

Letter Report

**RADIATION DOSE AND HEALTH RISK ESTIMATION: TECHNICAL  
BASIS FOR THE STATE-OF-THE-ART REACTOR CONSEQUENCE  
ANALYSIS PROJECT**

Prepared by

K.F. Eckerman  
Oak Ridge National Laboratory

January 30, 2012

# **Radiation Dose and Health Risk Estimation: Technical Basis for the State-of-the-Art Reactor Consequence Analysis Project**

## **1.0 Introduction**

This report describes the radiation dose and health risk assessment methodology used in the State-of-the-Art Reactor Consequence Analysis (SOARCA) project. The SOARCA project uses the MACCS2 computer code (MELCOR Accident Consequence Code System, version 2) to model offsite health effects by first calculating the radiological exposure to the population and then estimating resultant health effects using a dose-response model. The objective of the SOARCA analysis is to estimate a plausible range of delayed health effects with their associated uncertainties. For the accident scenarios analyzed, the risk of dying from prompt health effects (early fatalities) was considered to be minimal because the doses projected to be received by members of the public at offsite locations were below the threshold for prompt health effects.

The risk of dying of a latent cancer from radiation exposure resulting in lower doses was derived using three implementations of the dose-response model: a linear no-threshold (LNT) approach; exclusion of annual doses less than the U.S. annual dose from natural background and medical exposure (6.2 mSv or 620 mrem) (NCRP 2008); and exclusion of annual doses of less than 0.05 Sv (5 rem) with a lifetime limit 0.1 Sv (10 rem) (HPS 2010). The LNT model assumes that any dose can increase the probability of a radiogenic cancer death (an assumption within the regulatory framework of radiation protection). These approaches reflect the range of opinions of national and international experts on cancer mortality caused by chronic low-level radiation exposure.

The latent cancer fatality risk coefficients used in the SOARCA analysis are based on the technical approach described in the National Academy of Sciences/National Research Council report entitled, “Biological Effects of Ionizing Radiation (BEIR): Health Effects of Exposure to Low Levels of Ionizing Radiation” (BEIR V). Details regarding the implementation of the BEIR V models are given in the U.S. Environmental Protection Agency (EPA) report entitled, “Estimating Radiogenic Cancer Risk” (EPA 1994).

The focus of this report is on the methodology used in the assessment of radiation dose and latent health effects (cancer induction) within the population. Radiation dose quantities are involved in two different aspects of the post-accident situation: (1) the decision to allow the evacuated population to return to the site environs and (2) the estimation of the potential health effects in the population. The first issue falls within the scope of radiation protection and thus the dosimetric quantities of radiation protection are applicable. The second issue, projection of health effects, is outside the general scope of radiation protection and thus involves quantities and computational methods beyond those of radiation protection.

This report discusses the dose and risk coefficients used in the SOARCA health effects analysis and their associated uncertainties.

## 2.0 Dosimetric Quantities

The primary physical dosimetric quantity, absorbed dose, denoted as  $D$ , is defined as the ionizing energy absorbed per unit mass within an organ or tissue. The International System of Units (SI) unit<sup>1</sup> of absorbed dose is J/kg with the special name gray (Gy). The conventional non-SI unit of absorbed dose is the rad, 1 rad = 0.01 Gy. The absorbed dose in tissue  $T$  of the body is denoted as  $D_T$ .

The different radiations (electrons, photons, alpha particles, etc.) vary in their potential to cause damage in tissues of the body largely as a result of variation in the density of induced ionizations. The dosimetric quantity equivalent dose, denoted as  $H_T$ , is the primary radiation protection quantity and is defined as

$$H_T = \sum_R w_R D_{T,R} \quad (1)$$

where  $w_R$  is the radiation weighting factor for radiation  $R$  and  $D_{T,R}$  is the absorbed dose in tissue  $T$  delivered by radiation  $R$ . The radiation weighting factor is a subjective quantity that attempts to put all radiations on a common scale without regard to the particular tissue or to the specific biological response to the radiation. For radiations characterized by a low transfer of energy per unit path length (so-called linear energy transfer [LET]),  $w_R$  is assigned a value of 1. This group of radiations include beta particles, electrons, and x and gamma rays. In contrast, alpha particles deposit their energy over a short path and thus are classified as high LET radiations. Alpha particles are assigned a  $w_R$  value of 20. Neutrons also are referred to as high LET radiation and assigned  $w_R$  values that are a function of neutron energy. The SI unit of equivalent dose is J/kg with the special name sievert (Sv); the conventional non-SI unit of equivalent dose is the rem, 1 rem = 0.01 Sv.

The response of tissues of the body to radiation is dependent on the magnitude of the dose. The biological effects of the radiation can be classified as deterministic or stochastic. Deterministic effects only occur if the radiation dose exceeds the threshold for the effect. These effects include the acute radiation syndrome that requires prompt medical care and can lead to death. Within occupational radiation protection, the annual limit on dose is set such that the cumulative dose over a working lifetime is well below the threshold for such effects.

Stochastic effects exhibit a probability of occurrence, assumed to have no threshold, which is dependent on dose. These effects include genetic effects that are manifested in progeny of the exposed individual and radiogenic cancer induction. The effective dose quantity was introduced into radiation protection as a means to represent the risk (probability of occurrence) of stochastic effects resulting from exposure to radiation incident on the body (so-called external exposure) and from the intake of radionuclides (so-called internal exposure). The effective dose  $E$  is defined as

$$E = \sum_T w_T H_T \quad (2)$$

where  $w_T$  is the tissue weighting factor for tissue  $T$  and  $H_T$  is the equivalent dose in tissue  $T$ . The tissue weighting factor reflects the health detriment associated with irradiation of the specific tissue and includes subjective judgments regarding induction of a radiogenic cancer or morbidity (incidence), the mortality, loss of life expectancy, and impact on the quality of life. Thus, the tissue weighting factors do not just

---

<sup>1</sup> The MACCS2 software is written using SI units and thus the coefficients discussed in this report are expressed in SI units. Software development is aided greatly by adopting a coherent set of units.

reflect cancer mortality, which is the health measure adopted in the SOARCA project. The SI unit of effective dose is J/kg with the special name sievert (Sv). The conventional non-SI unit of effective dose is the rem, 1 rem = 0.01 Sv. The effective dose quantity used in the SOARCA project is based on the tissue weighting factors in Publication 60 of the International Commission on Radiological Protection (ICRP 1990). Table 1 lists these factors which are averaged over both genders and all ages. The nominal probability coefficient for cancer induction in the exposed population is 6.0 % Sv<sup>-1</sup> and 5.0 % Sv<sup>-1</sup> for fatal cancers. In the worker population, the nominal values are 4.8 % Sv<sup>-1</sup> for cancer induction and 4.0 % Sv<sup>-1</sup> for fatal cancers.

**Table 1. Tissue Weighting Factors of ICRP 60**

<b>Tissue</b>	<b><math>w_T</math></b>
Gonads	0.20
Active marrow	0.12
Lung	0.12
Colon <sup>a</sup>	0.12
Stomach	0.12
Breast	0.05
Thyroid	0.05
Bladder	0.05
Esophagus	0.05
Liver	0.05
Bone surface	0.01
Skin	0.01
Remainder <sup>b</sup>	0.05

<sup>a</sup>The colon is composed of the upper and lower large intestine.  
<sup>b</sup>The remainder is composed of adrenals, brain, small intestine, kidney, muscle, pancreas, spleen, thymus, and uterus.

Within the SOARCA project, the criteria for return of the evaluated population to the site environs are based on the annual effective dose. The assessment of cancer risk in the relocated population may be truncated when the annual effective dose falls below a truncation value (e.g., below the annual background dose). The effective dose is not used to assess health risk; rather, absorbed dose— modified by an applicable relative biological effectiveness factor (RBE) —is used to estimate the radiogenic cancer mortalities.

## 2.1 Absorbed Dose Coefficients

The dosimetric quantity used in assessment of cancer induction is the absorbed dose, modified by an RBE greater than one for high LET radiation component specific to cancer induction. The population is exposed to photon emission (low LET radiation) of radionuclides deposited on the ground surface and potential inhalation intakes of radionuclides resuspended from the ground surface that might involve both low and high LET radiations. The eight cancer sites considered in the SOARCA project are leukemia, bone, breast, lung, thyroid, liver, colon, and residual group. The residual group includes all other cancer sites not explicitly identified. The SOARCA project retained the conventional approach to cancer risk

estimation, that is, risk  $R_T$  is computed as the product of the absorbed dose in tissue  $T$ ,  $D_T$ , and a risk coefficient  $r_T$ :

$$R_T = r_T D_T . \quad (3)$$

The total risk is the sum of the risks over the eight individual cancer sites assuming that each cancer site is independent of the others. An alternative approach would be to use the nuclide-pathway specific risk coefficients of Federal Guidance Report 13 (FGR 13) (EPA 1999); however, that would have required a major revision of MACCS2 software. The restriction to address explicitly seven cancer sites plus “residual” cancers was inherited from an earlier version of the software.

### 2.1.1 Absorbed dose rate coefficients for exposure to deposited activity

The SOARCA analysis considers exposure to radiations emitted by the dispersing airborne radionuclides and that arising from their deposits on the ground surface. The absorbed dose coefficients for exposure to photons emitted by radionuclides external to the body used in the SOARCA analysis are those of Federal Guidance Report 12 (FGR 12) (EPA 1993). FGR 12 considered a number of external exposure pathways including submersion in a semi-infinite cloud, exposure to ground surface contamination, exposure to soil contaminated to various depths, and immersion in contaminated water. The dominant route of exposure in the SOARCA analysis is exposure to contaminated ground surfaces. In FGR 12, dose rate coefficients for the tissues and organs of an individual standing on the contaminated ground surface were calculated by Monte Carlo methods using a mathematical representation of an adult as defined in ICRP Publication 23 (1975) (Figures 1 and 2). Some age dependence in the organ dose rates is expected; however, the differences are minor for high-energy photons but approach a factor of 3 at photon energies below 100 keV (Yamaguchi 1994). Figure 3 illustrates the energy dependence of the effective dose as a function of age where the photons are incident normally on the body surface and thus minimize the shielding of organs by overlying tissues.

If one assumes a uniform concentration,  $C(t)$ , of the radionuclide on the ground surface at time  $t$  then the absorbed dose rate in tissue  $T$ ,  $\dot{D}_T(t)$ , is

$$\dot{D}_T(t) = C(t) \dot{d}_T \quad (4)$$

where  $\dot{d}_T$  is the nuclide specific dose rate coefficient for tissue  $T$  as tabulated in FGR 12. Only low LET radiations (photons and electrons) are sufficiently penetrating to contribute to the absorbed dose rate and their RBE is one. Integration of the dose rate over the exposure period of 1 year yields the annual absorbed dose. Appendix A lists the absorbed dose rate coefficients for the cancer sites and the effective dose rate coefficient for the radionuclides in the SOARCA project.

### 2.1.2 Uncertainty in absorbed dose rate coefficients for exposure to deposited activity

The ground plane dose coefficients of FGR 12 were derived assuming the radionuclide is deposited on a flat plane of infinite extent. The report included advice with regard to application of the coefficients. It is suggested, for example, that the coefficients be multiplied by 0.7 to account for the departure from the infinite flat plane due to “ground roughness.” Additional factors such as indoor residence time and household shielding also were discussed. MACCS2 includes consideration of these factors and their uncertainties in its assessment of external dose. The “ground roughness” factor is generally overwhelmed by the household shielding factor.

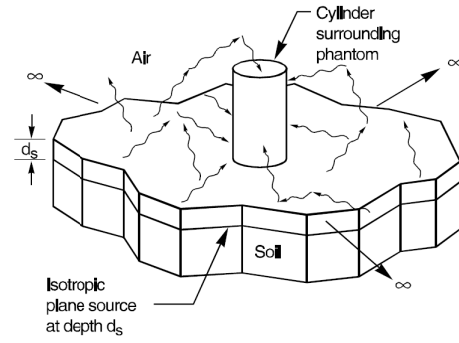


Figure 1. Calculation of radiation field due to contaminated ground plane on a cylinder surrounding the mathematical model of the body (computational phantom) (EPA 1993).

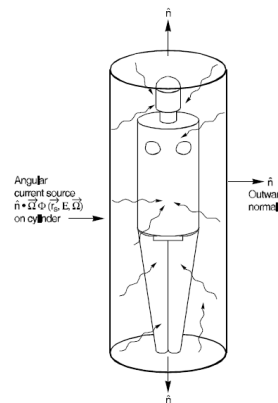


Figure 2. Calculation of organ absorbed dose from an angular current source on the cylinder surrounding the computational phantom (EPA 1993).

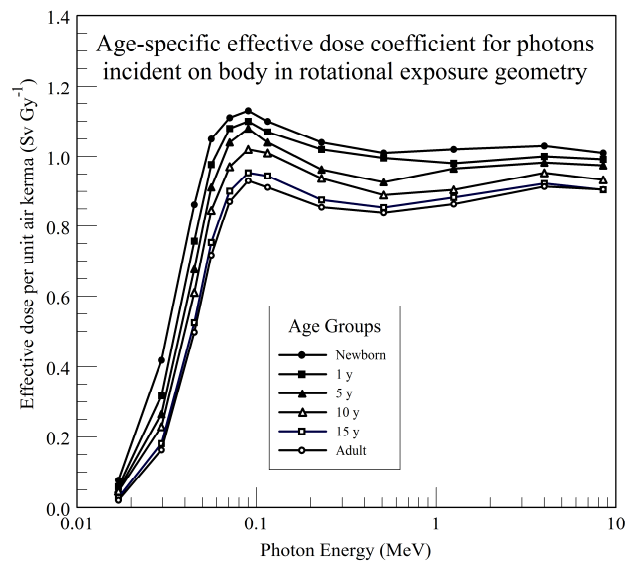


Figure 3. Illustration of age-dependence in effective dose for external irradiation (Yamaguchi 1994).

A multiplicative uncertainty in FGR 12's ground plane dose rate coefficient is assumed to be described by a triangular distribution, which increases linearly from 0.5 to 0.8 and then decreases to 1.5. The organ dose rate coefficients are highly correlated and thus a perfect correlation is assumed. The assigned triangular distribution and assumed correlation coefficient are based on subjective judgment.

As an illustrative exercise, assume that the deposition of Cs-137 on the ground surface following a release is  $1 \text{ MBq m}^{-2}$ . Further assume the shielding factor (SF) afforded the population can be described by a triangular distribution with a mode of 0.18, lower bound of 0.1 and upper bound of 0.5. The dose rate coefficient for induction of leukemia (target tissue is the active marrow of the skeleton),  $d_T$ , is  $5.58\text{E-}16 \text{ Gy/s per Bq/m}^2$ . Neglecting radioactive decay, the dose to the active marrow,  $D_T$ , resulting from a 1-year exposure to this contamination would be given by

(5)

where  $3.16 \times 10^7$  is the number of seconds in a year,  $C$  is the concentration of Cs-137 on the ground surface, SF is the shielding factor, and  $d_T$  is the dose rate coefficient for Cs-137. The uncertainties in both SF and  $d_T$  are considered. If one evaluates Eqn 5 with the mode values of SF and  $d_T$  then

(6)

or  $3.17 \times 10^{-3} \text{ Gy}$  or 0.32 rad. The result of a Monte Carlo simulation of Equation 5 with the uncertainties in SF and  $d_T$  is shown as a cumulative distribution in Figure 4. The median value of the simulation (5,000 trails) was  $3.95 \times 10^{-3} \text{ Gy}$  (0.39 rad) with 90 percent of the values lying between  $2.00 \times 10^{-3}$  to  $7.58 \times 10^{-3} \text{ Gy}$  (0.2 to 0.76 rad).

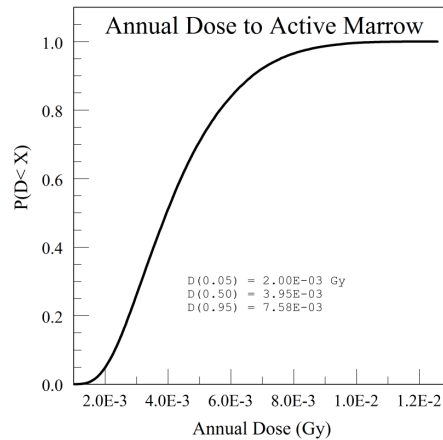


Figure 4. Cumulative distribution of the annual dose from a deposition of  $1 \text{ MBq m}^{-2}$  of Cs-137 due to uncertainties in the shielding and dose rate coefficient.

### 2.1.3 Absorbed dose coefficients for inhalation intakes of radionuclides

The behavior of a radionuclide entering the body is described in terms of three main types of mathematical models collectively referred to as biokinetic models. These include a respiratory tract model, a gastrointestinal tract model, and an element-specific systemic model. The generic respiratory tract model is used to describe the deposition and retention of inhaled material in the tract and its

subsequent clearance to blood (absorption) or to the gastrointestinal tract. The generic gastrointestinal tract model describes the movement of swallowed or endogenously secreted material through the stomach and intestines and, together with element-specific gastrointestinal absorption fractions ( $f_1$  values), the rate and extent of absorption from the small intestine to blood. Element-specific systemic biokinetic models describe the time-dependent distribution of the absorbed activity among the organs and tissues and the subsequent elimination of radionuclides from the body.

The ICRP respiratory tract model (1994) divides the respiratory system into extrathoracic (ET) and thoracic regions as shown in Figure 5. Materials deposited in the anterior nasal passages of the ET region are removed by extrinsic means (such as nose blowing) while materials in the posterior region are swallowed and enter the gastrointestinal tract. Materials deposited in the thoracic airways (bronchi - labeled *BB* in Figure 5, bronchioles- labeled *bb*, and alveolar region- labeled *AD*) may be cleared to the GI tract by mechanical processes (i.e., transported upward and swallowed) and to the regional lymph nodes by lymphatic channels. The mechanical clearance of particles indicated in Figure 5 is in addition to possible dissolution and absorption to blood, which depends on the chemical and physical form of the inhaled material. Particulate matter is generally assigned to one of three default absorption types: Type *F* (fast dissolution and high level of absorption to blood), Type *M* (an intermediate rate of dissolution and level of absorption to blood), and Type *S* (slow rate of dissolution and low level of absorption to blood). The fractional rate of absorption ( $d^{-1}$ ) assigned to the default types is

Type <i>F</i>	100
Type <i>M</i>	$10.0 e^{-100 t} + 0.005 e^{-0.005 t}$
Type <i>S</i>	$0.1 e^{-100 t} + 1.0 \times 10^{-4} e^{-0.0001 t}$

where  $t$  is the time (days) since deposition in the lung.

The model of the gastrointestinal (GI) tract (Figure 6) divides the GI tract into four segments or compartments: stomach (St), small intestine (SI), upper large intestine (ULI), and lower large intestine (LLI). Material is assumed to transfer from St to SI at the fractional rate of  $24 d^{-1}$ , from SI to ULI at  $6 d^{-1}$ , from ULI to LLI at  $1.8 d^{-1}$ , and from LLI to the compartment Feces at  $1 d^{-1}$ .

Absorption of ingested material to blood is generally assumed to occur only in SI and is described in terms of the fraction  $f_1$ . In the absence of radioactive decay, the fraction  $f_1$  of ingested material moves from SI to Blood, and the fraction  $1 - f_1$  moves from SI to ULI and eventually is excreted in feces. The transfer coefficient ( $d^{-1}$ ) from SI to Blood is  $6 f_1 / (1 - f_1)$  where  $6 d^{-1}$  is the transfer coefficient from the SI to ULI.

In the SOARCA project, ingestion of locally produced foodstuff is an exposure pathway assumed to be closed to the population upon return to the site environs. However, in the calculation of doses from inhalation of radionuclides, consideration is given to the absorption of material passing through the GI tract after clearance from the respiratory tract. A fraction of the inhaled particles enter the GI tract from which the absorption to blood depends on dissolution of the particle matrix as well as the elemental form of the radionuclide. In ICRP Publication 72 (ICRP 1995) and in FGR 13, element-specific  $f_1$  values applied to ingestion generally are also applied to inhalation of Type *F* compounds. For inhaled material of Type *M* or *S*, a default  $f_1$  value of 0.1 or 0.01, respectively, is applied unless a lower  $f_1$  value for that



The diagram illustrates a compartmental model of SARS-CoV-2 infection, divided into two main regions: Extrathoracic and Thoracic. The Extrathoracic region includes compartments for Anterior nasal (ET<sub>1</sub>), Naso-oro-pharynx (ET<sub>2</sub>), and larynx (ET<sub>SEQ</sub>). The Thoracic region includes compartments for Bronchi (BB<sub>1</sub>, BB<sub>2</sub>), Bronchioles (bb<sub>1</sub>, bb<sub>2</sub>), and Alveolar interstitium (Al<sub>1</sub>, Al<sub>2</sub>, Al<sub>3</sub>). The model shows the progression of infection from the environment through the respiratory tract to the alveolar interstitium. Key transitions and rates are: ET<sub>1</sub> to Environment (1), ET<sub>2</sub> to GI Tract (100), ET<sub>1</sub> to ET<sub>2</sub> (10), ET<sub>2</sub> to BB<sub>1</sub> (0.03), BB<sub>1</sub> to BB<sub>2</sub> (0.01), BB<sub>2</sub> to bb<sub>1</sub> (0.03), bb<sub>1</sub> to bb<sub>2</sub> (0.0004), bb<sub>2</sub> to Al<sub>1</sub> (0.001), Al<sub>1</sub> to Al<sub>2</sub> (0.02), Al<sub>2</sub> to Al<sub>3</sub> (0.0002), and Al<sub>3</sub> to LN<sub>TH</sub> (0.0002). LN<sub>TH</sub> is a central compartment in the Thoracic region. The model also includes a compartment for Sequestered in tissue (ET<sub>SEQ</sub>) in the Extrathoracic region, which transitions to ET<sub>1</sub> (0.001). The GI Tract is shown as a destination for ET<sub>2</sub> and has a downward arrow indicating further progression.

```

graph TD
    Ingestion --> St
    St -- 24 d⁻¹ --> SI
    SI -- 6 d⁻¹ --> ULI
    ULI -- 1.8 d⁻¹ --> LLI
    LLI -- 1 d⁻¹ --> Feces
    St --> RT[Respiratory Tract]
    RT --> Blood
    SI --> Blood
    Blood --> RT
  
```

The diagram illustrates the digestive tract and its connection to the blood. The process begins with Ingestion, leading to the Stomach (St). From the Stomach, material moves to the Small Intestine (SI) at a rate of 24 d<sup>-1</sup>. From the SI, it moves to the Upper Large Intestine (ULI) at 6 d<sup>-1</sup>, then to the Lower Large Intestine (LLI) at 1.8 d<sup>-1</sup>, and finally to Feces at 1 d<sup>-1</sup>. The Respiratory Tract is connected to the Stomach and the Blood. The Blood is a central component that receives input from the Stomach and the Small Intestine, and provides output to the Respiratory Tract.

In ICRP publications on dose coefficients for members of the public, physiologically based models were used to describe the systemic biokinetics of radioisotopes of calcium, iron, strontium, iodine, barium, lead, radium, thorium, uranium, neptunium, plutonium, americium, and curium. This model framework depicts loss of material by specific excretion pathways, feedback of material from organs to blood plasma, and certain physiological processes that are known to influence the distribution and translocation of the element from the body. The degree of biological realism incorporated into each of the models is limited by practical considerations regarding the amount and quality of information available to determine actual paths of movement and parameter values for specific elements. To illustrate the nature of these models, the structure of the biokinetic models for iodine and strontium are shown in Figures 7 and 8, respectively.

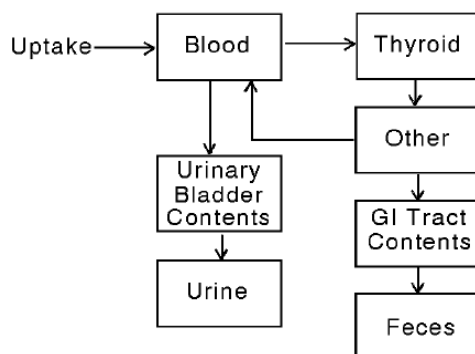


Figure 7. Structure of the ICRP biokinetic model for iodine (ICRP 1989).

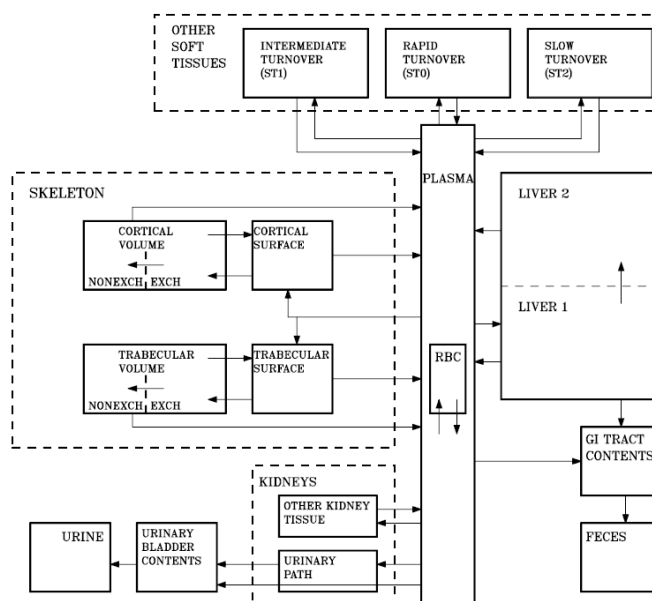


Figure 8. The ICRP generic model structure for strontium and other alkaline earth elements (ICRP 1993).

The biokinetic models of the respiratory tract, the gastrointestinal tract, and the systemic models define a set of coupled differential equations that describe the time-dependent distribution of the parent radionuclide and members of its decay chain in the tissues of the body. This set of equations is typically solved as an initial value problem using a numerical solver. The solver in the DCAL computational system (Eckerman et al., 1999) used in the production of FGR 13 is that of Leggett et al. (1993).

The dosimetric methodology used in FGR 13 is that of the ICRP and is generally consistent with the schema of the Medical Internal Radiation Dose Committee (MIRD) of the Society of Nuclear Medicine (Loevinger and Berman, 1968). The methodology considers two sets of anatomical regions within the body. A set of “source regions” represents anatomical regions specifying the location of the activity as indicated by the biokinetic models applicable to the inhaled material. The set of “target regions” consist of the organs and tissues for which the radiation dose is to be estimated.

The energy absorbed in target regions depends on the nature of the radiations emitted in the source regions, the spatial relationship between the source and target regions, and the nature of the tissues between the regions. The details of these considerations are embodied in the radionuclide-specific coefficient called the specific energy (SE). For any radionuclide, source region  $S$  and target region  $T$ , the specific energy is given by

$$SE(T \leftarrow S) = \frac{1}{M_T} \sum_i Y_i E_i AF_i(T \leftarrow S) \quad (7)$$

where  $Y_i$  is the yield of radiation of type  $i$  per nuclear transformation of the radionuclide,  $E_i$  is the average or unique energy of radiation  $i$ ,  $AF_i(T \leftarrow S)$  is the fraction of energy emitted in  $S$  that is absorbed in  $T$ , and  $M$  is the mass of  $T$ . The quantity  $AF_i(T \leftarrow S)$  is referred to as the absorbed fraction (AF), and division by the mass of  $T$ ,  $M_T$ , yields the specific absorbed fraction (SAF).

The absorbed dose in  $T$  includes contributions from each region in the body where the radionuclide resides. Thus, the absorbed dose rate in  $T$  at time  $t$ ,  $\dot{D}_T(t)$ , can be expressed as

$$\dot{D}_T(t) = c \sum_S \sum_j q_{S,j}(t) SE_j(T \leftarrow S) \quad (8)$$

where  $q_{S,j}(t)$  is the activity of radionuclide  $j$  present in source region  $S$ ,  $SE_j(T \leftarrow S)$  is the specific energy deposited in target  $T$  per nuclear transformation of the chain member  $j$  in source region  $S$ , and  $c$  is any numerical constant required by the units of  $q_{S,j}$  and  $SE$ . If  $q_{S,j}$  is normalized to a unit intake at time zero, then the left hand side of Equation 8 represents the absorbed dose rate in  $T$  per unit activity of the parent radionuclide inhaled at time zero or simply an absorbed dose rate coefficient and denoted by a lower case  $d_T$  as  $\dot{d}_T(t)$ . If the coefficient is integrated over time, post an acute intake, the resultant quantity is the committed dose coefficient  $d_T$ . In radiation protection, the commitment period is taken to be 50 years. In many instances, the integral converges due to the physical half-life of the radionuclide and biological elimination from the body as illustrated in Figure 9.

The equivalent dose is obtained by modified Equation 7 for  $SE$  by the radiation weighting factor applicable to radiation  $i$ . In a similar manner one can use a RBE applicable to the target tissue  $T$  and radiation  $i$  such that Equation 8 yields an RBE-weighted absorbed dose.

Except for breast cancer and leukemia, in FGR 13 EPA followed the ICRP recommendations and assumed an RBE for alpha particles of 20 relative to low-LET radiations at low dose and dose rate. For leukemia an alpha RBE of 1 is used and, for breast cancer, an alpha particle RBE of 10 is used (EPA 1994, 1999). As noted earlier the RBE is 1 for low LET radiations; i.e., photons and electrons.

Appendix B lists for inhalation, the committed absorbed dose coefficients for the cancer sites considered and the committed effective dose coefficient for the radionuclides in the SOARCA Project.

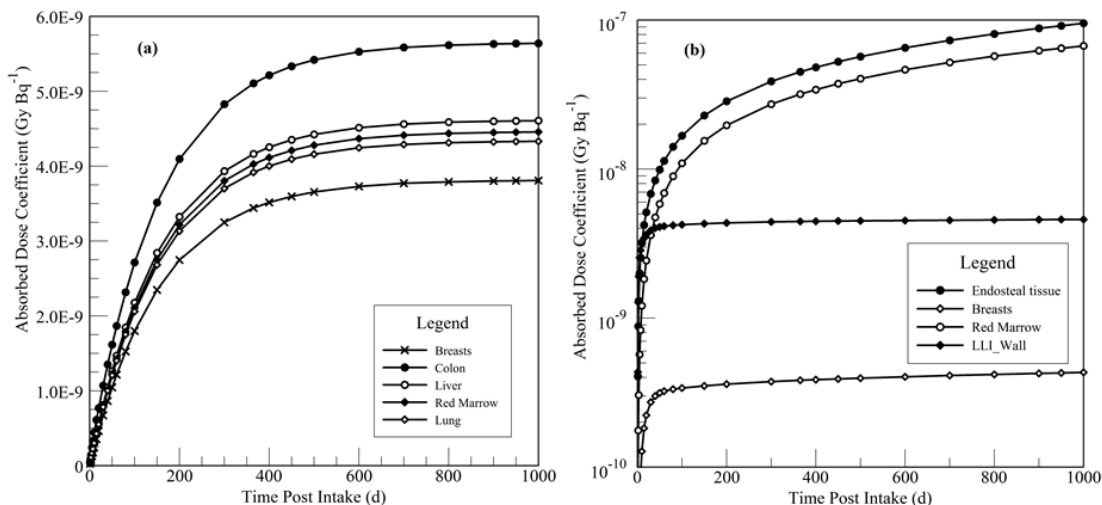


Figure 9. Temporal pattern in absorbed dose following an acute inhalation intake at time zero; the aerosol is assumed to be characterized by an activity median aerodynamic diameter (AMAD) of 1  $\mu\text{m}$ . (a) Absorbed dose delivered to selected tissues following inhalation of Cs-137. (b) Absorbed dose delivered to selected tissues following inhalation of Sr-90.

#### 2.1.5 Uncertainties in absorbed dose coefficients for inhalation intakes of radionuclides

The uncertainties in the inhalation dose coefficients were examined by Pawel et al. (2007) in their effort to evaluate the uncertainties in the cancer risk coefficients of FGR 13 (EPA 1999). The assessment of the uncertainty in dose was evaluated on the basis of sensitivity analyses in which various combinations of plausible biokinetic and dosimetric models were used to generate alternative dose coefficients. Choices of models, relevant parameters, and distributional assumptions for characterizing uncertainties in the biokinetics of radionuclides were derived in part by considering the extent to which assumptions and parameter values underlying ICRP models might be reasonably altered for a given element.

Assignment of uncertainty distributions to the dose coefficients of the various radionuclides is difficult because these values are end products of complex calculations involving a collection of uncertain biokinetic and dosimetric models and assumptions. Current biokinetic models for elements generally are not process-level models, and their parameter values often do not represent measurable quantities. Conversion from internally distributed activity to tissue doses involves the application of specific energies (SE values) for numerous pairs of target and source regions, and the uncertainty in a given SE value depends on the types and energies of emitted radiations. Even if the information were available to assign meaningful uncertainty distributions to all individual parameter values of the biokinetic and dosimetric models, a comprehensive uncertainty analysis would not be feasible due to the large number of parameters involved and generally inadequate information on their correlation.

In view of such difficulties, a systematic scheme was devised to produce computer-generated ranges of plausible dose estimates based on a limited number of alternatives to the dominant components in the biokinetic and dosimetric models. The methodology consisted of identification of those components of the models that are usually the dominant sources of uncertainty in absorbed dose estimates based on sensitivity analyses. For each radionuclide, a few substantially different but equally plausible variations of each of the dominant components were constructed. Absorbed dose coefficients were calculated using

each combination of the constructed dominant components, with all other aspects of the biokinetic and dosimetric models left unchanged.

The following components were judged to represent the dominant uncertainties in most situations: the fraction of inhaled material deposited in the respiratory tract, the rate of absorption from the respiratory tract to blood, the gastrointestinal absorption fraction ( $f_i$ ), the systemic biokinetic model, and SE values for certain combinations of source and target organs and radiation types. For inhalation of a radionuclide of a given absorption type, as many as 54 combinations were considered (i.e., all distinct combinations of 3 respiratory models, 3 systemic models, 3  $f_i$  values, and 2 sets of SE values). The set of 54 ( $3 \times 3 \times 3 \times 2$ ) dose coefficients may be thought of as a measure of uncertainty or, perhaps more appropriately, measures of imprecision in the dose coefficients arising from typically dominant uncertainties associated with the underlying models. For further details regarding this method, consult Pawel et al. (2007).

For each radionuclide considered in the SOARCA analysis, the lower and upper values of the inhalation dose coefficients were extracted from the files created as outlined above. These values were assumed to represent the 90 percent confidence interval for the dose coefficient for the eight target tissues considered in the cancer risk assessment. It was then assumed that the underlying probability distribution is a log-normal distribution, and their geometric standard deviations are listed in Appendix B. The log-normal distribution should be considered as bounded at appropriate percentiles.

In principle, the correlation of the dose coefficients for the target tissues could be estimated from the set of 54 derived dose coefficients. This procedure would yield an 8 by 8 matrix (null values on the diagonal), which is not workable within MACCS2. For inhalation intakes, it is suggested that the lung dose coefficient be viewed as negatively correlated with the coefficients for the systemic tissues. Depending on the details in the systemic biokinetic model, the dose coefficients for the systemic tissues might be positively or negatively correlated. We suggest assuming a positive correlation in the absence of specific information. A perfect correlation coefficient of 1.0 is assumed to be negative with respect to lung and any systemic tissue and positive among the systemic tissues.

The inhalation absorbed dose coefficients for cancer sites (geometric mean) and radionuclides considered in the SOARCA project and the effective dose coefficient are listed in Appendix B along with the geometric standard deviation of the log-normal distribution.

### **3.0 Estimation of Latent Cancer Risk**

The assessment of latent health effects (cancer mortality) in the SOARCA project is based on the BEIR V vintage (NAS 1990) risk models as detailed by EPA (1994) and used in Federal Guidance Report 13 (EPA 1999). The estimated mortality risk obtained using these models is influenced by a host of parameters including those defining the course of events within the plant and parameters associated with the offsite emergency response plan.

Parameters governing the radiation exposure of the population and used in the estimation of the latent cancer risk among members of the public were reviewed. The focus here is on the exposures of members of the public following their return to the vicinity of the plant. The operative pathways are long-term exposure to photons emitted from contaminated ground deposits (groundshine) and inhalation of resuspended activity.

### 3.1 Basis for Cancer Risk Estimation

It is common practice to estimate the cancer risk from intakes of a radionuclide or external exposure to its emitted radiations as the product of a “probability coefficient” and an estimated “effective dose” to a typical adult. For example, a “nominal cancer fatality probability coefficient” of  $0.05 \text{ Sv}^{-1}$  is given in ICRP Publication 60 (1991) for all cancer types combined. As ICRP noted, the resultant risk estimate may be inaccurate in situations where the tissues of the body are not uniformly irradiated and the irradiation is delivered through the life-span of the exposed individual. FGR 13 (EPA 1999) presents risk coefficients for specific radionuclides applicable to exposure of the U.S. population in the sense that the estimated risk is averaged over the age and gender distribution of the hypothetical closed “stationary” population whose life expectancies and cancer mortality rates are based on U.S. data. Specifically, the total mortality rates used in the calculations underlying FGR 13 are defined by a 1989-1991 U.S. decennial life table (NCHS, 1997), and the cancer mortality rate is defined by U.S. cancer mortality for the same period (NCHS 1992, 1993a, 1993b). This hypothetical population is referred to as “stationary” because the gender-specific birth rates and survival functions are invariant over time.

The calculation of radiogenic cancer risk coefficients are based on risk projection models for specific cancer sites. The cancer sites considered are esophagus, stomach, colon, liver, lung, bone, skin, breast, ovary, bladder, kidney, thyroid, red (active) marrow (leukemia), and residual (all remaining cancer sites combined). The risk projection models are based in large part on information from the Radiation Effects Research Foundation Life Span Study (LSS) cohort of Hiroshima and Nagasaki atomic bomb survivors. A statistically significant excess cancer mortality associated with radiation has been found among the bomb survivors. Results from other epidemiological models were used for some cancers (e.g., the risk models for thyroid and breast cancer were based primarily on epidemiological studies of medical exposures of these organs). For bone and liver, low LET risk estimates were extrapolated from results of epidemiological studies of humans exposed to Ra-224 and thorotrast, respectively (EPA 1994).

The epidemiological data assembled for the study population (e.g., the LSS cohort) is transported to the U.S. population by one of two basic types of risk projection models—an absolute risk model or a relative risk model. An absolute risk model assumes that the age-specific excess force of mortality due to the radiation dose is independent of the natural cancer mortality rates in the population. The relative risk model assumes that the age-specific excess force of mortality due to the radiation dose is the product of an exposure-age-specific relative risk coefficient and the natural baseline cancer mortality rate. In FGR 13, risk models for bone, skin, and thyroid cancer are based on an absolute risk model and those for other sites are based on a relative risk model.

All epidemiological studies used in development of radiation risk models involve subjects who experienced high radiation dose delivered in a relatively short time. Available evidence indicates that the response per unit dose at low doses and dose rates from low-LET radiations may be an overestimate if one extrapolates from observations made at high, acutely delivered doses. The degree of overestimation is commonly expressed in terms of a dose and dose rate effectiveness factor (DDREF). For example, a DDREF of 2 means that the risk per unit dose observed at high acute doses should be divided by 2 before being applied to low doses or low dose rates. “Low dose” is defined here as less than 0.2 Gy (20 rad), and “low dose rate” is defined as less than  $0.1 \text{ mGy min}^{-1}$  ( $0.6 \text{ rad h}^{-1}$ ) (UNSCEAR 1993, EPA 1999, ICRP 2005). In FGR 13, a DDREF of 2 was assumed for all cancers other than breast cancer, for which a value

of 1 was assumed. Taken together, current scientific data are generally indicative of a DDREF between 1 and 3 for human cancer induction, except for possibly higher values for lung. A DDREF of 2 has been recently adopted by ICRP as well as other organizations. Fig. 10 illustrates the uncertainty probability density function (pdf) of DDREF assumed in estimating the uncertainty in risk coefficients. Mathematically the functions are:

$$\text{Breast:} \quad (9.a)$$

$$\text{Other cancers:} \quad (9.b)$$

An upper bound on the DDREF of 3 and 8 can be assumed for the breast and other cancers, respectively.

Note: For high-LET radiations, the risk is assumed to increase linearly with dose; thus, the DDREF for these radiations is 1.0.

The DDREF of 2, assumed in FGR 13, is in agreement with the value assumed by the National Council on Radiation Protection and Measurement in their Report No. 126 (NCRP 1997); BEIR V and BEIR VII (NAS 1960, 2006); and the ICRP in its Publications 60 and 103 (ICRP 1991, 2007). The DDREF for breast cancer reflects the linearity of the dose response observed in several study populations and an apparent invariance in risk with dose fractionation (EPA 1994).

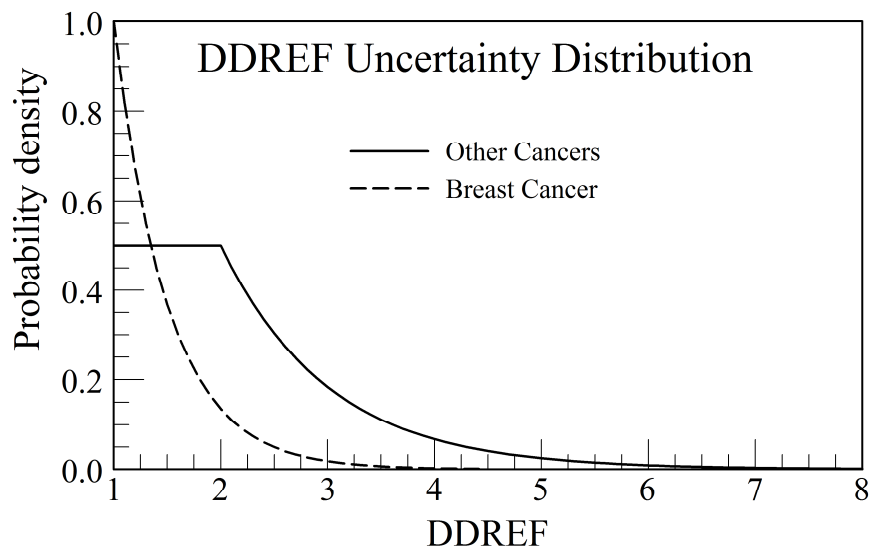


Figure 10. The probability distribution for DDREF applied in the uncertainty analysis for breast and other cancers.

The thyroid cancer risk model (EPA 1994) of FGR 13 suggest that a DDREF of 3 be used in estimating the risk for intakes of I-125, -129, and -131 while a value of 2 is applicable to all other irradiations of the gland. The higher DDREF was attributed to the lowered dose rate in the gland for these long-lived radioisotopes of iodine. It was not possible to use a pathway- and radioisotope-specific DDREF within the MACCS2 computer code and thus a DDREF of 2 was applied to all irradiations of the gland. If the projected thyroid cancer risk were solely due to inhalation intake of these radioiodines then MACCS2 over estimates the thyroid cancer risk by a factor of 1.5. Note the thyroid cancer risk tabulated below; e.g., in Table 2, reflect a DDREF of 2.

### 3.2 Cancer Mortality Risk Estimates

Table 2 lists the cancer mortality risk coefficients used in FGR 13. As noted above, these estimates reflect two different methods for transporting risks from observed population (for most cancer sites, the A-bomb survivors) to the U.S. via multiplicative (relative risk) and additive (absolute risk) methods. Using 1990 U.S. vital statistics, the risk models are applied to estimate organ-specific risks for a stationary population. The total risks listed in Table 2 are in general agreement with values of BEIR V, BEIR VII, ICRP 60, and ICRP 103.

### 3.3 Uncertainty in Cancer Mortality Risk Estimates

For purposes of quantifying the uncertainties in the estimated risks of Table 2, several sources of uncertainty can be identified. These include sampling variations, errors in dosimetry, errors in ascertainment of cause of death, modeling of the dependence on age at exposure and time since exposure, transport of risk estimates from a study group to the U.S. population, extrapolation to low DDREF, and, for high-LET radiation, uncertainty in RBE. The combined uncertainty due to all sources is calculated analytically using Monte Carlo methods (note that DDREF divides rather than multiplies the risk coefficient, and a DDREF of 2 is incorporated into the risk estimates of Table 2). The resultant distribution of the risk values is generated as

$$y_i = \frac{2}{f_0} f_1 \quad (8)$$

where  $f_0$  is random value from the DDREF distribution and  $f_1$  is the random value from the distribution defining the uncertainty in multiplicative terms of the risk coefficient. No correlation in DDREF and  $f_1$  values among the cancer sites is assumed.

**Table 2. Age-Averaged Site-Specific Cancer Mortality Risk Estimates (Cancer Deaths per Person-Gy) from Low-Dose, Low-LET Uniform Irradiation of the Body (EPA 1999)**

Site	Males	Females	Both
Esophagus	7.30E-04	1.59E-03	1.17E-03
Stomach	3.25E-03	4.86E-03	4.07E-03
Colon	8.28E-03	1.24E-02	1.04E-02
Liver	1.84E-03	1.17E-03	1.50E-03
Lung	7.71E-03	1.19E-02	9.88E-03
Bone	9.40E-05	9.60E-05	9.50E-05
Skin	9.51E-05	1.05E-04	1.00E-04
Breast	-	9.90E-03	5.06E-03
Ovary	-	2.92E-03	1.49E-03
Bladder	3.28E-03	1.52E-03	2.38E-03
Kidney	6.43E-04	3.92E-04	5.15E-04
Thyroid	2.05E-04	4.38E-04	3.24E-04
Leukemia	6.48E-03	4.71E-03	5.57E-03
Residual <sup>a</sup>	1.35E-02	1.63E-02	1.49E-02
Total	4.62E-02	6.83E-02	5.75E-02

<sup>a</sup> Residual is a composite of all radiogenic cancers that are not explicitly listed in the table.



Table 3 lists the age and gender average cancer mortality risk coefficients used in the SOARCA project. The 90 percent confidence intervals are based on the uncertainties in the risk model coefficients as assessed by nine researchers involved in an expert elicitation study (NRC-CEC 1998). As part of the elicitation, the nine experts were asked to provide 5 percent, 50 percent, and 95 percent quintiles of subjective probability distributions for the total number of radiation-induced cancer deaths and for the numbers of tissue-specific cancer deaths over a lifetime in a typical population of 100 million persons, each receiving a 1 Gy whole body dose of low LET radiation at a uniform rate over 1 min. Pawel et al. (2007) derived the confidence interval for the risk coefficients as the geometric mean of two derived values: (1) the interval based on judgments of all nine experts and (2) a smaller interval based on the opinions of eight experts. The geometric standard deviations of the log-normal distribution tabulated in Table 3 were derived from the 90 percent confidence interval noted by Pawel et al.

The 90 percent confidence intervals in the mortality risk coefficients tabulated in Table 3 reflect uncertainties in the coefficients and the DDREF. The risk model coefficients seem relatively well known for leukemia; moderately well known for lung, breast, colon, and residual (cancer sites not explicitly identified in the table); and relatively poorly known for thyroid, liver, and bone. These conclusions are based on the premise that inconsistency in expert judgment is a valid measure of uncertainty in a mortality risk coefficient.

Note the residual group in Table 3 includes, in addition to the residual group of Table 2, cancers of the esophagus, stomach, skin, ovaries, bladder, and kidney. As noted earlier, it is assumed that the mortality risk coefficients for the cancer sites are independent and not correlated.

**Table 3. Age-Averaged Cancer Mortality Risk Estimates and Uncertainties for Low-Dose, Low-LET Uniform Irradiation of the Body**

Site	Mortality Risk Coefficient <sup>a</sup>		DDREF	90% Confidence Interval	
	(Gy <sup>-1</sup> )	Uncertainty <sup>b</sup>		Risk05	Risk95
Leukemia	5.57E-03	1.23	Eqn(9.b)	2.42E-03	1.11E-02
Bone	9.50E-05	5.70	Eqn(9.b)	4.82E-06	1.72E-03
Breast	5.06E-03	2.21	Eqn(9.a)	1.01E-03	1.27E-01
Lung	9.88E-03	1.87	Eqn(9.b)	2.73E-03	3.23E-02
Thyroid	3.24E-04	4.15	Eqn(9.b)	2.70E-05	3.60E-03
Liver	1.50E-03	4.65	Eqn(9.b)	1.05E-04	1.98E-03
Colon	1.04E-02	2.26	Eqn(9.b)	2.22E-03	4.49E-02
Residual	2.46E-02	2.10	Eqn(9.b)	5.87E-02	9.53E-01
Total	5.75E-02			3.89E-02	2.42E-01

<sup>a</sup> Geometric mean - note, except breast, the values include a DDREF of 2.

<sup>b</sup> the geometric standard deviation of the log normal distribution.

#### 4.0 Conclusions

This report detailed the methodology underlying the radiation dose and risk assessment in the SOARCA project. The focus is on cancer mortality stemming from low-level chronic exposure of the population returning to the site environs. The exposure pathways involve external irradiation of the body from photons emitted by radionuclides present on the ground surfaces and inhalation intake of airborne radionuclides resuspended from the ground surfaces.

Appendix A tabulates the dose coefficients for the external pathway for the eight cancer sites considered in the project. These coefficients are to be used in an algorithm that accounts for household shielding, time spent outdoors, and attenuation of the radiation field by ground roughness. The uncertainty in the dose coefficients can be characterized by a probability distribution that is perfectly correlated among the organs and tissues of the body.

Appendix B tabulates the inhalation dose coefficients for the eight cancer sites and represents their uncertainties in terms of a log-normal distribution. The coefficients for the lung are assumed to be negatively correlated to the others, and the coefficients for non-lung organs are assumed to be positively correlated. The cancer mortality coefficients are based in part on BEIR V and tabulated in Table 3. The uncertainties in the mortality coefficients for the cancer sites are assumed to be independent (i.e., not correlated). Table 3 provides guidance with regard uncertainty in the DDREF and the risk coefficient itself to be considered in estimating the latent cancer mortality risk and its uncertainties associated with the chronic low-level radiation exposure of the population.

### References

Eckerman KF, Leggett RW, Cristy M, Nelson CB, Ryman JC, Sjoreen AL, Ward RC 1999. *User's Guide to the DCAL System*, ORNL/TM-2001/190 (Oak Ridge National Laboratory, Oak Ridge, TN).

EPA (1993). *External Exposure to Radionuclides in Air, Water, and Soil*. Federal Guidance Report No. 12, EPA 402402-R-93-081 (U.S. Environmental Protection Agency, Washington, DC).

EPA (1994). *Estimating Radiogenic Cancer Risks*, EPA 402-R-93-076 (U.S. Environmental Protection Agency, Washington, DC).

EPA (1999). *Cancer Risk Coefficients for Environmental Exposure to Radionuclides*. Federal Guidance Report No. 13, EPA 402-R-99-001 (U. S. Environmental Protection Agency, Washington, DC).

HPS (2010). Health Physics Society, *Radiation Risk in Perspective*, Position Statement of the Health Physics Society, PS010-2.

ICRP (1975). *Report of the Task Group on Reference Man*, ICRP Publication 23 (Pergamon Press, Oxford).

ICRP (1979). "Limits for Intakes by Workers," ICRP Publication 30, Part 1, *Ann. ICRP* 2(3/4).

ICRP (1991). "Recommendations of the International Commission on Radiological Protection," ICRP Publication 60, *Ann. ICRP* 21(1/3).

ICRP (1994). "Human Respiratory Tract Model for Radiological Protection," ICRP Publication 66, *Ann. ICRP* 24(1/3).

ICRP (1995). "Age-dependent Doses to the Members of the Public from Intake of Radionuclides – Part 5 Compilation of Ingestion and Inhalation Coefficients," ICRP Publication 72, *Ann. ICRP* 26(1).

ICRP (2002). "Basic Anatomic and Physiological Data for Use in Radiological Protection: Reference values," ICRP Publication 89, *Ann. ICRP* 32(3/4).

ICRP (2005). “Low-dose Extrapolation of Radiation-related Cancer Risk,” ICRP Publication 99, *Ann. ICRP* 35(4).

ICRP (2007). “The 2007 Recommendations of the International Commission on Radiological Protection,” ICRP Publication 103, *Ann. ICRP* 37(2/4).

Leggett, RW, Eckerman KF, Williams LR (1993). “An Elementary Method for Implementing Complex Biokinetic Models,” *Health Phys.* 64, 260-278.

Loevinger, R, Berman, M (1968). “A Schema for Calculating the Absorbed Dose from Biologically Distributed Radionuclides,” MIRD Pamphlet No 1: *J Nucl Med* 9 (suppl 1).

NAS (1990). *Health Effects of Exposure to Low Levels of Ionizing Radiation (BEIR V)* (National Academy of Sciences, National Academy Press, Washington, DC).

NAS (2006). *Health Effects of Exposure to Low Levels of Ionizing Radiation (BEIR VII)* (National Academy of Sciences, National Academy Press, Washington, DC).

NCHS (1992). *Vital Statistics Mortality Data, Detail, 1989*, NTIS tape:PB92-504554 (U.S. Department of Health and Human Services, Public Health Service, National Center for Health Statistics, Hyattsville, MD).

NCHS (1993a). *Vital Statistics Mortality Data, Detail, 1990*, NTIS tape: PB92-504777 (U.S. Department of Health and Human Services, Public Health Service, National Center for Health Statistics, Hyattsville, MD).

NCHS (1993b). *Vital Statistics Mortality Data, Detail, 1991*, NTIS tape:PB92-506889 (U.S. Department of Health and Human Services, Public Health Service, National Center for Health Statistics, Hyattsville, MD).

NCHS (1997). *U.S. Decennial Life Tables for 1989-91*, Vol. 1, No. 1. DHHS, PHS-98-1150-1 (National Center for Health Statistics: United States Life Tables. Public Health Service, Washington, DC).

NCRP (1997). Uncertainties in Fatal Cancer Risk Estimates Used in Radiation Protection, NCRP Report No. 126 (National Council on Radiation Protection and Measurements, Bethesda, MD).

NCRP (2009). *Ionizing Radiation Exposure of the Population of the United States*, NCRP Report No. 160 (National Council on Radiation Protection and Measurements, Bethesda, MD).

NRC (1991). “Chapter 3: Late Somatic Effects”, In: S. Abramamson, M.A. Bender, B.B. Borvker, E.S. Gilbert, and B.R. Scott. *Health Effects Models for Nuclear Power Plant Accident Consequence Analysis. Modification of Models Resulting from Recent Report on Health Effects Models*, NUREG/CR-4214, Rev. 1, Part II, Addendum 1, LMF-132, (US Nuclear Regulatory Commission, Washington, DC).

NRC (1992). “Chapter 3: Late Somatic Effects”, In: S. Abramamson, M.A. Bender, B.B. Borvker, E.S. Gilbert, and B.R. Scott. *Health Effects Models for Nuclear Power Plant Accident Consequence Analysis. Modification of Models Resulting from Addition of Effects of Exposure to Alpha-Emitting Radionuclides*,

*Part II; Scientific Bases for Health Effects Models*, NUREG/CR-4214, Rev. 1, Part II, Addendum 2, LMF-136, (US Nuclear Regulatory Commission, Washington, DC).

NRC-CEC (1997). Probabilistic Accident Consequence Uncertainty Analysis. Late Health Effects Uncertainty Assessment, NUREG/CR-6555; EUR 16774, SAND97-2322 (U S Nuclear Regulatory Commission, Washington, DC).

Pawel DJ, Leggett RW, Eckerman KF, Nelson CB (2007). Uncertainties in Cancer Risk Coefficients for Environmental Exposure to Radionuclides. ORNL/TM-2006/583.

UNSCEAR (1993). Sources, Effects and Risks of Ionizing Radiation. United Nations Scientific Committee on the Effects of Atomic Radiation. No. E.94.IX.2, United Nations, New York, NY.

Yamaguchi Y (1994). "Age-dependent effective dose for external photons," *Rad. Prot. Dosimetry* 55(2):123-129.

## Appendix A

Table A-1. Absorbed dose rate (Gy s<sup>-1</sup>) and effective dose rate (Sv s<sup>-1</sup>) per unit ground surface activity (Bq m<sup>2</sup>)

Nuclide	Leukemia	Bone	Breast	Lung	Thyroid	Liver	Colon	Residual	Effective
Co-58	9.29E-16	1.32E-15	9.51E-16	9.09E-16	9.46E-16	8.68E-16	8.85E-16	8.00E-16	9.22E-16
Co-60	2.33E-15	3.10E-15	2.34E-15	2.27E-15	2.25E-15	2.19E-15	2.25E-15	2.05E-15	2.30E-15
Rb-86	9.01E-17	1.25E-16	9.16E-17	8.78E-17	8.94E-17	8.45E-17	8.66E-17	7.84E-17	1.65E-16
Rb-88	5.98E-16	7.96E-16	5.96E-16	5.82E-16	5.57E-16	5.61E-16	5.73E-16	5.32E-16	7.48E-16
Sr-89	1.87E-18	5.46E-18	2.52E-18	1.99E-18	2.12E-18	1.85E-18	1.72E-18	1.57E-18	6.88E-17
Sr-90	1.93E-19	8.32E-19	3.50E-19	2.19E-19	2.49E-19	1.99E-19	1.71E-19	1.53E-19	1.64E-18
Sr-91	6.73E-16	9.38E-16	6.85E-16	6.57E-16	6.77E-16	6.30E-16	6.44E-16	5.83E-16	7.35E-16
Sr-92	1.24E-15	1.64E-15	1.24E-15	1.20E-15	1.18E-15	1.16E-15	1.19E-15	1.09E-15	1.22E-15
Y-90	4.57E-18	1.17E-17	5.71E-18	4.75E-18	5.01E-18	4.44E-18	4.23E-18	3.87E-18	1.10E-16
Y-91	4.84E-18	9.56E-18	5.52E-18	4.88E-18	5.01E-18	4.63E-18	4.58E-18	4.17E-18	7.43E-17
Y-91m	5.05E-16	7.49E-16	5.26E-16	4.99E-16	5.22E-16	4.71E-16	4.75E-16	4.32E-16	5.08E-16
Y-92	2.48E-16	3.50E-16	2.53E-16	2.43E-16	2.46E-16	2.33E-16	2.38E-16	2.16E-16	3.83E-16
Y-93	9.56E-17	1.43E-16	9.81E-17	9.38E-17	9.25E-17	8.99E-17	9.08E-17	8.39E-17	2.17E-16
Zr-95	7.00E-16	9.91E-16	7.18E-16	6.86E-16	7.15E-16	6.54E-16	6.67E-16	6.02E-16	6.96E-16
Zr-97	8.40E-16	1.19E-15	8.61E-16	8.23E-16	8.52E-16	7.85E-16	8.00E-16	7.24E-16	9.18E-16
Nb-95	7.31E-16	1.03E-15	7.48E-16	7.15E-16	7.45E-16	6.83E-16	6.97E-16	6.29E-16	7.26E-16
Nb-97	6.37E-16	9.19E-16	6.58E-16	6.27E-16	6.54E-16	5.95E-16	6.04E-16	5.47E-16	6.82E-16
Mo-99	1.41E-16	2.18E-16	1.47E-16	1.39E-16	1.44E-16	1.32E-16	1.34E-16	1.21E-16	1.77E-16
Tc-99m	1.10E-16	2.86E-16	1.21E-16	1.13E-16	1.14E-16	1.07E-16	1.03E-16	9.64E-17	1.14E-16
Ru-103	4.74E-16	7.19E-16	4.95E-16	4.69E-16	4.90E-16	4.42E-16	4.44E-16	4.05E-16	4.74E-16
Ru-105	7.11E-16	1.06E-15	7.39E-16	7.00E-16	7.29E-16	6.65E-16	6.73E-16	6.11E-16	7.48E-16
Ru-106	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rh-103m	3.14E-19	2.15E-18	2.36E-18	2.15E-19	7.30E-19	1.80E-19	4.98E-20	1.83E-20	8.26E-19
Rh-105	7.26E-17	1.24E-16	7.70E-17	7.22E-17	7.40E-17	6.82E-17	6.75E-17	6.25E-17	7.38E-17
Te-127	4.87E-18	8.61E-18	5.35E-18	4.88E-18	5.10E-18	4.59E-18	4.53E-18	4.16E-18	1.05E-17
Te-127m	4.29E-18	3.10E-17	1.78E-17	5.52E-18	9.58E-18	4.53E-18	2.19E-18	1.42E-18	8.54E-18
Te-129	5.78E-17	1.01E-16	6.66E-17	5.78E-17	6.18E-17	5.43E-17	5.35E-17	4.86E-17	1.16E-16
Te-129m	3.18E-17	6.42E-17	4.28E-17	3.22E-17	3.65E-17	3.01E-17	2.88E-17	2.57E-17	5.83E-17
Te-131	3.92E-16	6.73E-16	4.13E-16	3.89E-16	4.00E-16	3.70E-16	3.70E-16	3.39E-16	4.72E-16
Te-131m	1.37E-15	1.98E-15	1.40E-15	1.34E-15	1.37E-15	1.28E-15	1.30E-15	1.18E-15	1.36E-15
Te-132	2.01E-16	4.61E-16	2.45E-16	2.06E-16	2.16E-16	1.94E-16	1.84E-16	1.70E-16	2.13E-16
I-131	3.61E-16	5.91E-16	3.82E-16	3.59E-16	3.72E-16	3.39E-16	3.38E-16	3.10E-16	3.65E-16
I-132	2.15E-15	3.04E-15	2.20E-15	2.11E-15	2.17E-15	2.01E-15	2.05E-15	1.86E-15	2.18E-15
I-133	5.84E-16	8.62E-16	6.06E-16	5.76E-16	5.99E-16	5.46E-16	5.51E-16	5.02E-16	6.22E-16
I-134	2.45E-15	3.41E-15	2.49E-15	2.40E-15	2.45E-15	2.30E-15	2.35E-15	2.13E-15	2.49E-15
I-135	1.47E-15	1.96E-15	1.47E-15	1.43E-15	1.40E-15	1.37E-15	1.41E-15	1.29E-15	1.47E-15
Cs-134	1.48E-15	2.12E-15	1.53E-15	1.46E-15	1.52E-15	1.39E-15	1.41E-15	1.28E-15	1.48E-15
Cs-136	2.01E-15	2.87E-15	2.05E-15	1.96E-15	2.01E-15	1.88E-15	1.92E-15	1.74E-15	1.99E-15
Cs-137	2.03E-19	8.31E-19	3.54E-19	2.28E-19	2.57E-19	2.07E-19	1.80E-19	1.62E-19	3.13E-18
Ba-139	4.43E-17	1.04E-16	4.99E-17	4.54E-17	4.61E-17	4.28E-17	4.13E-17	3.84E-17	1.49E-16
Ba-140	1.70E-16	2.80E-16	1.84E-16	1.70E-16	1.78E-16	1.60E-16	1.59E-16	1.45E-16	1.91E-16
La-140	2.13E-15	2.88E-15	2.13E-15	2.08E-15	2.02E-15	1.99E-15	2.03E-15	1.87E-15	2.15E-15
La-141	2.98E-17	4.52E-17	3.09E-17	2.93E-17	2.89E-17	2.81E-17	2.84E-17	2.61E-17	1.40E-16
La-142	2.11E-15	2.76E-15	2.09E-15	2.05E-15	1.96E-15	1.97E-15	2.01E-15	1.89E-15	2.16E-15
Ce-141	6.52E-17	1.81E-16	7.61E-17	6.84E-17	7.03E-17	6.40E-17	6.07E-17	5.64E-17	6.97E-17
Ce-143	2.53E-16	4.95E-16	2.87E-16	2.57E-16	2.69E-16	2.41E-16	2.35E-16	2.15E-16	2.99E-16
Ce-144	1.56E-17	5.13E-17	1.99E-17	1.69E-17	1.78E-17	1.57E-17	1.44E-17	1.33E-17	1.73E-17
Pr-143	5.25E-19	1.91E-18	8.20E-19	5.77E-19	6.33E-19	5.30E-19	4.74E-19	4.31E-19	2.08E-17
Pr-144	3.39E-17	5.29E-17	3.54E-17	3.34E-17	3.30E-17	3.19E-17	3.21E-17	2.97E-17	1.61E-16
Nd-147	1.23E-16	2.91E-16	1.45E-16	1.29E-16	1.36E-16	1.20E-16	1.15E-16	1.05E-16	1.40E-16
Np-239	1.54E-16	3.88E-16	1.78E-16	1.58E-16	1.62E-16	1.49E-16	1.44E-16	1.34E-16	1.62E-16
Pu-238	1.84E-19	1.16E-18	1.84E-18	7.51E-20	3.22E-19	6.35E-20	2.71E-20	1.64E-20	5.99E-19
Pu-239	1.29E-19	6.34E-19	8.31E-19	8.66E-20	1.91E-19	7.78E-20	6.00E-20	5.15E-20	3.06E-19
Pu-240	1.77E-19	1.11E-18	1.74E-18	7.48E-20	3.08E-19	6.35E-20	2.87E-20	1.81E-20	5.68E-19
Pu-241	1.30E-21	4.23E-21	1.64E-21	1.37E-21	1.41E-21	1.28E-21	1.22E-21	1.14E-21	1.43E-21

Am-241	1.64E-17	9.01E-17	3.02E-17	2.00E-17	2.12E-17	1.80E-17	1.54E-17	1.39E-17	2.18E-17
Cm-242	2.18E-19	1.40E-18	2.03E-18	1.03E-19	4.22E-19	8.72E-20	3.07E-20	1.55E-20	6.68E-19
Cm-244	1.99E-19	1.21E-18	1.75E-18	9.97E-20	3.72E-19	8.55E-20	3.76E-20	2.38E-20	5.85E-19

-----

## Appendix B

**Table B-1. Committed Absorbed Dose Coefficients (Gy/Bq) and Effective Dose Coefficient (Sv/Bq) for Inhalation Intakes of Radionuclides.**

Nuclide	Type	Leukemia	Bone	Breast	Lung	Thyroid	Liver	Colon	Residual	Effective
Co-58	S	7.16E-10	5.26E-10	1.32E-09	1.27E-08	5.18E-10	1.06E-09	8.26E-10	9.27E-10	2.12E-09
Co-60	S	1.24E-08	9.36E-09	2.48E-08	1.83E-07	9.89E-09	1.98E-08	4.50E-09	1.63E-08	3.08E-08
Rb-86	F	1.43E-09	2.82E-09	7.32E-10	7.69E-10	7.46E-10	7.46E-10	1.28E-09	7.52E-10	9.37E-10
Rb-88	F	1.74E-12	1.86E-12	1.65E-12	3.14E-11	1.74E-12	1.66E-12	1.75E-12	1.96E-12	1.62E-11
Sr-89	F	4.33E-09	5.40E-09	1.77E-10	2.03E-10	1.78E-10	1.78E-10	2.26E-09	1.78E-10	1.01E-09
Sr-90	F	1.62E-07	3.69E-07	5.97E-10	6.19E-10	5.97E-10	5.97E-10	3.22E-09	5.97E-10	2.40E-08
Sr-91	F	1.31E-10	1.35E-10	2.07E-11	5.18E-11	2.64E-11	2.54E-11	4.64E-10	2.88E-11	1.57E-10
Sr-92	F	6.10E-11	8.89E-11	1.29E-11	3.70E-11	1.55E-11	1.49E-11	3.13E-10	1.76E-11	9.83E-11
Y-90	S	1.22E-12	1.22E-12	4.23E-14	7.85E-09	4.23E-14	1.20E-12	4.44E-09	4.23E-14	1.50E-09
Y-91	S	6.44E-11	6.42E-11	5.21E-12	6.97E-08	2.83E-12	6.43E-11	4.64E-09	3.93E-12	8.96E-09
Y-91m	S	9.41E-13	7.25E-13	8.01E-13	5.00E-11	6.78E-13	8.97E-13	5.26E-12	2.03E-12	1.14E-11
Y-92	S	1.83E-12	1.27E-12	1.43E-12	6.95E-10	1.09E-12	1.75E-12	4.94E-10	2.49E-12	1.77E-10
Y-93	S	1.77E-12	1.29E-12	1.27E-12	1.62E-09	8.99E-13	1.66E-12	1.63E-09	1.75E-12	4.22E-10
Zr-95	M	2.43E-09	1.26E-08	1.21E-09	3.07E-08	6.50E-10	1.05E-09	1.62E-09	1.05E-09	4.79E-09
Zr-97	M	7.77E-11	6.43E-11	3.78E-11	3.43E-09	2.77E-11	4.39E-11	2.81E-09	4.69E-11	9.23E-10
Nb-95	S	3.39E-10	2.49E-10	5.86E-10	1.19E-08	2.35E-10	4.94E-10	7.02E-10	4.23E-10	1.76E-09
Zr-97	M	7.77E-11	6.43E-11	3.78E-11	3.43E-09	2.77E-11	4.39E-11	2.81E-09	4.69E-11	9.23E-10
Mo-99	S	2.07E-11	2.12E-11	2.17E-11	6.02E-09	1.09E-11	2.74E-11	2.00E-09	2.06E-11	9.95E-10
Tc-99m	M	1.65E-12	2.32E-12	1.16E-12	7.60E-11	5.45E-12	1.66E-12	1.84E-11	2.30E-12	1.93E-11
Ru-103	S	2.36E-10	1.73E-10	4.06E-10	2.21E-08	1.64E-10	3.32E-10	1.08E-09	3.02E-10	2.96E-09
Ru-105	S	7.95E-12	5.59E-12	6.05E-12	7.99E-10	4.50E-12	7.41E-12	2.96E-10	1.02E-11	1.80E-10
Ru-106	S	6.70E-10	5.44E-10	1.12E-09	5.35E-07	5.43E-10	9.28E-10	1.28E-08	8.44E-10	6.62E-08
Rh-103m	S	1.32E-15	3.04E-15	1.64E-15	1.76E-11	7.05E-16	2.70E-15	9.37E-13	5.60E-15	2.73E-12
Rh-105	S	4.64E-12	3.87E-12	4.18E-12	2.31E-09	2.64E-12	4.49E-12	5.62E-10	4.61E-12	3.54E-10
Te-127	M	1.71E-12	1.74E-12	7.84E-13	7.50E-10	2.81E-12	8.01E-13	2.51E-10	8.20E-13	1.27E-10
Te-127m	M	2.31E-09	9.03E-09	5.84E-11	5.62E-08	8.62E-10	5.88E-11	1.91E-09	4.89E-11	7.43E-09
Te-129	M	3.51E-13	3.58E-13	3.00E-13	1.48E-10	3.53E-13	3.19E-13	2.06E-11	4.78E-13	3.70E-11
Te-129m	M	1.17E-09	2.71E-09	9.12E-11	4.83E-08	1.01E-09	8.69E-11	4.00E-09	8.02E-11	6.56E-09
Te-131	M	5.25E-13	5.36E-13	5.46E-13	9.37E-11	4.79E-11	5.37E-13	5.10E-12	1.06E-12	2.86E-11
Te-131m	M	8.85E-11	1.17E-10	7.11E-11	4.65E-09	2.69E-09	7.44E-11	1.46E-09	7.59E-11	1.07E-09
Te-132	M	2.22E-10	2.87E-10	2.30E-10	1.01E-08	4.36E-09	2.21E-10	3.34E-09	2.08E-10	2.06E-09
I-131	F	3.71E-11	4.89E-11	2.12E-11	5.98E-11	1.47E-07	1.67E-11	2.54E-11	1.82E-11	7.41E-09
I-132	F	1.16E-11	1.23E-11	8.59E-12	3.64E-11	1.36E-09	9.91E-12	1.18E-11	1.45E-11	9.39E-11
I-133	F	1.85E-11	2.00E-11	1.37E-11	4.24E-11	2.84E-08	1.38E-11	2.09E-11	1.57E-11	1.47E-09
I-134	F	5.51E-12	5.82E-12	4.40E-12	2.99E-11	2.59E-10	5.00E-12	5.39E-12	8.27E-12	4.52E-11
I-135	F	1.69E-11	1.84E-11	1.26E-11	4.04E-11	5.76E-09	1.35E-11	1.74E-11	1.73E-11	3.23E-10
Cs-134	F	6.37E-09	6.81E-09	4.81E-09	6.02E-09	6.34E-09	6.73E-09	7.20E-09	7.40E-09	6.71E-09
Cs-136	F	9.89E-10	1.12E-09	7.37E-10	9.68E-10	1.01E-09	1.05E-09	1.15E-09	1.17E-09	1.24E-09
Cs-137	F	4.47E-09	4.69E-09	3.82E-09	4.35E-09	4.46E-09	4.62E-09	5.22E-09	4.90E-09	4.69E-09
Ba-139	F	1.17E-11	1.20E-11	1.67E-12	2.91E-11	1.71E-12	1.73E-12	1.12E-10	1.81E-12	3.39E-11
Ba-140	F	1.46E-09	2.07E-09	8.97E-11	1.55E-10	1.21E-10	1.28E-10	5.45E-09	1.66E-10	1.03E-09
La-140	M	1.35E-10	9.67E-11	1.10E-10	3.96E-09	6.60E-11	3.23E-10	2.51E-09	1.23E-10	1.08E-09
La-141	M	3.07E-12	1.21E-11	1.33E-12	6.64E-10	1.13E-12	1.94E-11	3.67E-10	1.56E-12	1.47E-10
La-142	M	7.85E-12	6.06E-12	6.75E-12	2.45E-10	5.81E-12	9.79E-12	8.08E-11	1.34E-11	8.99E-11
Ce-141	S	3.71E-11	1.08E-10	5.97E-11	2.97E-08	2.26E-11	8.18E-11	1.29E-09	4.65E-11	3.76E-09
Ce-143	S	1.45E-11	1.38E-11	1.27E-11	4.98E-09	6.72E-12	1.64E-11	1.73E-09	1.38E-11	8.34E-10
Ce-144	S	1.19E-09	2.06E-09	2.74E-10	4.24E-07	1.44E-10	5.87E-09	1.14E-08	2.16E-10	5.29E-08
Pr-143	S	2.01E-12	2.01E-12	1.29E-14	1.82E-08	1.29E-14	1.60E-11	2.08E-09	1.29E-14	2.45E-09
Pr-144	S	1.58E-14	1.39E-14	1.71E-14	5.66E-11	1.58E-14	1.61E-14	1.28E-12	3.86E-14	1.83E-11
Nd-147	S	3.42E-11	9.92E-11	4.33E-11	1.80E-08	1.65E-11	6.26E-11	1.82E-09	3.35E-11	2.41E-09
Np-239	M	3.46E-11	5.16E-10	1.29E-11	6.35E-09	8.24E-12	3.34E-11	1.18E-09	1.46E-11	9.36E-10
Pu-238	S	4.14E-07	1.62E-04	1.38E-07	9.34E-05	2.77E-07	3.43E-05	2.88E-07	2.77E-07	1.61E-05
Pu-239	S	4.57E-07	1.84E-04	1.59E-07	8.76E-05	3.18E-07	3.91E-05	3.29E-07	3.18E-07	1.61E-05
Pu-240	S	4.58E-07	1.84E-04	1.59E-07	8.78E-05	3.18E-07	3.91E-05	3.29E-07	3.18E-07	1.61E-05
Pu-241	S	9.30E-09	4.08E-06	3.57E-09	4.58E-07	7.11E-09	8.60E-07	7.18E-09	7.12E-09	1.75E-07

Am-241	M	2.94E-06	1.70E-03	1.44E-06	3.72E-05	2.88E-06	1.05E-04	2.89E-06	2.88E-06	4.17E-05
Cm-242	M	9.32E-08	2.73E-05	1.73E-08	3.54E-05	3.47E-08	7.14E-06	4.39E-08	3.47E-08	5.22E-06
Cm-244	M	1.94E-06	9.22E-04	6.42E-07	3.93E-05	1.28E-06	7.53E-05	1.29E-06	1.28E-06	2.66E-05

---



Table B-2. Uncertainty in the committed absorbed dose coefficient for inhalation intakes of the radionuclides in the SOARCA Project. The geometric standard deviation of the log-normal distribution is tabulated.

Nuclide	Type	Lung	Leukemia	Bone	Breast	Thyroid	Liver	Colon	Residual
Co-58	S	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Co-60	S	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Rb-86	F	2.51	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Rb-88	F	1.50	1.50	1.50	1.50	1.50	2.56	1.50	1.50
Sr-89	F	5.40	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Sr-90	F	5.15	1.50	1.50	1.50	1.50	2.74	1.50	1.50
Sr-91	F	3.03	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Sr-92	F	2.88	1.56	1.76	1.50	1.50	1.50	1.50	1.50
Y-90	S	1.50	2.05	1.50	1.61	1.61	2.33	1.50	1.61
Y-91	S	1.50	4.51	3.25	1.50	1.65	4.95	1.50	1.50
Y-91m	S	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Y-92	S	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Y-93	S	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Zr-95	M	1.50	1.50	1.50	1.50	1.50	1.88	1.50	1.50
Zr-97	M	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Nb-95	S	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Nb-97	S	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Mo-99	S	1.50	1.61	2.05	1.50	1.50	3.05	3.00	1.50
Tc-99m	M	1.50	1.50	1.50	1.50	2.22	1.50	1.50	1.50
Ru-103	S	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Ru-105	S	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Ru-106	S	1.50	2.04	2.49	1.50	1.50	1.88	1.50	1.50
Rh-103m	S	1.50	1.50	1.50	1.50	1.50	1.50	3.08	1.50
Rh-105	S	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Te-127	M	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Te-127m	M	1.50	3.55	5.72	1.50	3.03	7.38	1.50	1.50
Te-129	M	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Te-129m	M	1.50	3.20	4.24	1.50	2.68	4.48	1.50	1.50
Te-131	M	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Te-131m	M	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Te-132	M	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
I-131	F	6.54	1.50	1.50	1.50	1.50	1.50	2.74	1.50
I-132	F	1.87	1.50	1.50	1.50	2.64	1.50	1.50	1.50
I-133	F	4.22	1.50	1.50	1.50	1.54	1.50	2.75	1.50
I-134	F	1.50	1.50	1.50	1.50	3.49	1.50	1.50	1.50
I-135	F	2.67	1.50	1.50	1.50	1.97	1.50	1.86	1.50
Cs-134	F	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Cs-136	F	2.04	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Cs-137	F	1.55	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Ba-139	F	1.72	4.25	3.31	2.25	2.47	2.09	1.50	1.85
Ba-140	F	6.28	1.50	1.50	1.50	1.50	3.68	1.50	1.50
La-140	M	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
La-141	M	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
La-142	M	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Ce-141	S	1.50	1.50	1.60	1.50	1.50	1.50	1.50	1.50
Ce-143	S	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Ce-144	S	1.50	4.57	3.39	1.50	1.50	3.32	1.50	1.50
Pr-143	S	1.50	3.88	2.39	5.49	5.49	2.24	1.50	5.49
Pr-144	S	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Nd-147	S	1.50	1.50	1.84	1.50	1.50	1.50	1.50	1.50
Np-239	M	1.50	1.50	1.50	1.50	1.50	1.50	1.50	1.50
Pu-238	S	1.50	4.32	2.97	3.15	3.15	3.10	3.17	3.15
Pu-239	S	1.50	4.32	2.95	3.13	3.14	3.08	3.16	3.13
Pu-240	S	1.50	4.32	2.95	3.14	3.14	3.08	3.16	3.14
Pu-241	S	1.84	4.29	2.77	2.96	2.97	2.92	2.99	2.96

Am-241	M	1.50	1.78	1.50	1.50	1.50	1.50	1.50	1.50
Cm-242	M	1.50	1.51	1.50	1.50	1.50	1.50	1.50	1.50
Cm-244	M	1.50	1.71	1.50	1.50	1.50	1.50	1.50	1.50

---