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MAXIMUM INTERNAL DOSE ESTIMATES FOR CERTAIN DOE COMPLEX CLAIMS

1.0 Purpose

The purpose of this Technical Information Bulletin (TIB) is to provide a method to facilitate timely processing of claims under the Energy Employee Occupational Illness Compensation Program Act (EEOICPA 2000), which involve cancer to an organ with little or no reported internal dose from internally deposited radionuclides that might be associated with work at DOE complex sites.

2.0 Background

Title 42, Part 82 of the Code of Federal Regulations (CFR) dictates the methods to be used for radiation dose reconstruction under the Energy Employees Occupational Illness Compensation Program Act of 2000. Section 82.10(k) summarizes the general philosophy to be adopted:

> *"Research and analysis will be determined sufficient if one of the following three conditions is met:*
>
> *(1) From acquired experience, it is evident the estimated cumulative dose is sufficient to qualify the claimant for compensation (i.e., the dose produces a probability of causation of 50% or greater);*
>
> *(2) Dose is determined using worst case assumptions related to radiation exposure and intake, to substitute for further research and analyses;*
>
> *(3) Research and analysis indicated under steps described in paragraphs (f) - (j) of this section have been completed.*
>
> *Worst-case assumptions will be employed under condition 2 to limit further research and analysis only for claims for which it is evident that further research and analysis will not produce a compensable level of radiation dose (a dose producing a probability of causation of 50% or greater), because using worst-case assumptions it can be determined that the employee could not have incurred a compensable level of radiation dose."*

"*Worst-case assumption*" is defined in Section 82.5(r) as:

> *"A term used to describe a type of assumption used in certain instances for certain dose reconstruction conducted under this rule [42 CFR 82]. It assigns the highest reasonably possible value, based on reliable science, documented experience, and relevant data to a radiation dose of a covered employee."*

A number of EEOICPA claims have been submitted that can be expedited based on paragraph (2) above. Cases were reviewed to identify those with cancer of an organ that does not concentrate any of the primary radionuclides expected to be found at DOE sites and those with little or no apparent internal dose to that organ.

In accordance with OCAS-IG-002 (Internal Dose Reconstruction Implementation Guideline, Rev. 0, August 2002), internal dose is assigned to employees who were monitored but had no detectable activity ("positive") in their samples and to employees who were not included in a bioassay program,

because there is some amount of intake and associated dose that is not detectable by an internal dosimetry program. To expedite the dose reconstructions, cases that met the criteria above and are thus considered unlikely to have significant internal radiation exposure, will first be evaluated with the method described in this TIB. If the outcome yields a probability of causation >50%, a dose reconstruction using more reasonable assumptions will be performed.

3.0 Implausible Undiscovered Intakes

3.1 ASSUMPTIONS

For claims where it is considered likely that the covered employee had no significant internal radiation exposure, a method to expedite claims has been developed in accordance with 42 CFR 82.10(k)(2). This method assumes the "largest reasonably possible value" of the source term comprised of radionuclides that are/were typically the more significant radionuclides (by either preponderance or by internal dose significance) on a site. For this "worst-case" estimate of internal dose, it is assumed that on the first day of the first year of employment, the covered employee had an acute inhalation intake of each of the radionuclides in the source term, in the amounts listed below.

Based on historical data, it is believed to be highly unlikely that such an intake could have occurred without being detected by workplace monitoring at the time. It is also believed that this is a significant overestimate of internal dose for an unmonitored covered employee or a covered employee with no internal monitoring above detection thresholds.

3.1.1 Additional Assumptions

Additional assumptions to develop this method are:

- All intakes are inhalations of standard 5 micrometer AMAD, except for I-131, which is assumed to be in vapor form (class SR-1).

- The most soluble form of the radionuclide specified in ICRP 1994a was used to maximize dose to systemic organs, except as noted below; dose to lung is not germane to this exercise.

- Because maximum permissible body burdens were the metric (actually uptake) for so many years, the assumed implausible uptake was based on a percent of the radionuclide-specific Maximum Permissible Body Burden (MPBB) for soluble chemical forms as defined by the National Committee on Radiation Protection (NCRP1959). It was assumed that an intake resulting in 10% of a MPBB would not have likely occurred to an unmonitored worker or would have likely resulted in a readily noticeable bioassay result in a monitored worker, readily noticeable air sample, or other indicators of personnel contamination. In other words, an event providing the possibility of an intake resulting in a body burden exceeding 10% of the MPBB would not have gone unnoticed and there would be some sort of indication in the worker's records. This assumption applies to bona fide DOE sites and national labs with active radiation protection programs, not to Atomic Weapons Employers (AWEs). The current ICRP methodology is used calculate doses from these implausible intakes.

- For type F and M materials the associated derived intake (i.e., intake resulting in a 10% MPBB) was assumed to be 10 and 20 times the 10% MPBB, respectively. The factors of 10 and 20 come from the current ICRP models and the differences between an intake and the activity that is present in the body after the initial clearance of the short-term compartments. These factors are used to relate the historical quantity of control, body burden, which was

based on ICRP 2 methods to the present quantity of control, intake, which is based on current ICRP methods. These factors were estimated from tables in the November 2002 issue of Health Physics that give the intake retention fraction for the whole body (without the ET region) as a function of time after acute intake for different elements and inhalation types. Because initial deposition in the lung was usually not considered by the ICRP (1959) to be part of the "body burden," the retention fractions used allowed some time for the rapid clearance components. Examples are listed in Table 3.1.1-1.

Table 3.1.1-1. Fractional Retention in Whole Body after Initial Clearance from Lung.[a]

Element	Fractional Retention in Whole Body After Initial Clearance from Lung			
	Inhalation Type F		Inhalation Type M	
	3 Days Post Intake	4 Days Post Intake	60 Days Post Intake	90 Days Post Intake
Strontium	0.227	0.119		
Cesium	0.449	0.440		
Cerium			0.0683	0.0654
Ruthenium	0.251	0.227	0.0392	0.0392
Barium	0.147	0.104		
Lanthanum	0.307	0.291	0.0683	0.0654
Zirconium	0.278	0.252	0.0496	0.0448
Niobium			0.0476	0.0409
Cobalt			0.0410	0.0337
Technetium	0.217	0.160	0.0321	0.0238
Europium			0.0605	0.0568

[a]. Source: Potter 2002

- The assumption of type S for Co-58 and Co-60 is used because it results in larger doses to the systemic organs because of the high-energy photons. Although the logic in the bullet above does not directly apply to type S material, the fractional retention in the whole body is similar for type M and type S at 60 and 90 days, so the derived intake is estimated as 20 times the 10% MPBB.

- Zn-65 is classified as type S in ICRP 68. As for Co-58 and Co-60 type S materials, the logic of determining a body burden to intake conversion factor does not directly apply, but use of the factor of 20 does not seem unreasonable.

- Mn-54 type M has a larger dose conversion factor for most organs/tissues and was generally more claimant favorable than type F.

- Intakes of tritium are not considered by this method and must be addressed by site, depending on the site-specific detection level. Tritium clears very rapidly from the body so multiple undetected intakes are possible, although H-3 is readily detected at very small intake levels by urinalysis. Tritium dose was often reported with an individual's external dose prior to 1989; in such cases no further H-3 analysis may be necessary.

- To be generic to most of the DOE sites, intakes were assumed to involve the most plausible radionuclides for all the sites, even though it is implausible that any one worker had intakes of all the radionuclides. Two groups of radionuclides were considered: sites having one or more reactors and sites without a reactor. The Nevada Test Site was considered a special case for which these two groups of radionuclides might not be appropriate. A review indicated that after 1971, the source term at NTS, except in the tunnels, was consistent with that chosen for reactor sites.

- For sites without a reactor, the actual ratio of radionuclides in mixtures is not relevant because each radionuclide is used at its maximum amount. For sites with reactors, the fission products were chosen for the list based on their relative abundance in fuel or contamination after 180 days of post-irradiation cooling time. This is very conservative for any contamination at facilities other than the reactors themselves.

- A review of the assumed intakes from the method described above and associated possible air concentrations for different periods indicate that the uranium intakes based on soluble uranium may not have been sufficiently conservative for uranium facilities that controlled their programs based on consideration of uranium exposure alone. The derived intakes for uranium-234 and uranium-238 were increased by a factor of 100 to ensure that this scenario will bound doses to those employees who worked at uranium facilities, but were not included in a bioassay program or had no detectable occupationally related activity in them or their samples. Derived intake values for U-234 and U-238 respectively may be lowered to 50 nCi and 5 nCi for employees who worked in areas that were controlled based on the possible presence of other alpha emitters, such as plutonium-238 or plutonium-239.

Table 3.1.1-2. Radionuclides of interest and respective maximum permissible body burdens, assumed intake absorption type, derived maximum plausible intake quantities, and absorption type, radiation type and IMBA type and version used for estimating organ doses at DOE sites.

Radio-nuclide	Soluble (S) MPBB (nCi) [a]	Assumed Intake Absorp-tion Type	Derived Intake[b] Used in IMBA[c] Runs (nCi)	Absorp-tion Type Used in IMBA[c] Runs	Radiation Type	IMBA[c] Type	IMBA[c] Version
Sites without a Reactor:							
Sr-90	2,000	F	2,000	F	electron > 15 keV	OCAS	3.0.48
Tc-99	10,000	F	10,000	F	electron > 15 keV	OCAS	3.0.63
Cs-137	30,000	F	30,000	F	electron > 15 keV	NIOSH	1.0.42
U-234	50	F	5,000[d]	F	alpha	NIOSH	1.0.42
U-238	5	F	500[d]	F	alpha	NIOSH	1.0.42
Am-241[e]	50	M	100	M	alpha	NIOSH	1.0.42
Cm-244	100	M	200	M	alpha	OCAS	3.0.48
Np-237	60	M	120	M	alpha	OCAS	3.0.48
Pu-238	40	M	80	M	alpha	NIOSH	1.0.42
Pu-239	40	M	80	M	alpha	NIOSH	1.0.42
Th-230[f]	50	M	100	M	alpha	OCAS	3.0.63
Cf-252	40	M	80	M	alpha	OCAS	3.0.63
Sites with a Reactor - All those above plus:							
Mn-54	20,000	M	40,000	M	Photon>250 keV	OCAS	3.0.63
Co-58	30,000	M	60,000	S	Photon>250 keV	OCAS	3.0.63
Co-60	10,000	M	20,000	S	Photon>250 keV	NIOSH	1.0.42
Fe-59	20,000	F	20,000	F	electron > 15 keV	OCAS	3.0.63
Zn-65	60,000	M	120,000	S	Photon>250 keV	OCAS	3.0.63
Y-91	5,000	M	10,000	M	electron > 15 keV	OCAS	3.0.63
Nb-95	40,000	M	80,000	M	electron > 15 keV	OCAS	3.0.59
Zr-95	20,000	F	20,000	F	electron > 15 keV	OCAS	3.0.59
Ru-103	20,000	F	20,000	F	electron > 15 keV	OCAS	3.0.59
Ru-106	3,000	F	3,000	F	electron > 15 keV	OCAS	3.0.59

Radio-nuclide	Soluble (S) MPBB (nCi) [a]	Assumed Intake Absorp-tion Type	Derived Intake[b] Used in IMBA[c] Runs (nCi)	Absorp-tion Type Used in IMBA[c] Runs	Radiation Type	IMBA[c] Type	IMBA[c] Version
I-131	700	F	700	F	electron > 15 keV	OCAS	3.0.63
Ce-141	30,000	M	60,000	M	electron > 15 keV	OCAS	3.0.63
Ce-144	5,000	M	10,000	M	electron > 15 keV	OCAS	3.0.63
Pm-147	60,000	M	120,000	M	electron > 15 keV	OCAS	3.0.63
Eu-154	5,000	M	10,000	M	electron > 15 keV	OCAS	3.0.59
Eu-155	70,000	M	140,000	M	electron > 15 keV	OCAS	3.0.59

a. Source: NCRP (1959).
b. Derived maximum plausible intake based on 10% of the soluble NCRP (1959) MPBB multiplied by factors of 10 and 20 for assumed intake absorption types of F and M, respectively.
c. Source: Birchall 2002, 2003, 2003a, and 2003b.
d. The uranium values for the general approach are increased to assure that this is a bounding approach for all facility-types. The lower values of 50 nCi for U-234 and 5 nCi for U-238 can be used to assign "worst-case" doses to individuals who did not work in areas where controls were based primarily on uranium exposure concerns.
e. Pu-241 is accounted for by the Am-241 intake value.
f. Th-230 can be used as a surrogate for Th-232.

After the intakes of the radionuclides were determined, the alpha radionuclides and the beta/gamma radionuclides were each summed to show the total intake of each for the reactor and non-reactor sites. These alpha and beta/gamma intakes were then used to consider what the average air concentrations would have been if the intakes had occurred chronically rather than acutely. A breathing rate of 2.4 E+09 milliliters per year, based on Reference Man, was used to calculate the air concentrations for 1, 10 and 30 years that would produce the given intakes. Site air concentration criteria (area controls, respirator usage, etc.) were based on the permissible air concentration of radionuclides assumed to be present in significant quantities. At many facilities, air concentration controls were specified for alpha, based on Pu-238/239/240, and beta/gamma, based on Sr-90. The typical air concentration controls listed in Table 3.1.1-3 are based on the radionuclide air concentration values cited in NCRP (1959), AEC (1968), and 10 CFR 835, which for most radionuclides did not change significantly over time. In later years, the beta/gamma "limit" may have increased to 2 E-9 uCi/ml, but the control criteria also changed and were based on 10% of the limit, which would equate to an air concentration of 2 E-10 uCi/ml.

It can be seen from the Table 3.1.1-3 that even chronic intakes at these levels would exceed typical air concentration controls that were being used at most sites since the 1950s to manage access to and work practices in areas that had increased radioactivity in air. Acute intakes are assumed for this worst-case approach, because these will result in larger assignments of dose.

Table 3.1.1-3 also shows that workers who spent significant time in areas where uranium air concentration controls were used to manage access and work practices do not appear to be good candidates for the non-uranium facility approaches. The non-uranium facility approaches can be used for reconstruction of doses for claims that did not involve exposure in uranium process environments for more than a few years. The uranium intakes are included with the alpha intakes with the alpha intakes, because it is unlikely that the measured alpha air concentration would have excluded uranium. Uranium is shown by itself for nonreactor facilities, because the air concentration control in a uranium facility was likely to be larger than the cited gross alpha air concentration control. For facilities with reactor source terms the uranium air concentration was not listed, because it is believed

that the control would have been on the gross alpha air concentration and because the uranium concentration will be the same as the comparable nonreactor facility.

Table 3.1.1-3. Total Intakes and Possible Average Air Concentrations

Facility Type	Radiation Type	Intake (uCi)	Air concentration based on 1 year intake (uCi/ml)	Air concentration based on 10 year intake (uCi/ml)	Air concentration based on 30 year intake (uCi/ml)	Typical air concentration controls (uCi/ml)
Non-Reactor and Non-uranium	Alpha	0.815	3.4E-10	3.4E-11	1.1E-11	2 E-12
	Beta/gamma	42	1.8E-08	1.8E-09	5.8E-10	3 E-10
	Uranium	0.055	2.3E-11	2.3E-12	7.6E-13	7 E-11
Reactor and Non-uranium	Alpha	0.815	3.4E-10	3.4E-11	1.1E-11	2 E-12
	Beta/gamma	776	3.2E-07	3.2E-08	1.1E-08	3 E-10
Non-Reactor and Uranium	Alpha	6.26	2.6E-09	2.6E-10	8.7E-11	2 E-12
	Beta/gamma	42	1.8E-08	1.8E-09	5.8E-10	3 E-10
	Uranium	5.5	2.3E-09	2.3E-10	7.6E-11	7 E-11
Reactor and Uranium	Alpha	6.26	2.6E-09	2.6E-10	8.7E-11	2 E-12
	Beta/gamma	776	3.2E-07	3.2E-08	1.1E-08	3 E-10

To encompass employees who may have worked in facilities where uranium was the primary radionuclide of concern for internal dose, the assumed uranium intakes were each increased by a factor of 100. The factor of 100 was chosen to assure that this approach adequately bounds the level of exposure that potentially could have gone unnoticed at uranium facilities.

To facilitate use of these data, two spreadsheets were created based on annual organ doses calculated with IMBA (Birchall 2002, 2003, 2003a, and 2003b) for the derived acute inhalation intakes shown in Table 3.1.1-2 and for the same derived intakes, except that the U-234 and U-238 intakes are lowered respectively to 50 and 5 nanocuries for non-uranium facilities. These spreadsheets allow the user to input claim specific data to obtain data necessary for the IREP program for the organs shown in Table 3.1.1-4. This approach may be also used for diagnosed conditions that are to have internal doses calculated using largest dose to a non-modeled organ as specified in OCAS-TIB-008.

Table 3.1.1-4. Organs Appropriate for Applying the Maximum Internal Dose for Certain DOE Complex Claims

Adrenals	Esophagus	Pancreas	Thymus
Brain	G. Bladder	Skin	U. Bladder
Breast	Heart	Spleen	Uterus
Colon	Muscle	Stomach	

4.0 Calculation of Internal Dose from Maximum Intakes

The calculation of organ dose assumes that the covered employee had maximum intakes of all of the radionuclides listed in Table 3.1.1-1. While individual exposures to a single radionuclide from the

mixtures defined in this method possibly could be underestimated, despite the attempt to overestimate each radionuclide intake, it is believed that the individual dose determined from this method will be overestimated, because at a minimum, intakes and doses from 12 radionuclides are considered.

Annual organ doses were calculated as follows. For each radionuclide listed in Table 3.1.1-1, the annual organ doses from inhalation intakes were calculated with IMBA (Birchall 2002, 2003, 2003a, 2003b). The absorption types for Table 3.1.1-1 radionuclides were selected based on the type that produced the largest non-metabolic annual organ doses. The ICRP 66 (ICRP 1994) respiratory tract model was used with the breathing rate of a standard worker. The assumed particle size distribution was 5 µm AMAD (default for worker exposures). The input summary is listed in Table 3.1.1-1.

To facilitate entry of organ doses into the IREP computer code, Excel® workbooks (entitled "Maximum Internal Dose Calculation Workbook.xls" and "Maximum Internal Dose Calculation Non-Uranium Facility Workbook.xls") were developed to create the IREP annual organ dose input data. Annual organ doses are summed over all radiation types and presented for each year that the covered employee was possibly exposed at the DOE site. Examples of annual organ dose output for the bladder are shown in Attachment A, Figures A.2.1-1 and A.2.1-2, for sites with and without reactors, respectively. In addition, Attachment A contains details of the dose calculations performed by the dose estimation Excel® spreadsheet.

5.0 Applications and Limitations

The following conditions must be met to apply this approach.

- The covered employee's initial hire date was 1970 or later.
- The covered employee was not included in a bioassay program or no detectable occupationally related activity was found in them or their samples.
- The target organ must be listed in Table 3.1.1-4 or must be an organ whose dose is based on the highest non-modeled organ dose.
- Employees who would have had no significant exposure to uranium can be assessed with the smaller uranium source term.
- The employment must have been at a DOE site or national laboratory with an active radiation protection program.
- Dose from Atomic Weapons Employers (AWEs) must be reconstructed by other methods.
- Dose from Nevada Test Site may be assigned by this approach if the employee worked at NTS after 1971, was not involved in tunnel work and meets the other conditions herein.
- Dose from tritium is to be considered separately.

6.0 References

10 CFR 835, Occupational Radiation Protection; Final Rule, December 14, 1993.

42 CFR 82, 2002, Methods for Radiation Dose Reconstruction under the Energy Employees Occupational Illness Compensation Act of 2000, May.

AEC (Atomic Energy Commission) Manual, Chapter 0524, Standards for Radiation Protection, AEC 0524-01, November 8, 1968.

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ICRP (International Commission on Radiological Protection), 1994, Human Respiratory Tract Model for Radiological Protection, ICRP Publication 66, Annals of the ICRP Vol. 24 (1-3), Pergamon Press, Oxford, England.

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Recommendation of the National Committee on Radiation Protection, National Bureau of Standards (NBS) Handbook 69 (1959), Maximum Permissible Body Burdens and Maximum Permissible Concentration of Radionuclides in Air or Water for Occupational Exposure, Superintendent of Documents, U.S. Department of Commerce, U.S. Government Printing Office, Washington, D.C., June 5, 1959.

ATTACHMENT A

A.1 PURPOSE

The purpose of this attachment is to provide a description of the Excel® workbooks entitled "Maximum Internal Dose Calculation Workbook.xls" and “Maximum Internal Dose Calculation Non-Uranium Facility Workbook.xls” that were developed to calculate annual organ doses in support of this Technical Information Bulletin (TIB). In addition, this attachment also provides documentation and a description of the IMBA (Birchall 2002, 2003, 2003a, 2003b) input and output files used to calculate and report organ doses and a description of the user workbook interface developed to assist the dose reconstructors in processing claims.

The workbooks and the IMBA input and output files used in support of the Maximum Internal Dose Estimates for Certain DOE Complex Claims TIB are supplied as a folder, “ORAUT-OTIB-0002 Rev 00 Attachment A,” on CD only.

A.2 WORKBOOK DESCRIPTION

The Excel® workbooks entitled "Maximum Internal Dose Calculation Workbook.xls" and “Maximum Internal Dose Calculation Non-Uranium Facility Workbook.xls” consist of multiple worksheets that are grouped according to function. These worksheets and their respective functions are described below.

A.2.1 Dose Reconstructor Interface

When a dose reconstructor uses the Maximum Internal Dose Calculation Workbook or Maximum Internal Dose Calculation Non-Uranium Facility Workbook, they will only see the worksheet entitled "UserPrompt."; all other worksheets contained within the workbook are hidden to assure the dose reconstructor enters the correct information in the appropriate location. This worksheet prompts the dose reconstructor for the following information:

- First year of employment
- Organ of interest
- Reactor or non-reactor site

Based on the responses of the dose reconstructor, when the "Open Template" button is pushed, the workbook will then calculate the annual organ doses and fill the data template that can be copied into an IREP template for the organ of interest. Figures A.2.1-1 and A.2.1-2 below show a typical output for a claimant whose first year of employment was 1975. The date of diagnosis of the cancer was 1985, so the values after this year were manually deleted. The information provided in these templates is suitable for direct input into the IREP computer code (NIOSH-IREP 2003) used for estimation of probability of causation. Although only 10 years of organ dose information is shown in Figures A.2.1-1 and A.2.1-2, the actual template contains 65 years of organ doses to accommodate differing years of employment and cancer diagnosis dates.

Figure A.2.1-1. Example output file of bladder dose for a covered employee who is assumed to have internal exposure beginning in 1975 at a non-uranium DOE complex site with a reactor and whose covered condition was diagnosed in 1985.

Enter first exposure year in cell H3, for IREP input format	1975	Copy internal dose data from the first year through the year of cancer diagnosis

Exposure Year	Exposure Rate	Radiation Type	Dose Distribution Type	Parameter 1	Parameter 2	Parameter 3
1975	chronic	alpha	Constant	9.85E-02	0	0
1975	chronic	electrons E>15keV	Constant	1.16E+00	0	0
1975	chronic	photons E>250keV	Constant	7.21E-01	0	0
1976	chronic	alpha	Constant	8.94E-02	0	0
1976	chronic	electrons E>15keV	Constant	1.28E-01	0	0
1976	chronic	photons E>250keV	Constant	1.20E-01	0	0
1977	chronic	alpha	Constant	8.40E-02	0	0
1977	chronic	electrons E>15keV	Constant	4.20E-02	0	0
1977	chronic	photons E>250keV	Constant	2.93E-02	0	0
1978	chronic	alpha	Constant	7.97E-02	0	0
1978	chronic	electrons E>15keV	Constant	2.50E-02	0	0
1978	chronic	photons E>250keV	Constant	9.48E-03	0	0
1979	chronic	alpha	Constant	7.69E-02	0	0
1979	chronic	electrons E>15keV	Constant	1.85E-02	0	0
1979	chronic	photons E>250keV	Constant	4.36E-03	0	0
1980	chronic	alpha	Constant	7.48E-02	0	0
1980	chronic	electrons E>15keV	Constant	1.47E-02	0	0
1980	chronic	photons E>250keV	Constant	2.55E-03	0	0
1981	chronic	alpha	Constant	7.36E-02	0	0
1981	chronic	electrons E>15keV	Constant	1.21E-02	0	0
1981	chronic	photons E>250keV	Constant	1.66E-03	0	0
1982	chronic	alpha	Constant	7.23E-02	0	0
1982	chronic	electrons E>15keV	Constant	1.01E-02	0	0
1982	chronic	photons E>250keV	Constant	1.13E-03	0	0
1983	chronic	alpha	Constant	7.14E-02	0	0
1983	chronic	electrons E>15keV	Constant	8.46E-03	0	0
1983	chronic	photons E>250keV	Constant	7.94E-04	0	0
1984	chronic	alpha	Constant	7.09E-02	0	0
1984	chronic	electrons E>15keV	Constant	7.13E-03	0	0
1984	chronic	photons E>250keV	Constant	5.67E-04	0	0
1985	chronic	alpha	Constant	7.06E-02	0	0
1985	chronic	electrons E>15keV	Constant	6.05E-03	0	0
1985	chronic	photons E>250keV	Constant	4.12E-04	0	0

Figure A.2.1-2. Example output file of bladder dose for a covered employee who is assumed to have internal exposure beginning in 1975 at a non-uranium DOE complex site without a reactor and whose covered condition was diagnosed in 1985.

Enter first exposure year in cell H3, for IREP input format	1975	Copy internal dose data from the first year through the year of cancer diagnosis

Exposure Year	Exposure Rate	Radiation Type	Dose Distribution Type	Parameter 1	Parameter 2	Parameter 3
1975	chronic	alpha	Constant	9.85E-02	0	0
1975	chronic	electrons E>15keV	Constant	7.14E-01	0	0
1975	chronic	photons E>250keV	Constant	0.00E+00	0	0
1976	chronic	alpha	Constant	8.94E-02	0	0
1976	chronic	electrons E>15keV	Constant	6.84E-02	0	0
1976	chronic	photons E>250keV	Constant	0.00E+00	0	0
1977	chronic	alpha	Constant	8.40E-02	0	0
1977	chronic	electrons E>15keV	Constant	7.08E-03	0	0
1977	chronic	photons E>250keV	Constant	0.00E+00	0	0
1978	chronic	alpha	Constant	7.97E-02	0	0
1978	chronic	electrons E>15keV	Constant	1.00E-03	0	0
1978	chronic	photons E>250keV	Constant	0.00E+00	0	0
1979	chronic	alpha	Constant	7.69E-02	0	0
1979	chronic	electrons E>15keV	Constant	3.59E-04	0	0
1979	chronic	photons E>250keV	Constant	0.00E+00	0	0
1980	chronic	alpha	Constant	7.48E-02	0	0
1980	chronic	electrons E>15keV	Constant	2.51E-04	0	0
1980	chronic	photons E>250keV	Constant	0.00E+00	0	0
1981	chronic	alpha	Constant	7.36E-02	0	0
1981	chronic	electrons E>15keV	Constant	2.05E-04	0	0
1981	chronic	photons E>250keV	Constant	0.00E+00	0	0
1982	chronic	alpha	Constant	7.23E-02	0	0
1982	chronic	electrons E>15keV	Constant	1.72E-04	0	0
1982	chronic	photons E>250keV	Constant	0.00E+00	0	0
1983	chronic	alpha	Constant	7.14E-02	0	0
1983	chronic	electrons E>15keV	Constant	1.45E-04	0	0
1983	chronic	photons E>250keV	Constant	0.00E+00	0	0
1984	chronic	alpha	Constant	7.09E-02	0	0
1984	chronic	electrons E>15keV	Constant	1.23E-04	0	0
1984	chronic	photons E>250keV	Constant	0.00E+00	0	0
1985	chronic	alpha	Constant	7.06E-02	0	0
1985	chronic	electrons E>15keV	Constant	1.05E-04	0	0
1985	chronic	photons E>250keV	Constant	0.00E+00	0	0

A.2.2 Maximum Permissible Body Burden

The worksheet entitled MPBBs (Maximum Permissible Body Burdens) contains information related to the estimation of the derived intakes used for calculation of maximum organ doses. The assumptions used for selection of solubility types and intake factors used to derive assumed intakes are provided in the main body of the TIB. The worksheet also specifies the radionuclides appropriate for consideration at both reactor and non-reactor sites. This worksheet is provided for information only and to document the derived intake calculations. The values contained within this worksheet are not linked to other worksheets in the workbook.

A.2.3 Radionuclide Intakes

The worksheet entitled "Intakes" lists the radionuclides (Column A, Rows 11 through 38), the IMBA absorption and radiation types (Columns B and C, Rows 11 through 38), and derived intake quantities for reactor and non-reactor sites (Columns D and E, Rows 11 through 38, respectively) used to calculate the maximum annual organ doses. The information related to the radiation-types, 65-year doses and radionuclide-specific dose fractions is provided for information only and to assist in the quality control process. Thus, this information-only data is not linked to other worksheets nor used directly by the worksheet for the organ dose calculations.

The annual dose calculations are based on the intake values located in Column F, Rows 11 through 38. These values are linked to the quantities in Columns D and E, Rows 11 through 38 for reactor and non-reactor sites, respectively by conditional statements based on user input from the UserPrompt worksheet, which is reproduced on the Intake worksheet at cell B2. The "Assumed Intake" values (derived intake quantities) in Column F, Rows 11 through 38 are then linked to the respective radionuclide sheets (at location E3) to provide the appropriate multiplier for the organ dose calculations (see discussion below).

A.2.4 Radionuclides Important to Internal Dose

Individual worksheets are included in the workbook for the 28 radionuclides determined to be important to internal dose (see Table 3.1.1-2 of the TIB). Each of these radionuclide-specific worksheets contains the derived intake values (cell E3), assumed intake adsorption type used for the IMBA runs (cell E2), radiation type (cell A1) and the IMBA type and version used for estimating organ doses (cell F1).

On each of these radionuclide worksheets, results of the IMBA runs (i.e., annual organ dose in rem), for a one-nanocurie (nCi) acute inhalation intake of the respective radionuclide, are captured in cells B77 through AJ141 for each of the organs of interest (see below). Each of these values is multiplied by the derived intake quantity (cell E3) and the products are recorded in cells B6 through AJ70. These values represent the annual organ dose resulting from the derived intake values shown in Table 3.1.1-2 of the TIB and are summed, by radiation type in the "organ" worksheets that are used for the IREP inputs. The IMBA input and output for all radionuclides and organs of interest are reproduced on the Attachment A CD.

A.2.5 Radiation Types

Because the IREP computer code requires specification of radiation type, three worksheets were developed to add the IMBA doses from the radionuclide worksheets for each radiation type. The radiation type worksheets are entitled "Alpha,'" "Electron>15 keV," and "Photon>250keV." Radionuclides contributing dose to each of these radiation type worksheets are as follows:

- Alpha - Cf-252, Am-241, Cm-244, Np-237, Pu-238, Pu-239, U-234, and U-238
- Electron>15keV - Sr-90, Cs-137, I-131, Ce-141, Eu-154, Eu-155, Nb-95, Ru-103, Ru-106, Zr 95, Y-91, Tc-99, and Pm-147
- Photon>250keV - Co-58, Co-60, Mn-54, and Zn-65

For each of the three radiation types, annual doses from the appropriate radionuclide worksheets (cells B6 through AJ70) were summed in cells B6 through AJ70 for each organ of interest. These annual doses summed by radiation type are then carried to the organ worksheets for preparation of the IREP input template.

A.2.6 Organs of Interest

Individual worksheets were included for each of the following organs of interest:

- Adrenals
- Esophagus
- Pancreas
- Thymus
- Brain
- G. Bladder
- Skin
- Thyroid
- Breast
- Heart
- Spleen
- U. Bladder
- Colon
- Muscle
- Stomach
- Uterus

These organ worksheets arrange the organ doses by radiation type from all of the radionuclides of interest into a format suitable for direct entry into the IREP computer code (see Figures A.2.1-1 and A.2.1-2 above). The annual dose information by radiation type is linked from the radiation type worksheets and presented in columns B, C, and D; Rows 6 through 70, representing 65 years of exposure. These doses are then arranged in the IREP input form (shown in various colors) with three entries (one for each radiation type) for each year of exposure. These radiation- and year-specific annual doses (rem) are shown in Column J, Rows 6 through 200. The first year that dose is calculated is shown in cell H3 and is set based on input provided by the dose reconstructor through the UserPrompt worksheet.

To setup the IREP input data, the dose reconstructor would copy cells F6 through L200 and paste these values into the identical cells in the IREP template. The actual number of yearly doses considered in the probability of causation calculation is set by the reconstructor in the "Number of Exposures" block in the IREP template.

A.3 REFERENCES

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