data is given below, and is included in Fig. IV-2.

Figure IV-2 Particle Retention on Skin (Ref. 9)

$$\Lambda_j = \frac{6.02}{\gamma_j 0.16} + \frac{1.14 \times 10^5}{\gamma_j 2.65} \quad (\text{hours}) \quad (\text{IV-26})$$

The mean number of particles that may be expected to stick for at least time, t , is given by the product of equations (IV-2) and (IV-7). The probability, then, that M_j particles will strike and reside on an individual for at least time t is

$$P(M_j, t) = \frac{[\bar{\Psi}(0)_j P(t)_j]^{M_j}}{M_j!} \exp \left[-\bar{\Psi}(0)_j P(t)_j \right] \quad (\text{IV-27})$$

where $\bar{\Psi}(0)_j$ is the mean number of particles retained by a clothed individual, see below.

3. Cultural Factors

The factors of apparent significance for external skin contact exposures, include the fraction of time spend indoors (under cover), the nature and extent of clothing, the frequency of bathing, and physical activity.

Certain cultural and hereditary aspects affecting the duration of epidermal exposure can only be considered if detailed information is available on a particular geographic locale. Any cultural factor that results in quick removal of particles from the body (i.e., bathing) will reduce retention time and the radiological dose to the skin. Such hereditary factors as hairiness or skin oiliness will increase retention. Since these factors cannot be quantitatively assessed without more detailed information, equations (IV-25) and (IV-26) are used for evaluation of skin exposure time in this report.

The development of Equation (IV-19) assumes a nude subject. For the vast majority of the potentially exposed populations, this is a highly conservative assumption, since very few tropical cultures, despite climatic conditions enjoy such complete freedom from the confinement of clothing. Thus, it becomes desirable to introduce a parameter which recognizes that effect in an appropriate manner. Several alternative approaches for the inclusion of this factor exist. The optimum approach would consider separately the probabilities of

exposure to bare and clothed skin areas, and include the computation of the radiological dose to both. Implicit in this approach is the use of data on the retention probability and exposure time for various types of clothing. These data are not presently available.

Clothing patterns for potentially affected geographic areas have been identified⁽¹⁾, and sketches of the four clothing models representative of these areas are presented in Figure IV-3. From these sketches and from data contained elsewhere⁽¹⁾ estimates have been made of bare and clothed body dimensions and areas which are tabulated in Table IV-2.

If the fraction of time spent outdoors and the fraction of exposed skin on body part k is considered then the mean deposition of particles of size j striking and sticking as given by equation (IV-28) is

$$\bar{\Psi}(0)_j = 10^{-4} f_o \bar{m}_j \left[f(H) a(H) + \frac{V(W)}{V(T)_j} \xi_j \sum_{k=1}^9 [f(V)_k a(V)_k r_{kj}] \right] \quad (IV-28)$$

where

f_o = Fraction of time spent outdoors

$f(H)$ = Fraction of horizontal body surface exposed

$f(V)_k$ = Fraction of vertical area, k, exposed

The incorporation of these factors into the complete skin dose - probability distribution is described in Section VI-B.

It should be noted that use of f_o assumes that no particles are deposited indoors. The effects of shielding by clothing is discussed in Section VI.

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Figure IV-3 Clothing Patterns

TABLE IV-2

BODY SURFACE AREAS (cm²)

COMPONENT DIMENSIONS, cm	SKETCH A		SKETCH B		SKETCH C		SKETCH D	
	BARE	CLOTHED	BARE	CLOTHED	BARE	CLOTHED	BARE	CLOTHED
Settling Area, a(H) 45 x 15	----	675	225	450	225	450	675	----
Impaction Area a(V) _k								
Head: 12 x 15	----	180	180	----	180	----	180	----
Face: 15 x 15	225	----	225	----	225	----	225	----
Arms: 60 x 10(2)	----	1200	800	400	800	400	1200	----
Hands: 15 x 10(2)	300	----	300	----	300	----	300	----
Legs: 80 x 15(2)	----	2400	----	2400	1200	1200	1200	1200
Trunk: 75 x 35	----	2625	----	2625	----	2625	1750	875
$\Sigma a(V)_k$	<u>525</u>	<u>6405</u>	<u>1505</u>	<u>5425</u>	<u>2705</u>	<u>4225</u>	<u>4855</u>	<u>2075</u>

D. LUNG EXPOSURE

1. Probability of Inhalation

The potential problems from inhalation may arise in two ways. For the smallest size class of particles (0 - 10 μ) settling velocities are low enough to permit a reasonably durable particle suspension to exist in air and inhalation of this suspension can occur. Larger wind-driven particles of up to perhaps 1 mm in diameter can impact on the breathing zone, and be inhaled via nose or mouth. Each of these mechanisms is considered in subsequent sections.

a. Suspension Inhalation

Only for the smallest particle size class (0 - 10 μ)* is inhalation of suspended particles a likely mechanism of interaction. A graphite particle with the log mean diameter of this size range (3.16 μ) would have a terminal velocity of about .06 cm/sec; a uranium carbide particle of this size, about 0.3 cm/sec.

Using the average ground deposition density (number of particles per unit area) for this size class, $\bar{m}_{j=12}$, and the settling velocities, $V(T)_{j=12}$, the mean time-integrated concentration, $\bar{X}_{j=12}$, can be calculated from:

$$\bar{X}_{j=12} = \frac{\bar{m}_j \text{ (particles/m}^2\text{)}}{V(T)_{j=12} \text{ (m/sec)}} \left(\frac{\text{Particle-Sec}}{\text{m}^3} \right) \quad (\text{IV-29a})$$

The number of particles inhaled per person is the product of $\bar{X}_{j=12}$ and the breathing rate. In view of the low settling velocity, any exposure would take place over an extended time period and consequently the breathing rate, B, of 20 m³/day or 2.31 x 10⁻⁴ m³/sec for the ICRP standard man is used. Thus, the mean number of particles of size class 12 inhaled per person, $\bar{\Psi}'(I)_{j=12}$, for suspended particles in this size range is given by

$$\bar{\Psi}'(I)_{j=12} = \frac{\bar{m}_j B}{V(T)_{j=12}} \quad (\text{IV-29b})$$

*Classification of particle sizes are given in Table II-1

b. Impaction Inhalation

The analysis in Section IV-C can be extended to provide estimates of the number of particles impacting in the breathing zone. Some modifications are required, however, due to the effect of the inspiration velocity on impaction efficiency, and the fraction of time during which inspiration takes place. Thus Equation (IV-20) is modified to produce the following relationship for size classes larger than class 12.

$$\bar{\Psi}_j'' = 10^{-4} \frac{V(W)}{V(T)_j} f(I) (f(N) \Gamma_j^* a^* + f(M) \Gamma_j^{**} a^{**}) \quad (IV-30)$$

where

- Γ_j^* = impaction efficiency for the nasal area
- Γ_j^{**} = impaction efficiency for the mouth area
- a^*, a^{**} = nasal and oral breathing areas respectively (cm²)
- $f(I)$ = fraction of time inhalation occurs
- $f(N)$ = fraction of inspired volume occurring via nose
- $f(M)$ = fraction of inspired volume occurring via mouth

Inhalation via impaction is considered to occur only with particles up to 1 mm in diameter ($j \leq 6$). Since inhalation occurs during about 45% of the respiratory cycle, a value of 0.45 was selected for $f(I)$. Γ_j^* is equal to 1 for $j = 9$ to 11 and equal to 0 for $j \leq 8$. Γ_j^{**} is equal to 1 for all size classes up to $j \leq 6$. Use of these impaction efficiency values assumes an effective rejection by the nasal passage of fragments larger than 100 μ , and trapping of all particles smaller than 1 mm entering the oral cavity. The values of 0.9 and 0.1 may be assumed for the fraction of time inhalation which occurs via nose or mouth, respectively; these values are arbitrary. However, the sensitivity analysis given in Appendix C shows that changing these ratios has little effect on the significance of the final lung dose calculations. Values of a^* and a^{**} of 10 and 15 cm², respectively, may be assumed to reflect a larger area of influence (created by the inspiratory velocity) than would be given by the physical orifice size of nose and mouth openings.

It should be noted at this point that the larger particles entering the nose and mouth do not contribute to the lung dose. As described in Section IV-D-2, particles much greater than about 10μ in diameter will not gain access to the deep lung, but may be swallowed. The dose from the larger particle sizes will be considered in Section IV-E on gastrointestinal exposure.

c. Total Inhalation Exposure

Since the nasal passage acts as a filter for the larger particle sizes it is convenient to calculate the particle exposure for each route of entry into the body separately for suspension and impaction inhalation. From equations IV-29 and IV-30 $\bar{\Psi}^*$ the mean number of particles inhaled via the nose is:

$$\bar{\Psi}^* = f(N) \left[\bar{\Psi}'(I)_{j=12} + f(I) \sum_{j=9}^{j=11} 10^{-4} \bar{m}_j \frac{V(W)}{V(T)_j} \Gamma_j^* a^* \right] \quad (IV-31a)$$

The mean number of particles inhaled via the mouth is

$$\bar{\Psi}^{**} = f(M) \left[\bar{\Psi}'(I)_{j=12} + f(I) \sum_{j=6}^{j=11} 10^{-4} \bar{m}_j \frac{V(W)}{V(T)_j} \Gamma_j^{**} a^{**} \right] \quad (IV-31b)$$

2. Lung Exposure

Not all particles inhaled particles reach the lung. A model for inhaled particles has been given⁽¹⁰⁾ which attempts to describe mathematically the fate of particles after they are inhaled. Figure IV-4 shows a compartmentalized model of the respiratory system showing the various routes of inhaled particles. Particles deposited in the nasal passages and upper respiratory system are assumed to be swallowed because of the ciliary action of the epithelium in these regions.

Pattle⁽¹¹⁾ measured the penetration of particles in the nasal and mouth cavity by drawing particle-laden air through the nose and out the mouth. From the data obtained, he found that the following empirical relationship represented a reasonable fit for particles of size j :

Figure IV-4 Schematic Representation of the Respiratory System

$$F_j^* = 0.95 \left[1 - 0.217 \ln \frac{\gamma_j^2 W}{B} \right] \quad (\text{IV-32})$$

where

F_j^* = fraction of particles of size j passing through the nose

γ_j = diameter of particle of size j , (microns)

W = air flow, (liters/min)

B = empirically derived constant $20.2 \frac{\text{micron squared liters}}{\text{min}}$

For the standard man the breathing rate is 20.8 liters/min during the working part of the day; thus, equation (IV-32) reduces to

$$F_j^* = 0.95 \left[1 - 0.434 \ln (1.015 \rho^{\frac{1}{2}} \gamma_j) \right] \quad (\text{IV-33})$$

where ρ is the specific gravity of the particles.

The specific gravity term is added to correct for particle density as suggested by the work of Landahl.⁽¹²⁾ Figure IV-5 is a graph of Equation (IV-33). It is of particular interest to note that for specific gravities of 10 and 2.1, the particle diameters corresponding to $F_j^* = 0$, are just over 3 and 6.7 microns, respectively. This supports the exclusion of size $> 10\mu$ when considering inhalation via the nasal passage.

Landahl⁽¹²⁾ calculated the deposition of particles in various parts of the respiratory system. Since the experimental results of Pattle was available, Landahl's calculation for deposition in the mouth-nose region was not used although the agreement is reasonable. Considerable experimental data for deposition in other parts of the human respiratory system than the mouth region exist⁽¹³⁾, however, it is difficult to formulate this data into a mathematical model. Therefore, recourse has been made to calculated values such as given in Landahl's analysis. Figure IV-6 shows the range of Landahl's calculations for deposition in the upper respiratory tract for four assumed tidal

Figure IV-5 Fraction of Particles
Passing Nasal Passages.

Figure IV-6 Deposition in Respiratory Tract
From Pharynx to Bronchi

volumes. The straight line was used to represent the bronchial deposition data for particles larger than 1 micron in diameter. Deposition of the particles smaller than 1 μ in the bronchial tract is neglected for simplicity. This results in a small overestimate of the lung dose. An equation has been fitted to the data above 1 micron.

$$F(\ell)_j = 1 - 0.33 \ln \frac{\gamma_j \rho_j^{\frac{1}{2}}}{1.1} \quad (\text{IV-34})$$

where $F(\ell)_j$ = fraction of particles not deposited in the upper respiratory tract. Again, the specific gravity term is inserted to correct from the unit density used by Landahl in the calculations.

An estimation of the fraction of particles passing into the lower respiratory tract that are deposited in the alveolar ducts and sacs can also be obtained from Landahl's work. Figure IV-7 shows the calculated deposition for four different tidal volumes in alveolar ducts and sacs. The trend of the data is shown by the line which represents a linear fit through the calculated points and is represented by

$$F(\text{av})_j = 0.222 \ln \frac{\gamma_j \rho_j^{\frac{1}{2}}}{0.14} \quad (\text{IV-35})$$

where $F(\text{av})_j$ = fraction deposited in the alveolar ducts and sacs.

The average number of particles of size j reaching the lung via the nose and deposited in the alveoli, $\bar{\Phi}_j^*$, is given by the products of the fractions given in Equations (IV-33) through (IV-35) and the mean number of particles of size j inhaled via the nose, i.e.

$$\bar{\Phi}_j^* = f_j^* \bar{\Psi}^* F_j^* F(\ell)_j F(\text{av})_j \quad (\text{IV-36})$$

where f_j^* is the fraction of inhaled particles $\bar{\Psi}^*$ of size j .

Figure IV-7 Retention in Alveolar Ducts and Sacs

Similarly, the average number of particles reaching the lung via the mouth and being retained in the alveoli is given by

$$\bar{\phi}_j^{**} = f_j^{**} \bar{\psi}^{**} F(\ell)_j F(av)_j \quad (IV-37)$$

where f_j^{**} is the fraction of inhaled particles $\bar{\psi}^{**}$ of size j .

Then, the average number of particles of size j reaching the lung via inhalation and being retained in the alveoli is the sum of Equations (IV-36) and (IV-37),

$$\bar{\phi}_j = \bar{\phi}_j^{*} + \bar{\phi}_j^{**} \quad (IV-38)$$

Figure IV-8 shows the combined retention in the lungs as a function of size for both nose and mouth entry. The lower curve gives the deposition for nose entry and is the product of Equations (IV-33), (IV-34) and (IV-35). The upper curve gives retention for mouth entry and is the product of Equations (IV-34) and IV-35).

For mouth-breathed particles, the maximum size retained in the lung is given in Figure IV-8 by $\gamma \sqrt{\rho} = 23\mu$. For a uranium carbide fuel particle with a specific gravity of about 10, this corresponds to 7.3μ ; for a particle of graphite the maximum size particle retained is 15μ . Thus, only size classes $j = 11$ and 12 need be considered in calculation of lung dose. If the particles are assumed to follow a Poisson distribution and $\bar{\phi}_j$ is the mean inhalation exposure for the population, the probability that an individual will inhale M_j particles is given by

$$P(M)_j \text{ Lung} = \frac{(\bar{\phi}_j)^{M_j}}{M_j!} \exp -\bar{\phi}_j \quad (IV-39)$$

For ease of calculation in comparing different specific missions only the mean particle exposure per person need be considered.

Figure IV-8 Fraction of Inhaled Particles Retained in Lung

3. Duration of Inhalation Exposure

For those particles reaching the lung, a biological half-life of 120 days given by the ICRP⁽¹⁴⁾ for insoluble materials is used. It is expected that although this would vary among individuals the assumption of a longer exposure time would not affect the significance of the calculated dose as shown in Section VI-C.

4. Cultural Factors

In general, the influence of cultural factors upon the lung dose to a population has not been considered because the reduction in the amount of debris entering a structure cannot be estimated. However, it is likely that structures would provide some protection from this mode of interaction.

E. INGESTION EXPOSURE

1. Contact Probability

Ingestion of radioactive rocket engine debris can occur from swallowing those particles that become trapped in the upper respiratory tract or by eating foods containing particles. The probability of ingesting a particle via the inhalation route is evaluated using the model developed for determining lung exposure discussed above and assuming that the particles retained in the nasal passages and upper respiratory system are swallowed because of the ciliary action of the epithelium in these regions.

$$\bar{\Psi}(s)_j = \Psi^*(1-F_j^*) + \Psi^*F_j^* (1-F(l)_j) + \Psi^{**}(1-F(l)_j) \quad (IV-40)$$

which reduces to:

$$\bar{\Psi}(s)_j = \Psi^*(1-F_j^* F(l)_j) + \Psi^{**}(1-F(l)_j) \quad (IV-41)$$

Figure IV-9 shows the likelihood of swallowing inhaled particles as a function of size. It is seen that all inhaled particles for which the parameter $\gamma \sqrt{r_0}$ is greater than 22 microns will be swallowed, i.e., graphite particles > 15 microns and fuel particles > 7 microns.

Direct exposure of the gastrointestinal tract from particles which have stuck to ingested plants can be estimated by considering the retention of particles on food plants, processing or cleaning of food before ingestion, the time lag between particle deposition on food and ingestion, and the estimated food consumption of the individual. The mean number of particles of size j ingested per person is given by⁽¹⁰⁾

$$\bar{\Psi}(H)_j = \sum_c \bar{m}_j f(r)_j Y_c f(p)_j \quad (IV-42)$$

where

$\bar{\Psi}(H)_j$ = mean number of size j particles ingested per person

$\bar{m}_j$ = average number of particles deposited per unit ground areas meters⁻²

$f(r)_{j,c}$ = fraction of particles of size j retained on foliage of crop c

Y_c = area planted in crop c needed to feed one person meters per person

$F(p)_{j,c}$ = reduction factor for processing (fraction of particles removed in preparing crop for eating such as peeling, washing, etc., varies with type of crop).

Equation (IV-42) applies chiefly to leafy type vegetable crops and crops such as berries, fruits eaten without peeling, green beans, etc., where the edible part is exposed directly to particles. The factor, $f(r)_j$ can be determined for certain crops from the Stanford Research Institute study⁽¹⁵⁾ of volcano fallout in Costa Rica, which includes particle removal effects of rain and wind. The factor, Y_c , has been estimated for regions of the world between 40°N and 40°S, country by country, in NUS-230.⁽¹⁾ The factor of $F(p)_j$ is expected to be extremely variable with culture, region, type of crop, etc., and is impossible to determine except through detailed study of specific localized areas and cultures. The total average exposure to particles

Figure IV-9 Fraction of Inhaled Particles Swallowed

is the sum of the contributions from inhalation and ingestion which is

$$\bar{\Psi}(g)_j = \bar{\Psi}(s)_j + \bar{\Psi}(H)_j \quad (\text{IV-43})$$

If the particles are again assumed to follow a Poisson distribution in areal deposition as before, then the probability that M_j particles of size class j reach the gastrointestinal tract of an individual is given by

$$P(M_j)(g) = \frac{[\bar{\Psi}(g)_j]^{M_j}}{M_j!} \exp \left[-\bar{\Psi}(g)_j \right] \quad (\text{IV-44})$$

However, for simplification of analysis, possible radiological doses can be calculated on the basis of the mean number of particles reaching the gastrointestinal tract per person.

2. Duration of Exposure

The transit time of the particles through the gastrointestinal tract of the individual will determine the duration of exposure to be used to calculate the radiological dose. Report of Committee II of the ICRP⁽¹⁴⁾ gives average transit times through portions of the gastrointestinal tract for the standard man as follows:

Stomach - 1 hr

Small Intestine - 4 hr

Upper Large Intestine (ULI) - 8 hr

Lower Large Intestine (LLI) - 18 hr

Studies of gross transit time were conducted at the Argonne Cancer Research Hospital.⁽¹⁶⁾ Table IV-3 shows the results following ingestion of insoluble ceramic spheres (30-40 microns diameter) containing Cs^{134} .

TABLE IV-3(32)

EXCRETION OF Cs¹³⁴ MICROSPHERES

Subject	Per cent of dose remaining on day +				
	2	4	6	8	10 days
1	54	.1	.1		
2	67	.1	.1		
3	69	2.5	.1	.1	
4	2	0.2	.1	.1	
5	8	0.3	.1	.1	
6	8	0.1	.1		
7	12	.1	.1	.1	
8	20	.1	.1		
9	53	2.5	.1	.1	
10	9	9.0	.1	.1	
11	100	83.0	26.0	6.0	0.2
12	94	12.0	-	-	0.9
Average	41	9.0	2.5	0.6	0.2

The average of the various values in Table IV-3 indicate that to a first approximation passage of material through the entire gastrointestinal tract can be adequately represented by a simple exponential expression, as shown in Figure IV-10.

Figure IV-10 Average Fraction of Microspheres Remaining in G.I. Tract

It is therefore assumed that the probability of a particle remaining in the gastrointestinal tract for at least t hours can be represented by

$$P'(g) = \exp [-\lambda t] \quad (\text{IV-45})$$

where:

$P'(g)$ = probability that a particle is retained for at least t hours in the gastrointestinal tract

λ = the excretion coefficient for an individual (hours^{-1})

t = time of residence in gastrointestinal tract, (hours)

Figure IV-11 shows the distribution of the values of λ for different individuals as calculated from the data contained in Table IV-3 assuming Equation (IV-45) holds for each individual. These data show that the retention times are approximately normally distributed with a mean value of $\bar{\lambda} = 0.033 \text{ hr}^{-1}$ and a standard deviation of (15.6 hrs). The average retention time of particles in the gastrointestinal tract then is $\frac{1}{\bar{\lambda}} = 30 \text{ hrs}$, in good agreement with the ICRP choice of 31 hours.

A recent report from LASL⁽¹⁷⁾ describes a similar study in which both high density UC_2 spheres and ceramic spheres with a specific gravity of 3.0 were fed simultaneously to human subjects. These spheres, with diameters of the order of 150 - 175 μ , were passed simultaneously by 52 out of 57 subjects. The mean gross transit time for these subjects was 34.5 hrs, with a standard error of 16.6 hours again in good agreement with both the ICRP and Argonne Cancer Research Hospital data.

3. Cultural Factors

The major cultural factor that influences the ingestion via the food route is the area of land devoted to the planting of a certain crop and required to feed a person. (It is not necessary to correct such figures for multiple cropping since only a single crop could be affected by any single mission failure.) The crop yield, i.e., the area planted in a given crop to feed one person, is given in Ref⁽¹⁾ for the geographical locales of interest.

Figure IV-11 Distribution of Individual Excretion Coefficients

F. METABOLIZED ISOTOPES

1. Exposure Probability

Human exposure to radionuclides dissolved from debris particles could occur by either of two routes: (1) direct entry via dissolution of inhaled or ingested particles; (2) indirect entry of nuclides which have been eluted from deposited debris and reach man via the water or food chain.

The exposure to soluble radionuclides by the direct route depends entirely upon the solubility of these nuclides in lung and gastrointestinal tract fluids. Some preliminary information is available on the leaching of nuclides from intact fuel beads and from the TNT test debris.* Additional experimental work, using simulated fragments exposed to the post-operational and re-entry environment, is being carried out to determine the solubility rates for individual nuclides of particular biological interest.

The probability of significant exposure to man from radionuclides via the indirect route seems low. Even if by weathering, these materials become available to man via his food chain, the additional time and inherent dilution involved in these processes would reduce the radioactivity reaching man to low levels. On this basis it is believed that only the direct route need be considered at this time; and, in the absence of data on the extent of dissolution and on the solubility rate the very conservative total inventory approach outlined below is used.

2. Maximum Exposure

An indication of the significance of the dissolution of individual radionuclides can be obtained by considering the maximum available inventory in a single particle that is representative of the largest size class particle which can be taken into the body. For this size class (500 - 1000 μ), the maximum inventory of selected nuclides for a 707 μ diameter fragment is presented in Table IV-4. This inventory was computed on the basis of averaging fission product abundances over the entire core and allowing 10^5 sec decay following a 10 minute reactor operating time at 1000 MW. Also presented in this table is the dose per ingested microcurie for a single intake of each of the

*Personal communication from E. Freiling, USNRDL.

TABLE IV-4

SOLUBLE NUCLIDE EVALUATION

Nuclide	Activity* (μ c)	Infinite Organ Dose** per Ingested μ Curie (mrad)		Infinite Organ Dose From Contained Activity (mrad)
Sr-89	1.6	132	(Bone)	212
Sr-Y-90	0.0088	1335	(Bone)	11.8 (first year dose)
Sr-Y-91	140	5.35	(Bone)	750
I-131	4.7	1850	(Thyroid)	8,700
I-133	110	520	(Thyroid)	57,000
Cs-Ba-137	0.0096	43.5	(Total Body)	0.416
Ba-La-140	8.3	16.6	(Bone)	138
Ce-Pr-144	0.34	0.485	(Bone)	0.165

*Average activity in 707 μ diameter particle 10⁵ seconds after shutdown following 10 minutes operation at full power and no fission product loss.

**Calculated for single intake (except for Sr-90 which is dose in first year) using ICRP values, see Section V-C-3.

nuclides (using equations developed in Section V-C-3) and the organ dose that would result if all of the radioactive material were soluble. It is obvious from the data in this table that only radioiodines could possibly create a potential problem of this kind; other nuclides examined (which include the more hazardous fission products) are not present in sufficient amounts to create a significant dose, even if completely soluble in body fluids. For small inhalable particles (<20 μ), the total inventory is too small for even radioiodines to present a thyroid dose problem.

Thus, under the conditions assumed for a significant organ dose to be delivered, a particle must survive post-operational and re-entry heating with a radioiodine inventory substantially intact, the particle must be large and must be ingested within a time period short compared to the radioiodine half life, and the radioiodine must be dissolved from the debris particle during the time of passage through the gastrointestinal tract.

Not enough information is presently available on the loss of radioiodine during the re-entry of explosively destroyed core material, nor on the dissolution rate and extent of iodine from such debris. For the present, therefore, it should be assumed that as an upper level each individual who would ingest a debris fragment would receive a thyroid dose from the total iodine inventory in the fragment at impact.

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SECTION V

RADIOLOGICAL DOSE MODELS

A. INTRODUCTION

This section describes the various dose models used to calculate the absorption of beta and gamma radiation in tissue and some of the limitations in applying these models to real exposure situations. Alpha particle dose models have not been developed for this study. For any reactor run of nominal duration the alpha activity of the uranium carbide fuel would be five to six orders of magnitude less than the fission product activity of debris and under no conceivable set of circumstances could the intake of uranium approach permissible concentration limits.

Figure V-1 shows schematically the dose models developed for this study. All of these models make use of nuclide abundance programs prepared by REWG member organizations to calculate fission product inventories. In addition to calculating the core inventory as a function of operating and decay time these programs also provide information on the total emitted beta and gamma energy as shown in Figure V-1 and described in Refs (1) and (2). The abundance and emission information is also available on a single nuclide basis to provide input information for considerations of metabolized isotopes. Outputs A and B in Figure V-1 refer to beta dose models developed for this study and described in Section V-C-1 and in Appendix D. Output C represents the total body gamma dose from external sources and is described in Section V-C-4. Output D represents the emitted beta and/or gamma energy averaged over the organ of interest (with appropriate corrections for non-absorbed radiation) as described in Sections V-C-2 and V-C-3.

B. SOURCE TERMS FOR DOSE CALCULATIONS

1. Beta Energy Source Term

As indicated in Figure V-1 and described in Section V-C-1, two

Figure V-1 Radiological Dose Models

programs are available within the Radiological Effects Program to compute the beta dose from a debris particle. Each of these codes requires a different type of input source term. The preliminary beta dose model computes the dose from each isotope separately and employs the time-integrated beta activity. This information is provided by the abundance programs described in Section V-A. The Combined Berger Spencer (CBS) dose model requires the electron spectrum emerging from the particle as input information. Using the isotope abundance information this spectrum is generated by three additional programs as shown in Figure V-2 and described below.

A Beta Spectrum Program computes the beta spectrum for each beta-emitting nuclide when given the end-point energies, beta branching fractions, and degree of forbiddenness of the beta transitions. Output from the program is a sequence of values, between $E = 0$ and $E = E_{\max}$ (the largest end-point energy), representing the probability that a beta will be emitted with an energy between E and $E + \Delta E$, where $\Delta E = 0.01$ Mev. Results of these calculations and spectra of the individual nuclides have been published.⁽³⁾ In practice these spectra are only generated once and then stored on tape for the composite spectrum programs as indicated in Figure V-2.

A Composite Spectrum Program produces a point source spectrum at a specified time from a mixture of fission products by summing the individual spectra with appropriate weighting for the activity of each contributing nuclide as determined by the nuclide abundance program. Output from this program is a sequence of values representing the number of betas per energy interval emitted by a point source mixture of fission products.

The electron spectrum from a piece of debris differs from that given by a point source calculation because the emergent beta particles are degraded in energy due to scattering and absorption processes within the fragment. Similar electron transport problems are being studied by the National Bureau of Standards Radiation Theory Section under contract R-80 with NASA, Office of Advanced Research and Technology (OART). Inasmuch as the mathematical techniques developed for the NASA shielding problems are also applicable to the REWG beta-ray dosimetry problems it was decided, on the basis of informal contacts and with the approval of OART, to carry out a joint study of electron transport from uranium carbide debris containing uniformly distributed beta sources. Consequently, the National Bureau of Standards performed a number of Monte Carlo calculations of the energy spectrum and angular distribution of electrons emerging from UC_2 spheres of various sizes using the methods discussed in Ref (4,5). The sources were taken to be either monoenergetic, or to have a spectrum corresponding to that of

Figure V-2 The CBS Beta Dose Model

fission-product beta rays at various times after fission. Typical results from the Monte Carlo calculations are shown in Figures V-3 and V-4. Two significant effects occur due to the finite size of high atomic number fuel particle containing beta emitters: 1) the average energy of the spectrum increases, and 2) the number of betas escaping from the fuel particle is decreased.

For the routine calculation of emergent beta spectra Monte Carlo computations require excessive machine time. Therefore, Monte Carlo data for monoenergetic electrons have been used to generate analytical energy degradation functions.* These functions describe transmitted spectra (number vs electron energy, in Mev) as a function of particle size. The adequacy of these degradation functions has been tested by comparing the calculated emergent spectrum against the corresponding Monte Carlo computations using the same point source spectrum data as input for both programs. Except for the first energy intervals which contribute very little to dose, transmitted spectra calculated by the two methods are essentially the same.

To obtain the beta dose in those exposure situations where the radioactivity is uniformly distributed throughout an organ knowledge of the total emitted beta energy is sufficient source information for the dose calculation.

2. Gamma Energy Source Term

The gamma energy source term is based on the activities calculated as a function of time^(2,3) and the gamma yields for individual nuclides as tabulated in Refs (6) and (7). For use in the gamma dose programs, the emitted gamma spectrum is divided into seven energy bands. For each particle size class the instantaneous emitted gamma energy flux in each of these bands is integrated as a function of time to obtain Q_γ , the integrated gamma emission (in Mev), as indicated in Figure V-1.

3. Experimental Verification of Source Terms

Several experimental studies of the beta and gamma emission from NERVA fuel are being performed at LASL, ORNL and USNRDL. Preliminary results of these studies are described in Appendix A, and indicate good agreement between calculations and measurements.

*Hogan, O., Beta Spectra IV. Fission Product Beta Spectrum Degradation by Uranium Carbide Particles, USNRDL Technical Report in preparation.

Figure V-3 Ratio of Beta Energy Transmitted as a Function of Initial Energy, E_0 for Uranium Carbide Spheres of Radius R .

Figure V-4 CBS Calculations of Beta Spectra and Dose Distributions. S_1 is a fission product beta spectrum 10^6 seconds after fission normalized to 1 emitted electron. S_2 is the corresponding degraded spectrum from a UC_2 fragment (60μ radius). D_1 and D_2 are dose distributions in tissue, with the inverse square attenuation removed; as a function of distance r from a point source, for source spectra S_1 and S_2 .

C. MODELS FOR CALCULATING BETA AND GAMMA ENERGY ABSORPTION IN TISSUE

1. Beta Dose Computational Models

Two models for calculating the beta dose rate from a debris particle have been developed. One, a preliminary model described in Appendix D, uses a modified point source function originally developed empirically by Loevinger for a homogeneous source and infinite absorber medium.⁽⁸⁾ The application of the Loevinger kernel to the debris problem is subject to question since it is an extension of the model to a problem involving two bounded media of quite different atomic number and density. For this reason a more sophisticated approach, the CBS model, which considers the degradation of the beta spectrum by the particle and finite absorber geometries, has been developed.⁽⁹⁾ At the present time use of the CBS model has been confined to single uranium carbide fragments; extension of the model to consider large debris particles containing several uranium carbide beads has not been completed. Because of programming limitations some of the beta dose calculations have been performed using the preliminary model. However, this model has been augmented (as described in Appendix D) so that its results essentially agree with the available CBS programs.

The CBS beta dose model treats the beta dose problem in three parts as shown in Figure V-2. Beta ray transport from various sized uranium carbide particles is calculated as described in Section V-B-1. Then the dose at various distances from a point source emitter having the Monte Carlo calculated energy spectrum is found using the methods developed by L. V. Spencer to calculate energy dissipation by fast electrons.⁽¹⁰⁾ Finally, numerical integration of the point source dose function over geometries of interest along with appropriate correction factors for finite source dimensions and possible non-infinite backscatter geometry are performed. The application of various parts of the CBS model is discussed below.

The "Energy Dissipation Factors" for fast electrons developed by L. V. Spencer⁽¹⁰⁾ are the result of moment method solutions of the electron diffusion equation and have been published with a short description in NBS Monograph #1 which lists the dissipation factors for various materials and electron energies.⁽¹¹⁾ Since polystyrene has usually been used as the material of choice for both calculational models and experimental measurement of beta ray depth dose it was selected as a tissue equivalent material for this study. Tables of the point source dissipation factor for this material were taken from Ref. (11), and graphed to determine dissipation factors at enough intermediate energies to allow for linear interpolation of the factors. Strictly

speaking, the point source moments method data is applicable only to cases where the scattering material completely surrounds the source. However, the boundary effects have been evaluated by means of Monte Carlo calculations using the methods outlined in Refs (5), (6) and the results applied as corrections to the appropriate moments method calculations. As shown in Figure V-5, these corrections are applicable near the interface but are less important with increasing penetration.

As indicated in Figure V-2 a computer program applies the moments method dissipation data to the calculated emergent spectrum. In this program a spectrum is divided into 24 energy intervals and the dose from each interval is calculated for up to 34 radial distances from the source. The dose rate (in rads per sec) as a function of distance (in microns) is summed for the entire 24 interval spectrum. This radial depth dose rate data can be integrated over a distance equal to the range of the beta rays to find the total beta energy emitted by a particle, the average dose to a given spherical volume, or the average dose delivered to a thin disk at any selected depth below the surface as shown in Figure V-2. The disk dose calculation is for a semi-infinite geometry, analogous to a particle on the skin, and can use the Monte Carlo corrections for this geometry discussed previously. Additional comments relative to beta dose absorption and geometrical considerations are presented in Section IV-D.

As an example of the model output, the dose distributions from point and finite fission product beta sources are shown in Figure V-4. It is seen that the initial electron spectra is degraded considerably by the uranium carbide particle and as a consequence the dose close to a source is appreciably reduced.

The angular distribution of electrons emerging from a spherical uranium carbide source has also been calculated by Monte Carlo methods. These calculations show that the energy degradation of the electrons is independent of their emergent angle and that the angular distribution is quite peaked in a direction normal to the surface of the spherical source. Since this angular distribution is not substantially different from that of a point isotropic source, the use of point source dissipation factors for small finite sources is appropriate if correction factors for inverse square attenuation are included. The inverse square corrections for a uniform spherical source, outlined in Ref (8) are used in the dose calculations.

2. Gamma Dose Computational Models for Internal Emitters

The gamma dose from an ingested or inhaled particle is derived from exact multiple scattering calculations. These calculations were

Figure V-5 Ratio of Beta Dose in a Half Space Media to the Dose in an Unbounded Absorber as a Function of Depth z . Composite curve for fission product beta spectra varying from 10^4 to 10^6 seconds decay.

carried out by the Monte Carlo method and formulated in terms of Absorbed Fractions. These fractions are the ratio of the energy absorbed in a given tissue volume to the total emitted energy and have been calculated for various anatomical models, source distributions and initial gamma ray energies.(12,13) The average absorbed dose delivered to the whole body, or a particular organ, can be determined by simply weighting the various fission product gamma spectra described in Section V-B-2 with appropriate values of the Absorbed Fraction. The general expression for performing this calculation is given in Equation (V-1).

$$D_j(0)_Y = \frac{1.602 \times 10^{-8}}{m} \sum_{E_i} Q_j(E_i) \times A.F. (m, E_i) \text{ rads} \quad (V-1)$$

where $D_j(0)$ is the average absorbed dose delivered to a particular organ or to the whole body and m is the mass in grams of the organ absorbing the gamma radiation. $Q_j(E_i)$ is the gamma energy (in Mev) emitted within the energy band E_i by a particle of size j . A.F. is the Absorbed Fraction and has been tabulated as a function of absorber mass and initial gamma ray energy for each of the individual exposure situations considered in this study (cf Section VI).

3. Metabolized Isotopes

a. Inhalation

Reference (15) presents a basis for deriving dose conversion factors for each biologically significant radionuclide.

$$[D \propto / A_T]_I = \frac{73.9 E T_e f(*)}{m} \quad (V-2)$$

where

- $[D \propto / A_T]_I$ = dose per μc intake of isotope I mrad
- $f(*)$ = fraction reaching critical organ via inhalation
- $f(**)$ = fraction reaching critical organ via ingestion
- E = effective energy, Mev/disintegration

T_e = effective half life, days

m = critical organ mass, g.

These factors give the dose to the critical body organ per inhaled unit of activity. The uptake fractions, effective energies, effective half lives, and organ masses used in preparing the dose conversion factors are those recommended by ICRP.⁽¹⁴⁾

The dose from a given isotope I to the critical organ from a particle of size j is

$$D^*(I)_j = \left[\frac{D}{A\tau} \right]_I [A^*(t)_j]_I \quad \text{rad} \times 10^{-3} \quad (V-3)$$

where $[A^*(t)_j]_I$ is activity in μci of isotope I for a particle of size j at the time of inhalation, t.

b. Ingestion

The dose from metabolized isotopes via ingestion is treated identically to the inhalation dose with the exception that the fractional uptake by ingestion $f(**)$ replaces the fractional uptake by inhalation in Equation (V-2).

4. External Gamma Ray Exposure Models

a. Exposure from a Single Particle

In estimating the external whole body gamma exposure the assumption is made that the effective height of the receptor is one meter above the particle deposition plane. If r is the distance of a source from the base of a receptor the distance from the source to effective height is:

$$d^2 = 1 + r^2$$

The exposure, $D_j(d)$, from a particle of size j at a distance d in meters from the receptor is given by

$$D_i(d) = \sum_{E_i} \frac{K(E_i) Q_j(E_i) B(E_i)}{4 \pi d^2} e^{-\mu(E_i)d} \quad (V-4)$$

where $Q_j(E_i)$ is the time integrated gamma energy emission (in Mev) emitted in energy level E_i by a particle of size j ; $\mu(E_i)$ is the linear absorption coefficient for air in reciprocal meters and K is an energy flux exposure conversion constant (roentgens per mev/meter squared).

$B(E_i)$ is a linear approximation to the buildup factor given by (16)

$$B(E_i) = 1 + \mu(E_i)d \quad (V-5)$$

Civil Defense investigators⁽¹⁷⁾ have found that buildup from simulated mixed fission product fields can be approximated by

$$B(E_i) = 1 + 0.85 \mu(E_i)d \quad (V-6)$$

indicating that ground roughness and other factors significantly reduce the scattered radiation contributing to the exposure dose. However, for the purpose of this analysis it was decided that this degree of precision is not warranted.

b. Uniformly Contaminated Plane

If a large number of particles of size j are distributed over the fallout plane such that $G_j(E_i)$ is the time integrated source strength per unit area, a , then

$$Q_j(E_i) = G_j(E_i) da = G_j(E_i) 2\pi r dr \quad (V-7)$$

and

$$D_j(\text{up}) = \sum_{E_i} \frac{k(E_i) G_j(E_i)}{2} \left[\int_0^{\infty} \frac{r}{d^2} \exp -\mu(E_i) dr + \int_0^{\infty} \frac{r\mu(E_i)}{d} \exp -\mu(E_i)d dr \right] \quad (\text{V-8})$$

where

$$d = \sqrt{r^2 + 1}$$

Performing the indicated integrations by substituting $\tau^2 = \mu^2(E_i) d^2$ and using the approximation that $e^{-\mu(E_i)} \approx 1$

$$D_j(\text{up}) = \sum_{E_i} \frac{k(E_i) G_j(E_i)}{2} \left[1 + \int_{\mu(E_i)}^{\infty} \frac{e^{-\tau}}{\tau} d\tau \right] \quad (\text{V-9})$$

Equation (V-9) is used when the exposure from a uniformly contaminated plane is to be calculated.

c. Energy Grouping

For calculating external gamma exposure the gamma decay energy is broken into seven energy groups. Exposure conversion coefficients and absorption coefficients are calculated for the average energy within each group. The energy grouping, linear absorption coefficients for air attenuation, and the exposure conversion constants are given in Table V-1.

TABLE V-1

EXPOSURE CONVERSION AND LINEAR ABSORPTION COEFFICIENTS
FOR FISSION PRODUCT GAMMA RAYS

Group	Energy Range (Mev)	$\mu(E_i), (m^{-1})$	$k(E_i) \left(\frac{\text{roentgens} \cdot m^2}{\text{Mev}} \right)$
1	0 - 0.4	1.56×10^{-2}	5.15×10^{-14}
2	0.41 - 0.90	1.01×10^{-2}	5.91×10^{-14}
3	0.91 - 1.35	7.40×10^{-3}	5.25×10^{-14}
4	1.36 - 1.80	6.10×10^{-3}	4.80×10^{-14}
5	1.82 - 2.20	5.45×10^{-3}	4.49×10^{-14}
6	2.21 - 2.60	4.95×10^{-3}	4.27×10^{-14}
7	2.61 - 5.13	4.03×10^{-3}	3.62×10^{-14}

This energy grouping is identical to that used by WANL and LMSC in the source term and NFEP programs.

The conversion constant $k(E_i)$ is determined for the midpoint energy of each group from data contained in reference (18).

D. GEOMETRICAL CONSIDERATIONS IN THE APPLICATION OF BETA DOSE DISTRIBUTION MODELS

The ICRP has established criteria for calculating the beta dose from radioactivity distributed uniformly throughout a critical organ.⁽¹⁴⁾ In this case the emitted beta energy is simply averaged over the entire organ mass. For a discrete beta emitting fragment the problem of selecting appropriate spatial parameters for averaging the dose is more complex and there are no established guidelines.

Because of the rapid inverse square attenuation from a small source, calculated beta doses will vary by orders of magnitude depending on what volume of tissue is specified in the computation. Table V-2 shows a typical set of dose calculations for a 3.3 day old fission product beta source formed during a 10 minute reactor operating time. It is

TABLE V-2

BETA DOSE DISTRIBUTION AROUND A POINT SOURCE OF FISSION PRODUCTS*

Radius of Sphere R cm	Per Cent of Energy Dissipated within R	Average Dose within the Sphere (Relative Units)
2	100	1
1	99	7.9
.1	50	3.5×10^3
.01	10	7.5×10^5
.001	0	7×10^7

*Computed with CBS model - 3.3 day decay following a ten minute reactor operating time.

seen that approximately half the emitted beta energy is absorbed within a one millimeter radius sphere and 99 percent within one centimeter. Inspection of Table V-2 shows that the average absorbed dose in these two cases varies by a factor of over 400. This difference in average dose is due mostly to the difference in the mass of the absorbing volumes. Clearly the dose is such a sensitive function of volume that a number of any magnitude can be generated by the selection of an appropriate volume of integration.

Unfortunately, applicable biological information does not, as yet, provide much help in specifying a critical volume. Repair processes play such an important role in somatic radiation effects that the size of the irradiated volume may, in some cases, be as important as absorbed dose. Little work has been done to analyze quantitatively differential radiation response as a function of the irradiated volume. Work done has been largely confined to beams of radiation and is not generally applicable to the problems considered in the Radiological Effects Program. It is hoped that eventually some of the experimental work being sponsored by the SNPO (cf Section VII) will provide information on the biological effects of very small discrete fission product sources. Until such information is available it would be speculative to put any biological significance to the volumes, areas, or distances selected in the various dose models discussed in Section VI. Arbitrary spatial parameters have been specified for each of the exposure situations considered to allow quantitative comparisons to be made between different accident models. In most cases an analysis of the effects of varying these spatial parameters is also included to indicate the range of variation possible in the physical dose calculations.

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SECTION V

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SECTION VI

COMPLETE DOSE MODELS

A. INTRODUCTION

This section of the report discusses the combined results of the interaction probability models (given in Section IV) and the dose functions (given in Section V). The combination of these two components constitutes the complete exposure model and, when the values for cultural parameters are inserted, yields relationships suitable for describing the doses received by various fractions of a population.

In addition to presenting the complete dose model, this section also contains a summary of the preliminary sensitivity analyses conducted to determine the effects on the calculated doses of some of the various biological, physical, and cultural parameters.* Such sensitivity analyses are useful in determining which of the many parameters in each dose model are most significant. Once the critical parameters have been identified, efforts to further define these values will be initiated.

B. EXTERNAL GAMMA DOSE MODEL

1. Possible Exposure Conditions

The external whole body gamma exposure dose and probability models were presented in Sections IV-B and V-C-4 of this report. Basically, two exposure conditions are considered. These are: (a) exposure to particles which are so widely spaced that significant doses are delivered to a receptor from only one particle; and (b) exposure to particles which are so spaced that significant doses are delivered from a number of particles. Since twelve particle size classes are considered, either condition may exist for each particle size class. Exposures must be calculated for each size class separately using the proper model equations and the results combined to obtain the total exposure.

*"Appendices to Rover Flight Safety Program Preliminary Review, Volume IV - Radiological Considerations in Nuclear Flight Safety" (U) USNRDL TR (In preparation)(C-RD).

A computer program calculates the Mobile Receptor Dose (MOREDO) for each particle size class and population group and combines the results to obtain: (a) the fraction of the population receiving at least a certain dose; and (b) the total population exposure. The MOREDO program listing is given in NUS-257.(1)

2. Required Parameter Values

The input information required to compute the whole body gamma dose is:

- a. total population considered
- b. area associated with this population
- c. fraction of this population in a given activity group (i.e. farmers, children, housewives, etc.)
- d. the fraction of time, f_i , during which a given value of the standard deviation of motion σ_i , applies
- e. number of particles in each size class per unit area
- f. time integrated particle activity from the time of arrival of the particle to the end of the desired exposure period in question
- g. shielding of the population by buildings.

These factors have been published for all countries between 40°N and 40°S latitude and are published in NUS-230.(2)

3. Summary of Sensitivity Analyses of Gamma Exposure Model

The MOREDO program has been used to determine the sensitivity of the mobile receptor gamma exposure model to changes in population density, the standard deviation of motion, and dose reduction (shielding) factors. Figure VI-1 shows the variation of whole body gamma exposure with distance between the source and receptor center of motion for various standard deviations of motion. As can be seen a greater degree of motion of the individual significantly reduces the exposure received for small source-receptor separation distances. However, as would be expected the effect is less pronounced as the separation distance becomes larger.

Figure VI-1 Relative Dose as a Function of Separation Distance for Various Degrees of Allowed Motion.

Figure VI-2 shows (a) the exposure distribution using input parameters averaged over the affected area of the world; (b) the exposure distribution obtained by averaging, on a population-weighted basis, the results of calculation of 42 countries, and (c) the range of the results for the 42 individual countries. As can be seen there is a negligible difference between the average world model input and the average results of 42 individual countries. Results for the 42 countries show considerable variation; the greatest variation between the highest and lowest results for all the countries is about a factor of 6. The results of the world average model and the average of 42 countries follows closely the highest results and is no less than 70% of the highest values encountered.

Of major interest is a comparison of the results obtained for deposition in a rural area, as opposed to deposition on an urban area. Examination of Equation (IV-10) indicates that if the deposition density, $\bar{m}_i$, is unchanged between urban and rural area, the distribution of dose within the exposed population will be unchanged except by those changes in activity distribution (i.e., f_i and σ_i for housewives/workers/school children, etc.) which would be expected. Equation (IV-13) indicates the same to be true for the population exposure as long as the total number of people is the same. The effect of expected reductions in the values of σ_i can be seen in Figure VI-1.

C. COMPLETE SKIN DOSE MODEL

1. Beta Skin Dose

The probability that a particle of a given size will strike and stick to bare skin for at least a minimum time can be determined from the relationships given in Section IV-C of this report. The number of individuals in a population group that will be similarly affected can also be determined by these relationships. Information relative to the dose delivered to the receptor from each particle can be determined by integrating the dose rate to a disk with respect to time. A method for obtaining beta dose rate is described in Section V-C. For representative purposes it is assumed that the radiation sensitive layer of the skin is at a depth of 100 microns.

The disk area over which the skin beta dose is calculated is taken as 1.0 sq cm. It is known that the biological effects of skin irradiation vary with the area exposed but it is not possible at this time to determine what exposure area if any is critical. The relative

Figure VI-2 Fraction of Population Receiving An Exposure Greater Than D (Average for 42 Individual Countries)

dose values for areas other than 1.0 sq cm can be found by referring to Table VI-3 in the sensitivity analysis as presented below.

By determining the probabilities and dose rates for the specific exposure conditions, the number of people receiving at least a certain dose can be determined. In determining probability of exposure, it is assumed for the present analysis that only the bare skin will receive a beta dose, i.e., the sticking probability and retention time for particles on clothing is negligible. If particles do stick to clothing for an appreciable time the maximum dose delivered to a point on the skin is considerably reduced due mainly to inverse square attenuation as shown by the sensitivity analysis in Section VI-C-3.

Figure IV-3 shows the general clothing categories used in reference (2). The weight of clothing (either light or medium) and fraction of the year during which each category applies for a particular country is also given in reference (2). Wind velocity also enters into the estimation of the mean number of particles striking and sticking to an individual (Equation (IV-28)). No estimate is available, by location, for this factor; however, it is expected that 5 meters/second will provide a reasonable estimate of a mean annual wind speed for most locations.

2. Recommended Equations for Skin Dose Model

The following equations have been derived and discussed in Sections IV and V of this report but are repeated here for summary purposes:

a. Mean exposure to particles of size j

$$\bar{\Psi}(0)_j = 10^{-4} f_0 \bar{m}_j \left[f(H) a(H) + \frac{V(W)}{V(T)_j} \xi_j \sum_{k=1}^9 [f(V)_k a(V)_k \Gamma_{kj}] \right] \quad (IV-28)$$

b. Probability that M_j particles will strike and reside for at least time, t

$$P(M_j t) = \frac{[\bar{\Psi}(0)_j P(t)_j]^{M_j}}{M_j!} \exp - [\bar{\Psi}(0)_j P(t)_j] \quad (IV-27)$$

c. Beta dose rate per particle

The beta dose to a layer 100 μ below the skin surface is calculated by the computer program for a semi-infinite medium. The flow sheet for this program is shown in Figure V-2.

3. Summary of Sensitivity Analysis of Skin Dose Model

In Appendix C results are presented on the sensitivity of the epidermal exposure model given in Section IV to variations in wind speed and fraction exposed skin area. Typical particle number ground deposition data used as input for the studies presented in Appendix C were generated using the LMSC NFEP Code.⁽³⁾ An abstract of the results for graphite particles and a nude subject is given in Table VI-1.

TABLE VI-1

PARTICLE NUMBER EXPOSURE AS A FUNCTION OF WIND SPEED FOR A DEPOSITION DENSITY OF ONE PARTICLE PER SQUARE METER (NUDE SUBJECT)

Wind Speed (mph)	Particle size class (log mean diameter in microns)						
	3.16	14.1	31.6	70.7	141	316	707
0	6.7×10^{-2}	6.7×10^{-2}	6.7×10^{-2}	6.7×10^{-2}	6.7×10^{-2}	6.7×10^{-2}	6.7×10^{-2}
2	6.7×10^{-2}	6.8×10^{-2}	7.5×10^{-2}	2.0×10^{-1}	1.6×10^{-1}	1×10^{-1}	7.5×10^{-2}
10	6.7×10^{-2}	1.3	5.3	2.7	8.5×10^{-1}	2.8×10^{-1}	1.2×10^{-1}
20	6.7×10^{-2}	1.2	21.6	7.5	1.85	5.1×10^{-1}	1.7×10^{-1}

All results have been normalized to a deposition density of 1 particle per square meter. Particle number exposure per unit deposition is seen to increase with wind speed for the larger sizes considered. Since impaction efficiencies are very low for small particles, there is little variation with the lowest wind velocity for these cases. For particles greater than 100 μ the particle exposure per unit deposition decreases with increasing particle size. This is of interest since particles in the larger size ranges may contain enough activity to deliver a significant dose.

All clothing patterns presented in Section IV-C have been examined for their effects on particle-skin contact. The variation of particle number exposure as a function of exposed skin area is non-linear, the decrease in exposure being less rapid than the change in the proportion of the body covered. An exception occurs at high wind velocities where impaction is an important mechanism for particle collection and the exposure becomes nearly proportional to the fraction of exposed vertical area. Details of these results are presented in the Appendix.

Some idea of the protection afforded by tight clothing can be gained from Table VI-2.

TABLE VI-2
BETA DOSE REDUCTION AFFORDED BY
DIFFERENT CLOTHING MATERIALS

Material	Thickness* (mgr/cm ²)	Dose Reduction to One square cm area	Reduction in Maximum Dose
taffeta	10.1	.89	.37
cotton shirting	17.0	.81	.22
heavy denim	34.0	.63	.10

*Ref. VI-4

The data are based on dose rate calculations for fission product debris from a 10 minute reactor run following 10^4 sec decay. Since the exposure geometry is essentially unchanged by a small increase in absorber thickness, the reduction factor afforded by clothing when the dose is averaged over one square centimeter is less than a factor of two. However, when the maximum dose at a tissue depth of 100 microns is considered, i.e. the dose to a point directly under the source, the dose reduction due to inverse square attenuation may be as much as a factor of ten. Loose clothing will reduce calculated doses to a greater extent because of the increased separation between source and skin.

The size of the area over which the skin dose is averaged is quite important. Table VI-3 (taken from Appendix D) shows how the dose varies with the size of the disk area specified in the dose calculation. These computations are based on the dose rate from fission product debris from a ten minute reactor run following 10^4 sec decay. As would be expected from the discussion given in Section V-D, the skin dose is approximately inversely proportional to the tissue area over which it is averaged.

TABLE VI-3
SKIN DOSE AS A FUNCTION OF THE DISK AREA
SPECIFIED IN THE DOSE CALCULATION

Disk Area cm ²	Average Dose Rate at 100μ depth, Arbitrary units
0*	180
.010	40
.100	7.3
.500	1.9
1.00	1.0

*radial depth dose at 100μ
10 min. reactor run at 1000 megawatts 10^4 sec decay

As shown in Figure IV-2 retention time on the skin for a given size particle is highly variable. As an example, a 71 micron particle has a mean retention time of 4.3 hours with an observed standard deviation of approximately ± 25 percent, which would cause the same variation in total dose (or less if the retention time were long compared to the effective half-life).

Available information on particle retention by clothing (Ref VI-4) indicates that skin exposure by particles trapped on clothing would be of less duration than by particles directly on skin. In the reference cited above it was shown that from 98.3 to 99.7 percent of simulated fallout particles ($\rho = 2.5$) were removed from clothing material by orienting them vertically. However, in cases where particles are entrapped by the clothing an increase in retention time might offset the effects of shielding and separation of the particle from the skin surface.

D. COMPLETE DOSE MODEL FOR THE LUNG

1. Probability and Duration of Exposure

The equations developed in Section IV-D-1 provide the basis for determining the expected number of particles in the lungs of an individual. These equations must be used in conjunction with the relationship which describes the dose rate to tissue per particle, and the residence time of the particle in the lung.

Following ICRP recommendations for the dose to critical organs the total emitted beta energy is assumed to be absorbed by the entire lung.⁽⁵⁾ Therefore, the average dose is the total emitted beta energy divided by the ICRP lung mass (1000g) for a 70 kilogram man. Since the size of particles retained in the lung does not greatly exceed 10 microns it is not necessary to include corrections for particle self-absorption and the time integrated beta energy emission rate from a point source may be used. This information is available from the abundance programs as described in Section V-A.

Averaging the particle dose over the mass of both lungs is not a correct approach if only one or a few particles are retained. However, significant exposure could only occur if many particles were retained since the fission product inventory in individual inhalable particles is quite small.

Dose rates to the lung are integrated from the time of particle earth impact, which is assumed to be the time the particle is inhaled, to the end of the exposure period. The exposure period is calculated using the 120 day biological half life recommended by ICRP for particulate matter in the lung.⁽⁵⁾

The method of absorbed fractions outlines in Section V-C-2 is used to calculate the average gamma dose to the lungs. Lung mass for these calculations is taken as 1000 grams as in the beta dose calculations and a uniform distribution of sources is assumed.

Absorbed fractions applicable to the lung are given in Table VI-4, for each of the seven gamma energy intervals listed in the gamma source term. Backscatter factors are included in the tabulated values.

TABLE VI-4
ABSORBED FRACTIONS FOR THE LUNG

Gamma Energy (Mev)	Absorbed Fraction (1000 gm)
0 - 0.4	.16
0.41 - 0.90	.15
0.91 - 1.35	.14
1.36 - 1.80	.13
1.81 - 2.20	.12
2.21 - 2.60	.11
>2.61	.10

The cultural parameter which affects the inhalation exposure probability is the fraction of time an individual is exposed to the settling debris. In some locations, persons within dwellings or other buildings may be effectively protected against inhaling small size particles; however, in most societies, no effective protection will be afforded by the typical structures. Therefore, for the inhalation exposure calculations, no reduction is considered for the fraction of time people are indoors.

2. Recommended Equations for Lung Dose Model

The following dose model equations have been developed and

discussed in Sections IV-C and V of this report. They are repeated here for summary purposes.

a. The mean number of particles inhaled via the nose is:

$$\bar{\Psi}^* = f(N) \left[\bar{\Psi}'(I)_{j=12} + f(I) \sum_{j=9}^{j=11} 10^{-4} m_j \frac{V(W)}{V(T)_j} \Gamma_j^* a^* \right] \quad (IV-31a)$$

b. The mean number of particles inhaled via the mouth is:

$$\bar{\Psi}^{**} = f(M) \left[\bar{\Psi}'(I)_{j=12} + f(I) \sum_{j=6}^{j=11} 10^{-4} m_j \frac{V(W)}{V(T)_j} \Gamma_j^{**} a^{**} \right] \quad (IV-31b)$$

c. The average number of particles reaching the lung via the nose and deposited in the alveolae is:

$$\bar{\Phi}_j^* = f_j^* \bar{\Psi}^* F_j^* F(l)_j F(av)_j \quad (IV-36)$$

d. The average number of particles reaching the lung via the mouth and deposited in the alveolae is:

$$\bar{\Phi}_j^{**} = f_j^{**} \bar{\Psi}^{**} F(l)_j F(av)_j \quad (IV-37)$$

e. Total lung particle exposure from particle of size j is given by:

$$\bar{\Phi}_j = \bar{\Phi}_j^* = \bar{\Phi}_j^{**} \quad (IV-38)$$

f. The probability for M_j particles being retained in the lung is:

$$P(M_j)(l) = \frac{\bar{\Phi}_j}{M_j} \exp -\bar{\Phi}_j \quad (IV-40)$$

g. The total beta dose averaged over the mass of the lung is calculated by the computer program described in Section V. The flow sheet for this program is shown in Figure V-1.

3. Summary of Sensitivity Analysis of Lung Dose Model

Using Equations (IV-36) and (IV-37) and a deposition density of one particle per square meter the mean number of particles retained in the lungs of an exposed individual as a function of particle size and density has been computed. Two breathing modes have been investigated: in one case 90% of the inhaled air is assumed to arrive via the nose and 10% via the mouth; in the second case these fractions are reversed. A summary of the results are given in Table VI-5. It can be seen that graphite particles larger than class 12 (i.e., 10-20 microns) are in effect excluded from the lung and that only the 3.16 μ (0 - 10 microns) class contributes to the exposure from uranium carbide particles.

TABLE VI-5

PARTICLE LUNG BURDENS FOR A DEPOSITION DENSITY OF ONE
PARTICLE PER SQUARE METER

Particle Size	Particle Material	Air Fraction Inhaled		Number to Lung Via Nose	Number to Lung Via Mouth	Total Number of Particles in Lung
		Via Nose	Via Mouth			
3.16	graphite	.90	.10	1.69×10^{-2}	8.9×10^{-3}	2.6×10^{-2}
14.11	graphite	.90	.10	None	8.4×10^{-5}	8.4×10^{-5}
3.16	graphite	.10	.90	1.88×10^{-3}	8.0×10^{-2}	8.2×10^{-2}
14.11	graphite	.10	.90	None	7.6×10^{-4}	7.6×10^{-4}
3.16	UC ₂	.90	.10	None	1.1×10^{-3}	1.1×10^{-3}
14.11	UC ₂	.90	.10	None	None	None
3.16	UC ₂	.10	.90	None	9.8×10^{-3}	9.8×10^{-3}
14.11	UC ₂	.10	.90	None	None	None

*Assuming 1 mph wind velocity - results are proportional to wind velocity

Using a nominal 10 min reactor run and minimum particle fall times the dose per particle to the lung has been calculated for two conditions of fission product distribution in the debris. For the calculations involving graphite particles, the limiting assumption was made that all the fission product activity was distributed uniformly throughout the reactor core prior to destruct. For calculations involving uranium carbide particles it has been assumed that all the activity is contained in the uranium carbide particle. Use of these two extreme contrasting assumptions provides a worst dose estimate for each type of particle. Average lung dose per particle is given in Table VI-6. Even in the worst case of uranium carbide debris the average dose per particle is quite small.

TABLE VI-6
AVERAGE LUNG DOSE PER PARTICLE

Particle Material	Particle Size, microns	Average Lung Dose Per Particle, rad
graphite	3.16	1×10^{-8}
graphite	14.1	8.5×10^{-7}
uranium carbide	3.16	2.5×10^{-5}

E., COMPLETE GASTROINTESTINAL TRACT DOSE MODEL

1. Probability and Duration of Exposure

The probability that particles will be swallowed, either as a result of inhalation or by direct ingestion of contaminated food can be estimated from relationships developed in Section IV-E of this report. By combining the dose relationships developed in Section V-C, the dose and the probability of receiving at least that dose can be estimated for various assumed transit times through the gastrointestinal tract.

The beta dose at a depth of 300 microns from the gut wall is chosen as the dose deposition distance for these calculations and is based on the fact that the dividing crypt cells in the walls of the intestines are located within this distance. Animal feeding experiments are now in progress to further define the critical tissue location for this type of exposure.

The method of Absorbed Fractions outlined in Section V-C is used to calculate the average gamma dose to the gastrointestinal tract. Because of the penetrating nature of gamma radiation, the gamma dose is averaged over the entire gastrointestinal tract rather than considering the dose to small and large intestine separately. The ICRP Committee II recommendation for a 70 kilogram man gives the mass of the gastrointestinal tract as 2000 grams. Since an ingested particle will move back and forth throughout its travels in the gastrointestinal tract, Absorbed Fractions (Section IV-C-2) for a uniform source are appropriate for the gastrointestinal tract dose calculation; these are given in Table VI-7.

TABLE VI-7

ABSORBED FRACTION FOR THE GASTROINTESTINAL TRACT

<u>Gamma Energy</u> <u>Mev</u>	<u>Absorbed Fraction</u> <u>(Uniform source - 2000 gm)</u>
0 -	.20
0.41 - 0.09	.19
0.91 - 1.35	.17
1.36 - 1.80	.16
1.81 - 2.20	.14
2.21 - 2.60	.14
2.61	.12

When determining the gamma dose to the gastrointestinal tract, the entire gastrointestinal tract transit time (cf Section IV-E-2) is used rather than that for the lower large intestine.

Significant contributions to gastrointestinal tract dose receipt may result from direct ingestion of contaminated food. Estimates of the food consumption of certain foods eaten raw or with a minimum of processing have been reported for each potentially affected country.⁽²⁾ These values for food consumption are used to estimate the probable number of particles ingested directly. As in the case of lung exposure, cultural characteristics are not considered for that portion of the ingestion exposure resulting from inhalation.

2. Recommended Dose Model Equations for Gastrointestinal Tract Dose Model

The following equations have been developed and discussed in Sections IV and V of this report and are repeated here for summary purposes:

- a. Mean number of particles of size j ingested via inhalation

$$\bar{\Psi}(s)_j = \bar{\Psi}^*(1-F_j^* F(\ell)_j) + \bar{\Psi}^{**}(1-F(\ell)_j) \quad (\text{IV-41})$$

- b. Mean number of particles of size j ingested via contaminated food

$$\bar{\Psi}(H)_j = \sum_c \bar{m}_j f(r)_j Y_c f(p)_j \quad (\text{IV-42})$$

- c. Total mean number of particles of size j ingested

$$\bar{\Psi}(g)_j = \bar{\Psi}(s)_j + \bar{\Psi}(H)_j \quad (\text{IV-43})$$

- d. Probability that M_j particles reach the gastrointestinal tract

$$P_g(M_j) = \frac{[\bar{\Psi}(g)_j]^{M_j}}{M_j!} \exp[-\bar{\Psi}(g)_j] \quad (\text{IV-44})$$

E. Beta dose rate per particle

The beta dose at a depth 300 microns beyond the gut wall is calculated by the computer program for an infinite medium. The flow sheet for this program is shown in Fig. V-2.

3. Summary of Sensitivity Analysis of Gastrointestinal Tract Dose Model

The model for determining the number of particles which reach the gastrointestinal tract considers the contributions resulting from inhalation and direct ingestion. These two contributions may not occur simultaneously since the contribution via the direct ingestion route may occur several months following deposition of the particles on food crops.

a. Contribution via Inhalation

The numbers of graphite particles expected to reach the gastrointestinal tract for various ground deposition density are calculated in Appendix C. The results depend upon the breathing habits assumed, (i.e., nose vs mouth breathing). Table VI-8 summarizes the results obtained for a wind speed of 1 mph and deposition density of 1 particle per square meter.

The effect of increasing particle size is to sharply decrease the amount of material reaching the gastrointestinal tract. Increasing the fraction of mouth breathing has little effect upon the number of small particles reaching the gastrointestinal tract; for large particles the effect of mouth breathing is more pronounced and can cause an increase by a factor of ten in the amount of debris reaching the gastrointestinal tract.

b. Contribution from Direct Ingestion

An examination was made of data available on volcanic ash fallout and contamination of vegetable crops.⁽⁶⁾ Average values of the ratio of contamination per gm of dry plant to contamination per sq ft of ground ranged from 0.04 sq ft/gm for cabbage to 0.16 sq ft/gm for lettuce. Assuming 90%⁽⁷⁾ moisture loss on drying and one plant per sq ft with a dry weight of 40 grams the value of f_r in Equation (IV-42) would vary between 0.16-6.4 for the particle sizes encountered. An examination of the data contained in NUS-230⁽²⁾ shows the value of Y_c varies from 2 to 90 square meters for the potentially contaminated areas of the world. Examination of the data for volcanic ash fallout⁽⁶⁾ indicates that the factor F_p may vary from 0.1 to almost 1.0. Thus, the expected number of particles directly ingested

TABLE VI-8

GRAPHITE PARTICLE INGESTION FOR A DEPOSITION DENSITY
OF ONE PARTICLE PER SQUARE METER*

Particle Size (Microns)	Air Fraction Inhaled via		Number to GI Tract via Nose	Number to GI Tract via Mouth	Total Number to GI Tract
	Nose	Mouth			
3.16	.9	.1	1.09×10^{-1}	1.02×10^{-2}	1.19×10^{-1}
14.1	.9	.1	1.42×10^{-2}	2.28×10^{-3}	1.65×10^{-2}
31.6	.9	.1	2.83×10^{-3}	4.75×10^{-4}	3.31×10^{-3}
70.7	.9	.1	5.15×10^{-4}	8.60×10^{-5}	6.01×10^{-4}
141.	.9	.1	-	2.68×10^{-5}	2.68×10^{-5}
316.	.9	.1	-	1.42×10^{-5}	1.42×10^{-5}
707.	.9	.1	-	6.99×10^{-6}	6.99×10^{-6}
3.16	.1	.9	1.20×10^{-2}	9.21×10^{-2}	1.04×10^{-1}
14.1	.1	.9	1.58×10^{-3}	2.05×10^{-2}	2.21×10^{-2}
31.6	.1	.9	3.14×10^{-4}	4.25×10^{-3}	4.56×10^{-3}
70.7	.1	.9	5.74×10^{-5}	7.73×10^{-4}	8.30×10^{-4}
141.	.1	.9	-	2.40×10^{-4}	2.40×10^{-4}
316.	.1	.9	-	1.27×10^{-4}	1.27×10^{-4}
707.	.1	.9	-	6.30×10^{-5}	6.30×10^{-5}

*For a 1 mph wind speed - results are proportional to wind speed for sizes $> 3.16\mu$.

per person may vary from about 0.32 to 576 particles per particle deposited per square meter on the ground.

c. Conclusion

Comparison of the expected numbers of particles reaching the gastrointestinal tract by both routes indicates that the direct ingestion route is the more significant route than the inhalation route. However, it must be remembered that a delay between deposition on the plants and ingestion which may approach 90 days or more is likely;

consequently in the case of prompt re-entry of the reactor debris, the importance of the inhalation route is enhanced by the lack of delay in this intake mode.

F. METABOLIZED ISOTOPES

The present unavailability of data on the degree of solubility of various nuclides in the rocket debris requires that a very general approach be utilized to assess the effect on various body organs. This approach (Section IV-F) resulted in the preliminary conclusion that the only nuclides of concern are the radioiodines and thus the critical organ is the thyroid. This conclusion is based on the assumption that no nuclides are lost from the particles during re-entry and that the material is completely soluble. Both of these assumptions are conservative although it is not possible at this time to assess to what degree the exposure is overestimated by this approach.

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SECTION VII

RADIATION EXPOSURE CRITERIA AND RADIOPATHOLOGY

A. INTRODUCTION

This section contains a summary of radiation protection criteria established by various national and international bodies, and a description of experimental data obtained in dose-effect studies of interest to the Radiological Effects Program. The criteria and data are intended to serve as a basis for comparison with the radiological effects calculated using the models presented in this volume.

1. Sources of Radiation Standards

The major organizations which establish radiation protection standards include:

- a. International Commission of Radiological Protection (ICRP).
- b. National Council on Radiation Protection and Measurements (NCRP) - USA.
- c. British Medical Research Council
- d. Federal Radiation Council (FRC) - USA.
- e. Atomic Energy Commission (AEC) - USA.

Standards of the first three organizations listed above are in the form of recommendations; those of the latter two have legal status.

2. Types of Standards

Primary standards are the maximum permissible doses (MPD) to the total body and to individual body organs. These standards are based on epidemiological studies and a judgment between the possible risks from radiation exposure and the reasons for allowing the exposure. Secondary standards include maximum permissible body burdens (MPBB) and maximum permissible concentrations (MPC) of radionuclides in air, food and water. These secondary standards are calculated from the

primary standards in accordance with accepted models. Many of these models are not applicable to the present situation.

Primary standards covering both normal and abnormal conditions associated with peacetime nuclear activities are presented in Section VII-B of this document. Although any radiological effects from aerospace nuclear propulsion applications would be the result of an accident situation, the doses probably would not exceed normal population radiation exposure criteria for all but a very small percentage of the exposed population. Nevertheless, in order to assess the significance of those cases which might exceed these levels, criteria are presented for occupational exposure, and for emergency situations involving the general public.

3. Special Considerations

The established standards do not deal precisely with some of the interactions considered in the Radiological Effects Program. As noted in Section VII-B primary (and secondary) standards are based on the assumption of uniform irradiation of the organ (or whole body) both in volume and time. Irradiation of highly localized areas or volumes of tissue rather than entire organs is not covered by established criteria; similarly, short period - high dose rate situations are not explicitly treated for many types of exposures.

In order to clarify these areas pertinent data from radiobiology experiments are described in Section VII-C. Most of the experimental results presented were available in the radiological effects literature; some are the preliminary results of the research effort established by SNPO to resolve present uncertainties in radiobiological effects of specific interactions of concern to this Program.

B. RADIATION CRITERIA APPLICATION

1. Basic Factors

a. Situation

Peacetime radiation protection standards can be broadly divided into two general types; (1) normal and (2) emergency conditions.

b. Classification of Persons

In US standards there are different limits for: (1) radiation workers (occupational exposure) and (2) the general population. This is also true of standards promulgated by the ICRP which also includes an intermediate special group consisting of adults

who work in the vicinity of controlled areas or who enter controlled areas occasionally in the course of their duties, but who are not radiation workers.

Different classifications are employed because exposure to ionizing radiation can result in injuries that manifest themselves in the exposed individual and in his descendants, i.e., somatic and genetic injuries, respectively. The objectives of radiation protection standards are to prevent or minimize somatic injuries to the workers and to minimize the deterioration of the genetic constitution of the population.

c. Types of Exposures

Two types of radiation exposures must be considered in establishing standards; external and internal. External radiation exposure is that which results from radiation sources located outside the body. Internal radiation exposure is that which results from radioactive materials within the body.

d. Body Organs

In a set of standards, exposure limits are specified for different body organs based on varying organ sensitivities to radiation. Present standards include the following groups of body organs:

- (1) whole body, head and trunk, active blood-forming organs, gonads, or lens of eye
- (2) skin of whole body, and thyroid
- (3) hands and forearms, feet and ankles
- (4) bone
- (5) other organs.

e. Dose Rate

It has been long recognized that the effects of radiation may be dependent not only on the accumulated dose received but also on the way in which this total is fractionated and on the dose-rate at which each fraction is given. It has been assumed in the recommendations of standards that when either the total dose or the dose rate is very low any effects will be directly proportional to the total dose and independent of dose-rate. In the present case, however, dose rates may be quite high with a resulting uncertainty in effect. The ICRP, for

example, discourages radiation workers from receiving the allowable limit for a given control period (e.e., 3 rems in 13 weeks) in a single exposure (e.g., over an hour or a day).¹

C. STANDARDS FOR NORMAL OPERATIONS

All of the organizations listed in the preceding have established radiation protection standards for both radiation workers and the general population, that are applicable for continuous exposure in normal peacetime operations. The standards of all four agencies are essentially in agreement.

1. Occupational Exposure

The FRC Radiation Protection Guides (RPG's) for radiation workers in normal operations are contained in Table VII-I. These are the same as those promulgated by the AEC both for their licensees and contractors.

TABLE VII-I

FEDERAL RADIATION COUNCIL RADIATION PROTECTION GUIDES
FOR OCCUPATIONAL EXPOSURE IN NORMAL OPERATIONS²

<u>Type of Exposure</u>	<u>Time Interval*</u>	<u>Dose (rem)*</u>
(1) Whole body, head and trunk, active blood-forming organs, gonads, or lens of eye	Life Time Dose 13 weeks	5 times the number years beyond age 18. 3
(2) Skin of whole body and thyroid	Year 13 weeks	30 10
(3) Hands and forearms, feet and ankles	Year 13 weeks	75 25
(4) Other Organs	Year 13 weeks	15 5
(5) Bone	Body Burden	0.1 microgram of radium-226 or its biological equivalent

*except as noted

As noted previously, the ICRP also recognizes a special intermediate group which comprises adults who work in the vicinity of controlled areas or who enter controlled areas in the course of their duties, but who are not themselves employed on work involving exposure to radiation. For persons in this special group, the ICRP¹ recommendations are

a. the total annual dose to the whole body, gonads and the blood-forming organs, shall not exceed 1.5 rems.

b. the annual dose to all other organs and tissues, and to the hands and forearms, feet and ankles, shall be limited to one-tenth of the corresponding annual occupational doses.

2. Population Exposure

The FRC Radiation Protection Guides (RPG's) for the general population from normal operations are contained in Table VII-2. These guides are followed by the AEC in establishing both primary and secondary regulatory standards.

TABLE VII-2

FEDERAL RADIATION COUNCIL RADIATION PROTECTION
GUIDES FOR EXPOSURE OF THE GENERAL POPULATION - NORMAL
PEACETIME OPERATIONS^{2,3}

<u>Organ</u>	<u>RPG for Individuals</u>	<u>RPG for Average of Suitable Sample of Exposed Population</u>
Whole Body	0.5 rem per year	0.17 rem per year
Gonads	5.0 rem per 30 years	0.17 rem per year
Thyroid	1.5 rem per year	0.5 rem per year
Bone marrow	0.5 rem per year	0.17 rem per year
Bone	1.5 rem per year	0.5 rem per year
Bone (alternate guide)	0.003 micrograms of Ra-226 in the adult skeleton or the bio- logical equivalent of this amount of Ra-226	0.001 micrograms of Ra-226 in the adult skeleton or the bio- logical equivalent of this amount of Ra-226

D. STANDARDS FOR PEACETIME EMERGENCIES

Of the groups involved in the preparation of standards and guides, only two, the Federal Radiation Council and the British Medical Research Council, (BMRC) have made recommendations for general population exposure limits in an emergency situation. The ICRP has endorsed the BMRC recommendations, but has not issued any of its own; neither

has the NCRP provided standards for peacetime emergency exposure limits to the general population. The ICRP and the NCRP have made recommendations for occupational exposure in emergencies which are not appropriate in the present context. The AEC has also prepared guidance for its contractors concerning allowable exposure of workers in emergency situations, but not for the control of exposure of the general public in similar circumstances.

1. Federal Radiation Council

The FRC has formulated recommendations for protective actions in abnormal circumstances. These affect the normal production, processing, distribution, and use of food products for human consumption when such products are radioactively contaminated as a consequence of an acute localized release. One set of recommendations⁽⁴⁾ concerns iodine-131; the other set⁽⁵⁾ concerns strontium-89, strontium-90 and cesium-137. Both sets of recommendations have been approved by the President.

These standards make use of the concept "Protective Action Guide" (PAG) which the FRC defines as the maximum projected dose to individuals in the general population which warrants protective action following a contaminating event. The projected dose is the dose that would be received by individuals if no protective action were taken following the contaminating event. FRC assumes the average projected dose to a suitable sample of an exposed population is defined as including 1 year old children if organ size is an important consideration, ref. (4).

The present PAG values are applicable to internal doses resulting from dietary intake (i.e., ingestion) of specific radionuclides. To date the FRC has not established equivalent standards for inhalation or direct irradiation, hence the application of PAG values to situations other than those specified by the FRC must remain speculative. However, it appears that the FRC, by not recommending that protective actions be taken when projected doses are below the PAG values, considers these values to be maximum allowable doses under emergency or accidental conditions.

Maximum projected doses to the general population calculated using the Federal Radiation Council's "Protective Action Guide" are presented in Table VII-3. The dose values derived from the "Protective Action Guide" are quite close to corresponding standards developed by the British Medical Research Council (BMRC) presented in Section VII-D-2.

TABLE VII-3

MAXIMUM PROJECTED DOSES TO THE GENERAL POPULATION
CALCULATED USING PROTECTIVE ACTION GUIDES ESTABLISHED
BY THE FEDERAL RADIATION COUNCIL

<u>Radionuclide</u>	<u>Organ</u>	<u>Maximum Projected Dose for Year Following the Event, rads</u>	<u>Maximum Permissible Accumulated Dose, rad</u>
Iodine-131	Thyroid	30	30
Strontium-89	} Bone Marrow	10	15
Strontium-90			
Cesium-137	Whole Body	10	15

2. British Medical Research Council

The BMRC has published emergency limits for both external⁽⁶⁾ and internal^(7,8) exposures from accidents to members of the general public. The external dose limits are listed in Table VII-4 for the whole body and superficial tissues. It is of particular interest to note in this listing the specification by BMRC of a localized beta skin dose limit of 300 rad at the germinal layer of the epidermis (average over 1 cm²) for an adult. This is the only known emergency exposure limit for skin which applies to the general population.

TABLE VII-4

BRITISH MEDICAL RESEARCH COUNCIL EMERGENCY EXTERNAL EXPOSURE LIMITS FOR MEMBERS OF THE GENERAL PUBLIC⁽⁶⁾

(1) The dose of gamma rays giving exposure to the whole body but measured in free air:

Children up to the age of 16 years and pregnant women	.20 roentgens (r)
Other persons	.30 roentgens

(In the circumstances envisaged, 30 r will give rise to about 25 rads in superficial tissues: 20 r will give rise to about 15 rads in superficial tissues and about 10 rads in an embryo or fetus)

(2) Subject to (1) above, the combined dose of beta and gamma rays giving exposure to the whole body surface and estimated in any superficial tissue:

Children under the age of 16 years	75 rads
Other persons	150 rads

(3) For small areas of contaminated skin in total not greater than 1/10 of the body surface, doses from beta rays averaged over 1 cm² additional to those specified in (2) above:

Children up to the age of 16 years	75 rads
Other persons	150 rads

(4) In order to allow essential duties to be performed, a special category of a few persons is envisaged, consisting of adult males (preferably in the older age groups) or of females above reproducing age. For this group, a dose of gamma rays giving exposure to the whole body but measured in free air, additional to the doses specified in (1), (2), and (3) above30 roentgens

NOTE: The dose limits specified above are to be considered as independent of the duration of the irradiation.

For ingested material the basic criterion used by BMRC⁽⁷⁾ is that the total doses resulting from continued intake during the course of the contamination should not exceed the ICRP-recommended annual occupational MPD values for the tissues involved. ICRP and the BMRC limits for ingested material are given in Table VII-5.

TABLE VII-5
BRITISH MEDICAL RESEARCH COUNCIL EMERGENCY INTERNAL
DOSE LIMITS FOR MEMBERS OF THE GENERAL POPULATION⁽⁷⁾

<u>Radionuclide</u>	<u>Total MPD (rad)</u>	<u>Annual MPD (Rad) for Radiation Workers-ICRP</u>
Iodine-131	25 (thyroid)	30
Strontium-89	15 (total, bone)	15
Strontium-90	1.5 (annual rate, bone)	15
Cesium-137	10 (whole body)	12 max; 5 (average)

The BMRC-recommended limits were derived on the basis that the number of persons exposed will be small compared with one-fiftieth of the whole population of the United Kingdom. Also taken into consideration was the possibility, however unlikely, that the same persons might be significantly exposed to a second accidental release of radioactive materials.

Another report⁽⁸⁾ from the British Medical Research Council deals with exposure from radioactive material which is inhaled during passage of a cloud of radioactive material. In determining the recommended maximum permissible contamination values for respirable air after an accidental release of radioactive materials, the BMRC used the same basic criteria (i.e., primary standards) for total doses (25 rems to thyroids, 10 rems to total body, etc.) as it used for the emergency dietary contamination limits. In addition, the Council used for the inhalation case a maximum total dose of 15 rads to the lungs. As in the dietary case, BMRC also took into account the small probability that the same persons would be involved in more than one emergency. In the opinion of the Council, the limits have been set sufficiently low that a second emergency exposure after a period of some years should not give rise to undue concern.

The 1961 BMRC report⁽⁸⁾ gives the average dose to the lungs resulting from inhaling the recommended total intake limits, assuming the radionuclides to be in an insoluble form. The reported values indicate that the BMRC total intake limits for iodine-131, strontium-89, and strontium-90 based on other critical organs will not result in a lung dose exceeding the emergency limit of 15 rads. Values computed for cesium-137 are higher, but are not considered significant because of the very low probability of cesium-137 appearing in an insoluble form.

3. Additional Comments

Dunning⁽⁹⁾ has stated that, although the thyroid dose resulting from ingesting fallout may be greater than the dose to the gastrointestinal tract, the greater sensitivity of the gastrointestinal tract may make it a controlling consideration in selecting emergency exposure levels. In the BMRC reports^(7,8), no mention was made of the gastrointestinal tract doses which would result from ingestion or inhalation at the recommended emergency limits.

Morgan, Snyder and Ford⁽¹⁰⁾ have reported a calculational method for estimating the maximum permissible intake (MPI) of radionuclides for a single exposure. Many radioactive materials are relatively insoluble, and the fraction entering the blood stream following ingestion or inhalation is so small that the gastrointestinal tract becomes the critical body tissue rather than the kidney, bone, or other body organ.

E. PATHOLOGIC EFFECTS OF RADIATION

1. Lung

In 1960, after extensively evaluating all available data, the National Academy of Science - National Research Council's Subcommittee on Inhalation Hazards of the Committee on Pathologic Effects of Atomic Radiation, reported on the biological effects of radionuclides deposited in the lung.⁽¹¹⁾ Briefly they concluded that: (1) the effects depend on the dose, sensitivity of the tissue, and type of radiation; (2) alpha emitters irradiate only nearby cells while beta-gamma emitters may irradiate the entire tissue; (3) the principal effects of radiation to the respiratory tract are pneumonitis, fibrosis, and carcinoma; (4) bronchial lymph nodes readily accumulate radioactive particles and may be severely damaged; (5) calculated mean doses to the lung should be used with caution in estimating biological effects since local variation in dose may be by orders of magnitude; (6) the effects of age and condition of the lung on retention and translocation of particles and the radiosensitivity of lung tissue are not well enough understood to be evaluated.

The Inhalation Hazards Subcommittee⁽¹¹⁾ reported that available data on humans was confined to alpha emitters (which are not a creditable hazard in the Radiological Effects Program) and the effects of external x-ray irradiation in producing pneumonitis and lung fibrosis.⁽¹²⁾ Minimal x-ray doses necessary to produce these effects in animals have been estimated at 800 r and 1800 r respectively.⁽¹³⁾ No human data on carcinogenetic effects of external gamma irradiation or internal beta irradiation was reported. Such data is confined to animal experiments.

A number of animals studies in which internally deposited beta and gamma were used to irradiate lung tissue are summarized in the report by the NAS-NRC Subcommittee on Inhalation Hazards.⁽¹¹⁾ More recent studies are described in the Proceedings of the Hanford Symposium on Inhaled Radioactive Particles and Gases.⁽¹⁴⁾ In the earlier work the radionuclides were given mostly by intra-tracheal injection. Beta and gamma emitting isotopes used included Co⁶⁰, Ru¹⁰⁶, Sr⁹⁰, S³⁵, and Ce¹⁴⁴. Table VII-6 below, has been abstracted from the NAS-NRC Subcommittee's report.⁽¹¹⁾

TABLE VII-6

TUMORS AND CALCULATED DOSE FROM LUNG DEPOSITED RADIOACTIVE MATERIALS⁽¹¹⁾

Material and Radiation(a)	Species	Results	No.	Calculated Dose (rad)(b)	Remarks
BaS ³⁵ O ₄ (beta)	Mice	Bronchiolar carcinoma	1/41	4000	100 days after 0.16µci
	Rat	Squamous cell carcinoma	2/16	12,000 to 20,000	300 days after 375 µci/week for 10 weeks
Ru ¹⁰⁶ O ₂ (beta)	Mice	Lymphosarcoma(c)	1/23	300	340 days after 0.15 µci
		Alveolar cell carcinoma	1/10	4000	422 days after 2 µci
		Non-differentiated tumor	1/11	9000	350 days after 4.5 µci
Ru ¹⁰⁶ (metal cylinder) (meta)	Rat	Bronchogenic carcinoma	5/26	200,000 to 900,000	224 to 337 days after implantation of 2.1 to 14 µci (dose calculated at 100 µ from source)
Sr ⁹⁰ (glass beads) (beta)	Rat	Lymphosarcoma	2	47,000 and 90,000	169 to 581 days after
		Squamous cell carcinoma	4	78,000 to 260,000	implantation of 1.1 to
		Lymphoma	1	----	59 µci beads in 23 rats
Ce ¹⁴⁴ F ₃ (gamma)	Rat	Carcinoma	1/27	2200	127 days after 5 µci
			1/23	5500	48 days after 15 µci
			7/28	8900	93 days after 25 µci
			4/15	15,000	83 days after 50 µci
Co ⁶⁰ (wire) (gamma)	Mice	Epidermoid carcinoma of bronchus	---	12,000 to 400,000	200 days after implantation

(a) All materials administered intratracheally or as otherwise indicated.

(b) Assuming uniform distribution and exponential loss unless otherwise indicated.

(c) Classed as incidental by author because autoradiograms showed no radioactivity in area of tumor.

An enlightening statement by the NAS/NRC Subcommittee concerning the interpretation of this data is quoted below:

"For each reported tumor, a dose was calculated assuming uniform distribution of the radionuclide in lung tissue and exponential loss from the lung. Neither of these two assumptions may represent the true state of affairs. In particular, uniform radiation to a tissue as a result of deposited radioactive particles is virtually impossible. Because of the difficulties inherent in the calculation of local radiation doses and because of the uncertainties in the interpretations of biological effects of various dose rates, the uniform dose concept has been generally applied. Where the author reported such a dose to the lung, this value is used in the table. The smallest lung doses associated with malignant tumors⁽¹⁵⁾ were 115 rads after administration of 0.003 μCi $\text{Pu}^{239}\text{O}_2$ and 300 rads after administration of 0.15 μCi $\text{Ru}^{106}\text{O}_2$. The genesis of the tumors could not be ascertained, however, because autoradiograms failed to show radioactivity in the area of the tumor. In other studies at least 2000 rads was the estimated dose to lungs that developed tumors. Such estimated doses are of questionable value in many cases because of the non-uniform distribution of the radionuclides. Autoradiograms showed that inhaled particulates were localized in discrete areas of the lung.⁽¹⁶⁾ In these cases the doses to micro-volumes of tissue could be considerably greater than those estimated by assuming uniform distribution. Therefore it should not be concluded from the dose estimates given in the table that the minimum dose required to induce lung cancer is necessarily as low as 2000 rads; it may, indeed, be much greater."⁽¹¹⁾

Some other recent data on carcinogenesis by beta emitters is summarized below. One undifferentiated tumor and one alveolar cell tumor were observed in 34 mice given 2 to 3 μCi $\text{Ru}^{106}\text{O}_2$ intratracheally.⁽¹⁵⁾ The calculated beta-gamma radiation dose was 4000 to 9000 rad. Laskin et al⁽¹⁷⁾ summarized their experience with induction of bronchogenic carcinoma in rats following the introduction of Ru^{106} and P^{32} plated cylinders and pellets into the bronchus. In these experiments the radiation dose received by the epithelial cells of the bronchi was calculated. In a serial sacrifice experiment, the lowest dose producing cancer was 4.3×10^5 rads accumulated in about 160 days. Tumors were seen in two of 13 animals. Earlier these authors had shown a dose related response, 7.3% tumor incidence in rats exposed to about 1000 rad and 65.6% in those exposed to 10^6 rads.⁽¹⁸⁾ Laskin et al⁽¹⁹⁾ also reported tumor incidences of 33% to 56% in rats implanted with beta emitting P^{32} plastic pellets, receiving beta radiation doses of 4×10^4 and 4×10^5 rads, respectively.

Cember⁽¹⁹⁾ reported a total of 101 rats with malignant neoplasms in 395 rats intratracheally injected with the beta-gamma emitter, Ce^{144} , at every dose level. The range of the average dose to the lung in these experiments was from 680 to 68,000 rads. Squamous metaplasia of the alveolar and bronchial epithelium was observed in many of the animals as well as fibrosis and pneumonitis. However, in the opinion of the NAS/NRC Subcommittee⁽¹¹⁾ since the studies described above were done with rats and mice which are known to have high frequency of respiratory pathology, it is difficult to ascribe all the changes to radiation. Also, since all but two or three of the experiments involved intratracheal injection of particles or implantation of specially prepared radiation sources, the results can be questioned as to their applicability to the inhalation situation where distribution of the radioactive particles may be quite unlike that obtained by the artificial methods of administration.

In a pilot study at Battelle Northwest Laboratory 11 dogs were exposed to $Ce^{144}O_2$ aerosols.⁽²⁰⁾ Initial alveolar deposition ranged from 0.05 to 2 mCi. One dog died eight months after deposition of 2 mCi due to severe fibrosis and pneumonitis. The remaining dogs show slight depressions of the circulating leukocytes, but no other clinical changes three years after exposure. In a similar study six dogs which deposited 20 to 35 μCi $Ru^{106}O_2$ show no significant changes two years after exposure.⁽²¹⁾

Aerosols consisting of large particles (up to 30 microns) have recently been used in inhalation studies at Battelle Northwest.⁽²²⁾ Prior studies in particle inhalation have usually been concerned with particles smaller than 5 microns due to the generally held opinion that larger particles would not reach the alveolar lung, see Section IV. The studies were performed using four beagle dogs exposed to ceramic microspheres labeled with Fe^{59} in two nominal size ranges (10 microns and 26 microns diameter). The results of this series of experiments are tentative due to their preliminary nature and the small number of animals involved (2 for each particle size class). However, the preliminary results are of interest in that it has been definitely shown that particles up to 30 microns in diameter are respirable, as evidenced by direct measurements in lung tissue sections, and are tenaciously retained. The fraction of the initially inhaled material was small (2 - 10 percent) but of this initially deposited material, 20 to 60 percent was retained for at least 2 months.⁽²²⁾ Studies on two of the exposed animals are continuing.

In summarizing the status of radioactive particle hazard evaluation the NAS/NRC Subcommittee on Inhalation Hazard concluded:(11)

"In view of the unknowns in the factors influencing the dose response, it is difficult to provide an overall assessment of the possible effects of radioactive materials upon the lung. Since experimental studies during the past few years have produced malignancies in the lungs of experimental animals, it now appears certain that radiation can produce lung cancers. The estimate of the dose to produce such an effect is more difficult because of both variability in the distribution through the lung and the uncertainty of extrapolating the few results available from animals to man. Present data indicate that the lung and respiratory tract do not differ markedly in radio-sensitivity from other soft tissues. The same dose-effect problems would apply to lung hazard evaluation as to other radiation hazard evaluations. 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."

2. Skin

The dose at the basal layer of the epidermis is of major interest in the evaluation of the significance of a skin exposure. Table VII-7, reproduced from reference⁽²³⁾, indicates that although the surface doses vary widely the dose required for microscopically recognizable transepidermal injury at the base of the porcine epidermal layer (estimated at 90 μ) is about 1300 $\pm$ 300 rad.

TABLE VII-7

AMOUNT OF TRANSEPIDERMAL RADIATION REQUIRED FOR
THE PRODUCTION OF RECOGNIZABLE INJURY (23)
(Porcine Skin)

<u>Isotope</u>	<u>Energy, Mev.</u>	<u>Surface Dose Required, rep</u>	<u>Estimated Dose At 0.09mm Depth, rad</u>
S-35	0.17	20,000	1100
Co-60	0.31	4,000	1500
Cs-137	0.55	2,000	1600
Y-91	1.53	1,500	1100
Sr-90, Y-90	0.61, 2.20	1,500	1300

It should be noted that data shown in Table VII-7 is for relatively large area exposures not single particles. Battelle Northwest has been performing animal experiments with small sources to further define the biological effects from this type of irradiation. Skin exposures of miniature white swine for 1 and 4 hour periods have been made using simulated NERVA debris spiked with Sr-89.(22,24). The particles were 1 mm in diameter and delivered a dose rate of about 4000 rad per hour at a depth of 50 mg/cm² in tissue. Following the one hour exposure (4000 rad) a slight localized erythema developed between 10 and 15 days. After the four hour exposure (16,000 rad), a slight erythema could be seen immediately. A more severe erythema developed after the second week followed by necrosis of an area about 3 mm in diameter and the appearance of a crusted ulcer by 30 days. Healing of the lesions was slow, being incomplete after ten weeks.

Since the radiation exposure of various parts of the tissue surrounding a deposited particle depends on the distances from the particle, there exists a possibility that the resultant damage to the skin may be greater than it would be if the same amount of radioactivity were uniformly dispersed over a large area. An experiment by Passoneau⁽²⁵⁾ that may bear on this was done on rat skin. In this experiment, rats were exposed for periods of two days to 1500μci of Sr⁹⁰-Y⁹⁰ which was either spread uniformly over an area or divided equally into 10, 20, or 50 small beads. Malignant and benign tumors were formed in all cases, but fewer tumors were formed by exposure from point sources than from the same activity uniformly distributed. The number of tumors produced per bead was largest for the beads with the most activity, but the number of tumors per microcurie was less for the stronger beads than for the weaker. The NAS/NRC Subcommittee has reviewed Passoneau's results and noted that they were obtained with extremely large doses with consequent killing of cells.⁽¹¹⁾ The lessened tumor production observed for localized sources may not be applicable in all cases of particle exposure for as the Subcommittee stated "(Passoneau's results) 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".⁽¹¹⁾

Some information of the response of human skin to beta radiation is available from radiation accidents. Four cases of beta ray burns of the hands, which occurred after an atomic bomb test at Eniwetok, have been reported by Knowlton, et al.⁽²⁶⁾ It was calculated that two of the men received beta ray doses of 4,500 - 9,000 rad, another received between 7,000 - 15,000 rad and the fourth received 2,700 - 4,200 rad. For all but the smallest dose, skin damage was so extensive that grafts were required. There was loss of mobility of some of the fingers. In one case serious ulcers persisted for periods greater than 100 days

after the exposure. The effects of the smallest dose were less pronounced; however, the damage persisted for a period greater than 50 days.

An accident resulting in skin burns to six people from scattered 1.2 Mev electrons is described by Robbins, et al.⁽²⁷⁾ The resulting burns covered large areas of skin. Actual doses received could not be determined, but careful dosimetry following the accident placed the maximum exposures somewhere between 900 to 1,800 rad. The burns were analogous to, but different from, thermal burns and x-ray reactions, in that only superficial layers of tissue were involved. In one patient a skin graft was necessary on the 190th day after exposure due to a 4mm area on the index finger that broke down and began to granulate. At ten months after exposure the hand had healed (93 days after grafting).

A recent report by Conrad⁽²⁸⁾ on the ten year study of the Marshallese people exposed to fallout radiation indicates that during the past several years an increased number of pigmented nevus-like lesions have been noted in previously irradiated areas of the skin but these have appeared to be quite benign. Neither chronic radiation dermatitis nor evidence of cancer of the skin has been noted. It should be pointed out however that the latent period for skin cancer following irradiation in humans is probably as long as 15-20 years.

Short term effects of ionizing radiation on skin as a function of dose have been summarized by Alpen.⁽²⁹⁾ Table VII-8 taken from this summary is reproduced below.

TABLE VII-8

ACUTE EFFECTS OF IONIZING RADIATION ON SKIN⁽²⁹⁾

Estimated Acute Dose

0-600 rad	No acute effects
600-2000 rad	Moderate early erythema
2000-4000 rad	Early erythema under 24 hours Skin breakdown* in 2 weeks
4000-10,000 rad	Severe erythema in <24 hours Severe skin breakdown* in 1-2 weeks

Estimated Acute Dose

10,000-30,000 rad	Severe erythema in <4 hours
	Severe skin breakdown* in 1-2 weeks
30,000-100,000 rad	Immediate skin blistering (less than 1 day)

*When the dose is due to beta radiation, the skin epithelium may suffer severe breakdown without destruction of the dermis (true skin). More penetrating radiation destroys the true skin and causes ulceration.

Long term effects of skin irradiation include carcinogenesis. Severe skin burns are not a prerequisite for the production of skin tumors. Table VII-9, taken from a recent review by Hulse⁽³⁰⁾ indicates that carcinogenetic doses may produce only a moderate skin erythema.

TABLE VII-9

INCIDENCE OF MALIGNANT TUMORS OF THE SKIN FOLLOWING EXTERNAL IRRADIATION WITH BETA PARTICLES OR ELECTRONS⁽³⁰⁾

Source	Species	Dose, rads	Tumors per 100 cm ²	Percentage Sarcomas
Passonneau et al (1952) ⁽⁷⁵⁾	rats	5,500-8,200	2.6	50
Boag and Glucksmann (1956) ⁽⁷⁶⁾	rats	2,000 or 2,300 12,000	1.8 6.1	83 93
Albert et al (1961) ⁽⁷⁷⁾	rats	230-2,500 2,750-10,000	0.8 5.8	10 2
Glucksmann and Boag (1954) ⁽⁷⁸⁾	mice	8,000	19	0
Cloudman et al (1955) ⁽⁷⁹⁾	mice	4,600-9,300	2.0	50
Hulse (1962) ⁽⁸⁰⁾	mice	1,900-2,800 3,300-24,000	0.4 2.1	67 98

The area irradiated is important in determining the incidence of skin carcinoma. Figure VII-1 taken from reference (3) shows the tumor incidence per 100 sq cm irradiated area as a function of the estimated beta dose range from P^{32} and Sr^{90} for several strains of mice. Much of the available information with isotopes is difficult to interpret because of the uncertainty in the beta dose. More precise work performed by Hulse using Tl^{204} is shown in Figure VII-2.(30) Hulse has found that for doses as low as 375 rad the tumor incidence is clearly above the control level. The effects of multiple exposures upon the tumor incidence is also being investigated by Hulse. His experimental results are shown in Table VII-10.

TABLE VII-10
THE EFFECTS OF FRACTIONATION ON THE INCIDENCE OF
MALIGNANT TUMORS OF THE SKIN FOLLOWING EXTERNAL
BETA IRRADIATION⁽³⁰⁾

<u>No. of Fractions</u>	<u>Fractionation</u>	<u>Tumors/100 cm²</u>	
		<u>6,000 rads</u>	<u>3,000 rads</u>
1	--	6.2	4.9
20	daily	2.9	1.5
4	weekly	6.9	2.3
4	monthly	7.5	3.5
12	weekly	2.9	0

The data given in Table VII-10 indicate no significant decrease in tumor incidence with fractionation of the total dose into four smaller doses spread. Fractionation of the total dose into 12 weekly and 20 daily doses however, does show some decrease in tumor incidence. Hulse's results from both sets of experiments suggest a plateau in tumor incidence with dose; in one experiment this occurred between 3000 and 20,000 rad and in the other between 3000 and 6000 rad. There is no evidence of a decline in incidence at higher dose levels.(30)

A summary of Biological Effects of Radiation on Skin abstracted from Alpen⁽²⁹⁾ is given in Table VII-11.

Figure VII-1 Incidence of Malignant Tumors of the Skin Following Beta Irradiation from P³² and Sr⁹⁰ (Ref. 30)

Figure VII-2 Incidence of Malignant Tumors of the Skin in CBA Mice Following External Beta Irradiation From a Tl^{204} Source (Ref. 30)

TABLE VII-11

SUMMARY OF BIOLOGICAL EFFECTS OF RADIATION ON THE SKIN⁽²⁹⁾

Effects of beta radiation are of four types:

- a. Immediate -- appearing from 0 to 48 hours after exposure.
 1. Erythema (reddening of the skin, as in severe sunburn) and itching. Estimated dose required: 600-1000 rads. If 600 rads, will probably appear within 48 hours; if 1000 rads, will probably appear within 24 hours.
 2. Vesication (formation of blisters) between 30,000 and 100,000 rads.
- b. Delayed -- appearing from one to five weeks after irradiation.
 1. Second wave erythema. 600-1000 rads.
 2. Vesication and Desquamation (loss of skin). 2500 rads.
 3. Epilation (loss of hair). 300-700 rads.
- c. Persistent Changes
 1. Radiation Dermatitis. Persistent ulceration in which skin repeatedly breaks down. Requires replacement of skin. More than 600-100 rads.
 2. Vascular Changes. Visible spiderwebbing of surface begins. May contribute to dermatitis. 500 rads, to the blood vessels.
 3. Atrophic Changes. Skin becomes very thin and easily damaged. No estimate as to dose required.
- d. Long-Term -- appearing after one year.
 1. Tumor induction. 1000-2000 rads. A statistically significant increase in tumors has occurred in irradiated animals with doses as low as 375 rad. Not predictable on an individual basis. A genetic effect at the cellular level.
 2. Less severe radiation dermatitis. No estimate as to those required.
 3. Cataract formation. 2000 rads to lens (may be 600 rad).

3. Gastrointestinal Tract

A literature survey on the exposure of the gastrointestinal tract to "insoluble" beta emitters has been made by Fish et al.⁽³¹⁾ Much of the information that follows is abstracted from Fish's survey. Primary interest is in the effects to man resulting from the ingestion of small (~1 mm dia.) insoluble radioactive particles. References concerning human exposures of this type are scarce in the open literature; thus, it is usually necessary to extrapolate from animal data. Furthermore, the sources of radiation used in the reported exposures are not, in general, comparable with small discrete sources.

The gastrointestinal tract has long been considered as one of the more radiosensitive organs⁽³²⁾ and, as Morgan⁽³³⁾ points out, ICRP has used it as the critical organ in over 70 percent of the maximum permissible concentration (MPC) values listed for radioisotopes contained in food or water. A complete catalog of the biological effects of radiation on the gastrointestinal tract is beyond the scope of this summary. A review of the normal patterns of physiological changes is given by Conard⁽³²⁾, and the work of Ricketts⁽³⁴⁾ indicates the usual response pattern in humans to x-radiation of the fundus and body of the stomach.

Many investigators have reported a decrease in the rate of gastric emptying following whole body x-radiation of rats^(35,36,37); presumably other radiation would have the same type of effect. This decrease in emptying time, together with entrapment of radioactive particles in radiation damaged areas of the gut wall, would increase the total dose received by these organs, and the consequent damage.

Beta irradiation of the gastrointestinal tract using liquid solutions of isotopes, given orally, has been a routine procedure for many years, and the effects studied are varied.^(38-46 inclusive) Other irradiation techniques using beta sources have been (1) exteriorizing of the bowel and the use of solid radioactive applicators^(47,48,49), (2) external irradiation of mice with high energy beta particles⁽⁵⁰⁾, and (3) use of intragastric balloons either filled with radioactive solution⁽⁵¹⁾ or with the active material incorporated in a thin layer of latex on the balloon.^(52,53) Edema, degenerative changes in the epithelial cells, hyperemia, atrophy of the gastric mucosa and concurrent metabolic disturbances are commonly seen after a total beta dose of a few thousand rads. These changes may not be reversible depending upon the total dose received by the cells, the number of cells involved and the time over which the dose was delivered. Further damage may result in necrosis of the area. Latent effects observed include shortened median survival times due mainly to occurrences of superficial ulcerative lesions and to hyperplastic changes in mucosa.^(50,54,55)

One of the earlier studies on the effect of internal irradiation on the gastrointestinal tract was made by Jenkins and McGeorge⁽⁵⁶⁾ using radium applicators inside the stomach. These and subsequent attempts to reduce gastric acidity in man by this technique are reported by McGeorge.⁽⁵⁷⁾ Pierce⁽⁵⁸⁾ has given the results of the exposure of small numbers of rats to beta and gamma emitters and to mixed fission products introduced by gavage into the animals. Doses in the range 23 to 28 μ ci of fission products per gram of body weight caused death in the four rats tested, and in these one perforated stomach ulcer was seen. Seven rats dosed with from 0.4 to 7 μ ci/g body weight in a single dose showed no effects at one month. Of two rats receiving Sr-89, one died 11 days after a dose of 18 μ ci/g body weight, and the other was moribund 19 days after receiving 9 μ ci/g body weight. No histopathological data were given. Five μ ci/g body weight of Y-91 given as a single dose produced "minimal" crypt cell damage. Daily doses of Y-91 gave the results shown in Table VII-12. One stomach ulcer was observed in a control rat that had been given daily doses of a saline solution.

TABLE VII-12

EFFECTS OF CHRONIC INGESTION OF Y-91 IN RATS⁽⁵⁸⁾

Daily Dose of Y-91 μ ci/g body wt	No. of Rats	Duration (months)	Effects*
.03	3	6	None
1.0	3	6	Damage in order of increasing severity was seen in: cardiac area of stomach, large bowel, ileum, fundic area of stomach, duodenum
2.0	2	3	Similar to 1.0 μ c dose; stomach showed more severe damage

*The relative insensitivity of the large bowel indicated by these data apparently is in conflict with reports of Lisco⁽⁵⁹⁾, Sullivan⁽⁶⁹⁾, and Nold.⁽⁵⁴⁾

With the advent of large-scale radioisotope production, other investigators looked at the effect of nonabsorbable internal emitters on the gastrointestinal tract. Lisco, et al⁽⁵⁹⁾ gave both single oral dose of insoluble emitters (1 to 6 mci) and multiple doses (0.6 to .46 mci) of Y-91 to rats. Pathological changes observed varied from superficial ulcerative lesions to adenocarcinoma and damage to the colon.

Stomach irradiation of dogs using intracavitary balloons and P^{32} ^(60,61,62), I^{131} ⁽⁶³⁾, or radioactive noble gases⁽⁶⁴⁾ was shown to reduce gastric acidity, usually only temporarily. Doses ranged from 230 to 23,000 rads. The dog receiving the largest dose was found to have a mucosal ulcer three months post-irradiation.

Fox and Littman⁽⁶⁵⁾ used solid sources inside balloons (a method similar to the technique of McGeorge) to test the effect of several individual isotopes on the stomach mucosa of dogs. Aside from the reduction in gastric acidity, the results are summarized in Table VII-13. Further work by this group indicated that gastrectomy appeared to have prevented death in three of five dogs that received "lethal" doses of radiation to the stomach wall.⁽⁶⁶⁾

TABLE VII-13
EFFECT OF RADIATION ON THE GASTRIC MUCOSA⁽⁶⁵⁾

Isotope	No. of Dogs	Dose ^(a)	Effect
Ra	3	1,500-9,000 mg-hr	Death at highest level in 3-1/2 mos.
Ru ¹⁰⁶ -Rh ¹⁰⁶	3	1,600 rbe ^(a)	Chronic gastric ulcer in one
	4	4,800-18,000 rbe ^(a)	All had large gastric ulcers, death within 3 mos.
Sr ⁹⁰	3	.75-2.1 curie hrs	None
	5	3-6 curie hrs	Chronic gastric ulcer
	2	12 curie hrs	Chronic ulcer; death within 1.5 mos.

(a) Author's units: rbe = Roentgen beta equivalents (approximately equals one rad)

Rothe and Tuttle⁽⁶⁷⁾ used an essentially insoluble compound, CrPO_4 labeled with P^{32} , in a study with rats that yielded information both on transit time through portions of the gastrointestinal tract and on clinical and histopathological changes. A portion of their results dealing with the stomach is summarized in Table VII-14. By an extrapolation, which the authors agree is not normally permissible, they estimate that man would incur only a minor risk of gastrointestinal tract damage by the daily ingestion of 445 μCi of insoluble beta-emitting fission products over a 10-day exposure period. Assumptions and equations used in calculating the radiation dose to the gut wall were taken from a paper by Thompson and Hollis.⁽⁶⁸⁾ The latter group had used tracer amounts of Ru-106 to derive estimates of transit times and doses to the gastrointestinal tract of rats from beta-emitting isotopes.

TABLE VII-14

EFFECTS OF CHRONIC P-32 INGESTION ON RAT STOMACH⁽⁶⁷⁾

<u>μCi of P-32/day</u>	<u>Duration of Ingestion Days</u>	<u>No. of Rats</u>	<u>Total Dose (rep)</u>	<u>Effect on Stomach</u>
100	10	4	600	None
	20	3	1,200	None
	30	2	1,800	None
200	5		475	None
	10		950	Minimal
400	5	9	1,300	Abnormal epithelial cells
	10	6	2,600	Degeneration and death of epithelial cells

Oral doses of Y-91 in the range of 10 to 50 mci/kg body weight gave an LD 50/30 for rats of about 17 mci/kg in a study of Sullivan, et al.⁽⁶⁹⁾ Hematological changes differing slightly from those seen in whole body exposures were evaluated. At all levels the greatest damage was to the large intestine, presumably due to the longer retention time of the radioactivity in that portion of the gut. Changes seen were comparable to those following x-irradiation.⁽⁷⁰⁾ A similarity of effect of x-irradiation and internal beta irradiation of the gastrointestinal tract on the depression of serum albumin of rats was noted by Palmer and Sullivan.⁽⁷¹⁾

Preliminary results are available of work done at Battelle Northwest using simulated reactor debris (1 mm diameter) spiked with Sr-89 and delivering about 4000 rad/hour at 50 mg/cm tissue depth.⁽²²⁾ Gastric mucosal exposures were accomplished by cementing six particles to gastric balloons which were inserted into the stomachs of Hanford miniature white swine and inflated. Three swine were so exposed for a one hour period, and sacrificed at 2, 3 and fifteen days for examination. A second group of three were exposed for four hours, two of which were sacrificed after 1 and 3 days. In all cases, a patchy congestion of the gastric mucosa was observed which could not be correlated with the actual irradiated sites due to lack of knowledge of precise particle location. Focal lesions were not observed and congestion was absent 5 weeks after four hour exposure.

The simulated reactor fuel described above was also given to rats by gastric intubation.⁽²²⁾ Significant amounts of radioactive leaching, the large size of the particles relative to the lumen of the rat gut, and uncertainties in the dosimetry make definitive interpretation of results difficult. However, Drs. Sullivan and Mahoney who performed these studies felt that qualitative interpretation of the experiments was possible. Prolonged retention of particles was observed, (this may have been due to the relatively large size of the particles). Of interest is the fact that prolonged retention did not correlate with a high radiation level. Gross systems of gastrointestinal irradiation effects were observed with doses of the order of 500 rads, as determined with glass dosimeters adjacent to the gut wall. Histopathologic examination provided definite evidence of focal damage. This damage was attributed to lodged particles. The lesions were sharply localized and covered less than one third of the mucosal circumference. Damaged areas were seen most frequently in the colon, lower ileum, cecum and dependent glandular stomach. The authors stated "the relatively high incidence of focal lesions was surprising since one might expect that the particles might be imbedded within, and shielded by, the intestinal contents. Examination of the excreted fecal pellets containing Sr-89 labeled particles disclosed that in nearly every instance these particles were located at the surface of the pellet".

Observations by Nold et al on transit time and biological effect of Y-90 fed to dogs and goats give further evidence for the selection of cecum and colon as the more critical segments of the gastrointestinal tract.^(54,41) This conclusion holds true when dealing with insoluble beta emitters homogeneously mixed with intestinal contents and with the mass moving through the tract on a known, normal time schedule.

Recent work by Bell⁽⁷²⁾ on the radiotoxicity of Ce-Pr-144 in sheep has shown that some individual animals are remarkable resistant to the lethal effects of massive oral doses of this material. Of 12 sheep fed 1.2 curies over a 60-day period, 4 showed no clinical effects. Necropsies on 6 animals that died showed maximum injury to the omasum and abomasum. This was partially attributable to the lower moisture content of the material as it passed through these segments and the consequent more intimate contact of gut wall and radioactivity.

Daily doses of from 91 to 410 μ ci of Y-90 per rat over a 60-day period via drinking water resulted in a surprisingly high incidence of oral carcinomas and uterine adenocarcinomas as reported by Sullivan et al.⁽⁷³⁾ Only two intestinal tumors were found over the subsequent lifetime of the experimental animals. Eve⁽⁷⁴⁾ gives a brief discussion of this work and the work of others, concluding that beta irradiation is an unlikely carcinogenic agent in the large intestine.

In summary, beta rays of sufficient number and energy will have a deleterious effect on the lining of the gastrointestinal tract and the damage is recognized in three succeeding steps: (1) Destruction of the epithelia cells with consequent loss of regenerative capacity, (2) Loss or reduction of normal organ function, and (3) A delayed onset of overfunction or overgrowth as in neoplasias. Additional work is required (and is underway) however, to define the relative effectiveness of small discrete source in producing such damage.

F. SUMMARY OF PATHOLOGICAL EFFECTS

Of the kinds of potential exposures resulting from nuclear propulsion reactor malfunctions, only internal exposures from discrete particles appear to suffer a major lack of significant experimental data. Generally accepted criteria reflecting a substantial body of information on the somatic effects of extensive whole body radiation are available. The skin dose from small source irradiations present some questions engendered by the localization of delivered dose but on the whole more information is available on skin effects than for internal exposures. Since projected average lung doses exposures from NERVA debris appear

to be quite small (see Section VI) the uncertainties in the consequences of lung exposure are limited to effects of high local doses in very small volumes. Perhaps the greatest uncertainties in dose-effect evaluation remain at present in the localized dose potentially delivered by small ingested debris particles to the gut wall, including the possible trapping effects by local ulceration, and any delayed effects of high localized irradiation of the gut wall.

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SECTION VIII

CONCLUSIONS AND RECOMMENDATIONS

A. INTRODUCTION

In this section is presented a critique of the adequacy of the models developed, the available criteria for interaction assessment, and the state of pertinent radiobiological research. Areas are identified in which additional refinement is necessary or desirable, and specific recommendations for such work are included. It must be recognized, however, that judgements of areas for further work must in some instances be viewed as preliminary since the importance of the interaction models may depend on their application to specific mission evaluations. Specific mission evaluations are presented in other volumes in this series.

B. INTERACTION MODELS

As discussed in Section III the nature of nuclear propelled rocket vehicles is such that conventional approaches to radiological safety analysis must be modified. Reliance must be placed in extensive probability analyses to obtain an insight into the degree of interaction between nuclear rocket debris and people. This probabilistic approach requires consideration of the behavior of the engine debris itself, the demographic and cultural characteristics of the world's people and the behavior of particles in man's environment.

The degree of precision with which the behavior and location of the debris is known from its generation in space until it is deposited in the biosphere is outside the scope of this report but is treated elsewhere in two volumes of this series. Once the debris reaches the biosphere certain interactions between debris and people have been assumed and are treated in terms of their probability of occurrence. Possible limitations of these particle-human interaction models are discussed below.

1. External Gamma Exposure

The external gamma exposure model assumes that all receptors are effectively one meter above the particle deposition plane. While this model is adequate for describing the exposure for many rural areas of the world it is of limited usefulness when large metropolitan areas are considered because such areas usually consist of multistory buildings. This affects not only the geometry but the distribution of people through the area.

The present model considers only those exposure situations where the gamma field is from either a single particle or an infinite plane source. Clearly for some ground deposition densities neither model is an adequate approximation. Related to this problem is the conditional probability of a person receiving a dose from several particles.

The sensitivity of the model to the use of population input data which does not account for inhomogeneities in the population density has not been tested. This may be important when urban population groups are considered.

The available cultural data used in the debris interaction model is based on estimates in many cases and further refinement in population totals and other parameters may be warranted.

2. External Skin Dose

Results obtained using the external skin dose contact probability model should be compared with the limited experimental data available to determine the adequacy of the basic impaction model. Further, experimental data to define sticking probability and sticking time for debris particles are required. The effect of clothing on these parameters has been neglected in this report because of a lack of data; this effect should be investigated. The model, however, is adequate to provide estimates of the importance of this dose route.

3. Lung Dose

The present model describing particle deposition in the lungs is overly conservative in that it is assumed that the impaction efficiency of particles into the inhalation air stream does not decrease with particle size. The present models do not account for the recently observed retention of large particles in lung reported in Section VII-E-1. The importance of that finding however rests on the likelihood of large particles being inhaled. It will be necessary

to look more carefully at the model if further research indicates that initial deposition of large particles in lung could be significant. To some extent adequacy of the lung interaction model depends on the relative importance of the expected doses. In terms of average lung doses the projected averages given in Section VI are so small that an extensive model revision is hardly warranted. If, however, significant information on the biological effects of highly localized doses in lung tissue is developed, a general revision of the model may be warranted.

4. Gastrointestinal Tract

The model for predicting the particle interaction with the gastrointestinal tract is believed to be adequate. The major route appears to be via the ingestion of contaminated foods. Cultural data regarding the food intake should be further refined. Improved data for the fraction of particles retained on certain food plants, the degree of particle removal by natural causes and food preparation, and the agricultural practices and experience of various cultures are required. The exposure model should be developed further to include motion and location of the particle in the gastrointestinal tract.

5. Metabolized Isotopes

No model for metabolized isotopes has been developed here because of the lack of solubility data for representative fuel material. It has been shown that solubility rate data for iodine are most important and should be obtained. Further consideration of this dose route is warranted if solubility rate data show that rapid release of fission products is possible.

C. RADIOLOGICAL DOSE MODELS

The dose models developed in this study to calculate the energy absorption from beta and gamma radiation are adequate for their intended purposes. The beta dose model is particularly unique in that effects of both the self-absorption in the source and a finite media are adequately considered. Unless biological information is developed that centers interest on the dose distribution within a few microns of a source, the present beta models are sufficiently accurate for any future safety analyses.

However, from the standpoint of user utility the CBS dose model requires much repetitive calculational effort. As now constituted it treats each exposure situation as unique depending upon the reactor history, age of debris and time exposure. With very little effort it should be possible to normalize dosage distributions so that they can be tabulated and stored on tape. With suitable interpolation codes, machine calculational effort would be minimized and the problem largely reduced to a table look-up procedure.

As mentioned in Section V the CBS model is limited to uranium carbide debris. The effects of self-absorption are much less in the case of graphite debris but should be investigated and the model extended to include large pieces of such debris. Lastly, there is a paucity of good experimental data with which to check the dosage calculations.

The narrow beam dose reduction factors now used in the gamma dose model are not entirely appropriate to the exposure situations considered in this report. Broad beam and infinite plane source dose reduction factors are being developed for civil defense application and could be substituted. The gamma dose to the skin and gut wall is not included in the present analysis mainly because these effects would be small compared to the beta dose. However, information on the x-ray and soft gamma spectra from debris material is lacking and this point cannot be proven until such data is available.

D. CRITERIA

As described in Section VII, radiation protection criteria have been evolved which consider both the probable effects on the exposed individual or group and the reason for the exposure. Emergency criteria for the general population have been developed by the British Medical Research Council which include the doses arising from external sources and from ingestion or inhalation of soluble materials released in an accident at a nuclear facility. The Federal Research Council has also promulgated Protective Action Guides from which emergency acceptable doses to bone, thyroid and whole body can be inferred.

Of the types of interactions considered to be significant in nuclear rocket flight operations, some guidance on exposure criteria is available for all but the gastrointestinal tract. Whole body gamma dose, small area skin exposure and lung dose have been treated by either FRC or the BMRC. Similar guidance for localized irradiation of the gastrointestinal

tract wall is required. Additional guidance for highly localized doses to the skin and lung would be helpful in safety calculations.

Consideration of the applicability of criteria developed for other purposes to the nuclear rocket program is indicated for several reasons. First, the geographic distribution of these doses would be much greater than that contemplated in the established emergency exposure criteria. Next, the established criteria implicitly assume an official knowledge of the exposure and hence an ability to reduce exposures by controlling exposed individuals; this may not be the case in a nuclear rocket accident.

E. RADIOBIOLOGICAL EFFECTS

The considerations of appropriate criteria for this application described in the preceding section, depend upon several factors: the calculated distribution of various exposure levels over the affected population, the number of people so affected, and the biological consequences of the range of doses for each of the interactions. Whole body dose effects from acute exposure are reasonably well understood. Additional information on effects of protracted irradiation would be helpful in safety assessments. In view of the anticipated small doses information on lung dose effects is sufficient to permit a preliminary assessment of the significance of effects expected from this application.

Further work is required (and is in progress) to assess the effects of highly localized doses to skin, lung and to the gastrointestinal tract. More information on skin interactions is available than is the case for the gastrointestinal tract; the latter would also appear to be more critical in the sense of its importance to overall body function. Continued work in these areas with emphasis on the gastrointestinal tract response would appear to be highly desirable.

F. RECOMMENDATIONS

In view of the comments presented above with regard to the Radiological Effects Program, some recommendations are made below. The evaluation of certain of these recommendations however must also consider the importance of the interaction as shown by the application of Program models to specific missions performed by other SNPO Working Groups.

1. The external gamma dose model is of limited validity when applied to urban areas. It is recommended that the model be extended to treat such areas more realistically and include the case of multiple interactions from several particles.

2. The parameters and modeling concepts employed in the skin interaction model require a great deal of additional experimental support. It is recommended that research efforts on impaction efficiencies, retention probability and duration be extended to verify assumptions used in the present model. The effects of clothing on particle retention should be studied.

3. Gastrointestinal tract exposure remains semi-enigmatic; the sensitivity analysis indicates that ingestion of debris fragments on fresh vegetables is the major route of entry. It is recommended that additional emphasis be placed on evaluating experimental evidence concerning the deposition and retention of such fragments by appropriate food crops, particularly data on these factors as a function of particle size in order to support or reject the factors presently used. The probable decay period before ingestion takes place should also be considered.

4. Radionuclide solubility rate data is almost entirely non-existent for debris fragments exposed to a simulated system failure and re-entry environment. Although a preliminary evaluation indicates that only the radioiodines should provide a significant dose if the fission products are metabolized, it is recommended that solubility rate data be obtained for all biologically significant (i.e., low MPC) fission products.

5. The radiological dose calculation models are sufficiently accurate to treat the present situations. However, if a large number of safety analyses are to be made using these schemes it is recommended that the CBS model be simplified to make its use more economic, and that it be extended to treat graphite debris as well as uranium carbide. Further, it is recommended that experimental work to verify the calculated beta depth dose distributions be continued.

6. A major requirement for this program is applicable criteria for mission evaluation. Although in an accident, the vast majority of the exposed population should receive doses well below current guides for normal peacetime application, a finite probability may exist of a few individuals receiving greater doses. It is recommended that guidance be sought from appropriate agencies for criteria applicable to this situation.

7. Support for the safety criteria and a safety assessment program must derive from knowledge of biological effects of the calculated interactions. It is considered that the greatest present lack in this area is that associated with the gastrointestinal tract exposure. It is recommended that present studies of this exposure route be extended. Studies on skin and lung exposure should be continued at the present level of effort.

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