COMPARISON OF EPA, NRC AND IAEA RECYCLE ASSESSMENTS

INTRODUCTION

EPA, NRC and the International Atomic Energy Agency (IAEA) have each performed analyses of the maximum doses that exposed individuals might reasonably be expected to experience as a result of the unrestricted clearance of scrap metals from nuclear facilities. The EPA analyses were initially reported in the "Technical Support Document—Potential Recycling of Scrap Metal from Nuclear Facilities, Part I: Radiological Assessment of Exposed Individuals" (TSD) of September, 1999. The NRC reported its findings in NUREG-1640: "Radiological Assessments for Clearance of Equipment and Materials from Nuclear Facilities," Draft Report for Comment, March 1999. The IAEA analyses were reported in an unpublished document: "DRAFT Annex to Safety Guide: Clearance Levels for Solid Materials," January, 2000.

EPA'S UPDATED ASSESSMENTS

A comparison of the EPA assessments with those of IAEA and foreign governments showed differences due to the use of different sets of dose conversion factors (DCFs) for internal exposure. The TSD analyses utilized the DCFs in Federal Guidance Report No. 11, which is consistent with ICRP Publication No. 30 of the International Commission on Radiation Protection (ICRP 30). The analyses performed by the European Commission (EC), IAEA, several European countries and Japan use the DCFs in the Basic Safety Standards (BSS), adopted by The Council of the European Union and published in EURATOM 1996. EPA has recalculated the normalized doses from the TSD scenarios, using the DCFs in the Basic Safety Standards for workers, which are based on ICRP 68.[1]

The TSD analyses showed that the maximum exposure scenario for three radionuclides—C-14, I-129 and Np-237—was the landfill disposal of aluminum dross. Following a recent re-evaluation of the aluminum dross disposal scenario, described in Appendix A of this report, the normalized doses from these three nuclides were revised—the disposal of aluminum dross is no longer the critical pathway for these nuclides. The new maximum exposure scenario for two of these nuclides—C-14 and I-129—is exposure of a nearby resident to airborne effluent emissions from an electric arc furnace facility recycling scrap steel. This scenario was

[1] EPA took an exception to the IAEA worker inhalation pathway analysis, which assumes that the inhalable particles have an AMAD of 5 μm. The size distribution of inhalable particles produced during the handling of cleared scrap metal is typically skewed towards the low end: the median AMAD diameter is closer to 1 μm than 5 μm. In order to perform a conservative assessment, we assumed either a 1 μm or 5 μm AMAD, depending on which had the higher DCF, for each lung clearance class for each nuclide.

also re-evaluated, as discussed in Appendix B, resulting in lower normalized doses from these two nuclides.

REVISION OF NRC DOSE FACTORS

In harmony with EPA's reanalysis, the NRC's critical group dose factors for the 40 radionuclides addressed in the TSD have also been revised. First, three of the scenarios in draft NUREG-1640 have been re-evaluated, as described in Appendix C. The dose factors of five radionuclides—Zn-65, Tc-99, Cs-134 and 137, and Np-237—for which these scenarios constituted potentially critical pathways were recalculated and were found to be lower than originally reported. As a result, the individuals exposed to these nuclides in other scenarios in NUREG-1640 emerge as the critical groups.

Next, the NRC dose factors were adjusted to reflect the ICRP 68 DCF's, assuming that the changes would be proportional to the changes to the EPA's normalized doses for the radionuclides in question. Each NRC dose factor was multiplied by the ratio DB:DF, where DB is the EPA normalized dose using BSS DCFs, while DF is the normalized dose from the same nuclide, using FGR 11 DCFs. The results of this adjustment are shown in Table 1.

Comparison of EPA Normalized Doses with NRC Dose Factors

Table 1 also presents a comparison of the revised EPA normalized doses to the RME individual with the NRC's revised dose factors for the average member of the critical group. As shown by the ratios of the two sets of values, which are listed in the last column, the values for all 40 radionuclides agree within about a factor of three. This constitutes substantial agreement, given the uncertainties in the analyses.

Table 1. Comparison of EPA and NRC Doses (μSv/a per Bq/g)

Nuclide	EPA dose		NRC Dose Factor		EPA ÷ NRC
	FGR 11	ICRP 68	FGR 11	ICRP 68	
C-14	5.78e-02	5.95e-02	1.6e-02	1.7e-02	3.6e+00
Mn-54	2.73e+01	2.73e+01	8.5e+01	8.5e+01	3.2e-01
Fe-55	9.70e-04	1.12e-03	1.0e-03	1.2e-03	9.7e-01
Co-60	1.26e+02	1.26e+02	2.5e+02	2.5e+02	5.0e-01
Ni-59	6.92e-04	4.65e-04	4.5e-04	3.0e-04	1.5e+00
Ni-63	1.67e-03	1.09e-03	1.2e-03	7.9e-04	1.4e+00
Zn-65	1.93e+01	2.02e+01	6.0e+01	6.3e+01	3.2e-01
Sr-90+D	5.68e+00	4.22e+00	1.0e+01	7.6e+00	5.5e-01
Nb-94	6.35e+01	6.32e+01	1.6e+02	1.6e+02	3.9e-01
Mo-93	1.16e-02	1.16e-02	1.4e-02	1.4e-02	8.3e-01
Tc-99	2.81e-03	5.05e-03	3.5e-03	6.3e-03	8.0e-01
Ru-106+D	6.97e+00	6.97e+00	2.1e+01	2.1e+01	3.4e-01
Ag-110m+D	8.51e+01	8.51e+01	2.8e+02	2.8e+02	3.0e-01
Sb-125+D	1.68e+01	1.68e+01	4.1e+01	4.1e+01	4.2e-01
I-129	5.24e+01	7.73e+01	6.1e+01	9.0e+01	8.6e-01
Cs-134	4.97e+01	5.19e+01	1.6e+02	1.7e+02	3.1e-01
Cs-137+D	1.79e+01	1.88e+01	6.2e+01	6.5e+01	2.9e-01
Ce-144+D	2.32e+00	2.25e+00	3.3e+00	3.2e+00	7.0e-01
Pm-147	1.69e-02	9.57e-03	1.1e-02	6.4e-03	1.5e+00
Eu-152	4.59e+01	4.59e+01	1.1e+02	1.1e+02	4.1e-01
Pb-210+D	2.95e+02	1.51e+02	1.5e+02	7.9e+01	1.9e+00
Ra-226+D	8.11e+01	8.11e+01	1.7e+02	1.7e+02	4.7e-01
Ra-228+D	4.65e+01	5.22e+01	8.4e+01	9.4e+01	5.5e-01
Ac-227+D	9.11e+02	3.27e+02	3.4e+02	1.2e+02	2.7e+00
Th-228+D	1.79e+02	1.18e+02	1.3e+02	8.4e+01	1.4e+00
Th-229+D	5.76e+02	1.11e+02	4.3e+02	8.4e+01	1.3e+00
Th-230	8.46e+01	2.02e+01	6.5e+01	1.5e+01	1.3e+00
Th-232	3.73e+02	3.00e+01	2.9e+02	2.3e+01	1.3e+00
Pa-231	3.14e+02	6.62e+01	2.3e+02	4.9e+01	1.4e+00
U-234	4.19e+01	1.01e+01	3.3e+01	7.9e+00	1.3e+00
U-235+D	4.38e+01	1.41e+01	3.2e+01	1.0e+01	1.4e+00
U-238+D	3.86e+01	9.70e+00	2.9e+01	7.3e+00	1.3e+00
Np-237+D	1.95e+02	3.32e+01	1.4e+02	2.4e+01	1.4e+00
Pu-238	9.11e+01	2.17e+01	7.1e+01	1.7e+01	1.3e+00
Pu-239	9.76e+01	2.37e+01	7.6e+01	1.8e+01	1.3e+00
Pu-240	9.73e+01	2.37e+01	7.6e+01	1.8e+01	1.3e+00
Pu-241+D	1.57e+00	4.30e-01	1.2e+00	3.3e-01	1.3e+00
Pu-242	9.27e+01	2.22e+01	7.3e+01	1.7e+01	1.3e+00
Am-241	1.55e+02	4.86e+01	1.1e+02	3.6e+01	1.4e+00
Cm-244	8.62e+01	3.11e+01	6.3e+01	2.3e+01	1.4e+00

REVIEW OF IAEA CLEARANCE LEVELS

Draft IAEA clearance levels were prepared in the course of a meeting of three consultants, drawn from Germany, the United Kingdom and the U.S. NRC, at IAEA headquarters in Vienna in November, 1999. These clearance levels received a critical review during an IAEA Technical Committee Meeting held in Vienna February 14-18, 2000, which was attended by representatives and technical experts of IAEA, EPA and NRC, EC, other European nations and Japan. This review included a detailed re-examination of the scenarios and assumptions for the assessment of the clearance of metals by a working group comprising one representative each from the U.S. EPA, the United Kingdom and Japan. The metals working group recommended that the enveloping scenario "ING-B"—ingestion of cleared material by a one-to-two-year-old child—be modified. While the ingestion of 100 g/a of soil by a young child is plausible, the assumption that this material has the same level of contamination as cleared metal is not. It was suggested that the concentration be reduced by at least factor of 10 to 100, with the higher concentration applied to the isotopes of elements that partition to the slag, where they would be reconcentrated, while the lower applies to the nuclides that remain in the metal. To envelop the consumption of food and water contaminated by the cleared material, an additional ingestion pathway—"ING C"—was recommended. This scenario assumes the consumption of 100 kg/a of food by the child. The dilution factor (radionuclide concentration in food ÷ radionuclide concentration in cleared metal) would depend on the root uptake factor (B_{iv}) and on whether the nuclide partitioned primarily to the slag or to the metal. The recommended dilution factor is

$$f_d = f_1 f_2$$

f_d = dilution factor of radionuclides in food

f_1 = 10^{-3} ($B_{iv} > 0.1$)
= 10^{-4} ($(B_{iv} \leq 0.1$)

f_2 = 1 (if element partitions to slag)
= 0.1 (if element partitions to metal)

A table showing the recalculation of the IAEA's draft normalized doses and the revised clearance levels is found in Appendix D.

Comparison of EPA Normalized Doses with Revised IAEA Clearance Levels

Table 2 presents the results of a comparison of the revised EPA doses from the 40 nuclides assessed in the TSD to IAEA's clearance values. The last two columns in Table 2 list IAEA's

revised clearance values and the resulting doses to the RME individual, using the TSD normalized doses and assuming (as IAEA did) that the average specific activities in all cleared metals were equal to the IAEA values. (It should be noted that this assumption is made to be consistent with IAEA's assumption. It does not account for the range of specific activities of the various radionuclides which are likely to be found in cleared materials. In actual practice, the average concentrations of radioactive contaminants in cleared materials are well below the clearance levels.) The values in the last column are calculated by multiplying the normalized dose in μSv/y per Bq/g, listed in column 4, by the clearance value in Bq/g in column 5.

The IAEA derived its clearance levels on the basis of the limiting pathways in a small set of exposures scenarios. The specific activity that corresponded to an annual dose of 10 μSv was determined for each nuclide. This calculated activity was rounded to the nearest power of ten, using a logarithmic rather than an arithmetic approach—i.e., if an activity x fell in the range $3 \times 10^{n-1} < x \leq 3 \times 10^{n}$, the corresponding clearance level was assigned the value 10^{n}. In keeping with that philosophy, if the calculated doses are within a factor of three of the target dose of 10 μSv, the IAEA values are considered to be in substantial agreement with EPA's analyses.

The last column in Table 1 shows that the doses from 26 of the 40 radionuclides included in the TSD lie between 3 and 30 μSv/y—the IAEA clearance values for these nuclides are therefore consistent with the revised EPA analyses. The doses from the two nuclides that exceed the 30-μSv criterion are listed in bold type. The doses from 12 other nuclides are less than 3 μSv/y—i.e., the limits on these nuclides are more restrictive than required, according to EPA's analyses. These values are listed in an open outline font. Only three doses differ by more than a factor of ten from the target dose of 10 μSv/y—these values are underlined.

Table 2. Comparison of IAEA Clearance Levels with Revised EPA Normalized Doses

Nuclide	Maximum Scenario	Metal	EPA Dose (μSv/a per Bq/g)	IAEA	
				Clearance Bq/g	EPA Dose (μSv/a)
C-14	Airborne effluents	steel	5.95e-02	100	6
Mn-54	Lathe operator	steel	2.73e+01	1	27
Fe-55	Slag handler	Cu	1.12e-03	1000	1.1
Co-60	Hull-plate on ship	steel	1.26e+02	0.1	13
Ni-59	Scrap cutter	steel	4.65e-04	1000	0.46
Ni-63	Scrap cutter	steel	1.09e-03	1000	1.1
Zn-65	EAF dust processor	steel	2.02e+01	1	20
Sr-90+D	Slag leachate	steel	4.22e+00	1	4.2
Nb-94	Slag pile worker	steel	6.32e+01	0.1	6.3
Mo-93	Hull-plate on ship	steel	1.16e-02	100	1.2
Tc-99	Slag handler	Cu	5.05e-03	100	0.51
Ru-106+D	Lathe operator	steel	6.97e+00	1	7.0
Ag-110m+D	Lathe operator	steel	8.51e+01	0.1	8.5
Sb-125+D	Hull-plate on ship	steel	1.68e+01	1	17
I-129	Airborne effluents	steel	7.73e+01	1	**77**
Cs-134	EAF dust processor	steel	5.19e+01	0.1	5.2
Cs-137+D	EAF dust processor	steel	1.88e+01	1	19
Ce-144+D	Slag pile worker	steel	2.25e+00	10	23
Pm-147	Slag pile worker	steel	9.57e-03	1000	1.0
Eu-152	Slag pile worker	steel	4.59e+01	0.1	4.6
Pb-210+D	EAF furnace operator	steel	1.51e+02	0.1	15.1
Ra-226+D	Slag pile worker	steel	8.11e+01	0.1	8.1
Ra-228+D	Slag pile worker	steel	5.22e+01	0.1	5.2
Ac-227+D	Scrap cutter	steel	3.27e+02	0.01	3.3
Th-228+D	Slag pile worker	steel	1.18e+02	0.1	12
Th-229+D	Slag pile worker	steel	1.11e+02	0.1	11
Th-230	Scrap cutter	steel	2.02e+01	0.1	2.0
Th-232	Slag pile worker	steel	3.00e+01	0.1	3.0
Pa-231	Scrap cutter	steel	6.62e+01	0.01	0.66
U-234	Slag pile worker	steel	1.01e+01	1	10
U-235+D	Slag pile worker	steel	1.41e+01	1	14
U-238+D	Slag pile worker	steel	9.70e+00	1	9.7
Np-237+D	Slag pile worker	steel	3.32e+01	1	**33**
Pu-238	Scrap cutter	steel	2.17e+01	0.1	2.2
Pu-239	Scrap cutter	steel	2.37e+01	0.1	2.4
Pu-240	Scrap cutter	steel	2.37e+01	0.1	2.4
Pu-241+D	Scrap cutter	steel	4.30e-01	10	4.3
Pu-242	Scrap cutter	steel	2.22e+01	0.1	2.2
Am-241	Slag pile worker	steel	4.86e+01	0.1	4.9
Cm-244	Slag pile worker	steel	3.11e+01	0.1	3.1

Appendix A

REANALYSIS OF ALUMINUM DROSS BURIAL SCENARIO

Last fall, EPA completed its "Technical Support Document: Potential Recycling of Scrap Metal from Nuclear Facilities, Part I: Radiological Assessment of Exposed Individuals." One scenario in the TSD analyses, the burial of aluminum dross in a dedicated landfill, resulted in the maximum doses to the RME individual from three of the radionuclides addressed in the study. In the absence of detailed information on the waste disposal practices of the secondary aluminum industry, the analysis employed a bounding scenario: dross generated during the six-year period that a single facility would be smelting residually contaminated scrap would be buried in a landfill which contains no other wastes, and which is closed at the end of this period. This led to the maximum radiological impacts from the subsequent failure of the cap covering the landfill.

We have recently obtained additional information from Wabash Alloys, whose smelter in Dickson, Tennessee was the reference facility for the aluminum recycling analysis. Until recently, as reported in Appendix B of the TSD, the Dickson facility sent its dross to the main Wabash Alloys plant in Wabash, Indiana, for recovery of metallic aluminum contained in the dross. Wabash Alloys informed us that the Indiana plant also processes the dross generated at its own facility and receives dross from several other Wabash smelters. The Dickson dross thus represented about 13 percent of the total dross processed by that facility. The wastes that are the residue from the aluminum recovery process are buried in an on-site landfill, which is currently licensed to operate for about 25 years. However, Wabash may request an extension of this operating life.

Consequently, under this revised scenario, the Dickson wastes generated during the time that that facility is postulated to process the aluminum scrap from the Paducah Gaseous Diffusion Plant would undergo a further dilution, resulting in an additional dilution factor of about 0.03 ($0.13 \times 6\ y \div 25\ y = 0.031$). Rather than performing a detailed analysis using these parameters, we make the following bounding argument. As can be deduced from the material balance presented on page 8-1 of the TSD, halide salts, presumed to be highly soluble, constitute 40 % of the dross. Once the metallic aluminum is recovered, these salts would constitute 44% of the wastes. By multiplying 0.037, the scrap dilution factor at the Dickson facility, by the concentration factor in the dross—about 7.1 for those radionuclides that strongly partition to the dross—and by the dilution factor of Dickson dross, we find that mass activity concentration of these radionuclides in the buried wastes to be about 0.008 pCi/g, normalized to 1 pCi/g in the residually contaminated scrap ($0.031 \times 0.037 \times 7.1 = 8.2 \times 10^{-3}$).

To calculate the maximum radionuclide concentrations in the drinking water, we note that, according to EPA drinking water standards, water containing more than 10,000 ppm of total dissolved solids (TDS) is not considered potable. Consequently, a person drinking two liters of water per day would not consume more than 7,300 grams of TDS per year (2 L × 365 d × 10 g/L). Assuming that all the TDS came from the halide salts and that these salts leached from the waste in the same proportion as the most mobile radioactive contaminants, the RME individuals could not consume more than 136 pCi/y ($7300 \times 8.2 \times 10^{-3} \div 0.44 = 136$), normalized to 1 pCi/g in the cleared scrap. Taking I-129 as an example, this would lead to a dose of 0.0375 mrem/y per pCi/g in scrap, which is less than one tenth the normalized dose from the airborne effluents from the melt-refining of carbon steel scrap. Similar consideration apply to the two other nuclides for which the original buried dross scenario was the limiting pathway. The recalculated normalized doses from all three radionuclides are shown in the table below.

Nuclide	Maximum Exposure Scenario		Revised Buried Dross Scenario		
	Description	Dose (mrem/y per pCi/g in scrap)	Intake (pCi/y per pCi/g in scrap)	DCF—ingestion (mrem/pCi)	Dose (mrem/y per pCi/g in scrap)
C-14	Airborne effluents	4.31e-04	136	2.09e-06	2.84e-04
I-129	Airborne effluents	3.96e-01	136	2.76e-04	3.75e-02
Np-237+D	Slag pile worker	7.23e-01	136	4.44e-03	6.04e-01

The current practice at the Wabash Alloys Dickson, Tennessee plant is to send the dross to the company's Benton, Arkansas plant, where it is processed together with the dross generated at that facility as well as dross from other Wabash plants in the same geographical region. The resulting wastes are then conveyed to a commercial waste company, such as BFI or Waste Management, for landfill burial. These wastes would be mixed with other commercial waste streams, minimizing the potential for radioactive contamination of groundwater supplies.

Appendix B

RE-ANALYSIS OF EAF EFFLUENT EMISSIONS SCENARIO

The impact of fugitive airborne emissions from the furnace on nearby residents was modeled by means of EPA's Clean Air Assessment Package, using the computer code CAP88-PC.

To calculate the effects of airborne effluent emissions on the RME individual, the map showing locations of EAF facilities and commercial nuclear power plants and shutdown dates was used to identify seven EAF facilities which could receive the D&D scrap from two or more nuclear plants in a single year. The meteorological data accompanying CAP88-PC were used to locate the meteorological station nearest to each of these seven facilities.[2] CAP-88 analyses for releases of C-14 and I-129 were performed using each of the seven meteorological data sets, as shown in Table B-1—the highest individual doses from each of the two nuclides from the seven runs were used in the TSD analysis. The RME individual was assumed to reside 1 km from the emission point—default CAP-88 values for the rural residential scenario were used for all other parameters.

Table B-1. Locations and Results of CAP-88 Analyses

Location		Nuclide							
		I-129				C-14			
CAP88 ID	State	Dose mrem/y	Cancer Risk	X/Q sec/m^3	Rank	Dose mrem/y	Cancer Risk	X/Q sec/m^3	Rank
PVD0560	Rhode Island	3.90e-01	2.23e-06	7.25e-07	7	3.01e-04	7.36e-09	2.65e-06	7
HAR0631	Pennsylvania	6.94e-01	4.07e-06	1.58e-06	2	8.66e-04	2.11e-08	9.37e-06	1
ILG1058	Delaware	4.48e-01	2.59e-06	9.01e-07	4	4.81e-04	1.18e-08	4.67e-06	6
TYS1328	Indiana	4.86e-01	2.82e-06	1.02e-06	3	8.28e-04	2.02e-08	8.36e-06	2
MLI0269	Illinois	3.97e-01	2.28e-06	7.73e-07	6	5.72e-04	1.40e-08	5.64e-06	3
LAX0304	California	7.91e-01	4.65e-06	1.84e-06	1	4.86e-04	1.19e-08	4.65e-06	4
ORD0452	Illinois	4.16e-01	2.39e-06	8.12e-07	5	4.84e-04	1.18e-08	4.55e-06	5

The fractions of foods produced on the individual's own land was based on a 1965-66 U.S. Department of Agriculture (USDA) survey. More recent information obtained from the USDA indicates that there are very few farms which produce the variety of food needed to supply a family. The TSD assumptions thus led to a highly conservative analysis: the RME was assumed to live in the worst possible location of the seven localities analyzed *and* to raise most of his own

[2] Although the NRC's schedule of termination of operating licenses has been revised since these seven facilities were selected, their locations, in different geographical regions, constitute a representative range of meteorological conditions.

food. The revised analyses of the two nuclides assume that the individual lives in the median location—rank 4 in Table B-1—rather than in the worst location—rank 1.

Appendix C

RE-EVALUATION OF NRC CRITICAL GROUP DOSE FACTORS

NRC has re-evaluated three of the potentially critical scenarios described in draft NUREG-1640.

Transport of Refinery Dust

The model of the truck used to transport dust generated during melt-refining at either an electric arc furnace (EAF), used to refine scrap steel, or a copper converter furnace was a simple box truck—the same as that used to model the scrap transport scenario. The earlier draft report with the same title, prepared by SAIC for NRC and dated September 23, 1997, modeled a specialized dust transporter for this scenario. Recalculation of the doses to the truck driver from three of the strong γ-emitting nuclides—Zn-65, and Cs-134 and 137—that concentrate in the dust, using the geometry factors (i.e., dose rates from external exposure) from the earlier report, show about a seven-fold reduction of the dose factors from this scenario. The revised dose factors are shown in Table D-1.

Processing of EAF Dust for Disposal

The analysis of the EAF dust processing scenario in NUREG-1640 assumed year-long exposure to dust generated by a single EAF facility. The EAF described in NUREG-1640 would generate approximately 6,800 Mg of dust per year. A typical metal recovery facility used to process EAF dust has an annual capacity of 68,000 Mg (see Appendix G to the TSD). Consequently, it is unlikely that a single worker would be processing dust from a single EAF mill for an entire year. It is more likely that this dust would undergo a dilution of about 1:10 with dust from other EAF mills which are not recycling metal from nuclear facilities during that time. (Appendix G of the TSD, which has a detailed analysis of the flow of contaminated dust to processing facilities, draws a similar conclusion.) Consequently, the dose factors for this scenario were decreased by a factor of 10.

Table C-1. Recalculation of Dose Factors (μSv/y per Bq/g) for 3 Nuclides in Two Scenarios

Nuclides	NUREG-1640			Recalculated		Scenario	Dose	
	Scenario	GF (mrem/hr)	Dose	GF (mrem/hr)	Dose		Old	New
Zn-65	Transport of EAF dust	2.87e-4	210	4.19e-5	30.7	Processing EAF dust for disposal	69	6.9
Cs-134		7.63e-4	660	1.12e-4	96.9		310	31
Cs-137+D		2.93e-4	260	4.27e-5	37.9		130	13

Storage of Slag at the Refinery

The analysis of the leaching of radionuclides from a slag pile at the refinery modeled the leach rate using the equations for the leaching from soil. Both experimental and empirical data show the leach rate from slags to be considerably slower (see Appendix I of the TSD). This scenario is therefore not used to determine critical group dose factors.

Appendix D

REVISION OF IAEA CLEARANCE LEVELS

The table below shows the draft IAEA normalized doses for the 40 radionuclides addressed in the TSD, by enveloping scenario. The scenario labeled ING-B' is a modification of ING-B, with a dilution factor of 0.1 for radionuclides that primarily partition to slag and a factor of 0.01 for those that partition to the metal. The scenario labeled ING-C is the child's ingestion of contaminated food, as described on page 4 of the present report.

Table D-1. Revision of IAEA Clearance Levels

Nuclide	Normalized Dose (μSv/a/Bq/g)											Limiting Pathway	CL (Bq/g)	
	External			Inhalation		Ingestion				Skin	Maximum			
	M-EXT-A	M-EXT-B	M-EXT-C	M-INH-A	M-INH-B	M-ING-A	M-ING-B	M-ING-B'	M-ING-C	M-SKIN			exact	rounded
C-14	0.0e+00	0.0e+00	0.0e+00	1.3e-03	8.7e-05	1.2e-02	1.6e-01	1.6e-02	1.6e-01	2.4e-02	1.6e-01	M-ING-C	6.3e+01	100
Mn-54	2.7e+01	3.2e+01	2.3e+01	2.6e-03	7.9e-05	1.4e-02	2.1e-01	2.1e-02	2.1e-01	1.6e-03	3.2e+01	M-EXT-B	3.1e-01	1
Fe-55	0.0e+00	0.0e+00	0.0e+00	7.1e-04	2.0e-05	6.6e-03	2.1e-01	2.1e-03	2.1e-03	4.3e-04	6.6e-03	M-ING-A	1.5e+03	1000
Co-60	8.4e+01	1.0e+02	1.2e+02	1.5e-02	4.4e-04	6.8e-02	2.5e+00	2.5e-02	2.5e-02	5.3e-02	1.2e+02	M-EXT-C	8.3e-02	0.1
Ni-59	0.0e+00	0.0e+00	0.0e+00	2.0e-04	8.3e-06	1.3e-03	3.4e-02	3.4e-04	3.4e-03	4.0e-06	3.4e-03	M-ING-C	2.9e+03	1000
Ni-63	0.0e+00	0.0e+00	0.0e+00	6.7e-04	2.6e-05	3.0e-03	8.4e-02	8.4e-04	8.4e-03	4.9e-04	8.4e-03	M-ING-C	1.2e+03	1000
Zn-65	1.9e+01	2.3e+01	1.4e+01	6.0e-03	8.0e-05	7.8e-02	9.9e-01	9.9e-03	9.9e-02	2.4e-03	2.3e+01	M-EXT-B	4.3e-01	1
Sr-90	0.0e+00	0.0e+00	0.0e+00	6.8e-02	1.5e-03	6.1e-01	9.2e+00	9.2e-01	9.2e+00	1.4e-01	9.2e+00	M-ING-C	1.1e+00	1
Nb-94	5.0e+01	6.0e+01	7.7e+01	1.6e-02	4.5e-04	3.4e-02	9.7e-01	9.7e-02	9.7e-02	6.1e-02	7.7e+01	M-EXT-C	1.3e-01	0.1
Mo-93	7.4e-07	0.0e+00	4.9e-25	3.2e-03	9.5e-05	5.4e-02	7.8e-01	7.8e-03	7.8e-03	1.7e-05	5.4e-02	M-ING-A	1.9e+02	100
Tc-99	7.8e-06	1.9e-07	2.5e-06	6.9e-03	1.8e-04	1.6e-02	4.8e-01	4.8e-03	4.8e-02	4.3e-02	4.8e-02	M-ING-C	2.1e+02	100
Ru-106	4.0e+00	4.8e+00	3.5e+00	3.7e-02	1.5e-03	1.4e-01	3.5e+00	3.5e-02	3.5e-02	7.7e-02	4.8e+00	M-EXT-B	2.1e+00	1
Ag-110m	8.7e+01	1.1e+02	6.6e+01	1.6e-02	4.8e-04	5.6e-02	8.8e-01	8.8e-03	8.8e-03	2.6e-02	1.1e+02	M-EXT-B	9.1e-02	0.1
Sb-125	1.2e+01	1.4e+01	1.4e+01	8.3e-03	2.4e-04	2.5e-02	6.5e-01	6.5e-03	6.5e-03	5.6e-02	1.4e+01	M-EXT-C	7.1e-01	1
I-129	1.3e-02	0.0e+00	1.6e-05	1.1e-01	7.6e-04	2.2e+00	2.2e+01	2.2e+00	2.2e+01	1.8e-02	2.2e+01	M-ING-C	4.5e-01	1
Cs-134	4.8e+01	5.8e+01	5.7e+01	2.1e-02	1.2e-04	3.8e-01	1.4e+00	1.4e-01	1.4e-01	5.2e-02	5.8e+01	M-EXT-B	1.7e-01	0.1
Cs-137	1.8e+01	2.2e+01	2.7e+01	1.4e-02	9.3e-05	2.6e-01	1.2e+00	1.2e-01	1.2e-01	6.9e-02	2.7e+01	M-EXT-C	3.7e-01	1
Ce-144	1.3e+00	1.3e+00	9.3e-01	5.0e-02	2.0e-03	1.1e-01	2.6e+00	2.6e-01	2.6e-01	1.2e-01	1.3e+00	M-EXT-B	7.7e+00	10
Pm-147	6.6e-05	1.5e-05	2.9e-05	7.6e-03	2.2e-04	5.2e-03	1.7e-01	1.7e-02	1.7e-02	3.4e-02	1.7e-02	M-ING-C	5.9e+02	1000
Eu-152	3.6e+01	4.2e+01	5.3e+01	5.8e-02	1.2e-03	2.8e-02	7.2e-01	7.2e-02	7.2e-02	4.5e-02	5.3e+01	M-EXT-C	1.9e-01	0.1
Pb-210	7.3e-03	3.2e-04	5.7e-04	6.9e+00	2.0e-01	1.8e+01	1.2e+03	1.2e+02	1.2e+02	7.1e-02	1.2e+02	M-ING-C	8.3e-02	0.1
Ra-226	3.3e+01	3.8e+01	4.9e+01	1.2e+01	3.5e-01	2.3e+01	1.2e+03	1.2e+02	1.2e+02	1.6e-01	1.2e+02	M-ING-C	8.3e-02	0.1
Ra-228	5.9e+01	6.8e+01	8.2e+01	4.7e+01	1.2e+00	1.5e+01	6.0e+02	6.0e+01	6.0e+01	8.7e-02	8.2e+01	M-EXT-C	1.2e-01	0.1
Ac-227	9.3e+00	8.5e+00	1.0e+01	3.5e+02	6.7e+00	2.4e+01	4.2e+02	4.2e+01	4.2e+01	1.8e-01	3.5e+02	M-INH-A	2.9e-02	0.01
Th-228	4.9e+01	5.6e+01	5.8e+01	7.4e+01	1.8e+00	2.8e+00	9.0e+01	9.0e+00	9.0e+00	1.8e-01	7.4e+01	M-INH-A	1.4e-01	0.1
Th-229	7.6e+00	7.0e+00	8.9e+00	1.3e+02	2.8e+00	1.2e+01	2.4e+02	2.4e+01	2.4e+01	2.3e-01	1.3e+02	M-INH-A	7.7e-02	0.1
Th-230	1.4e+00	1.6e+00	2.1e+00	1.6e+01	4.3e-01	5.0e+00	8.2e+01	8.2e+00	8.2e+00	2.9e-03	1.6e+01	M-INH-A	6.3e-01	0.1
Th-232	8.0e+01	9.2e+01	1.2e+02	1.0e+02	2.5e+00	2.1e+01	7.2e+02	7.2e+01	7.2e+01	5.1e-03	1.2e+02	M-EXT-C	8.3e-02	0.1
Pa-231	9.8e+00	8.9e+00	1.1e+01	5.3e+02	8.7e+00	3.7e+01	5.4e+02	5.4e+01	5.4e+01	4.1e-03	5.3e+02	M-INH-A	1.9e-02	0.01
U-234	1.3e-03	2.3e-04	5.3e-04	1.5e+01	3.5e-01	1.0e+00	1.3e+01	1.3e+00	1.3e+00	2.7e-04	1.5e+01	M-INH-A	6.7e-01	1

Table D-1 (continued)

Nuclide	Normalized Dose (μSv/a/Bq/g)											Limiting Pathway	CL (Bq/g)	
	External			Inhalation		Ingestion				Skin	Maximum			
	M-EXT-A	M-EXT-B	M-EXT-C	M-INH-A	M-INH-B	M-ING-A	M-ING-B	M-ING-B'	M-ING-C	M-SKIN			exact	rounded
U-235	3.5e+00	1.9e+00	2.6e+00	1.4e+01	3.3e-01	9.9e-01	1.4e+01	1.4e+00	1.4e+00	6.8e-02	1.4e+01	M-ING-B	7.1e-01	1
U-238	5.5e-01	5.5e-01	7.3e-01	1.2e+01	3.1e-01	9.5e-01	1.5e+01	1.5e+00	1.5e+00	2.1e-01	1.2e+01	M-INH-A	8.3e-01	1
Np-237	5.6e+00	4.8e+00	5.9e+00	3.2e+01	4.6e-01	2.2e+00	2.2e+01	2.2e+00	2.2e+00	9.4e-02	3.2e+01	M-INH-A	3.1e-01	1
Pu-238	1.7e-04	4.3e-08	1.2e-05	6.5e+01	8.2e-01	4.6e+00	4.0e+01	4.0e+00	4.0e+00	2.9e-03	6.5e+01	M-INH-A	1.5e-01	0.1
Pu-239	9.6e-04	1.7e-04	4.7e-04	6.9e+01	8.4e-01	5.0e+00	4.2e+01	4.2e+00	4.2e+00	3.9e-05	6.9e+01	M-INH-A	1.4e-01	0.1
Pu-240	1.7e-04	1.2e-10	1.2e-05	6.9e+01	8.4e-01	5.0e+00	4.2e+01	4.2e+00	4.2e+00	2.8e-06	6.9e+01	M-INH-A	1.4e-01	0.1
Pu-241	4.8e-03	6.4e-08	4.3e-04	3.0e+00	3.2e-02	2.1e-01	1.6e+00	1.6e-01	1.6e-01	4.3e-08	3.0e+00	M-INH-A	3.3e+00	10
Pu-242	1.7e-04	1.5e-10	1.4e-05	6.7e+01	8.0e-01	4.8e+00	4.0e+01	4.0e+00	4.0e+00	2.3e-06	6.7e+01	M-INH-A	1.5e-01	0.1
Am-241	1.6e-01	2.2e-06	1.5e-02	5.8e+01	7.7e-01	4.0e+00	3.7e+01	3.7e+00	3.7e+00	1.9e-03	5.8e+01	M-INH-A	1.7e-01	0.1
Cm-244	1.1e-04	1.0e-10	8.9e-06	3.7e+01	6.5e-01	2.4e+00	2.9e+01	2.9e+00	2.9e+00	5.9e-05	3.7e+01	M-INH-A	2.7e-01	0.1