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Maximum Internal Dose Estimates for Savannah River Site (SRS) Claims

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."*

To facilitate timely processing of Savannah River Site claims under the Energy Employee Occupational Illness Compensation Program Act (EEOICPA), cases were reviewed to identify those with 1) little or no apparent internal dose and 2) cancer of an organ that does not concentrate internally deposited radionuclides that might be associated with work at the Savannah River Site. The cases were further screened to find those that met the following criteria:

- No detectable activity in *in vitro* bioassay samples, other than H-3.
- No detectable activity in chest counts.
- No detectable activity in whole body counts other than Cs-137, Co-60, or Eu-152.

(Note that although Eu-152 was indicated as detected in a few whole body counts, it was not considered to be due to an intake. Eu-152 was in the whole body counter library because it was used in the daily check source of the body counter and was not otherwise expected to be found on site.)

For the purposes of the compensation program, 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 the 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 will be evaluated with the method described in this paper. The selected values were assigned per 42 CFR 82.10(k)(2) to be worst-case estimates.

These hypothetical intakes were based on recorded internal doses at SRS and were assumed to be composed of the radionuclides contributing the majority of the recorded internal dose at the Savannah River Site, except for tritium (assignment of tritium dose is discussed at the end of this paper). All recorded inhalation intakes in the history of the site were reviewed and the largest intakes for each radionuclide were selected. Hypothetical inhalation intakes, based on the recorded intakes, were used to calculate organ doses for each calendar year from the start of employment at the Savannah River Site through the year of cancer diagnosis. The IMBA-NIOSH and IMBA-Expert OCAS dose estimating computer codes were used in these calculations. The calculated organ doses were compiled into a spreadsheet for use by the dose reconstructors on the selected cases.

Hypothetical Intake

Internal dose was assigned based upon a hypothetical intake with the following characteristics:

- All radionuclides for which internal deposition by inhalation was calculated by the Savannah River Site were reviewed, except for tritium, which is addressed separately.
- The amount of the inhalation intake for each radionuclide is the average (mean) of the five largest documented intakes, or the average of all intakes if there were fewer than five intakes reported for a radionuclide.
- An acute inhalation intake was assumed to have occurred on January 1 in the first year of employment.
- ICRP 66 and 68 modeling and default parameter values were used to determine dose.
- The material type resulting in the largest dose to the organ or tissue of interest was used. This was typically the most soluble form of the material because it would clear from the lung more rapidly than insoluble material, thus depositing in the organ or tissue sooner.

The largest inhalation intakes reported by SRS are summarized in Tables 1 and 2, and include the material classes used in the SRS calculations. Several complicating factors arose in the use of the intake amounts to be applied to these cases. Intakes and doses at SRS were calculated using regulatory-prescribed ICRP 30 methodologies rather than the newer ICRP methodology prescribed for this dose reconstruction effort. The material classes used in the calculations were based on workplace source term information or the class that provided the best fit to the bioassay data; the most claimant favorable class was not necessarily selected. In addition, several of the radionuclides identified in SRS intakes were not included in the IMBA software used to perform the annual dose calculations for the dose reconstruction.

Table 1: Largest intakes assigned at SRS (radionuclides available in IMBA)

Nuclide	Intake Date	Activity (nCi)	Class	Average (nCi)	Nuclide	Intake Date	Activity (nCi)	Class	Average (nCi)
Am-241	11-Dec-61	30.78	Y		Pu-239	24-Apr-62	267.87	Y	
	29-Apr-81	12	Y	14.20		7-Dec-70	178.58	Y	129.1
	4-Sep-84	10	Y			24-May-77	85.718	Y	
	8-May-72	9.2	Y			7-Sep-56	59.824	Y	
	6-May-69	9	W			20-Sep-79	53.574	Y	
Cm-244	10-Mar-70	200	W		Pu-241	24-Apr-62	3867	Y	
	7-Feb-65	124	W	91.8		7-Dec-70	2578	Y	1863.9
	7-Oct-64	72	W			24-May-77	1237.44	Y	
	4-Aug-71	33	Y			7-Sep-56	863.63	Y	
	19-Feb-71	30	W			20-Sep-79	773.4	Y	
Co-60	14-Nov-85	430	W	430	Sr-90	1-Mar-59	392	D	
Cs-137	19-Nov-67	664.06	D			25-Aug-59	213.9	W	158.18
	10-Sep-79	410	D	361.39		5-Nov-86	69	D	
	28-Feb-74	330	D			5-Nov-86	64	D	
	3-Mar-73	202.9	D			5-Nov-86	52	D	
	13-Jan-75	200	D		U-234	1-Jan-59	172.58	D,W,Y	
Np-237	6-Mar-91	3	W			1-Jan-71	97.9	W,Y	105.4
	27-Sep-99	1.26	W	1.17		1-Jul-64	77	W	
	1-Oct-61	0.82	W			17-Jul-69	95	W,Y	
	10-Aug-99	0.534	W			15-Oct-69	84.68	W, Y	
	5-Dec-98	0.24	W		U-238	5-Jan-53	87.4	W	
Pu-238	1-Jul-67	330	Y			14-Dec-69	8.87	D	20.95
	11-Apr-67	270	W	250		25-Apr-70	3.23	D	
	15-Jun-81	260	Y			10-Mar-62	2.8037	D	
	3-Dec-69	200	Y			2-Apr-71	2.43	D	
	22-Jul-77	190	W, Y		U-235	21-Mar-85	0.14	D	0.14
						19-Jul-90	0.13	Y	

Table 2: Largest inhalation intakes assigned at SRS (radionuclides not available in IMBA)

Nuclide	Intake Date	Activity (nCi)	Class	Average (nCi)
Ce-144	13-Jan-75	1200	W	623.96
	5-Nov-86	560	W	
	5-Nov-86	520	W	
	5-Nov-86	420	W	
	17-May-67	419.8	W	
Cf-252	27-Jan-72	20	D/W	7.28
	27-Jan-72	1.3	W	
	9-Apr-98	0.55	W	
Cm-242	6-Apr-72	24	Y	13.35
	23-Aug-79	2.69	W	
Nb-95	20-Oct-83	650	W	650
Ru-106	21-Nov-68	595	W	306.6
	20-Oct-83	325	W	
	25-Mar-74	266	W	
	17-May-67	217	W/Y	
	28-Feb-74	130	Y	
Zn-65	3-Aug-89	700	Y	700
Zr-95	25-Mar-74	599	W	359.72
	11-Apr-74	410	W	
	21-Nov-68	333	W	
	20-Oct-83	325	W	
	17-May-67	131.6	D/W	

Because these values are being applied as a large overestimate of the dose likely received by the Covered Employee, it is not necessary to use the exact values determined by SRS but it must be shown that the values are indeed a likely overestimate. To demonstrate this, the intake retention fractions (IRFs) for the radionuclides of interest from ICRP 30 and ICRP 68, for the applicable material classes/absorption types, are compared for several times following an intake. Tables 3 through 10 list the intake retention fractions for five specified times following an acute inhalation intake for the material type assumed here for dose reconstruction purposes and the material class(es) applied for the SRS-calculated intakes in Tables 1 and 2. ICRP 30 IRF values were taken from NUREG/CR-4884, Interpretation of Bioassay Measurements. ICRP 68 values are from Health Physics, Volume 83, No. 5. Tables 3 through 8 contain 24-hour urinary (systemic when urine values not available) excretion values while Tables 9 and 10 contain whole body retention values. The relative intake, determined from a single bioassay measurement at a given point in time, of the ICRP 30 methodology to the ICRP 68 methodology, is calculated in the final column(s). When IRFs were available in the references for the SRS material classes, the relative intakes are weighted for the multiple classes assumed in the SRS calculations. For example, of the top five Pu-238 intakes, one was assessed as W, three as Y and one as a mixture of W and Y, so the normalization for Pu-238 included three parts class W and seven parts class Y.

Table 3: Plutonium IRFs and relative intakes (ICRP 30/68) following acute intake of unit activity

	IRFs			Relative intake ICRP 30/68	
Days since intake	**Type M**	**Class W**	**Class Y**	**Pu-238**	**Pu-239, Pu-241**
5	3.91E-05	5.39E-05	2.87E-06	2.50E+00	1.36E+01
10	1.54E-05	2.54E-05	1.29E-06	2.10E+00	1.19E+01
50	8.52E-06	1.45E-05	7.57E-07	2.03E+00	1.13E+01
600	3.09E-06	2.95E-06	8.68E-07	2.23E+00	3.56E+00
2000	1.64E-06	1.62E-06	8.72E-07	1.55E+00	1.88E+00

Table 4: Americium IRFs and relative intakes (ICRP 30/68) following acute intake of unit activity

Days since intake	Type M IRF	Class W IRF	Class Y IRF	Relative intake ICRP 30/68
5	7.08E-05	5.24E-05	2.21E-06	5.78E+00
10	4.84E-05	4.97E-05	1.87E-06	4.23E+00
50	2.03E-05	3.48E-05	1.91E-06	2.39E+00
600	5.09E-06	3.58E-06	2.23E-06	2.04E+00
1000	3.81E-06	3.48E-06	2.23E-06	1.54E+00

Table 5: Uranium IRFs and relative intakes (ICRP 30/68) following acute intake of unit activity

	IRFs				Relative intake ICRP 30/68	
Days since intake	**Type F**	**Class D**	**Class W**	**Class Y**	**U-238**	**U-234**
5	4.18E-03	1.31E-02	2.69E-03	1.31E-04	3.79E-01	1.71E+00
10	2.67E-03	7.26E-03	1.75E-03	8.42E-05	4.34E-01	1.77E+00
50	3.00E-04	6.67E-04	4.80E-04	2.34E-05	4.76E-01	9.23E-01
600	2.85E-06	1.61E-06	1.05E-06	1.78E-05	1.90E+00	3.94E-01
1000	2.31E-06	1.50E-06	4.58E-07	1.58E-05	1.79E+00	3.75E-01

Table 6: Neptunium-237 IRFs and relative intakes (ICRP 30/68) following acute intake of unit activity

Days since intake	Type M IRF	Class W IRF	Relative intake ICRP 30/68
5	3.35E-04	3.02E-05	1.11E+01
10	1.27E-04	2.56E-05	4.96E+00
50	6.15E-05	1.78E-05	3.46E+00
600	6.91E-06	1.87E-06	3.70E+00
1000	4.67E-06	1.81E-06	2.58E+00

Table 7: Curium-242* IRFs and relative intakes (ICRP 30/68) following acute intake of unit activity

Days since intake	Type M IRF	Class W IRF	Class Y# IRF	Relative intake ICRP 30/68 (unweighted)
5	6.93E-05	5.13E-05		1.35E+00
10	4.64E-05	4.77E-05		9.73E-01
50	1.64E-05	2.81E-05		5.85E-01
600	4.01E-07	2.79E-07		1.44E+00
1000	5.53E-08	4.94E-08		1.12E+00

* Cm-244 is the primary curium isotope at SRS, based on the intake values in Tables 1 and 2, but Cm-244 is not included in NUREG-4884. The class/type comparisons for Cm-244 and Cm-242 will be identical because the decay correction will cancel out of the equation.

Not included in NUREG-4884.

Table 8: Strontium-90 IRFs and relative intakes (ICRP 30/68) following acute intake of unit activity

Days since intake	Type F IRF	Class D IRF	Class W# IRF	Relative intake ICRP 30/68 (unweighted)
5	9.06E-03	2.45E-02		3.70E-01
10	4.11E-03	1.04E-02		3.95E-01
50	3.26E-04	1.94E-04		1.68E+00
600	9.90E-06	2.27E-05		4.36E-01
1000	6.44E-06	1.16E-05		5.55E-01

Not included in NUREG-4884.

Table 9: Cobalt-60 IRFs and relative intakes (ICRP 30/68) following acute intake of unit activity

Days since intake	Type S IRF	Class W IRF	Relative intake ICRP 30/68
5	7.99E-02	2.06E-01	3.88E-01
10	6.53E-02	1.63E-01	4.00E-01
50	4.71E-02	9.78E-02	4.82E-01
600	1.87E-02	7.33E-03	2.55E+00
1000	1.16E-02	4.44E-03	2.61E+00

Table 10: Cesium-137 IRFs and relative intakes (ICRP 30/68) following acute intake of unit activity

Days since intake	Type F IRF	Class D IRF	Relative intake ICRP 30/68
5	4.35E-01	5.72E-01	7.60E-01
10	4.10E-01	5.43E-01	7.55E-01
50	3.16E-01	4.19E-01	7.54E-01
600	8.86E-03	1.27E-02	6.98E-01
1000	7.49E-04	9.94E-04	7.54E-01

It can be seen in Tables 3 through 6 that intakes of alpha emitters are, for the most part, larger when using ICRP 30 versus ICRP 68 IRFs (note that this is based only on single IRF values and not based on the fitting of data). Plutonium, americium and neptunium are significantly overestimated. Uranium-238 intakes are smaller when based on samples collected within the first year but the hypothetical intake of U-238 is a factor of five lower than that of U-234, so this will have little impact on the uranium dose and even less on the dose from the total hypothetical intake. Tables 7 and 8 show that Sr-90 and curium can be underestimated at various times following an intake but these ratios are somewhat low because the more insoluble classes used for the SRS intakes are not included in NUREG-4884. The inclusion of ICRP 30 methodology IRFs for the more insoluble form of these materials would result in an increase in the relative intakes. Co-60 and Cs-137 are also underestimated at times, but not significantly, and their relative contributions to the total dose are quite small.

Worst-case internal doses assigned from these hypothetical intakes are likely to be significant overestimates of dose because they are comprised of the sum of the largest intakes for all radionuclides at SRS. Additionally, the SRS calculations used in determining these intakes applied the ICRP 30 methodology, which for the larger dose contributing radionuclides appear to over-predict the intakes that would have been calculated using the ICRP 68 method. It is very unlikely that an intake so large would go undetected and even more unlikely that a single individual would have been exposed to this entire mixture of radionuclides.

Calculations of Internal Dose from Hypothetical Intakes

Annual organ doses were calculated as follows. For each radionuclide listed in Table 1, except U-235, the annual organ doses from inhalation intakes were calculated with IMBA. IMBA-NIOSH, Version 1.0.42 was used to calculate the annual organ dose, unless a radionuclide was unavailable in its library. For these radionuclides, IMBA-OCAS Expert, Version 3.0.48 was used. The absorption type for Table 1 radionuclides was selected based on the type that produced the largest non-metabolic annual organ doses. The ICRP 66 respiratory tract model was used with the ventilation 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 11. The U-235 intake was ignored because it is about three orders of magnitude less than the U-234 intake and U-235 has a smaller dose conversion factor.

Doses were categorized by radiation type for IREP input, alpha or electron > 15 keV. Although most of the radionuclides have associated photon emissions, selection of the radiation type based on the associated particle emissions is claimant favorable (and avoids having to determine which energy category to select for photon emissions). Although Pu-241 is a beta emitter, it has a half-life of 14.4 years and decays to Am-241, an alpha emitter. Therefore, the claimant favorable decision was made to include Pu-241 doses with the alpha emitters.

Table 11: Hypothetical inhalation intake assumptions

Radionuclide	Absorption Type	Intake (nCi)	Radiation Type
Cs-137	F	361.5*	Electron >15 keV
Sr-90	F	158.18	Electron >15 keV
Cm-244	M	91.8	Alpha
Co-60	S	430	Electron >15 keV
U-238	F	20.9	Alpha
U-234	F	105.4	Alpha
Pu-238	M	250	Alpha
Am-241	M	14.2	Alpha
Pu-239	M	129.1	Alpha
Np-237	M	1.17	Alpha
Pu-241	M	1863.9	Alpha

* The intake used for this dose calculation was slightly larger than the calculated average. Because it was claimant favorable and had a small impact on overall dose, the calculations were not redone.

Doses for intakes of the radionuclides in Table 2 could not be calculated using either version of the IMBA code. Therefore, the Table 2 radionuclides were associated subjectively with three of the Table 1 radionuclides with similar irradiation characteristics. These Table 1 radionuclides are referred to as surrogate radionuclides in this paper. Cs-137 was assigned as surrogate for Zn-65 and Zr-95. Sr-90 was the surrogate for Ru-106, Ce-144 and Nb-95. Cm-244 was the surrogate for Cm-242 and Cf-252.

An effective dose for each Table 2 radionuclide was calculated by multiplying its intake by the largest, inhalation, 5 μm AMAD, effective dose coefficient listed in ICRP 68 for the given radionuclide. Each surrogate radionuclide intake was multiplied by its ICRP 68 inhalation, 5 μm AMAD, effective dose coefficient for the absorption type noted in Table 11. The effective doses were summed for each of the three radionuclide groups. These sums were divided by the respective surrogate radionuclide effective dose to determine a dose adjustment factor for each surrogate that accounts for the associated radionuclides' assumed dose contribution. The results appear in Table 12.

Table 12: Assumptions used to account for dose from radionuclides not available in IMBA

Surrogate Radionuclide	Associated Radionuclides	Intake (nCi)	Effective Dose Coefficient (Sv/Bq)	Effective Dose (rem)	Surrogate Dose Adjustment Factor
Cs-137		361.39	6.70E-09	8.96E-03	2.43E+00
	Zn-65	700	2.80E-09	7.25E-03	
	Zr-95	359.72	4.20E-09	5.59E-03	
Total Effective Dose from Group				2.18E-02	
Sr-90		158.18	3.00E-08	1.76E-02	7.25E+00
	Ru-106	306.6	3.50E-08	3.97E-02	
	Ce-144	623.96	2.90E-08	6.70E-02	
	Nb-95	650	1.30E-09	3.13E-03	
Total Effective Dose from Group				1.27E-01	
Cm-244		91.8	1.70E-05	5.77E+00	1.09E+00
	Cm-242	13.35	3.70E-06	1.83E-01	
	Cf-252	7.28	1.30E-05	3.50E-01	
Total Effective Dose from Group				6.31E+00	

Annual organ doses from the IMBA runs for the surrogate radionuclides were multiplied by the Surrogate Dose Adjustment Factor to account for the doses from the intakes of the associated Table 2 radionuclides.

Annual organ doses for all of the Table 1 alpha radionuclides and Pu-241 were summed after the surrogate doses were adjusted for intakes of the associated Table 2 radionuclides. The resulting table provides the annual alpha doses by organ for the hypothetical SRS worst-case intake.

Annual organ doses for all of the Table 1 radionuclides categorized as electron >15 keV were summed after the surrogate doses were adjusted for intakes of the associated Table 2 radionuclides. The resulting table provides the annual electron doses by organ for the hypothetical SRS worst-case intake.

The alpha and electron annual organ dose spreadsheets were copied into a new workbook to create a spreadsheet of annual internal organ doses by year. For each organ, a worksheet was set up to display the organ doses from the assumed alpha emitters and electron emitters by year. The data was then reorganized into a format that is compatible with the IREP input spreadsheet. The data were set up so that the reconstructor may input the first year of exposure to calculate the years of exposure for a given case. All internal doses are assumed to be chronic exposure rates. The dose distribution type is constant.

Instructions to input claim-specific data from the SRS Lookup Table into an IREP spreadsheet follow.

- Click on the appropriate organ spreadsheet.
- Type the first year of exposure in the green-highlighted cell.
- Select the yellow-highlighted data for the years of interest (always beginning with the top row) and click Copy.
- Click on the row in the Exposure Year Column of the IREP spreadsheet where the data are to be pasted
- For the old IREP template, select Paste Special. Select “Value and number formats” and click on OK or
- For the new IREP template, click on Paste.

Assignment of Tritium Dose

Use of Savannah River Site (SRS) Whole Body Dose Equivalent Records for Tritium

A review of the SRS Internal Dosimetry Technical Basis Manual (TBM), Revision 8, was conducted to identify the methods and assumptions used to calculate tritium doses in order to provide assurance that the recorded tritium doses can be used as found and that dose reconstruction is not needed for reported doses that exceed the calculated minimum reported dose. In addition, for cases where tritium is always recorded as a less than value, information regarding the possible missed recorded dose per year is discussed.

Method(s) of Tritium Dose Calculation

According to the SRS TBM, significant production of tritium began at SRS in 1953 and urinary excretion of tritiated water (HTO) has “always been evaluated in terms of whole body dose equivalent that is added to whole body dose equivalent from external sources” (SRS 2001).

At the SRS, it is assumed that HTO is the most dosimetrically important form of tritium. The dose is calculated from urinary excretion of HTO following an acute intake by fitting a single exponential to the urine concentration data, integrating the curve, and multiplying by a constant to obtain dose. Calculating the dose from HTO excretion provides the dose whether the material inhaled was HTO, tritium gas (T2), or a mixture. Organically bound tritium (OBT) historically has been ignored for occupational dose assessment and SRS assumes that there are no significant quantities of stable metal tritides (SMT). In the past, tritium at SRS has been called “P-10”.

The present constant for dose equivalent rate H_{st} to soft tissue from HTO in body water used at SRS has been in use since 1986:

H_{st} = 1.946E-04 rem/day per µCi/L in body water

This is based on ICRP 30, while earlier methods were based on ICRP 2. The key differences between the present and earlier calculations are the:
• target tissue (43 kg of body water then, 63 kg of soft tissue now),
• default mass of body water (43 kg then, 42 kg now),
• quality factor for tritium (1.7 then, 1.0 now), and
• mean energy of tritium beta particle (6.1 keV then, 5.7 keV now).

With regard to the dates that these changes occurred, the SRS TBM states that “the ICRP 30 methodology for the calculation of tritium dose, especially the quality factor and target tissue, was adopted in 1986” and that “changes in the default biological half-life and mean beta energy were probably made before 1981”. Using this information, the dose rate to the body water for the earlier method can be calculated to be higher than the dose rate to the soft tissue by a factor of approximately 2.7 (this factor may have been slightly lower for some periods, depending on which parameters were modified, but it was never less than one):

H_{bw} = 5.3E-04 rem/day per µCi/L in body water

Another difference between present and the past is that the default biological half-life for HTO was 12 days then, and is 10 days now. The SRS TBM notes that the default half-life was only used to interpolate between samples under a chronic intake pattern, and that the biological removal half-lives observed at SRS have ranged from 4 to 18 days. In days prior to computer evaluation of the data, only results greater than 5 µCi/L were evaluated (SRS TBM 1992). Since computers became available, all reported results are used in the evaluation using an exponential interpolation method.

A questionnaire received by Savannah River in June 1982 indicated that computer evaluation of dose probably was not occurring at that time, but a memorandum dated June 6, 1983 describes a procedure for the a tritium bioassay computer program. Based on this limited information, it is assumed that reported tritium results that were less than 5 µCi/L were being included in dose calculations by 1984.

<u>H-3 Missed Dose</u>
From startup until 1958, the minimum detectable activity (MDA) was 1 µCi/L, measured with a vibrating reed electrometer. The SRS TBM notes that tritium urinalysis results have always been reported in units of µCi/L and that a denominator of 1.5 L was never used. Liquid scintillation counting (LSC) was implemented in 1958 and has improved since then, but the 1 µCi/L was retained as a reporting level for some time. The SRS TBM states that the reporting level was eventually lowered, first to 0.5 µCi/L and later to 0.1µCi/L, which is the level in use today, but no dates for these changes are provided. A review of the Savannah River Site Internal Dosimetry Technical Basis Manual, Revision 1, published in 1992

indicates that the reporting level of 0.1 µCi/L was in use at the time. The current MDA for LSC is given as 0.02 µCi/L.

Currently the routine sampling frequency for HTO is monthly and SRS assumes the annual minimum detectable dose (MDD) is 12 times the MDD for the monthly frequency, which gives an annual MDD of 10 mrem. For a point of reference, the SRS TBM notes that no doses over 100 mrem/year have been assigned to any SRS tritium workers in the last 10 years.

With regard to missed dose in the past, an estimate may be made using the current SRS dose rate factor and assuming that the HTO concentration in urine is equal to the MDA of 1 µCi/L for 365 days per year. The assumption of a steady state concentration in the urine is consistent chronic exposure. Because SRS notes that chronic intakes prior to 1980 were evaluated using only results greater than 5 µCi/L, the annual missed dose for the earliest years is assumed to equal 1.946e-04 rem/day per µCi/L multiplied by 5 µCi/L multiplied by 365 days/year, which is 3.55e-1 rem. The annual missed dose assumed from 1980 through 1991 is based on 1 µCi/L and is 7.1e-2 rem per year. For later years, a steady state urine concentration of 0.1 µCi/L, equal to the minimum reporting level, is used. The annual missed dose for later years is then equal to 7.1e-3 rem.

Assignment of H-3 Dose

For the purposes of this compensation program, a tritium dose was assigned for years when an individual was not monitored for tritium exposure or when their reported tritium dose was less than the calculated annual missed dose. Because it is not clear what tritium urine concentrations were used in dose calculations for what period of time, it was assumed that the minimum tritium urine concentration used in dose calculations was 5 µCi/L prior to 1984, 1 µCi/L from 1984 through 1991 and 0.1 µCi/L thereafter. Based on the above assumptions, the H-3 missed dose assignment is 0.355 rem for the years 1953 through 1983, 0.071 rem for the years 1984 through 1991, and 0.0071 rem for all subsequent years. For years in which monitoring was performed and a dose greater than the missed dose was reported, no additional dose is included. For years where the assigned dose is less than the missed dose, the missed dose is assigned. Because of the ubiquitous distribution of tritium in the body, all organ doses will be identical to the annual dose.

Table 13: Annual missed dose to all organs due to H-3

Years	Annual dose (rem)
1953 - 1983	0.355
1984 - 1991	0.071
1992 - present	0.0071

For IREP, all internal doses are assumed to be “chronic” exposure rates. The radiation type for tritium is “electron < 15 keV”. The dose distribution type is “constant”. The annual dose is put in the parameter 1 column. Parameters 2 and 3 are not used in the IREP calculations.

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Attachment A:

Spreadsheets and IMBA Input and Output Files Used In Support of the Hypothetical Intake Calculations

Note: Attachment A is supplied as a folder, “ORAUT-OTIB-0001 Rev 00 Attachment A,” on CD only.