

RADIOLOGICAL RISK ASSESSMENT OF THE TRANSPORT OF TANTALUM RAW MATERIALS



Prepared For:

TIC

**Tantalum-Niobium
International Study Center**

Prepared By:

SENES Consultants Limited

April 2007

RADIOLOGICAL RISK ASSESSMENT OF THE TRANSPORT OF TANTALUM RAW MATERIALS

Prepared for:

**Tantalum-Niobium International Study Center (T.I.C.)
40 Rue Washington
1050 Bruxelles
Belgium**

Prepared by:

**SENES Consultants Limited
121 Granton Drive, Unit 12
Richmond Hill, Ontario
L4B 3N4**

April 2007

Printed on Recycled Paper Containing Post-Consumer Fibre



EXECUTIVE SUMMARY

The Tantalum-Niobium International Study Center (T.I.C.) has commissioned a study of the transport of tantalum raw materials. Tantalum raw materials can be defined as encompassing the tantalite mineral concentrates [Fe, Mn (Ta, Nb)₂O₆] where the tantalum (Ta) content is greater than the niobium (Nb) content, as well as slag materials which are a by-product of smelting operations (e.g. tin smelting) and which contain varying levels of Ta. Due to the presence of natural uranium and thorium, tantalum raw materials are considered to be naturally occurring radioactive material (NORM).

The current IAEA exemption value (IAEA 2005a) for the transport of NORM (below which the requirements of the transport regulations do not apply) is 10 becquerels per gram (Bq/g, U-238 + Th-232 combined, decay products in radioactive equilibrium), provided such materials are not intended to be processed for the use of the naturally occurring radionuclides.

The NORM exemption values are 10 times the exemption values for other radioactive materials. The rationale for this factor of 10 is not explicit but rather appears to be an arbitrary consensus that balances radiological protection concerns and the impracticality and inconvenience of regulating large amounts of low activity NORM. A 2003 IAEA International Conference on the Safety of Transport of Radioactive Material suggested that “... *the full impact of and technical basis for the ‘factor of 10’ exemption be thoroughly researched.*” (IAEA 2005b).

The main objectives of this study were to determine the radiological characteristics of tantalum raw materials and to evaluate the potential radiological exposures associated with the transport of these materials during normal transport and in the event of an accidental spill. The study was carried out by SENES Consultants Limited (SENES), supported by Alfred H. Knight International Ltd. (AHK) for the physical and chemical analysis of the tantalum raw materials and, in certain instances, for carrying out gamma radiation surveys at T.I.C. member company sites.

The main conclusions from the study are as follows:

- Radioactive equilibrium in the uranium (U-238) and thorium (Th-232) decay series was found to be a reasonable assumption for tantalum raw materials for dose assessment purposes.
- A range of about a factor of 10 in radioactivity concentrations was measured in 67 shipments of tantalite and slag, with an average activity concentration (U-238 + Th-232 combined) of about 20 Bq/g for tantalite and about 25 Bq/g for slag. The majority (78%)

of tantalite shipments and 45% of the slag shipments had concentrations exceeding 10 Bq/g.

- Exposure scenarios that considered both duration and location of exposure were established for several types of transport workers and for members of the public. In addition, exposures to facility workers (who are not considered to be transport workers) during loading and unloading were assessed in this study to provide perspective on potential exposures from tantalum raw materials. Based on an evaluation of potential exposure pathways, exposure to gamma radiation was determined to be the only significant exposure pathway.
- Doses from exposure to spilled materials due to potential accidents were calculated and determined not to be a regulatory concern, as the resulting doses were less than 10 μ Sv/y.
- An assessment of potential dose rates around the transport containers was conducted using the range of measured radioactivity concentrations and modelling of the associated gamma radiation doses using the MicroShield model. The modelling approach overestimated the measured dose rates, primarily due to the assumption that the transport containers were always considered to be a full 1 tier or 1.5 tier load whereas in practice the loading pattern varied.
- Calculated doses to transport workers and members of the public based on the distribution of measured concentrations are shown in Table ES.1.

**TABLE ES.1
SUMMARY OF DOSES CALCULATED FOR NORMAL (NON-ACCIDENTAL)
TRANSPORT ACTIVITIES**

Exposure Scenario	Mean Dose ^{a, b} (mSv/y)	
	Slag	Tantalite
Transport Worker - Truck Driver	0.24	0.16
Transport Worker - Dockworker	0.032	0.02
Transport Worker - Seaman	0.0041	0.0026
Transport Worker - Trainman	0.019	0.012
Public - Living Adjacent to Road	0.00038	0.00024
Public - Living Adjacent to Rail	0.00017	0.0001

- a) Mean annual dose from shipments of tantalum raw materials analyzed in this study.
 b) For perspective, doses to facility workers (who are not considered transport workers) were 0.49 and 0.31 mSv/y from slag and tantalite, respectively.

- Doses to members of the public from the transport of these materials were found to be insignificant, that is, much less than 10 $\mu\text{Sv}/\text{y}$ (0.01 mSv/y).
- The calculated doses to transport workers were well within the internationally accepted dose limit of 1 mSv/y for non-radiation workers. If it were assumed that the tantalum raw materials considered in this study reliably represent the likely range of tantalum raw materials in general, then the expected (mean) dose to the most exposed group of transport workers would be about 0.24 mSv/y to truck drivers from the transport of slag (Table ES.1). Thus, there is considerable allowance for truck drivers who transport tantalum raw materials to transport other materials containing elevated levels of naturally occurring radioactivity without exceeding a cumulative annual dose of 1 mSv .

On the basis of the analyses of doses arising from the transport of tantalum raw materials described in this report, there is no apparent reason with regards to radiological dose for an exemption value as restrictive as the current value of 10 Bq/g for these materials. Even in the absence of an exemption value, no one would be expected to receive a dose above 1 mSv/y arising from the transport of tantalum raw materials.

Notwithstanding the low doses from the transport of tantalum raw materials, specific numerical exemption values are required for the practical, real-world application of the transport regulations. Moreover, the radioactivity levels of future shipments of tantalum raw materials may differ from the distribution of levels from which the conclusions of this study were derived. If it were conservatively assumed that the radioactivity levels of all future exempted shipments of tantalum raw materials were at a selected specific exemption value, the resultant annual doses due to exempted shipments under actual transport situations would be overestimated. The calculated doses to transport workers under this assumption are summarized in Table ES.2 for various potential exemption values.

**TABLE ES.2
SUMMARY OF ANNUAL DOSES (mSv/y) TO TRANSPORT WORKERS AT VARIOUS
POTENTIAL EXEMPTION VALUES**

Potential Exemption Value (Bq/g)	Material	Annual Dose (mSv/y) ^a			
		Truck Driver	Dock Worker	Seaman	Trainman
10	Tantalite	0.12	0.016	0.0021	0.0097
	Slag	0.12	0.016	0.0021	0.0098
30	Tantalite	0.31	0.040	0.0052	0.024
	Slag	0.37	0.049	0.0062	0.029
50	Tantalite	0.48	0.062	0.0080	0.038
	Slag	0.54	0.071	0.0090	0.043
70	Tantalite	0.65	0.084	0.011	0.051
	Slag	0.71	0.093	0.012	0.056
90	Tantalite	0.82	0.11	0.014	0.064
	Slag	0.89	0.12	0.015	0.070
100	Tantalite	0.91	0.12	0.015	0.071
	Slag	0.97	0.13	0.016	0.077

a) Assumes maximum load (1.5 tiers) of material and all loads at specific exemption value.

To account for the possibility of other transport-related exposures, an annual dose constraint of 0.3 mSv/y, for example, might be considered. (To allow for potential multiple exposures in the context of radioactive waste disposal, the ICRP (2007) recommends a 0.3 mSv/y dose constraint for members of the public for a single waste site.) Considering the conservatism in the dose calculations summarized in Table ES.2, an exemption value of 30 Bq/g (U-238 + Th-232) would result in doses that would be unlikely to exceed 0.3 mSv/y to the most exposed transport workers. Thus, on the basis of a 0.3 mSv/y reference dose, an exemption value of at least 30 Bq/g is considered appropriate for the transport of tantalum raw materials.

Irrespective of the exemption value selected, the radiological dose assessments described in this report should provide assurance to the tantalum industry and to its shippers that the doses arising from the transport of tantalum raw materials are low and well within international norms for both transport workers and members of the public.

ABBREVIATIONS

AHK	Alfred H. Knight International Ltd.
Al ₂ O ₃	Aluminum Trioxide
AMAD	Activity Median Aerodynamic Diameter
BSS	Basic Safety Standards
Bq	Becquerel
CRP	Coordinated Research Programme
DC	Dose Coefficient
Fe	Iron
Fe ₂ O ₃	Iron Trioxide
F - 2 Tier	Full Two Tier
f ₁	Fractional Absorption in the Gastrointestinal Tract
F – 1 Tier	Full One Tier
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
Inhalation Type F	Fast
Inhalation Type M	Moderate
Inhalation Type S	Slow
MeV	Megaelectron volt
Mn	Manganese
MnO	Manganese Oxide
NAA	Neutron Activation Analysis
Nb	Niobium
Nb ₂ O ₅	Niobium Pentoxide
NF – 1 Tier	Partially Full One Tier
NF – Trailer	Partially Full Trailer
NF – 2 Tier	Partially Full Two Tier
NORM	Naturally Occurring Radioactive Material
PIC	Pressurized Ion Chamber
Ppm	Parts Per Million
SENES	SENES Consultants Limited
SiO ₂	Silicon Dioxide
SnO ₂	Tin Dioxide
Sv	Sievert
Ta	Tantalum
Ta ₂ O ₅	Tantalum Pentoxide
Th-nat	Natural Thorium
ThO ₂	Thorium Dioxide
T.I.C.	Tantalum-Niobium International Study Center
TiO ₂	Titanium Dioxide
TRANSSC	Transport Safety Standards Committee (of the IAEA)
U-nat	Natural Uranium
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
U.S. EPA	United States Environmental Protection Agency
U ₃ O ₈	Triuranium Octaoxide (used to express uranium content or ore grade)
XRF	X-Ray Fluorescence
ZrO ₂	Zirconium Dioxide

Radiological Risk Assessment of the Transport of Tantalum Raw Materials

Ac-228	Actinium-228
Bi-210	Bismuth-210
Bi-212	Bismuth-212
Bi-214	Bismuth-214
Co-60	Cobalt-60
Cs-137	Cesium-137
Pa-234	Protoactinium-234
Pb-206	Lead-206
Pb-210	Lead-210
Pb-212	Lead-212
Pb-214	Lead-214
Po-210	Polonium-210
Po-212	Polonium-212
Po-214	Polonium-214
Po-216	Polonium-216
Po-218	Polonium-218
Ra-223	Radium-223
Ra-224	Radium-224
Ra-226	Radium-226
Ra-228	Radium-228
Rn-220	Radon-220
Rn-222	Radon-222
Th-227	Thorium-227
Th-228	Thorium-228
Th-230	Thorium-230
Th-232	Thorium-232
Th-234	Thorium-234
Tl-208	Thallium-208
U-234	Uranium-234
U-235	Uranium-235
U-238	Uranium-238

TABLE OF CONTENTS

	<u>Page No.</u>
EXECUTIVE SUMMARY	ES-1
ABBREVIATIONS	AC-1
1.0 INTRODUCTION	1-1
2.0 REGULATORY IMPLICATIONS OF RADIOACTIVITY ON TRANSPORT OF TANTALUM RAW MATERIALS	2-1
2.1 Introduction.....	2-1
2.2 Exemption Values.....	2-2
3.0 RADIOLOGICAL CHARACTERIZATION OF TANTALUM RAW MATERIALS	3-1
3.1 Data Collection	3-1
3.1.1 Data Requirements.....	3-1
3.1.2 Shipment Characteristics	3-2
3.1.3 Sample Collection and Gamma Radiation Survey.....	3-3
3.1.4 Laboratory Analyses of Concentrations.....	3-5
3.1.5 Modelled Exposure Rates	3-5
3.1.6 Data Completeness.....	3-6
3.2 Radioactivity Concentrations.....	3-6
3.2.1 Uranium and Thorium Content.....	3-6
3.2.2 Tantalum Content and Density	3-8
3.2.3 Other Radionuclide Concentrations.....	3-9
3.2.4 Discussion	3-9
3.3 Gamma Radiation Exposure Rates	3-10
3.3.1 Measured Gamma Radiation Rates.....	3-10
3.3.2 Modelled Gamma Radiation Rates.....	3-11
3.3.3 Comparison between Measured and Modelled Gamma Radiation Dose Rates	3-12
3.4 Prediction of Gamma Radiation Exposure Rates for Transport Scenarios.....	3-18
3.5 Summary.....	3-19
4.0 DOSES ARISING FROM NORMAL TRANSPORT ACTIVITIES.....	4-1
4.1 General Assumptions	4-1
4.2 Predicted Gamma Radiation Rate for Exposure Scenarios.....	4-3
4.2.1 Gamma Radiation Exposure Rates during Normal Transport	4-3
4.3 Annual Doses	4-3
4.4 Uncertainty.....	4-5
4.5 Potential Exemption Value for the Transport of Tantalum Raw Materials	4-6
4.5.1 Calculated Doses at Various Potential Exemption Values	4-6
4.5.2 Selection of an Exemption Value for the Transport of Tantalum Raw Materials.....	4-10

5.0	DOSES ARISING FROM TRANSPORT SPILLS AND ACCIDENTS	5-1
5.1	General Assumptions	5-1
5.2	Ingestion and Inhalation Dose	5-1
6.0	OBSERVATIONS AND CONCLUSIONS	6-1
7.0	REFERENCES	7-1
ANNEX A	GAMMA RADIATION SURVEY AND SAMPLE COLLECTION	
	PROTOCOL	A-1
A.1	Radiation Measurement Protocol.....	A-1
ANNEX B	RESULTS FROM RADIATION SURVEYS	B-1
B.1	Data Management	B-1
	B.1.1 Data Capture	B-1
	B.1.2 Relational Structure	B-1
	B.1.3 Data Completeness.....	B-1
	B.1.4 Data Processing.....	B-2
B.2	Description of Shipments.....	B-3
B.3	Concentration Data	B-6
	B.3.1 Sampling of Materials.....	B-6
	B.3.2 Laboratory Measurements	B-6
B.4	Dose Rate Measurements.....	B-9
	B.4.1 Instrumentation	B-9
	B.4.2 Measured Dose Rates.....	B-9
ANNEX C	MICROSHIELD MODELLING	C-1
C.1	Shielding	C-1
C.2	Source Density	C-2
C.3	Source Composition.....	C-2
C.4	Source Concentration.....	C-3
C.5	Source Dimensions	C-3
	C.5.1 Sea-land Container with One Full Tier.....	C-3
	C.5.2 Sea-land Container with One Full Tier and One Partial Tier	C-4
	C.5.3 Trailer with a Full Load	C-4
C.6	MicroShield Dose Factors.....	C-4
C.7	Sensitivity Analysis	C-11
	C.7.1 Shielding Thickness	C-11
	C.7.2 Source Density	C-12
	C.7.3 Source Composition.....	C-13
	C.7.4 Location of Load Inside Sea-land Container	C-14
	C.7.5 Contact Distance from Sea-land Container.....	C-14
C.8	References.....	C-15

ANNEX D	ANALYSIS OF MEASUREMENT DATA	D-1
D.1	Concentrations	D-1
D.1.1	Summary of Measurements Reported by the Laboratories.....	D-1
D.1.2	Data Quality Review and Interlaboratory Comparisons.....	D-4
D.1.3	Uranium and Thorium Content.....	D-7
D.1.4	Equilibrium Conditions.....	D-10
D.1.5	Tantalum Content and Density	D-12
D.2	Modelled and Measured Gamma Radiation Exposures.....	D-14
D.2.1	Predicted Dose Rates	D-14
D.2.2	Measured Gamma Radiation Dose Rates	D-16
D.2.3	Comparison between Modelled and Measured Dose Rates.....	D-17
D.2.4	Summary	D-18
D.3	References.....	D-19
ANNEX E	DOSE SCENARIO FACTORS AND CALCULATIONS.....	E-1
E.1	Annual Dose Factors.....	E-1
E.2	Probabilistic Simulation of Annual Dose	E-1

LIST OF TABLES

	<u>Page No.</u>
ES.1 Summary of Doses Calculated for Normal (Non-Accidental) Transport Activities.....	ES-2
ES.2 Summary of Annual Doses (mSv/y) to Transport Workers at Various Potential Exemption Values.....	ES-4
3.1 Material Types and Loading Configuration.....	3-3
3.2 Completeness of Data.....	3-6
3.3 Summary of U-238 and Th-232 Activity Concentrations (Bq/g).....	3-7
3.4 Summary of Measured Attributable Dose Rates (μ Sv/h) by Distance and Geometry Relative to the Container.....	3-10
3.5 Summary of Predicted Dose Rates (μ Sv/h) for Shipments with Measured Concentrations.....	3-12
3.6 Percent Difference (%) Between Mean Modelled and Measured Attributable Exposure Rates by Loading Configuration.....	3-15
3.7 Percent Difference (%) Between Mean Modelled and Measured Attributable Exposure Rates by Location.....	3-16
3.8 Percent Difference (%) Between Mean Modelled and Measured Exposure Rates by Location for the Risk Assessment.....	3-18
4.1 Parameter Values used for Normal Transport Activities.....	4-2
4.2 Annual Dose Factors for Normal Transport Activities.....	4-3
4.3 Summary of Annual Average Concentrations for Normal Transport Activities.....	4-4
4.4 Summary of Dose (mSv/y) for Normal Transport Activities.....	4-5
4.5 Summary of U-238 and Th-232 Annual Doses (mSv/y) for Potential Exemption Values.....	4-7
4.6 Summary of Combined Annual Doses (mSv/y) at Potential Exemption Values.....	4-9
4.7 Annual Doses (mSv/y) to Truck Driver from Shipments at Potential Exemption Values.....	4-10
5.1 Ingestion and Inhalation Dose Factors for Workers for Uranium and Thorium Series Radionuclides.....	5-3
5.2 Ingestion and Inhalation Doses to Workers from Clean-up of Spilled Materials.....	5-4
6.1 Summary of Doses Calculated for Normal (Non-Accidental) Transport Activities.....	6-3
6.2 Summary of Annual Doses (mSv/y) to Transport Workers at Various Potential Exemption Values.....	6-4

LIST OF TABLES (Cont'd)

	<u>Page No.</u>
A.1 Gamma Radiation Survey Location Description	A-1
B.1 Completeness of Data	B-2
B.2 Conversion Factors	B-2
B.3 Shipment Characteristics and Available Information	B-3
B.4 Material Types and Loading Configuration.....	B-5
B.5 Laboratory Measurements of Concentration	B-7
B.6 Meters Used for Gamma Radiation Surveys	B-9
B.7 Baseline Gamma Radiation Dose Rates	B-10
B.8 Gamma Radiation Dose Rates for Each Shipment	B-13
C.1 Concentrate and Slag Composition.....	C-2
C.2 MicroShield Dose Factors for Each Loading Scenario.....	C-5
C.3 Comparison of Gamma Dose Rates with Different Iron Shield Thicknesses.....	C-11
C.4 Comparison of Gamma Dose Rates with Different Densities	C-13
C.5 Comparison of Gamma Dose Rates with Different Source Compositions.....	C-13
C.6 Comparison of Gamma Dose Rates with Load at Different Locations in Sea-land Container.....	C-14
C.7 Comparison of Gamma Dose Rates at Different Distances that are Representative of Contact Dose Rates.....	C-15
D.1(a) Summary of Concentrations in Slag Materials Reported by the Laboratories	D-2
D.1(b) Summary of Concentrations in Tantalite Materials Reported by the Laboratories	D-3
D.2 Summary of U-238 and Th-232 Activity (Bq/g)	D-8
D.3 Summary Statistics on Ratio Between Decay Series Radionuclides and Parent Radionuclide	D-11
D.4 Summary of MicroShield Dose Factors ($\mu\text{Sv/h}$ per Bq/g)	D-15
D.5 Median Variability (%) of Gamma Radiation Measurements for Shipments.....	D-16
E.1 Calculation of Dose Factors for Each Exposure Pathway with Maximum Load	E-2

LIST OF FIGURES

	<u>Page No.</u>
1.1 The Uranium-238 Decay Series.....	1-2
1.2 The Thorium-232 Decay Series.....	1-3
3.1 Sample of Survey Results Form	3-4
3.2 Uranium (U-238) and Thorium (Th-232) Concentrations	3-8
3.3 Modelled and Measured Attributable Dose Rates	3-13
3.4 Example Load Configuration of Sea-land Container.....	3-14
3.5 Summary of Modelled and Measured Exposure Rates from the Side of the Container	3-17
A.1 Background Data Sheet.....	A-10
A.2 Dimensions and Loading Geometry Sheet	A-11
A.3 Survey Results Sheet.....	A-12
D.1 Interlaboratory Comparisons of Uranium and Thorium Measurements.....	D-5
D.2 Comparison Between Th-232 and Th-228 Measurements	D-6
D.3 Comparison Between Uranium (U-238) and Thorium (Th-232) Concentrations	D-9
D.4 Ratio Between Uranium (U-238) and Thorium (Th-232) Concentrations	D-10
D.5 Comparison Between Pb-210 and U-238 Concentrations (Bq/g).....	D-12
D.6 Tantalum Concentration (%) and Bulk Density in Shipments.....	D-13
D.7 Comparison Between Modelled and Measured Exposure Rates	D-18
E.1 Procedure for the Probabilistic Simulation of Annual Dose and Concentration	E-10

1.0 INTRODUCTION

The Tantalum-Niobium International Study Center (T.I.C.) has commissioned a study of the transport of tantalum raw materials. Due to the presence of natural uranium and thorium, tantalum raw materials are considered to be naturally occurring radioactive material (NORM). The main objectives of this study were to determine the radiological characteristics of tantalum raw materials and to evaluate the potential radiological exposures associated with the transport of these materials during normal transport and in the event of an accidental spill. The study was carried out by SENES Consultants Limited (SENES), supported by Alfred H. Knight International Ltd. (AHK) for the physical and chemical analysis of the tantalum raw materials and, in certain instances, for carrying out gamma radiation surveys at T.I.C. member company sites.

Tantalum raw materials can be defined as encompassing the tantalite mineral concentrates [Fe, Mn (Ta, Nb)₂O₆] where the tantalum (Ta) content is greater than the niobium (Nb) content, as well as slag materials which are a by-product of smelting operations (e.g. tin smelting) and which contain varying levels of Ta. Naturally occurring radioactivity contained in tantalum raw material (concentrates and slags) is uranium, thorium and their radioactive decay products. Natural uranium (U-nat) consists primarily of uranium-238 (U-238) (approximately 99.3% of U-nat by mass) and a series of radioactive decay products, including U-234, which terminates in non-radioactive lead. U-nat also contains the U-235 radioactive series (approximately 0.7% of U-nat by mass), but U-235 is usually not considered to be environmentally or occupationally significant in NORM. Natural thorium (Th-nat) consists of Th-232 (essentially 100% by mass) and a corresponding series of radioactive decay products that terminates in non-radioactive lead. The decay series for U-238 and Th-232 are provided in Figures 1.1 and 1.2, respectively.

As shown in Figures 1.1 and 1.2, the decay of U-238 and Th-232 occurs by alpha or beta radiation (particle radiation). The radioactive decay products (radionuclides) also emit gamma radiation (electromagnetic radiation). Gamma radiation is the most significant radiation for the transport of the tantalum materials since gamma radiation can penetrate the transport packages/drums and transport containers. In addition to the decay schemes, Figures 1.1 and 1.2 include the half-life of each radioactive decay product. The half-life of a radionuclide is the amount of time required for half of the radionuclides to decay. In undisturbed raw materials such as tantalum ore, the activity of each radioactive decay product in the U-238 and Th-232 decay series will normally be equal within each series (i.e. in radioactive equilibrium). However, as a result of thermal processing, the U-238 and Th-232 decay series may not be in equilibrium in slags, another source material for tantalum. Radiological analyses, such as done in this study, are used to determine if equilibrium exists.

FIGURE 1.1
THE URANIUM - 238 DECAY SERIES

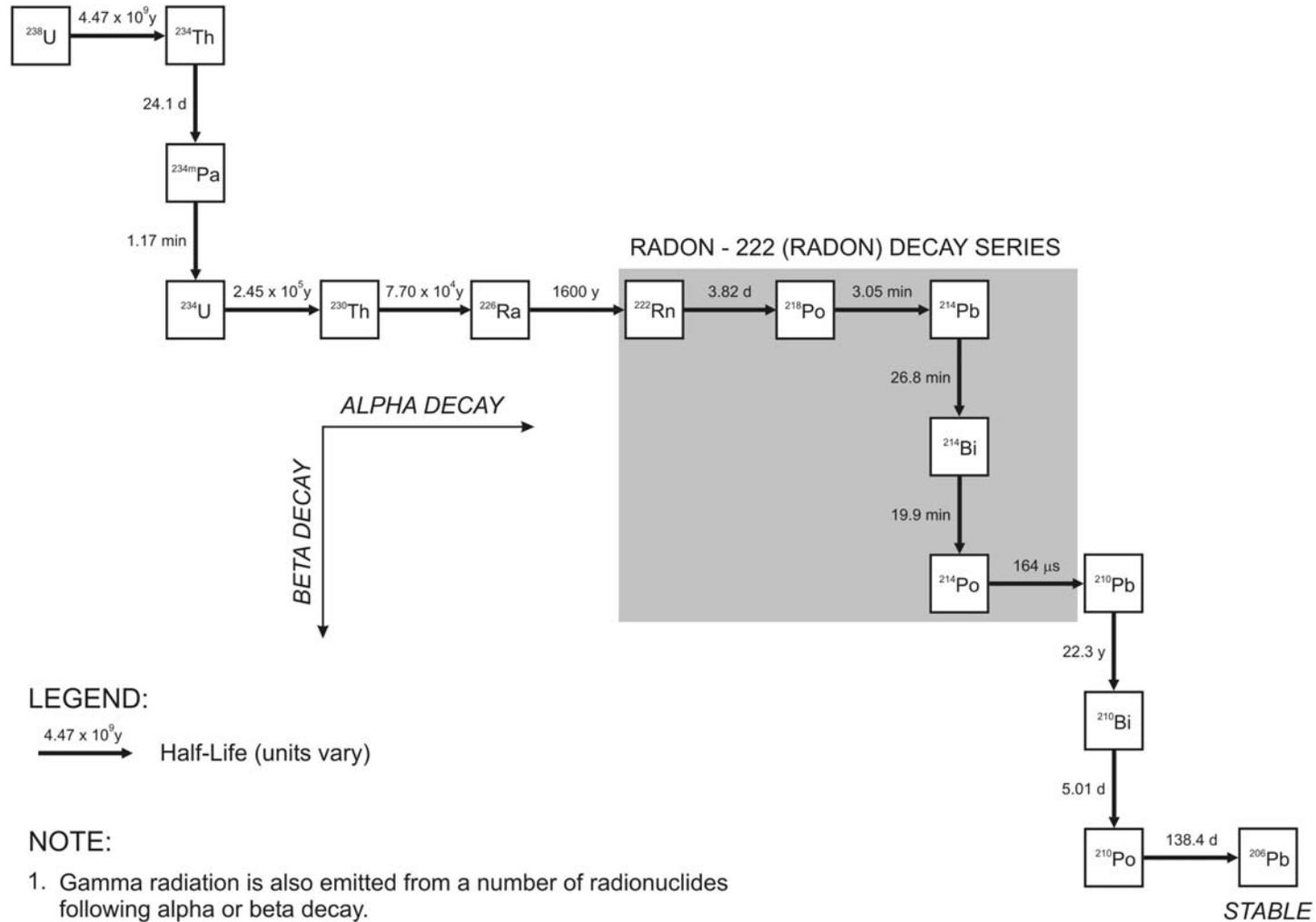
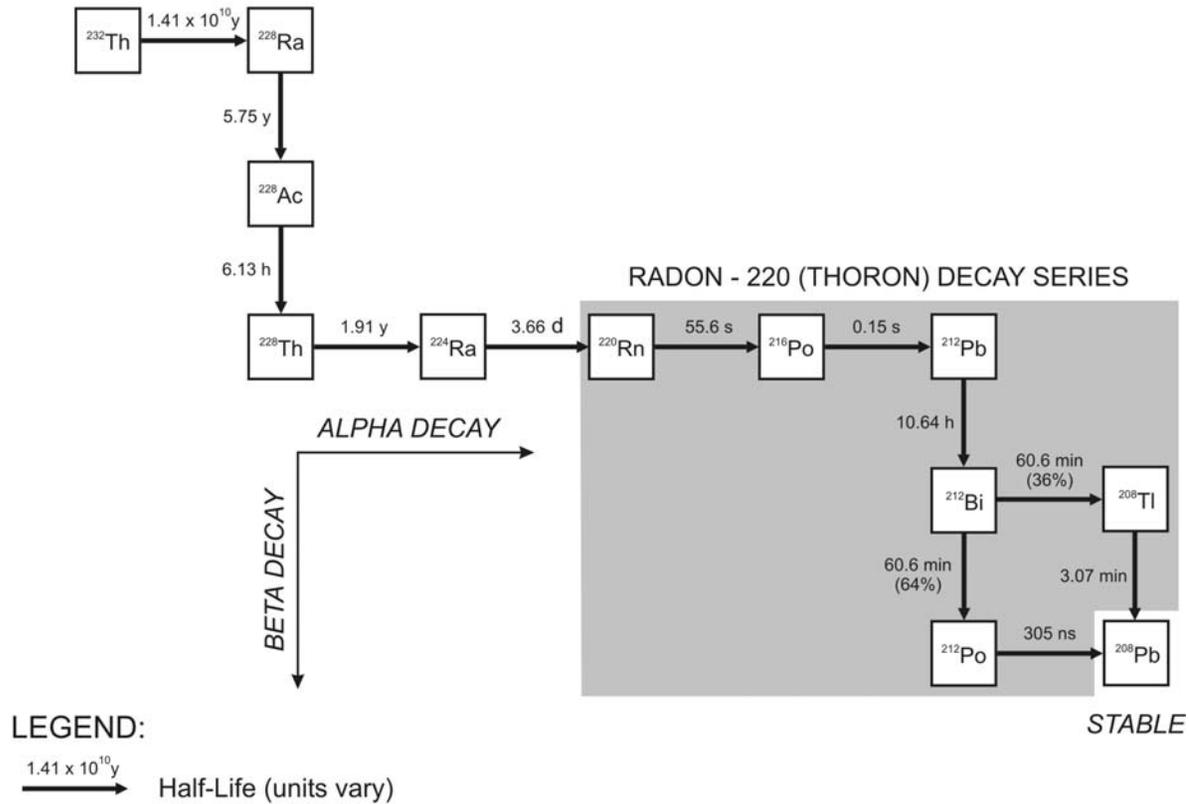


FIGURE 1.2
THE THORIUM - 232 DECAY SERIES



NOTE:

1. Gamma radiation is also emitted from a number of radionuclides following alpha or beta decay.

The transport of NORM is an international concern and the International Atomic Energy Agency (IAEA) issued its first safety series on international and national transport of radioactive material by all modes in 1961. By 1969, almost all of the international organizations concerned with transport and IAEA Member States had adopted the IAEA Guidance. The most recent version of the transport regulations (IAEA 2005a) includes amendments to the 1996 Edition from the second cycle of the biennial review and revision process. Many countries have to date adopted the 1996 regulations in their national laws.

The IAEA safety standards are based on information from a variety of sources including findings of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), recommendations from expert international bodies (especially the International Commission on Radiological Protection (ICRP)) and experience from various other agencies and organizations. IAEA maintains the safety standards current by having them undergo review five years after publication to determine if any revisions are required.

A 2003 International Conference on the Safety of Transport of Radioactive Material:

“...identified a need for additional research to relieve unnecessary regulatory burdens related to the transport of very low activity NORM. Since the 1996 edition of the IAEA Transport Regulations introduced radionuclide-specific exemption levels in lieu of the single 70 Bq/g [becquerels per gram] value, ores, tailings, and backfill from large mining operations (e.g. phosphate, coal, gold and monazite) have been brought within the scope of the Regulations. To address this situation, the 1996 Regulations included an allowance for a factor of 10 higher than the exemption quantities for naturally occurring materials, provided they are not intended to be processed to extract the naturally occurring radionuclides. The Conference noted the potential inconsistency between this provision and the developing international guidance on the more general issue of the scope of the regulatory control in RS-G-1.7, the problems associated with determining the ultimate use of the material, and the inconsistency of excepting doses associated with some types of source (e.g. naturally occurring radioactive material – NORM) but not doses of the same magnitude from other types of source. The Conference suggested that the full impact of and technical basis for the “factor of 10 exemption be thoroughly researched.” (IAEA 2005b)

The IAEA has established a Coordinated Research Programme (CRP) related to the transport of NORM. The T.I.C. has become concerned with the implications of the IAEA transport regulations on the tantalum industry and submitted a research proposal to the IAEA via the CRP to conduct a study on NORM transport. The Canadian Competent Authority to TRANSSEC has agreed to sponsor the T.I.C. study.

This study “Radiological Risk Assessment of the Transport of Tantalum Raw Materials” was commissioned to SENES in March of 2005. The objectives of the SENES study were:

1. Establish the relationship between measured uranium (U-nat) and thorium (Th-nat) concentrations of tantalum raw materials and the measured dose rates on the outside of a transport container.
2. Assess doses to handling and transport workers and the general public during routine transport conditions.
3. Carry out a risk assessment comparing these doses to relevant safety standards.
4. Identify the range of potential exposure risks from hypothetical accidents (including spillage).
5. Use the results of the study to determine the level of regulation appropriate to the estimated risk found in order to ensure safe transport.
6. Provide results to educate the carriers that currently deny shipments, about the actual quantified risks.

The study consisted of an evaluation of radiation exposures associated with the transport of tantalum ore concentrates and slags, from source to processor. The first stage of the study involved the radiological characterization of tantalum raw materials. In order to develop this characterization, SENES and AHK defined a reference gamma radiation survey protocol for raw materials in their normal shipping configurations. Subsequently, a number of T.I.C. member companies carried out radiation measurements of a variety of tantalum raw materials. In addition, AHK carried out radiation surveys at a number of sites.

While radiation surveys have been carried out on a reasonable range of tantalum raw materials, it is not possible to make measurements for all possible source materials and transport configurations. Hence, theoretical calculations have been carried out for a variety of source materials and transport configurations. The radiation survey data collected in this study were used to “benchmark” the theoretical radiation dose rate calculations. These relations were then used as the basis for estimating potential gamma radiation exposures for a variety of transport scenarios. Finally, the potential radiation exposures to transport workers and the general public were calculated for transport under normal operations and for transportation accidents.

This report is organized as follows:

Chapter 2: Regulatory Implications of Radioactivity on Transport of Tantalum Materials.

Chapter 3: Radiological Characterization of Tantalum Raw Materials.

Chapter 4: Doses Arising from Normal Transport Activities.

Chapter 5: Doses Arising from Transport Spills and Accidents.

Chapter 6: Observations and Conclusions.

Chapter 7: References.

Annex A – Gamma Radiation Survey and Sample Collection Protocol.

Annex B – Results from Radiation Surveys.

Annex C – MicroShield Modelling.

Annex D – Analysis of Measurement Data.

Annex E – Dose Scenario Factors and Dose Calculation.

2.0 REGULATORY IMPLICATIONS OF RADIOACTIVITY ON TRANSPORT OF TANTALUM RAW MATERIALS

2.1 INTRODUCTION

The transport of radioactive materials is regulated internationally by the International Atomic Energy Agency (IAEA). The IAEA “regulations” are considered to represent the general international consensus on transport issues, which only officially become regulations when they are adopted into national and international laws by countries and international agencies.

The transport of NORM is an international concern and the IAEA issued its first safety series on international and national transport of radioactive material by all modes in 1961. Reviews conducted with Member States and international organizations concerned with transport resulted in six separate revisions being published in 1964, 1967, 1973, 1985, 1996 and now 2005. After the first revision (1964), the regulations were applied to all IAEA and IAEA-assisted operations, and by 1969, almost all of the international organizations concerned with transport and Member States had adopted the regulations. Initially, the IAEA published two companion standards that provided the advisory and explanatory material relating to the regulations; however, in support of the 1996 edition of the regulations (known as TS-R-1), the IAEA published a companion volume that included both advisory and explanatory material (Safety Guide No. TS-G-1.1 (ST-2), IAEA 2002).

The most recent version of the transport regulations (IAEA 2005a) includes amendments to the 1996 edition from the second cycle of the biennial review and revision process. The 2005 regulations (also TS-R-1) are essentially similar to the 1996 regulations. Because of the normal time delays in regulatory processes, many countries have yet to adopt the 2005 regulations, but have to date adopted (in whole or with minor revisions) the 1996 regulations and revisions (IAEA 2000, 2004a) in their national laws.

The transport regulations are part of the IAEA safety standard series, which is intended to provide radiation protection guidance for both people and the environment. These safety standards “*reflect an international consensus on what constitutes a high level of safety for protecting people and the environment*” (IAEA 2005a) and have three different categories: safety fundamentals, safety requirements and safety guides. The safety standard for the transport of radioactive material is classified as a safety requirement. Safety requirements establish “*the requirements that must be met to ensure the protection of people and the environment, both now and in the future. The requirements, which are expressed as ‘shall’ statements, are governed by the objectives, concepts and principles of the Safety Fundamentals. If they are not met, measures must be taken to reach or restore the required level of safety. The Safety Requirements use regulatory language to enable them to be incorporated into national laws and regulations.*” (IAEA 2005a).

The safety standards are based on information from a variety of sources including findings of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), recommendations from expert international bodies (especially the International Commission on Radiological Protection, ICRP) and experience from various other agencies and organizations. The IAEA keeps the safety standards current by having the standards undergo review five years after publication (biennially for transport regulations) to determine if any revisions are required.

2.2 EXEMPTION VALUES

The TS-R-1 transport regulations (IAEA 2000, 2004a, or 2005a) provide radionuclide-specific activity concentration (Bq/g) and radionuclide-specific total activity (Bq) exemption values below which the regulations do not apply. (Both the concentration and total activity limits have to be exceeded before the transport regulations apply. The total activity limits apply on a per consignment¹ rather than on a per package² basis.) According to the IAEA (2002), these exemption values were initially derived for inclusion in the overall IAEA Basic Safety Standards (BSS) for radiation protection (BSS 115, IAEA 1996) on the basis that the dose to an individual would not exceed 10 microsieverts per year ($\mu\text{Sv/y}$) under normal conditions. The 10 $\mu\text{Sv/y}$ dose value was considered to represent an insignificant, trivial or *de minimis* level of risk. The IAEA refers to this dose rate as being “... *sufficiently low as to be of no regulatory concern*” (IAEA 1996). Another basis for derivation of the exemption values was that the collective dose associated with the values (i.e. the summed dose to all impacted individuals) would not exceed 1 person-Sv. However, it has since been generally concluded (IAEA 2004b) that the individual dose would almost always be the limiting factor.

The scenarios used to derive the exemption values in the BSS were not specifically related to transport situations. Subsequent calculations for transport scenarios were performed and it was found that the derived limits were similar to the BSS values. To avoid potential complications, the exemption values derived for the BSS were adopted for the transport regulations. For radionuclides not in the BSS, exemption values were calculated on the same basis (IAEA 2002).

Examples of the exemption values are 1 Bq/g activity concentration for both natural uranium (U-nat) and natural thorium (Th-nat) and 1000 Bq total activity (IAEA 2005a, Table 1). For radioactive decay series in equilibrium, the limits apply to the parent radionuclides, i.e. U-238 or Th-232 (See Figures 1.1 and 1.2 in Chapter 1). For materials containing both uranium and thorium, the limits apply to the sum of the U-238 and Th-232 activities.

¹ *Consignment shall mean any package or packages, or load of radioactive material, presented by a consignor for transport (IAEA 2005a).*

² *Package shall mean the packaging with its radioactive contents as presented for transport (IAEA 2005a).*

However, in regards to NORM and industries such as the tantalum industry, the limits are different. The transport regulations do not apply to natural materials and ores containing naturally occurring radionuclides that are not intended to be processed for the use of these radionuclides provided that the activity concentration of the material does not exceed 10 times the exemption value (IAEA 2005a). According to the IAEA (2002), if this were not the case, “...the Regulations would have to be applied to enormous quantities of material that present a very low hazard.” The activity concentration exemption values for the tantalum raw materials (NORM) are therefore 10 Bq/g for both U-nat and Th-nat.

Despite the higher exemption values for NORM, the NORM radioactivity has caused significant practical difficulties to the operations of T.I.C. member companies. It should be noted that the appropriateness of this factor of 10 is not obvious as there is apparently no specific radiological basis for choosing “10” as the factor. According to the IAEA,

“... a factor of 10 times the exemption values for activity concentration was chosen as providing an appropriate balance between the radiological protection concerns and the practical inconvenience of regulating large quantities of material with low activity concentrations of naturally occurring radionuclides.”
(IAEA 2002, para. 107.4)

It was one objective of this study that the information presented and analyzed here would provide some useful data for determining if the “factor of 10” was appropriate and if not, then to suggest an alternative factor.

3.0 RADIOLOGICAL CHARACTERIZATION OF TANTALUM RAW MATERIALS

The dose arising from transport of the tantalum raw materials depends on several factors including the concentrations of radioactivity, the amount of material and the configuration of the material during shipment. This chapter describes the collection of information from actual tantalum shipments and the radiological characterization of these shipments. In total, there were 71 shipments³ with information on tantalum raw materials available for this assessment.

Uranium and thorium concentrations were found to be variable between shipments as were the gamma radiation exposure rates measured in the vicinity of the containers. More than half of the tantalite shipments had a total (i.e. U-238 + Th-232) activity concentration exceeding 10 Bq/g with mean total (U-238 + Th-232) activity concentrations of 25.2 and 17.7 Bq/g for slag and tantalite materials, respectively. Exposure rates from slag shipments were on average nearly twice as high as the exposure rates from tantalite shipments. There were a higher proportion of tantalite shipments relative to the number of slag shipments.

Based on the characterization data, dose factors relating the concentration to the dose received per hour of exposure were developed using the MicroShield model (Grove Software 2005). These model predictions were consistent with the measured gamma radiation dose rates.

3.1 DATA COLLECTION

3.1.1 Data Requirements

The dose from transport of tantalum raw material depends on several factors. First, since the exposure rate depends on the concentration of radioactivity in the shipment, a representative mean and range of concentrations of uranium and thorium in the shipments is required. Measurements of other radionuclides in the decay series are necessary since much of the dose can come from these radionuclides. It is important to know what the concentrations of these radionuclides are relative to the uranium or thorium parent. Characteristics of the materials such as density and the composition of other elements in the materials may also affect the amount of gamma radiation emitted from the material. These attributes were characterized through sample collection and laboratory analyses.

Gamma radiation exposure rates also depend on the type of material (e.g. slag or tantalite), the type of transport container (e.g. trailer or sea-land container) and the loading configuration within the container (e.g. full or partially full). This information, including the dimensions of the

³ Only the 67 shipments with the primary laboratory measurements were used in the assessment. See Section 3.1.6.

container, was requested from the companies. Some characteristics, for example the thickness of the container walls, were gathered elsewhere. This information was used for modelling gamma radiation exposure rates using the MicroShield model.

Direct measurements of gamma radiation exposure rates around the shipment container provide another estimate of gamma exposure rates at selected locations and provide validations of the MicroShield modelling predictions.

3.1.2 Shipment Characteristics

T.I.C. requested that companies voluntarily participate in the characterization of tantalum materials through collection of samples, gamma radiation measurements and providing other characteristics of the shipments. The participating companies provided information on tantalum shipments that were “on-hand” at the facilities owned or operated by T.I.C. member companies. It was assumed that the radiological characteristics of these shipments were representative of tantalum raw material shipments in general.

The shipments varied by the type of material and packaging within the shipment. Slags were packaged within one tonne bags and tantalite concentrates were typically contained in drums. These packages were placed in either sea-land containers or on trailers (tantalite only) with various configurations. For example, there could be one or two tiers (layers) of packages within a sea-land container. The first layer could be completely or partially full and was denoted as F - 1 Tier (full one tier) and NF - 1 Tier (partially full one tier). It was noted that due to the weight restrictions, a sea-land container could not contain 2 full tiers of tantalum material (personal communication with T.I.C. Transport Committee, Dec. 2005). Therefore, the maximum loading considered for this study was a full first layer and a half-full second layer, which has been denoted as Maximum Load. There were a number of shipments reported as full two-tier and some as not full two-tier; therefore, all such loads have been assumed to be Maximum (full first layer and half-full second layer) Loads.

Table 3.1 summarizes the number of shipments by type of material and the shipment loading configuration. The majority, 48, of shipments were tantalite concentrate, while 23 of the shipments were slag materials. Approximately one half of the 71 shipments were Maximum loads; however, the loading configuration was unspecified for the 12 shipments without gamma radiation survey measurements.

**TABLE 3.1
MATERIAL TYPES AND LOADING CONFIGURATION**

Configuration	Type of Tantalum Raw Material		
	All Shipments	Tantalite	Slag
All	71	48	23
Maximum Load	39	25	14
F - 1 Tier	11	6	5
NF - 1 Tier	5	4	1
NF Trailer	2	2	0
On ground	2	2	0
Unknown	12	9	3

Notes:

Maximum (for modelling purposes, assumes maximum load was full 1st tier and half-full 2nd tier).

F- Full (Assumes no empty spaces within load).

NF – Not Full (Assumes empty spaces within load).

Unknown – Configuration could not be determined since gamma survey results were not provided.

3.1.3 Sample Collection and Gamma Radiation Survey

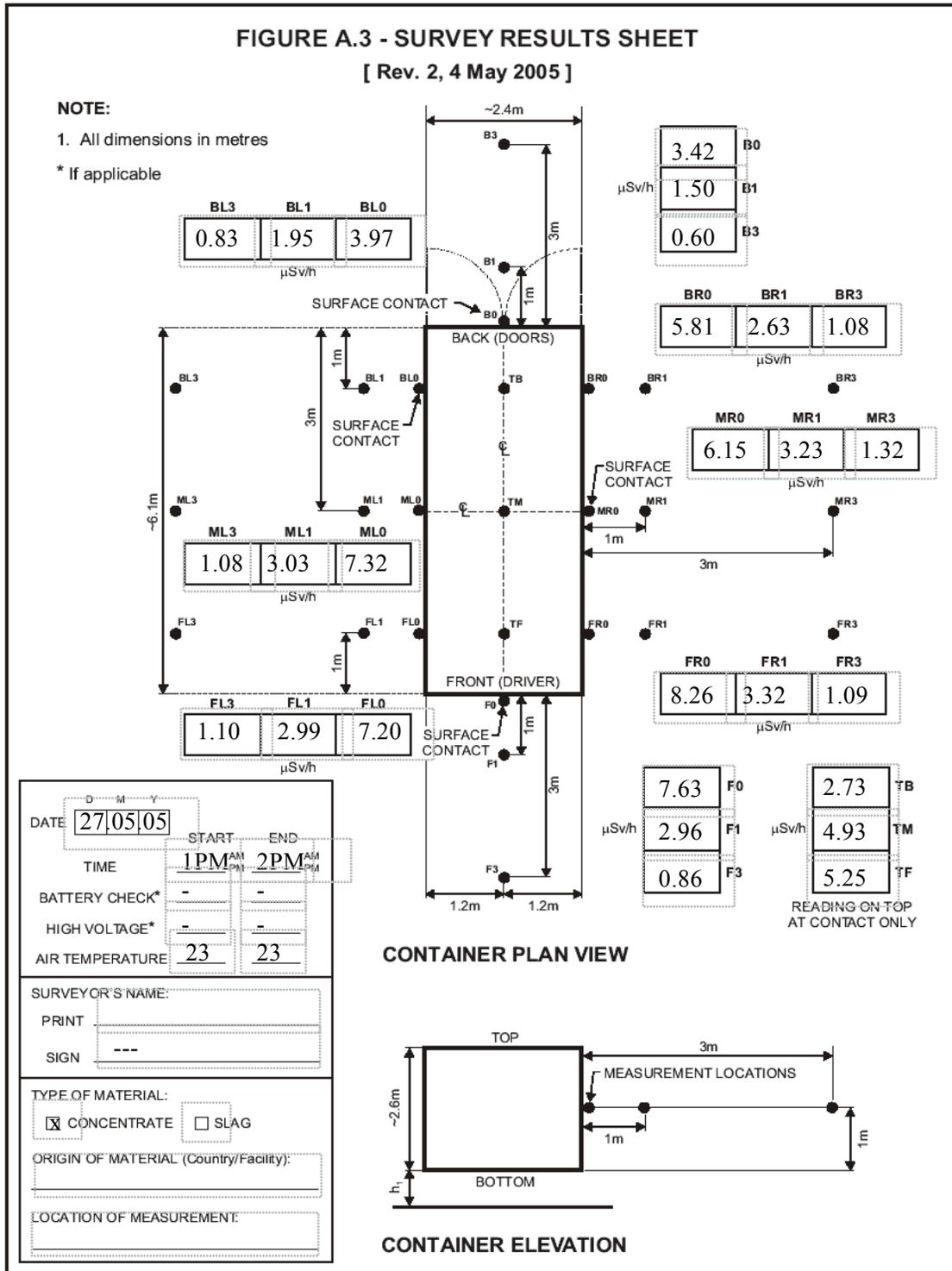
A composite sample of the material in each shipment was collected following an established protocol. In most cases, these samples were shipped to the primary laboratory for sample preparation and laboratory analyses of uranium and thorium concentration. The protocol for sample collection is provided in Annex A.

Participating companies measured gamma radiation exposure levels at pre-specified locations around the shipment according to a protocol developed for this study. Since gamma radiation is ubiquitous, baseline gamma radiation surveys were conducted in the absence of the shipment at the same location. The model of the gamma radiation meter used for the survey was recorded.

Figure 3.1 is a sample of a completed survey form showing the measurement results. For reasons of confidentiality, the company identifiers have been removed.

The gamma radiation survey protocol is described in greater detail in Annex A. The results from the radiation surveys are provided in Annex B.

**FIGURE 3.1
SAMPLE OF SURVEY RESULTS FORM**



3.1.4 Laboratory Analyses of Concentrations

Most samples were sent to the primary laboratory for sample preparation and laboratory analyses. The samples were ground to a homogeneous finely divided powder form. A portion of the sample was analyzed for thorium and uranium content by X-ray Fluorescence spectrometry with concentrations reported as ThO_2 and U_3O_8 , respectively. Additional analyses included bulk density determination and chemical analysis (i.e. Ta_2O_5 , Nb_2O_5).

Split samples were sent to an independent laboratory for analyses of the thorium and uranium concentrations using neutron activation analyses (NAA). These concentrations provide the ability for an interlaboratory comparison with the primary laboratory measurements of uranium and thorium concentrations. The independent laboratory also measured the concentrations of other radionuclides in the uranium and thorium decay series using gamma spectroscopy methods. The majority of dose from the uranium and thorium series typically comes from radionuclides other than uranium or thorium; therefore, it is important to know the concentrations of the radionuclides in the decay series relative to the uranium and thorium content.

Samples from one company's shipments could not be sent outside the country to the primary laboratory; therefore, the concentrations of thorium and uranium in these samples were measured by an alternate laboratory. An interlaboratory comparison could not be conducted for these samples; however, it has been assumed that these measurements were appropriate for radiological characterization and dose assessment.

3.1.5 Modelled Exposure Rates

The theoretical model, MicroShield (Grove Software 2005), was used to predict gamma radiation exposure rates at the same locations where gamma radiation exposure rates were measured on the actual shipments using the gamma radiation survey protocol. Inputs to the model include characteristics of the material including the radionuclide concentrations, the density of the material, the elemental composition of the material along with the physical size and configuration of the material emitting the gamma radiation. The thickness, density and elemental composition of any shielding is considered. The locations of interest relative to the gamma-emitting material and the shielding are also important inputs to the model.

The values of these parameters were developed in part through information provided by the companies, laboratory analyses of the material and professional judgement. The MicroShield model was used to predict dose factors (i.e. $\mu\text{Sv/h}$ (dose rate) per Bq/g (concentration)) for the different types of material (i.e. slag or tantalite) and selected shipment loading configurations (e.g. Full 1 Tier). These dose factors were relatively insensitive to variations in density or elemental composition of the materials.

A more detailed explanation of the MicroShield modelling is provided in Annex C.

3.1.6 Data Completeness

The information was entered into a relational database to facilitate further modelling and analysis, and the amount of information is summarized in Table 3.2. There were 71 shipments: of these, there were 59 shipments with gamma radiation survey results and 67 shipments with concentrations of uranium and thorium measured by the primary laboratory. There were 55 (i.e. 6 + 49) shipments with both gamma radiation survey measurements and laboratory analyses of uranium and thorium content. There were 61 (i.e. 49 + 12) shipments with measurements from both the primary and independent laboratories available.

**TABLE 3.2
COMPLETENESS OF DATA**

Number of Shipments	Gamma Radiation Survey	Primary Laboratory Measurements	Independent Laboratory Measurements
12	No	Yes	Yes
4	Yes	No	No
6	Yes	Yes	No
49	Yes	Yes	Yes
Total			
71	59	67	61

3.2 RADIOACTIVITY CONCENTRATIONS

3.2.1 Uranium and Thorium Content

Uranium and thorium concentrations are typically expressed on a mass basis (e.g. ppm, %); however, the regulations relative to transport are based on activity with units of Bq/g. In addition, radioactive dose is calculated based on this activity. For this report, concentrations originally measured on a mass basis have been converted to activity concentrations. For example, the U-238 activity concentration is used for uranium content and the Th-232 content is used for thorium content. Annex D provides a summary of uranium and thorium concentrations on both a mass basis and an activity basis.

A data quality assessment was conducted including a review of the laboratory precision and an interlaboratory comparison of uranium and thorium measurements. There were no U-238 measurements reported as being below the laboratory reporting limit (i.e. “<”); however, there

was a limited number of “<” values reported for Th-232. A comparison between the primary and independent laboratory measurements of split samples showed good agreement over the range of U-238 concentrations and for Th-232 concentrations, above a few Bq/g. See Annex D for more details on this comparison. Potentially, both laboratories could underestimate the Th-232 concentrations at low levels (when compared to the Th-228 concentrations – see Annex D). However, based on the close agreement in concentrations at higher radioactivity levels, the uranium and thorium measurements by the primary laboratory were considered appropriate for this dose assessment.

Table 3.3 shows a summary of the activity concentrations of U-238 and Th-232 along with the total (U-238 + Th-232) activity concentration in the shipments using the reported concentrations from the primary laboratory with conversion from mass basis to activity basis. The concentrations were assumed equal to the reporting limit for concentrations reported as “<” by the laboratory. Overall, the concentrations of U-238 are higher than the concentrations of Th-232 with median concentrations of 16.4 and 1.3 Bq/g respectively, in tantalite materials. The Th-232 concentrations tend to be higher in slag materials compared to concentrations in tantalite materials with a mean concentration of 6.5 Bq/g for Th-232. The U-238 mean concentration in slag of 18.8 Bq/g was similar to the mean concentration in tantalite materials.

The concentrations were variable ranging, for example, from a minimum of 2.4 to a maximum of 92.2 Bq/g for U-238 in slag materials and from 0.2 to a maximum of 11.1 Bq/g for Th-232 in tantalite materials. The mean total activity concentrations were 17.7 and 25.3 Bq/g for tantalite and slag materials, respectively. The shipments of tantalite were more likely (i.e. 78% vs. 45%) to exceed 10 Bq/g of total activity than the shipments of slag.

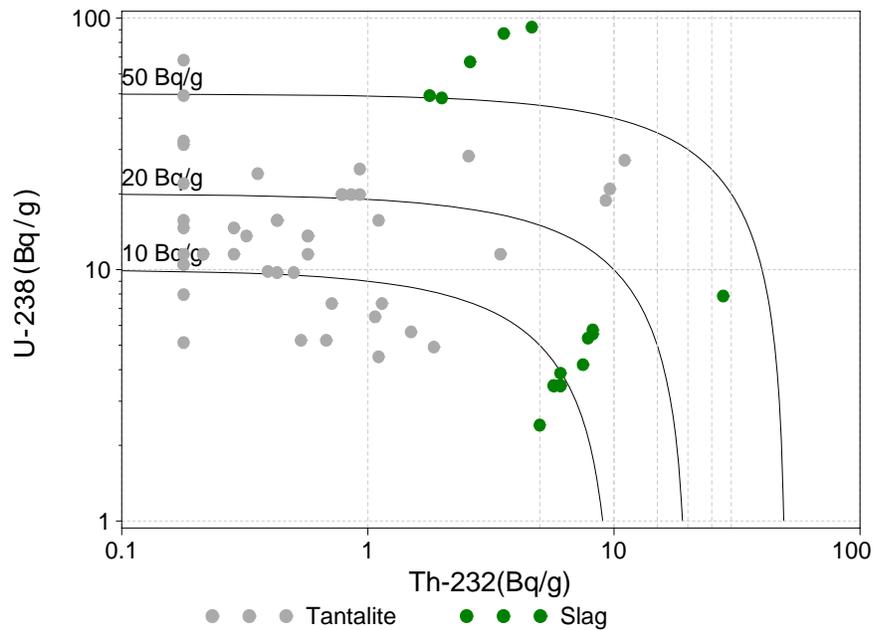
**TABLE 3.3
SUMMARY OF U-238 AND TH-232 ACTIVITY CONCENTRATIONS (Bq/g)**

Radionuclide	Material Type	Number of Shipments	Reported as “<” (%)	Median (Bq/g)	Mean (Bq/g)	Min. (Bq/g)	Max. (Bq/g)	Proportion > 10 Bq/g (%)
Th-232	Slag	22	0	5.9	6.5	1.8	27.8	5
Th-232	Tantalite	45	24	0.5	1.3	0.2	11.1	2
U-238	Slag	22	0	3.7	18.8	2.4	92.2	23
U-238	Tantalite	45	0	13.6	16.4	4.5	68.1	71
Total	Slag	22		9.7	25.3	7.4	96.8	45
Total	Tantalite	45		14.2	17.7	5.3	68.3	78

Figure 3.2 shows a scatter plot of the concentrations of the U-238 and Th-232 in the shipments, with the lines showing total activities of 10 Bq/g, 20 Bq/g and 50 Bq/g. On an overall basis, there is little overall correlation between the concentrations of U-238 and Th-232 concentrations

although there are some patterns in the scatter plot. The slag shipments appear to form two groups. The Th-232 concentrations tend to be higher in slag shipments than in tantalite shipments; however, there is one group of slag shipments with relatively high U-238 concentrations and another group with relatively low U-238 concentrations.

FIGURE 3.2
URANIUM (U-238) AND THORIUM (Th-232) CONCENTRATIONS



3.2.2 Tantalum Content and Density

As part of data review, an investigation into the relationship between tantalum (Ta_2O_5) content and density of material was conducted. Slag materials, in general, had lower bulk density and lower tantalum content compared to tantalite materials; however, a few shipments varied from this typical pattern. There were two tantalite samples with low bulk-density and low tantalum content and there were five slag shipments with a moderate bulk-density and with a tantalum content typical of tantalite shipments. The difference in characteristics of slag shipments are potentially due to difference in mineralization in the feed material and differences in pyrometallurgical processes that created the slag.

3.2.3 Other Radionuclide Concentrations

The concentrations of other radionuclides for each shipment are shown in Annex B and have been summarized in Annex D. The concentrations of these other radionuclides in the uranium and thorium decay series were found to be generally comparable to the concentrations of the parent radionuclides (i.e. U-238 and Th-232, respectively); that is, there were no radionuclides with concentrations consistently and substantially higher or lower than the concentrations of the parent radionuclide. An exception to the general equilibrium condition was the five slag samples with relatively high tantalum content and high density compared to the other slags. These samples had substantially lower Pb-210 concentrations than would be expected with equilibrium conditions and this may be due to volatilization of lead during the type of process that created these slags.

An assumption of equilibrium within each of the uranium and thorium decay series is reasonable for dose assessment purposes since the majority of the dose from raw material arises from gamma radiation to which Pb-210 contributes little. This assumption means that the concentrations of other radionuclides in the uranium and thorium decay series can be assumed equal to the concentrations of U-238 and Th-232, respectively.

3.2.4 Discussion

Measured uranium and thorium concentrations were available for 67 shipments. The uranium and thorium measurements by the primary laboratory compared well in an interlaboratory comparison and are appropriate for dose assessments. An assumption of equilibrium for other radionuclides in the uranium and thorium decay series is considered appropriate for the dose assessment.

The U-238 and Th-232 concentrations in the tantalum materials are variable, varying by a factor of about 50 and, overall, there is little correlation between the concentrations of U-238 and Th-232. Slag materials had average concentrations of U-238 that were similar to average U-238 concentrations in tantalite; however, the Th-232 concentration tended to be higher in slags than in tantalites. The majority, 78%, of tantalite shipments with about half, 45%, slag shipments had a total activity concentration greater than 10 Bq/g. The mean total activities were 17.7 and 25.3 Bq/g for tantalite and slag materials, respectively.

3.3 GAMMA RADIATION EXPOSURE RATES

3.3.1 Measured Gamma Radiation Rates

Gamma exposure rates were measured for 59 tantalum raw material shipments. Based on symmetry, there are nine combinations of geometry and distance for the measurement locations. For example, the two locations at a distance of 1 metre (m) from each end of the container would be expected to have the same gamma radiation exposure level if the material were uniformly placed within the container.

Table 3.4 summarizes the gamma radiation exposure rate attributable to the tantalum raw materials for the 57 tantalum raw material shipments with gamma radiation surveys⁴. The attributable amount was calculated by subtracting the baseline gamma radiation level from the gamma radiation levels measured with the loaded container.

**TABLE 3.4
SUMMARY OF MEASURED ATTRIBUTABLE DOSE RATES ($\mu\text{Sv/h}$) BY DISTANCE
AND GEOMETRY RELATIVE TO THE CONTAINER**

Statistic	Side			Corner			End		
	Contact	1 m	3 m	Contact	1 m	3 m	Contact	1 m	3 m
Slag Shipments (n= 20)									
Median	3.5	1.5	0.5	3.0	1.1	0.4	2.0	0.8	0.2
Mean	4.7	1.9	0.6	6.1	1.3	0.4	2.6	0.9	0.3
Maximum	16.5	5.9	1.5	26.2	4.1	1.3	9.9	3.9	0.7
Tantalite Shipments (n= 37)									
Median	2.0	0.8	0.3	1.9	0.7	0.3	1.3	0.5	0.2
Mean	2.4	0.9	0.3	2.0	0.8	0.3	1.6	0.6	0.2
Maximum	6.6	3.0	1.1	6.2	2.6	0.9	5.4	2.1	0.6

Note: Two gamma radiation surveys with the materials in bags on the ground (not in a container) have been excluded from this summary.

The mean exposure rates attributable to the tantalum raw materials decrease by about a factor of ten from contact with the container to a distance of 3 m. The mean contact measurements for tantalite shipments range from 1.6 to 2.4 $\mu\text{Sv/h}$ depending on the location. The mean contact exposure rates are higher for slags and range from 2.6 to 6.1 $\mu\text{Sv/h}$ depending on the location. For both slags and tantalite materials, the lower contact exposure rates tend to be at the ends of the container.

⁴ Two gamma radiation surveys that were conducted with the material in bags on the ground but not inside a container have been excluded from this analysis.

This pattern of higher exposure rates for slag shipments compared to tantalite shipments is expected for a number of reasons; first, slag materials tended to have higher total activity concentrations than tantalite. Second, gamma exposure rate per Bq/g of Th-232 is higher than the gamma exposure rate per Bq/g of U-238 and the slag materials have a higher proportion of Th-232 to U-238 compared to tantalite materials. In addition, slag materials have a lower density than tantalite materials; therefore, the slag shipments tend to be more fully loaded and hence there is a larger volume of material emitting gamma radiation than for tantalite shipments.

3.3.2 Modelled Gamma Radiation Rates

Information on the types of material, loading configuration and the type of containers were used with the MicroShield model to estimate the gamma radiation dose factors (i.e. $\mu\text{Sv/h}$ (dose rate) per Bq/g (concentration)) at the measured locations. The modelled gamma radiation exposure can be calculated by multiplying the concentration in the material by the appropriate dose rate factor for the type of material, type of container and loading configuration for the shipment.

Dose factors were modelled for two types of materials, slag and tantalite, and for the following three loading configurations; i) Full Trailer; ii) Full 1 Tier; and, iii) Maximum (i.e. 1 full tier and a half-full 2nd tier) loads. Some of the loads were partially full; however, precise information on the configuration was not available to model all the variation in loads. In addition, the configuration was unknown for the 12 shipments that did not provide gamma radiation surveys. The modelled dose rate factors are likely to be overestimates of actual dose rate for loads that are partially full.

The dose factors were higher for Maximum load shipments compared to Full 1 Tier shipments as expected since there is more material emitting gamma radiation. Gamma radiation exposure factors are about 50% higher for the thorium series than for uranium series and the dose factors for slags were slightly higher than for tantalite materials. More detailed discussion on the MicroShield modelling is provided in Annex C and Annex D.

The dose factors were matched with the material type, loading configuration and the measured concentrations in the individual shipments. The model predictions of dose rate for the 67 shipments with measured concentrations are summarized in Table 3.5. The overall average for contact measurements on slag material shipments ranges from 6.7 to 7.7 $\mu\text{Sv/h}$ depending on the location with the higher mean exposure rates measured at the end of the container. The predicted mean contact dose rates for slag materials is about 50% higher than the predicted mean dose rate for tantalite shipments which range from 4.5 to 5.2 $\mu\text{Sv/h}$. As discussed earlier, this arises because slag materials tend to have higher activity levels on average, a higher proportion of thorium content relative to uranium content and also tend to be more fully loaded compared to tantalite materials.

**TABLE 3.5
SUMMARY OF PREDICTED DOSE RATES ($\mu\text{Sv/h}$) FOR SHIPMENTS WITH
MEASURED CONCENTRATIONS**

Statistic	Side			Corner			End		
	Contact	1 m	3 m	Contact	1 m	3 m	Contact	1 m	3 m
Slag Shipments(n= 22)									
Median	4.0	2.0	0.7	3.8	1.7	0.6	4.8	1.5	0.3
Mean	6.9	3.5	1.1	6.7	3.0	0.9	7.7	2.6	0.5
Maximum	18.9	9.6	3.0	18.9	8.3	2.5	18.9	7.3	1.5
Tantalite Shipments(n= 45)									
Median	3.6	2.0	0.6	3.6	1.7	0.5	4.0	1.5	0.3
Mean	4.6	2.5	0.8	4.5	2.1	0.7	5.2	1.9	0.4
Maximum	20.2	10.7	3.6	19.5	9.2	3.0	23.4	8.2	1.6

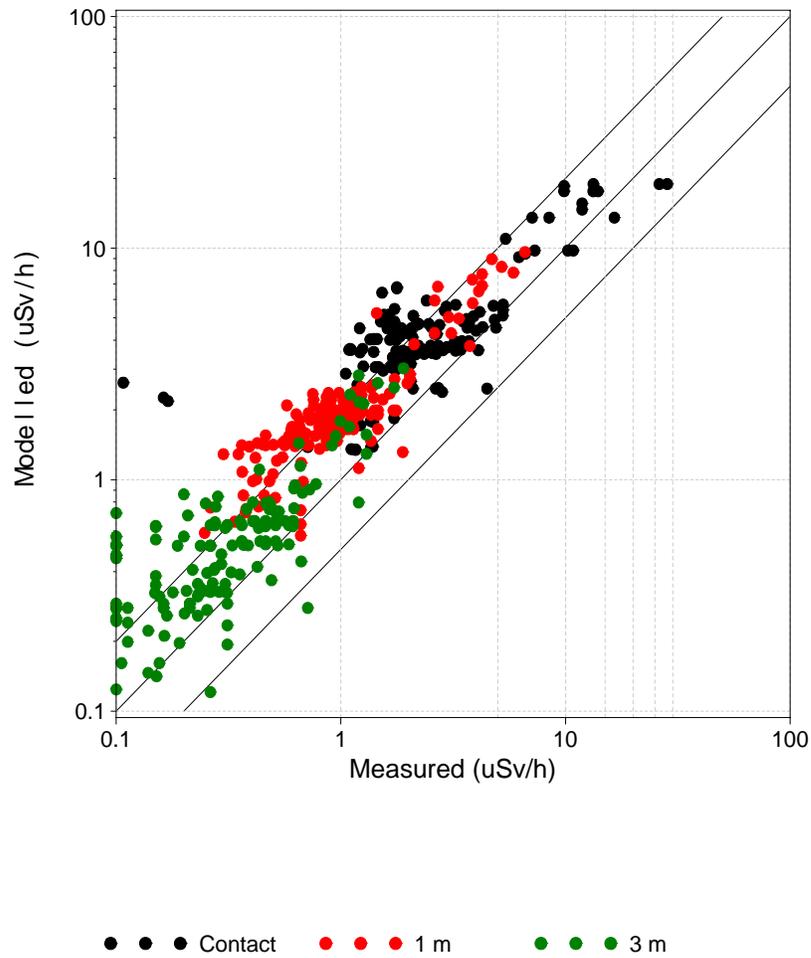
3.3.3 Comparison between Measured and Modelled Gamma Radiation Dose Rates

A comparison between measured and modelled gamma radiation dose rates for the same shipment was conducted. Figure 3.3 shows a scatter-plot of the modelled dose rate against the measured dose rates for the same shipment⁵. The middle diagonal line indicates perfect agreement between the modelled and measured dose rates with the two other diagonal lines showing agreement within a factor of two. For almost all comparisons, the modelled dose rate is higher than the measured dose rate. There is better agreement at a distance of 3 m than at 1 m or at contact⁶. Due to the variability in loading configuration (i.e. there can be empty spaces within the container), the gamma dose rate at different positions along and in contact with the container can give very different gamma dose rates, depending on whether the measurement is made immediately adjacent to an empty or filled space. Measurements at a distance from a container would be less sensitive to the loading configuration. A sample of a load configuration with empty spaces within the sea-land container is provided in Figure 3.4.

⁵ Does not include the two shipments where the bags were placed on the ground but not within a container.

⁶ Note that “contact” is contact with the sea-land container, not with a bag or drum.

FIGURE 3.3
MODELLED AND MEASURED ATTRIBUTABLE DOSE RATES



Based on primary laboratory measured concentrations of U-238 and Th-232. The figure does not include the two shipments that were in bags on the ground and not within a container.

FIGURE 3.4
EXAMPLE LOAD CONFIGURATION OF SEA-LAND CONTAINER

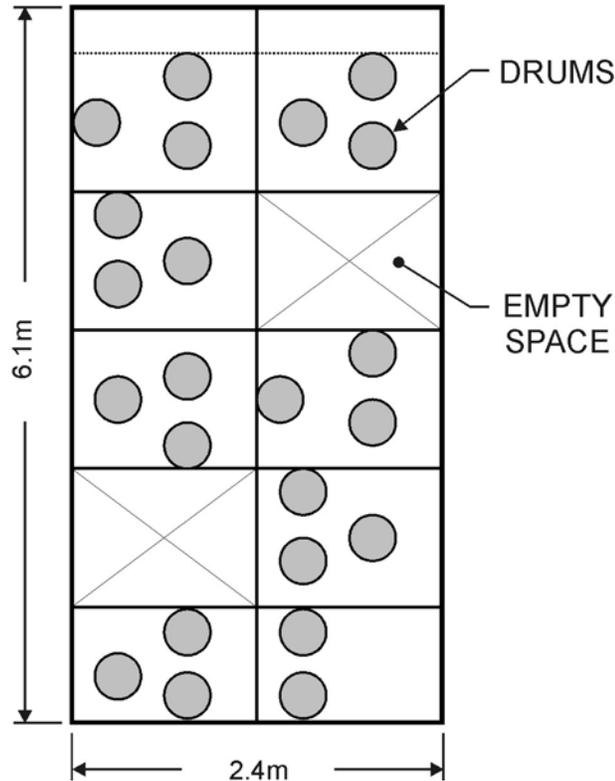


Table 3.6 summarizes the overall differences, irrespective of location or distance from the containers, between modelled and measured exposure rates. The mean modelled exposure rates at all locations for each loading configuration (configurations as reported by the companies) were compared to the mean measured exposure rates for all locations for the corresponding loading configuration. The table shows the percent difference of the modelled result relative to the measured result (i.e. $(\text{modelled} - \text{measured}) / \text{measured} * 100\%$) for the 53 shipments with both measured concentrations and measured gamma radiation exposure rates. On average, the modelled exposure rates are 45% higher than measured exposure rates for slag shipments and 84% higher than measured exposure rates for tantalite shipments. The closest agreement is for the F-1 Tier shipments modelled as full 1 Tier Loads with 25% and 28% overprediction relative to measured exposure rates for slag and tantalite shipments, respectively. The modelled exposure rates overestimate the measured exposure rates to a higher degree when a partial shipment i.e. (NF-1 Tier) was modelled as a full tier.

**TABLE 3.6
PERCENT DIFFERENCE (%) BETWEEN MEAN MODELLED AND MEASURED
ATTRIBUTABLE EXPOSURE RATES BY LOADING CONFIGURATION**

Statistic	Loading Configuration Used For Modelling			
	All	F - Trailer	F - 1 Tier	Maximum Load ^a
Slag				
All (n= 19)	45		35	56
F - 1 Tier (n= 4)			25	
F - 2 Tiers (n= 10) ^b				57
NF - 1 Tier (n= 1)			79	
NF - 2 Tiers (n= 4) ^b				56
Tantalite				
All (n= 34)	84	137	51	93
F - 1 Tier (n= 6)			28	
NF - 1 Tier (n= 4)			74	
NF - 2 Tiers (n= 22) ^b				93
NF Trailer (n= 2)		137		

a) Full first layer and half-full second layer.

b) As discussed in Section 3.1.2, since a sea-land container cannot contain two full tiers of tantalum material due to weight restrictions, all shipments reported as full two-tier (F-2 Tier) or partially full two-tier (NF - 2 Tier) were compared against the modelled exposure rates for Maximum Load (n = number of shipments).

Table 3.7 shows the percent difference in mean modelled and measured exposure rates by location using the modelled loading configurations identified in Table 3.6. The largest overestimation of measured exposure rates occurs on contact with the ends of the shipment containers with better overall agreement at a distance of 3 m.

**TABLE 3.7
PERCENT DIFFERENCE (%) BETWEEN MEAN MODELLED AND MEASURED
ATTRIBUTABLE EXPOSURE RATES BY LOCATION**

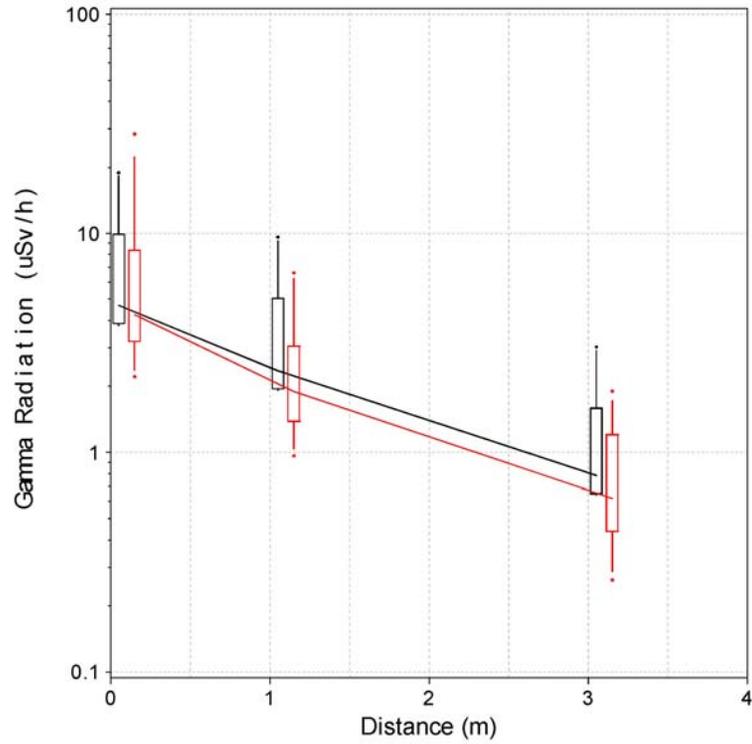
Statistic	Side			Corner			End		
	Contact	1 m	3 m	Contact	1 m	3 m	Contact	1 m	3 m
Slag Shipments									
All (n=19)	8	54	54	20	66	48	107	104	42
F - 1 Tier (n=4)	-13	61	71	2	73	55	59	97	51
F - 2 Tiers (n=10) ^a	23	42	50	39	56	58	135	96	21
NF - 1 Tier (n=1)	70	107	28	37	69	3	172	176	66
NF - 2 Tiers (n=4) ^a	19	36	51	33	63	63	157	92	41
Tantalite Shipments									
All (n=34)	46	97	84	66	99	75	128	135	76
F - 1 Tier (n=6)	-3	72	43	9	72	35	34	99	48
NF - 1 Tier (n=4)	56	89	23	62	76	12	124	133	15
NF - 2 Tiers (n=22) ^a	43	95	120	75	106	127	159	139	90
NF Trailer (n=2)	162	166	92	130	137	49	122	166	290

a) As discussed in Section 3.1.2, since a sea-land container cannot contain two full tiers of tantalum material due to weight restrictions, all shipments reported as full two-tier (F-2 Tier) or partially full two-tier (NF - 2 Tier) were compared against the modelled exposure rates for Maximum Load (n = number of shipments).

The MicroShield modelling tends to overestimate the gamma radiation exposure rates relative to gamma radiation exposure rates measured for the same shipment. The general overprediction of exposure rates is due to incomplete loading of the shipments compared to the loading assumed with modelling and the overestimation is more pronounced for tantalite materials compared to slag materials. The effect is most pronounced at the ends of the container.

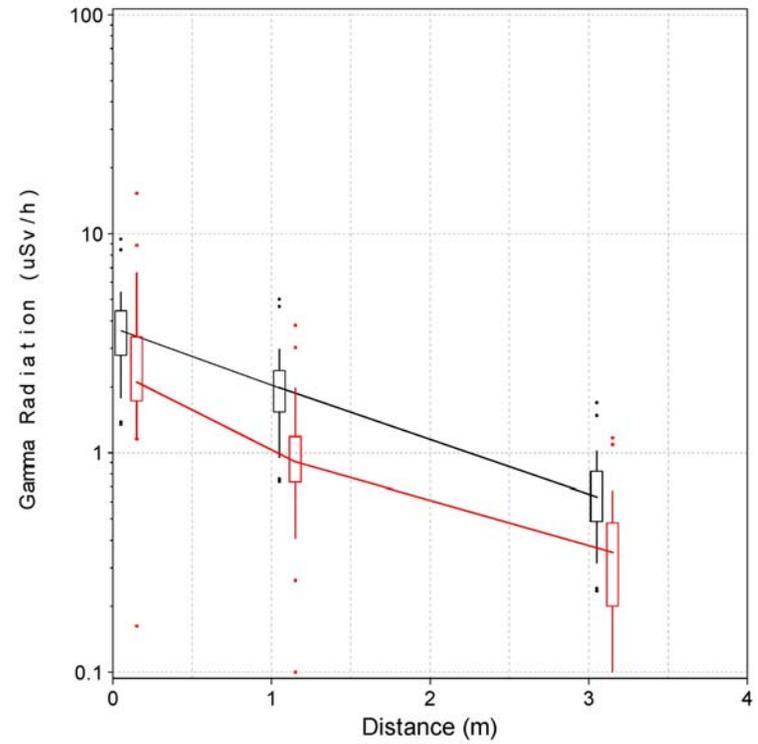
Figure 3.5 shows a comparison between the summary statistics for modelled and measured exposure rates for measurements at the side of the shipment. The figure shows that, as expected, gamma exposure rates decrease with distance from the container and that the modelled exposure rates tend to be higher than the measured exposure rates at all three distances. The figure also shows that the agreement between modelled and measured exposure rates at the side of the container is closer for slag materials than for tantalite materials and, to a lesser extent, that exposure rates tend to be higher for slag materials than for tantalite materials.

FIGURE 3.5
SUMMARY OF MODELLLED AND MEASURED EXPOSURE RATES
FROM THE SIDE OF THE CONTAINER



--* Microshield Model *-*-* Measured

a) Slag material



--* Microshield Model *-*-* Measured

b) Tantalite Material

3.4 PREDICTION OF GAMMA RADIATION EXPOSURE RATES FOR TRANSPORT SCENARIOS

Exposure rates for the transport risk assessment were developed assuming that all loads were fully (maximum) loaded (i.e. assumes a full first layer and a half-full second layer). As discussed in the previous section, the MicroShield modelling predicts exposure rates that tend to be higher than the measured exposure rates when a loading configuration more closely matching the actual loading was used. Assuming that all shipments have a maximum load configuration when modelling with MicroShield will further increase the average over-estimation of exposure rates (i.e. a conservative assumption).

Table 3.8 shows the percent difference between mean modelled exposure rates using the exposure factors for maximum loading configuration compared to the mean measured gamma radiation exposure factors. The overestimation tends to be higher for contact and 1 m distances at the ends of the container. This is thought to arise from the uncertainty in the location of tantalite materials within the container for partial loads.

**TABLE 3.8
PERCENT DIFFERENCE (%) BETWEEN MEAN MODELLED AND MEASURED
EXPOSURE RATES BY LOCATION FOR THE RISK ASSESSMENT**

Statistic	Side			Corner			End		
	Contact	1 m	3 m	Contact	1 m	3 m	Contact	1 m	3 m
Slag Shipments									
All (n=19)	47	109	115	58	120	104	206	172	84
F - 1 Tier (n=4)	45	168	202	61	178	169	218	220	130
F - 2 Tiers (n=10) ^a	23	42	50	39	56	58	135	96	21
NF - 1 Tier (n=1)	186	244	126	116	172	79	443	348	153
NF - 2 Tiers (n=4) ^a	19	36	51	33	63	63	157	92	41
Tantalite Shipments									
All (n=34)	83	145	127	106	147	123	204	192	112
F - 1 Tier (n=6)	68	187	154	80	182	141	168	226	126
NF - 1 Tier (n=4)	170	215	118	169	188	99	347	283	75
NF - 2 Tiers (n=22) ^a	43	95	120	75	106	127	159	139	90
NF Trailer (n=2)	292	298	145	238	263	129	300	307	449

a) As discussed in Section 3.1.2, since a sea-land container cannot contain two full tiers of tantalum material due to weight restrictions, all shipments reported as full two-tier (F-2 Tier) or partially full two-tier (NF - 2 Tier) were compared against the modelled exposure rates for Maximum Load (n = number of shipments).

3.5 SUMMARY

Gamma radiation exposure rates were measured on samples of material collected from 71 shipments of tantalum materials. Most of the materials were tantalite (concentrate); however there were a number of slag shipments. The shipments were primarily in sea-land containers with a small number of trailers. The slag shipments tended to have a lower bulk density compared to tantalite materials but a higher thorium (Th-232) content than tantalite materials. There was a tendency for sea-land containers to be more fully loaded with slag material than tantalite material probably due to the lower density of the slag material compared to tantalite

An interlaboratory comparison showed consistent measurements over the range of uranium concentrations and good agreement at high thorium concentrations. Measurements indicated that the activity concentrations of other radionuclides in the decay series have concentrations that are equal, or nearly equal, to the parent radionuclide (i.e. U-238 and Th-232). Based on the findings of good agreement between laboratories and the equilibrium with other radionuclides, the measurement of uranium and thorium alone by standard chemical analyses was considered adequate for radiological characterization of these materials.

The measured gamma radiation exposure rates around the shipments were variable as would be expected given the range of concentrations and loading configurations between shipments. These measured gamma radiation exposure rates were compared to modelled exposure rates at the same location using the MicroShield model and the measured concentrations in the particular shipment. Typically, the modelled gamma radiation exposure rates were higher than the measured exposure rates. This was due, in part, because the shipments were not completely loaded. Available information on the loading and limitations in the MicroShield model did not allow for modelling of these incomplete loads; therefore, the MicroShield modelling assumed the first layer was fully loaded. This leads to overestimation of exposure rates.

The MicroShield model was selected to estimate gamma radiation exposure rates for tantalum material transport in general based on the information from the sample of shipments. All shipments would be assumed to have the most fully loaded configuration (i.e. 1.5 tier, Maximum load). This is conservative in that it overestimates exposure rates compared to measured exposure rates; in many cases, this is a factor of two or more. However, the approach is reproducible and can be applied to varying concentrations of material and for calculation of exposure rates at locations and geometries not measured during the surveys of actual shipments.

4.0 DOSES ARISING FROM NORMAL TRANSPORT ACTIVITIES

4.1 GENERAL ASSUMPTIONS

As mentioned in Chapter 3, shipments of tantalum raw materials vary based on the type of material and packaging technique used by the shipper. The standard method of shipping tantalites is in drums on pallets in either sea-land containers or trailers, while slags are typically shipped in one tonne bags in sea-land containers.

The drums or bags can be placed in various configurations within the sea-land containers; however, for conservative purposes, a maximum load was assumed to be present in the sea-land container or trailer (trailer was assumed to be full) for calculating the doses from normal transport activities. The maximum load for the sea-land container was assumed to consist of a full bottom tier and a half-full 2nd tier, as 2 full tiers would exceed the weight restrictions for the sea-land container. The 2nd tier was assumed to be half the width of the 1st tier but was the entire length of the sea-land container, in order to maintain a balanced load.

The tantalum raw material (tantalites and slags) shipments can be transported by road (truck), rail or sea⁷. Therefore, exposure scenarios for transport workers, on-site facility workers⁸ and members of the public were evaluated in this assessment. The receptors assessed in this study were:

- Transport Worker – Truck Driver
- Transport Worker – Dockworker
- Transport Worker – Seaman
- Transport Worker – Trainman
- Facility Worker⁸ – Shipping & Receiving
- Public – Living Adjacent to Road
- Public – Living Adjacent to Rail

The parameter values used for each receptor exposed to a maximum load in the sea-land container or trailer are provided in Table 4.1.

⁷ A small portion of tantalum raw material shipments has been transported via air; however, there is limited information on this mode of transportation and the information was not available for this assessment.

⁸ For this assessment, workers associated with the loading and unloading of the sea-land containers were considered to be part of the on-site facility operations and were not considered to be transport workers. Their radiation exposures were considered to be subject to the appropriate regulatory requirements of the facilities. However, doses to these workers were assessed in this study to provide perspective on potential doses from tantalum raw materials.

**TABLE 4.1
PARAMETER VALUES ^a USED FOR NORMAL TRANSPORT ACTIVITIES**

Scenario	# of Containers per Month (containers/month)	Time Spent per Container (h/container)	Fraction of Time at 1 m from container (Unitless)	Fraction of Time within 1 m from tantalite/slag (Unitless)	Trips per Month (per driver) (trips/month)	Time Loaded per Trip (h/trip)	Time Within 3 m of container (min/container)	Time within 1 m from container (min/container)	Time Stopped at Traffic Light (min/truck)	# of Trucks ^b per Month (trucks/month)	Fraction of Trucks Stopped at Traffic Light (Unitless)	# Trains ^c per Month (trains/month)	Time Stopped on Rail (min/train)	Fraction of Trains Stopped on Rail (Unitless)
Transport Worker - Truck Driver	-	-	-	-	3	10	-	-	-	-	-	-	-	-
Transport Worker - Dockworker	6	-	-	-	-	-	5	5	-	-	-	-	-	-
Transport Worker - Seaman	6	-	-	-	-	-	5	-	-	-	-	-	-	-
Transport Worker - Trainman	6	-	-	-	-	-	10	2	-	-	-	-	-	-
Facility Worker - Shipping & Receiving	6	1	0.5	0.5	-	-	-	-	-	-	-	-	-	-
Public - Living Adjacent to Road	-	-	-	-	-	-	-	-	3	6	0.5	-	-	-
Public - Living Adjacent to Rail	-	-	-	-	-	-	-	-	-	-	-	6	5	0.5

- a) All parameter values provided in this table were agreed upon during discussions with the T.I.C. Transport Committee (December 2005 and January 2007).
 b) Assumes 1 sea-land container per truck, but 2 drivers (i.e. 6 trucks/month corresponds to each driver transporting 3 sea-land containers per month by truck).
 c) Assumes 1 sea-land container per train (i.e., 6 trains/month corresponds to transporting 6 sea-land containers per month by train).

4.2 PREDICTED GAMMA RADIATION RATE FOR EXPOSURE SCENARIOS

4.2.1 Gamma Radiation Exposure Rates during Normal Transport

The only significant exposure pathway during transport is from gamma radiation exposure. For each transport dose scenario, the locations where the person would be exposed to the gamma radiation were identified. MicroShield modelling was used to predict the dose factors at these locations for the combinations of two types of materials and the two radionuclide series (U-238 and Th-232). These dose factors are described in Annex E.

The annual dose factors per annual average Bq/g in the shipments were calculated by multiplying the hourly dose rate factors by the duration of time spent each year at that location. The parameter values for duration of time and the subsequent annual dose factors are provided in Tables 4.1 and 4.2, respectively. The annual dose factors are slightly higher for slag than those for tantalite due to the different source compositions of the materials.

**TABLE 4.2
ANNUAL DOSE FACTORS FOR NORMAL TRANSPORT ACTIVITIES**

Activity	Effective Shipments per Year	Material Type	Annual Dose Factor (mSv/y per Bq/g)	
			U-238	Th-232
Transport Worker - Truck Driver	36	Tantalite	8.47E-03	1.24E-02
		Slag	8.55E-03	1.24E-02
Transport Worker - Dockworker	72	Tantalite	1.09E-03	1.59E-03
		Slag	1.12E-03	1.62E-03
Transport Worker - Seaman	72	Tantalite	1.41E-04	2.07E-04
		Slag	1.43E-04	2.07E-04
Transport Worker - Trainman	72	Tantalite	6.60E-04	9.66E-04
		Slag	6.74E-04	9.81E-04
Facility Worker - Shipping & Receiving	72	Tantalite	1.68E-02	2.46E-02
		Slag	1.72E-02	2.51E-02
Public - Living Adjacent to Road	36	Tantalite	1.31E-05	1.93E-05
		Slag	1.35E-05	1.97E-05
Public - Living Adjacent to Rail	36	Tantalite	5.67E-06	8.36E-06
		Slag	5.84E-06	8.56E-06

Note: Effective shipments are the number of shipments with potential for exposure – see Table 4.1 for assumptions. Facility workers are not considered to be transport workers (see footnote 8).

4.3 ANNUAL DOSES

The annual dose received during transport is the sum of doses from multiple shipments during the year. The exposure scenarios assume a number of shipments and the duration of time an individual would be exposed at various locations around the shipment and these factors, along

with MicroShield model predictions, were used to develop dose factors that quantify the dose received per year per Bq/g of either U-238 or Th-232 in the shipment.

In general, the annual dose was calculated by multiplying the mean concentration in those shipments by the annual dose factor. For example if a truck driver transported 36 shipments of tantalite with a U-238 concentration of 10 Bq/g, the calculated annual dose for the truck driver would be 0.085 mSv/y (i.e. 0.0085 mSv/per Bq/g from Table 4.2 times 10 Bq/g). Since the uranium and thorium concentrations in the shipments are variable, the mean concentrations in the 36 shipments will also be variable. To account for this variability, probabilistic simulation was used to develop the distribution of mean concentrations in the groups of shipments encountered during a year and, by multiplying this average by the dose factors, to quantify the distribution of annual doses. Additional details on the probabilistic simulation are provided in Annex E.

Table 4.3 shows a summary of the average U-238, Th-232 and total concentrations for the two numbers of shipments (36 or 72) considered in the dose scenarios. For example, the upper bound for average U-238 concentrations in 36 slag shipments is 27 Bq/g compared to the mean of 19 Bq/g.

**TABLE 4.3
SUMMARY OF ANNUAL AVERAGE CONCENTRATIONS FOR NORMAL
TRANSPORT ACTIVITIES**

	U-238 (Bq/g)	Th-232 (Bq/g)	Total (Bq/g)
Mean			
Slag	19	6.5	25
Tantalite	17	1.3	18
Upper Bound (36 shipments)			
Slag	27	8.0	33
Tantalite	20	2.0	21
Upper Bound (72 shipments)			
Slag	24	7.6	31
Tantalite	19	1.8	20

Note:

- i) Upper bound is the 95th percentile from the probabilistic distribution of annual average concentration in the shipments.

Table 4.4 summarizes the upper 95th percentile of doses calculated in the probabilistic trials arising from the variation in radioactivity content between shipments. For transport workers, facility workers and members of the public with equal likelihood of being exposed from materials shipped from anywhere in the world, the 95th percentile is the dose that would be unlikely to be exceeded (e.g. only one time in 20). The table includes, in brackets, the calculated mean doses. Annual doses from slag shipments tend to be higher than the annual dose from tantalite due, in part, to the higher total activity in slag materials and the higher proportion of thorium content compared to uranium content in the slag materials. The highest dose to transport workers or members of the public was to the truck drivers; however, these doses are well below 1 mSv.

TABLE 4.4
SUMMARY OF DOSES (mSv/y) FOR NORMAL TRANSPORT ACTIVITIES

Receptor Scenario	Slag	Tantalite
Transport Worker - Truck Driver	0.31 (0.24)	0.19 (0.16)
Transport Worker - Dockworker	0.038 (0.032)	0.023 (0.02)
Transport Worker - Seaman	0.0048 (0.0041)	0.0030 (0.0026)
Transport Worker - Trainman	0.023 (0.019)	0.014 (0.012)
Public - Living Adjacent to Road	0.00048 (0.00038)	0.00029 (0.00024)
Public - Living Adjacent to Rail	0.00021 (0.00017)	0.00013 (0.0001)

Notes:

- a) 95th percentile doses based on 2,000 simulations.
- b) Values in brackets are mean values.
- c) Seaman could spend an extra 5 minutes per trip checking the lashings on the containers; however, this would result in an insignificant increase to the dose.
- d) Warehouse staff at the docks were included in the dockworker scenario.
- e) Doses to facility workers (who are not considered transport workers) were calculated to be 0.58 (0.49) mSv/y and 0.35 (0.31) mSv/y from slag and tantalite, respectively.

4.4 UNCERTAINTY

The dose calculations are somewhat uncertain due to a number of factors. Conservatism, or a tendency to overestimate the doses, has been introduced by the MicroShield modelling. The loading configuration has been assumed to be the maximum loading; however, many shipments do not have as much material inside as modelled for this configuration. As a result, the modelled exposure rates will tend to be higher than the actual exposure rates for many loads. The overestimation is more pronounced for tantalite compared to slags as tantalite materials have a higher density; consequently, the tantalite shipments are even less likely than shipments of slag materials to be fully loaded..

The exposure locations and time spent at those locations have been assumed. It was felt that these assumptions were not likely to underestimate the dose; for example, given the small total

quantity of tantalum produced annually, from all the different places where it is mined, it is not possible for one driver to be occupied full time transporting only tantalum materials.

The range of concentrations of radioactivity in the sampled shipments is well known based on the number of shipments measured and the good agreement in the interlaboratory comparison of measurements of uranium and thorium.

The tantalite shipments that provide the information for this study were not selected using a statistical sampling methodology. Nonetheless, it is the opinion of the T.I.C. Transport Committee that the information from these shipments is reasonably representative of the range of concentrations for all tantalum shipments and suitable for the dose assessment.

Based on these considerations of sources of uncertainty, the calculated doses for normal transport activities are unlikely to be exceeded.

4.5 POTENTIAL EXEMPTION VALUE FOR THE TRANSPORT OF TANTALUM RAW MATERIALS

4.5.1 Calculated Doses at Various Potential Exemption Values

U-238 and Th-232 Doses

As mentioned previously, the current IAEA exemption value for the transport of tantalum raw materials is 10 Bq/g (U-238 + Th-232 combined, with decay products in radioactive equilibrium within each decay series). In order to determine the appropriateness of this exemption value, the annual doses from the transport of slag and tantalite were calculated at different activities of U-238 and Th-232 (separately and combined), with each activity representing a potential exemption value. The potential exemption values (i.e. activities) used were 10, 30, 50, 70, 90 and 100 Bq/g. The annual doses from the transport of tantalite or slag were calculated by multiplying the exemption value by the corresponding annual dose factor per Bq/g (Table 4.2). The calculated annual doses for U-238 and Th-232 at each potential exemption value are provided in Table 4.5. The doses to members of the public from transport of tantalum raw materials were much lower than those to transport workers (see Table 4.4) and therefore were not included in the development of potential exemption values.

TABLE 4.5
SUMMARY OF U-238 AND Th-232 ANNUAL DOSES (mSv/y) FOR POTENTIAL EXEMPTION VALUES

Potential Exemption Value (Bq/g)	Material	Annual U-238 Dose (mSv/y) at Potential Exemption Value ^{a, b, c}				Annual Th-232 Dose (mSv/y) at Potential Exemption Value ^{a, b, c}			
		Truck Driver	Dock Worker	Seaman	Trainman	Truck Driver	Dock Worker	Seaman	Trainman
10	Tantalite	0.085	0.011	0.0014	0.0066	0.12	0.016	0.0021	0.0097
	Slag	0.086	0.011	0.0014	0.0067	0.12	0.016	0.0021	0.0098
30	Tantalite	0.25	0.03	0.0042	0.020	0.37	0.048	0.0062	0.029
	Slag	0.26	0.03	0.0043	0.020	0.37	0.049	0.0062	0.029
50	Tantalite	0.42	0.05	0.0071	0.033	NA	NA	NA	NA
	Slag	0.43	0.06	0.0071	0.034	NA	NA	NA	NA
70	Tantalite	0.59	0.08	0.0099	0.046	NA	NA	NA	NA
	Slag	0.60	0.08	0.010	0.047	NA	NA	NA	NA
90	Tantalite	0.76	0.10	0.013	0.059	NA	NA	NA	NA
	Slag	0.77	0.10	0.013	0.061	NA	NA	NA	NA
100	Tantalite	0.85	0.11	0.014	0.066	NA	NA	NA	NA
	Slag	0.86	0.11	0.014	0.067	NA	NA	NA	NA

- a) Assume 1.5 tiers of material.
- b) The annual dose to members of the public (Living Adjacent to Road, Living Adjacent to Rail) for all cases (Annual U-238 Dose, Annual Th-232 Dose & Combined Dose) at each potential exemption value is less than 10 µSv.
- c) Doses to facility workers (who are not considered transport workers) ranged from 0.17 to 1.7 mSv/y and 0.25 to 0.75 mSv/y for U-238 and Th-232, respectively.
- NA - Not applicable since the maximum Th-232 concentration (Bq/g) in all shipments was less than 30 Bq/g (27.8 Bq/g).

As noted in Table 4.5, there were no shipments with Th-232 measuring over 30 Bq/g; therefore, the annual doses from Th-232 for potential exemption values over 30 Bq/g were not calculated. However, the annual dose from U-238 was calculated for each potential exemption value since measurements of the shipments analyzed in this assessment showed that the U-238 activity could exceed 90 Bq/g.

Combined Annual Doses

Since tantalum materials contain both U-238 and Th-232, the annual doses from both U-238 and Th-232 at each potential exemption value for tantalites and slags were calculated. Furthermore, since the annual dose factor per Bq/g of Th-232 is larger than that for U-238, to be conservative, the combined annual dose was calculated based on the maximum Th-232 activities measured in this assessment.

For example, for slags, the maximum Th-232 activity in the shipments was measured at 27.8 Bq/g (or nominally 30 Bq/g). Therefore, for the potential exemption values of 10 and 30 Bq/g, the combined activity was assumed to be 100% Th-232, which would result in the highest annual dose. The combined activities for higher potential exemption values (50 Bq/g, 70 Bq/g, 90 Bq/g and 100 Bq/g) were calculated assuming 30 Bq/g of Th-232 with the remaining activity from U-238. For example, for a potential exemption value of 50 Bq/g, the activity was assumed to consist of 30 Bq/g of Th-232 plus 20 Bq/g of U-238.

For tantalites, the maximum Th-232 activity in the shipments was measured at 11.1 Bq/g (nominally 15 Bq/g). Therefore, for the exemption value of 10 Bq/g, the total activity was assumed to be 100% Th-232, which would result in the highest annual dose. The combined activities for the higher potential exemption values (30 Bq/g, 50 Bq/g, 70 Bq/g, 90 Bq/g and 100 Bq/g) were calculated assuming 15 Bq/g of Th-232 with the remaining activity from U-238.

The combined annual doses for shipments of tantalite and slag at each potential exemption value are summarized in Table 4.6.

TABLE 4.6
SUMMARY OF COMBINED ANNUAL DOSES (mSv/y) AT POTENTIAL EXEMPTION VALUES

Potential Exemption Value (Bq/g)	Material	% of Shipments That Exceed Potential Exemption Value	Combined Annual Dose (mSv/y) at Potential Exemption Value ^{a, b, c}			
			Truck Driver	Dock Worker	Seaman	Trainman
10	Tantalite	78	0.12	0.016	0.0021	0.0097
	Slag	45	0.12	0.016	0.0021	0.0098
30	Tantalite	16	0.31	0.040	0.0052	0.024
	Slag	27	0.37	0.049	0.0062	0.029
50	Tantalite	2	0.48	0.062	0.0080	0.038
	Slag	23	0.54	0.071	0.0091	0.043
70	Tantalite	0	0.65	0.084	0.011	0.051
	Slag	9	0.71	0.093	0.012	0.056
90	Tantalite	0	0.82	0.11	0.014	0.064
	Slag	9	0.89	0.12	0.015	0.070
100	Tantalite	0	0.91	0.12	0.015	0.071
	Slag	0	0.97	0.13	0.016	0.077

- a) Assume 1.5 tiers of material.
- b) The annual dose to members of the public (Living Adjacent to Road, Living Adjacent to Rail) for all cases. (Annual U-238 Dose, Annual Th-232 Dose & Combined Dose) at each potential exemption value is less than 10 µSv.
- c) Doses to facility workers (who are not considered transport workers) ranged from 0.25 (at 10 Bq/g) to 1.8 mSv/y (at 100 Bq/g) and 0.25 (at 10 Bq/g) to 2.0 mSv/y (at 100 Bq/g) for tantalite and slag, respectively.

As shown in Table 4.6, the highest combined annual doses (U-238 and Th-232) to any of the transport workers were to the truck driver, ranging from 0.12 mSv/y (for 10 Bq/g) to 0.97 mSv/y (for 100 Bq/g). It should be noted that these combined annual doses are very conservative since they were based on the assumption that each exempted shipment the transport worker handles is at the potential exemption value; under actual conditions, each transport worker would handle a variety of exempted shipments at different exemption activity concentrations less than the exemption value. Therefore, in addition to the combined annual doses, Table 4.6 includes the cumulative percentage of the shipments that exceed the potential exemption values. For example, 27% of the slag shipments had activity levels greater than 30 Bq/g, while only 16% of the tantalite shipments had activities greater than 30 Bq/g.

To illustrate the effect of this conservative assumption, the doses for the most exposed transport worker, the truck driver, were re-calculated using specific shipment information obtained in this assessment. The same probabilistic simulation methodology used to estimate the total dose from normal transport activities in Section 4.3 was used to estimate the doses from those shipments encountered during the year with total concentrations below potential exemption values.

A probabilistic sample of concentrations in shipments was selected for each trial from the overall distribution of concentrations and the dose from those loads with total concentrations at or below the potential exemption values was determined. A summary of the doses from these shipments and the mean activity in those loads for the truck driver are shown in Table 4.7. For example, the upper 95th percentile on dose from slag shipments at or below 10 Bq/g was 0.07 mSv/y, while the upper 95th percentile on the mean activity in these loads was 9.4 Bq/g.

TABLE 4.7
ANNUAL DOSES (mSv/y) TO TRUCK DRIVER FROM SHIPMENTS AT
POTENTIAL EXEMPTION VALUES ^{a, b}

Potential Exemption Value (Bq/g)	Slag		Tantalite	
	Dose (mSv)	Mean Activity (Bq/g)	Dose (mSv/y)	Mean Activity (Bq/g)
10	0.07 (0.05)	9.4 (9.2)	0.02 (0.01)	7.6 (6.9)
20	0.09 (0.08)	10.8 (10.2)	0.08 (0.06)	12.3 (11.1)
30	0.09 (0.08)	10.8 (10.2)	0.12 (0.10)	15.4 (13.6)
40	0.12 (0.10)	13.7 (11.7)	0.16 (0.13)	18.2 (15.9)
50	0.12 (0.10)	13.7 (11.7)	0.17 (0.14)	19.4 (16.6)
60	0.18 (0.14)	19.9 (15.8)	0.17 (0.14)	19.4 (16.6)
70	0.21 (0.17)	23.9 (18.6)	0.19 (0.16)	21.3 (17.8)
80	0.21 (0.17)	23.9 (18.6)	0.19 (0.16)	21.3 (17.8)
90	0.21 (0.17)	23.9 (18.6)	0.19 (0.16)	21.3 (17.8)
All	0.31 (0.24)	33.1 (25.4)	0.19 (0.16)	21.3 (17.8)

a) Values are the upper 95th percentiles based on 2,000 simulations.

b) Values in brackets are the mean values from the simulations.

The results shown in Table 4.7 indicate that at any exemption value, the expected mean dose to truck drivers transporting tantalum raw materials would be 0.24 mSv/y, with an upper 95th percentile of 0.31 mSv/y.

4.5.2 Selection of an Exemption Value for the Transport of Tantalum Raw Materials

From the analyses described above, no transport worker would be expected to receive a dose exceeding 1 mSv/y arising from the transport of tantalum raw materials. On this basis, the current exemption value of 10 Bq/g (U-238 + Th-232 combined, where decay products are in equilibrium) seems to be too restrictive for the transport of tantalum raw materials. Indeed, even in the absence of an exemption value, no one would be expected to receive a dose in excess of 1 mSv/y arising from the transport of tantalum raw materials based on the current measured concentrations.

However, since the radioactivity levels of future shipments of tantalum raw materials may not be similar to the distribution examined in this assessment, the practical application of the transport regulations requires an exemption value for such materials. The following reasoning was used for selecting a potential exemption value.

The method that was used to select an exemption value was first to conservatively assume that the radioactivity levels of all future exempted shipments of tantalum raw materials would be at the selected specific exemption value. This conservative assumption would result in an overestimate of the annual doses for actual transport situations (i.e. different exempted shipments would have different (lower) radioactivity concentrations). The calculated doses to transport workers under this assumption are as summarized in Table 4.6.

In order to account for the possibility of other transport-related exposures (those not related to the transport of tantalum raw materials), a dose constraint lower than the dose criterion of 1 mSv/y should be considered as a reference dose level. For present purposes, a dose constraint of 0.3 mSv/y is suggested as being appropriate to account for such potential exposures⁹. Table 4.6 shows that an exemption value of at least 30 Bq/g would result in doses to truck drivers (the most exposed transport workers) of 0.31 to 0.37 mSv/y, if all the exempt tantalum raw materials were at 30 Bq/g, an unlikely assumption. Therefore, an exemption value of 30 Bq/g (U-238 + Th-232) would result in doses that would be unlikely to exceed 0.3 mSv/y to the most exposed transport worker. For that reason, an exemption value of at least 30 Bq/g is considered appropriate for the transport of tantalum raw materials.

⁹ Recently, the ICRP (2007) supported its previous recommendation (ICRP 1997) of using 0.3 mSv/y as a dose constraint for members of the public with respect to multiple exposures in the context of radioactive waste disposal. The final ICRP recommendations, approved 21 March 2007, are expected to be published in the Annals of the ICRP.

5.0 DOSES ARISING FROM TRANSPORT SPILLS AND ACCIDENTS

5.1 GENERAL ASSUMPTIONS

Through discussions with the T.I.C. Transport Committee, it was confirmed that there have never been any reported experiences with a sea-land container transporting tantalum raw material breaking open and having drums (or bags) break open and spill out of the container. The only accidents that have occurred in the past have been drums (or bags) breaking open within the sea-land container. Therefore, this section assesses the radiological dose to transport workers from this type of accident.

During the clean-up of the tantalum materials, the transport workers will receive radiological doses from external gamma radiation, ingestion and inhalation. The clean-up of the materials will be for a relatively short time period (assumed here to be 5 hours); therefore, the external gamma radiation received during this time period will be insignificant, compared to annual doses calculated in Chapter 4 (most exposed transport worker, truck driver, exposed for 360 h/y). A discussion on the radiological dose received by the transport workers from the ingestion and inhalation of the tantalum materials is provided in Section 5.2.

5.2 INGESTION AND INHALATION DOSE

The transport workers were considered to have the potential to ingest or inhale the tantalum materials under accident conditions. The ingestion or inhalation of tantalum materials by the transport workers would occur while cleaning up the opened drums (or spilled bags) inside the sea-land container. For the development of the ingestion and inhalation doses under accident conditions to the transport workers, it was necessary to accommodate the varying concentrations (Bq/g) of the tantalum materials as well as the different hours of exposure for the various workers along the transportation route.

The ingestion and inhalation dose factors were derived from the dose coefficients (DCs) of the International Commission on Radiological Protection (ICRP). The ICRP DCs are internationally-accepted, including by the IAEA (1996), and are used throughout the world for radiation exposure calculations. The calculation of the ingestion and inhalation DCs for workers for the natural uranium and thorium series radionuclides of interest to this study are shown in Table 5.1. Other radionuclides in the U-238 and Th-232 series were not included in Table 5.1 because they are very small contributors ($\ll 1\%$) to the total inhalation and ingestion DCs.

The DCs calculated for uranium and thorium assume that the radionuclides in the U-238 and U-235 decay series in natural uranium and in the Th-232 series in natural thorium are in radioactive equilibrium within their respective decay series; that is, the activity concentration (Bq/g) of each radionuclide is equal. The radiological analysis of the tantalum materials (Section

3.2) generally supports this assumption. (The small disequilibrium in the samples, wherein the concentrations of some of the radionuclides are lower (by 10-20%) than the U-238 or Th-232 concentrations, would result in lower doses than the assumption of equilibrium used in this analysis.) The inhalation DCs also assume that the tantalum materials are relatively insoluble as characterized by the ICRP inhalation types S (slow), M (moderate) and F (fast) which relate to the removal rate of the radionuclides from the lungs. This assumption generally results in larger DCs. The ICRP default particle size of 5 microns (5×10^{-6} m) for occupational exposures was also assumed (as characterized by the activity median aerodynamic diameter or AMAD). The f_1 factors for ingestion in Table 5.1 are also for the relatively insoluble forms of the radionuclides and refer to the transfer fraction from the gut to the blood of the ingested radionuclides. The f_1 factors are related to the doses ultimately received from the inhaled and ingested radionuclides.

The DCs were converted to unit exposure duration and unit concentration by assuming nominal worker ingestion rates of 100 mg per day (assumed 8 h to determine an hourly rate) and a respirable airborne dust concentration of $100 \mu\text{g}/\text{m}^3$ (Hofmann et al. 2000), a relatively dusty situation, for workers in proximity to the concentrates. [The tantalite materials are relatively large grained (non-respirable) and not dusty. Based on measurements by a T.I.C. member company of the particle size distribution in two tantalite samples, the respirable component ($<10 \mu\text{m}$) is $\leq 1\%$.] Relative to ingestion, the U.S. EPA (1997) recommends using an ingestion rate of 50 mg/day for industrial settings, but has used 100 mg/day for agricultural scenarios. The inhalation rate of $1.2 \text{ m}^3/\text{h}$ is the ICRP default inhalation rate for workers. While these ingestion and dust concentration values are necessarily uncertain, they are considered to be reasonably conservative approximations for worker exposures, especially considering the relatively coarse-grained nature of the tantalum materials.

TABLE 5.1
INGESTION AND INHALATION DOSE FACTORS FOR WORKERS FOR URANIUM
AND THORIUM SERIES RADIONUCLIDES

Radionuclide	Inhalation ^a		Ingestion ^a	
	Type ^d	DC(Sv/Bq)	f ₁ ^c	DC(Sv/Bq)
U-238	S	5.7E-06	0.002	7.6E-09
Th-234	S	5.8E-09	2.0E-04	3.4E-09
U-234	S	6.8E-06	0.002	8.3E-09
Th-230	S	7.2E-06	2.0E-04	8.7E-08
Ra-226 ^d	M	2.2E-06	0.2	2.8E-07
Pb-210	F	1.1E-06	0.2	6.8E-07
Bi-210	M	6.0E-08	0.05	1.3E-09
Po-210	M	<u>2.2E-06</u>	0.1	<u>2.4E-07</u>
U-238 series		2.5E-05		1.3E-06
U-235	S	6.1E-06	0.002	8.3E-09
Pa-231	S	1.7E-05	5.0E-04	7.1E-07
Ac-227	S	4.7E-05	5.0E-04	1.1E-06
Th-227	S	7.6E-06	2.0E-04	8.4E-09
Ra-223	M	<u>5.7E-06</u>	0.2	<u>1.0E-07</u>
U-235 series		8.3E-05		1.9E-06
U-nat ^e		2.91E-05		1.40E-06
Dose (mSv per hour of exposure per Bq/g U-238) ^{f, g}		3.5E-06		1.7E-05
Th-232	S	1.2E-05	2.0E-04	9.2E-08
Ra-228	M	1.7E-06	0.2	6.7E-07
Ac-228	M	1.2E-08	5.0E-04	4.3E-10
Th-228	S	3.2E-05	2.0E-05	3.5E-08
Ra-224	M	2.4E-06	0.2	6.5E-08
Pb-212	F	3.3E-08	0.2	5.9E-09
Bi-212	M	<u>3.9E-08</u>	0.05	<u>2.6E-10</u>
Th-232 series		4.82E-05		8.69E-07
Dose (mSv per hour of exposure per Bq/g Th-232) ^{f, g}		5.8E-06		1.1E-05

a. DCs from ICRP 68 (1994). Inhalation DCs for 5 micron AMAD particles.

b. ICRP inhalation types S (slow), M (moderate) and F (fast).

c. ICRP gut-to-blood transfer factor.

d. Revised inhalation DC for Ra-226 from ICRP 72 (1996), Annexe B.

e. Based on U-235 activity = 4.6% U-238 activity in natural uranium (U-nat).

f. Hourly inhalation DC based on 100 µg/m³ and 1.2 m³/h inhalation rate.

g. Hourly ingestion DC based on 100 mg/ 8 h ingestion rate.

The dose factors (mSv per hour of exposure per Bq/g) provided in Table 5.1 can be used to calculate the annual inhalation and ingestion doses to transport workers that participate in the clean-up activities of tantalum material. These dose factors would be combined with the corresponding clean-up time, number of clean-ups per year and the activity of the tantalum material. The annual ingestion and inhalation doses to transport workers who participate in clean-up of tantalum materials of 30 Bq/g for 5 hours in one year is provided in Table 5.2.

TABLE 5.2
INGESTION AND INHALATION DOSES TO WORKERS FROM CLEAN-UP OF
SPILLED MATERIALS

Radionuclide	Annual Dose (mSv/y)	
	Inhalation	Ingestion
U-238 ^a	5.3E-04 (0.53 µSv/y)	2.6E-03 (2.6 µSv/y)
Th-232 ^a	8.7E-04 (0.87 µSv/y)	1.7E-03 (1.7 µSv/y)

a) Includes radioactive decay products in equilibrium.

As shown in Table 5.2, the annual doses to transport workers that clean up 30 Bq/g (U-238 or Th-232 and decay products) of tantalum material for 5 hours per year is less than 10 µSv/y. Therefore, the inhalation and ingestion doses to transport workers from the clean-up of tantalum materials should not be a regulatory concern and can be considered insignificant.

6.0 OBSERVATIONS AND CONCLUSIONS

T.I.C. member companies require the transport of tantalum raw materials (mainly tantalite and tin slag). These materials contain varying levels of naturally occurring radioactivity, namely uranium and thorium and associated radioactive decay products. The current IAEA exemption value for the transport of NORM is 10 Bq/g (U-238 + Th232 combined, with decay products in radioactive equilibrium) provided such materials are not intended to be processed for the use of the naturally occurring radionuclides. Below this value, the IAEA transport regulations do not apply. This value is 10 times larger than the 1 Bq/g exemption value in the IAEA Basic Safety Standards (BSS, IAEA 1996) derived on the basis that the dose to any member of the public would not exceed 10 μ Sv/y (0.01 mSv/y). The initial derivation of the BSS exemption values did not specifically address transport scenarios; however, additional calculations for transport scenarios showed relatively small differences that did not justify different exemption values in the transport regulations (IAEA 2002). The 10 μ Sv/y dose rate is defined in the BSS as being “...sufficiently low as to be of no regulatory concern”.

From our investigations, the rationale for the factor of 10 used to derive exemption values for the transport of NORM is not explicit but rather appears to be an arbitrary consensus that balances radiological protection concerns and the impracticality and inconvenience of regulating large amounts of low activity NORM. An IAEA Coordinated Research Programme (CRP) related to the transport of NORM refers to a 2003 IAEA International Conference on the Safety of Transport of Radioactive Material noting “*The Conference suggested that the full impact of and technical basis for the ‘factor of 10’ exemption be thoroughly researched.*” (IAEA 2005b)

The International Commission on Radiological Protection (ICRP) has suggested a 1 mSv/y dose criterion for regulating doses to members of the public “...*incurred as the result of practices.*” (ICRP 1990). Recently, in its latest draft recommendations (ICRP 2007), the ICRP refers to additional guidance on the scope of radiological protection regulations (ICRP 2006). This includes a discussion of the basis for exempting radionuclides from regulatory control practices. The ICRP (2006) suggests “*For exemption of situations involving naturally occurring radioactive material...the dose criterion could justifiably be established in the order of 1 mSv in a year.*”

For the present study, both the radioactivity content of the various tantalum materials and the potential doses to the transport workers and members of the public arising from the transport of tantalum material were evaluated.

A characterization program was conducted to determine the radioactivity present in the tantalum materials and the gamma radiation levels associated with shipments of these materials. This was

done not only to characterize the radioactivity in the materials but also to support the estimation of the doses arising from the transport of these materials.

The observations and conclusions from the study are as follows:

- Based on analyses of the tantalum raw materials being transported, radioactive equilibrium within each of the natural uranium (U-238) and thorium (Th-232) decay series was found to be a reasonable assumption for dose assessment purposes.
- A range of about a factor of 10 in radioactivity concentrations was measured in 67 shipments of tantalite and slag, with an average activity concentration (U-238+ Th-232 combined) of about 20 Bq/g for tantalite and about 25 Bq/g for slag. The majority (78%) of tantalite shipments and 45% of the slag shipments had concentrations exceeding 10 Bq/g.
- Exposure scenarios that considered both duration and location of exposure were established for several types of transport workers (e.g. truck driver, trainman) and for members of the public. Based on an evaluation of potential exposure pathways, exposure to gamma radiation was determined to be the only significant exposure pathway.
- Doses from exposure to spilled materials due to potential accidents were calculated and determined not to be a regulatory concern, as the resulting doses were less than 10 μ Sv/y.
- An assessment of potential dose rates around the transport containers was conducted using the range of measured radioactivity concentrations and modelling of the associated gamma radiation doses using the MicroShield model. The modelled results were compared to measurements. The modelling approach overestimated the measured dose rates, primarily due to the assumption that the transport containers were always considered to be a full 1 tier or 1.5 tier load (whereas in practice the loading pattern varies, Sections 3.1.2 and 3.3.3).
- Calculated doses to transport workers and members of the public based on the distribution of measured concentrations are shown in Table 6.1.

TABLE 6.1
SUMMARY OF DOSES CALCULATED FOR NORMAL (NON-ACCIDENTAL)
TRANSPORT ACTIVITIES

Exposure Scenario	Mean Dose ^{a, b} (mSv/y)	
	Slag	Tantalite
Transport Worker - Truck Driver	0.24	0.16
Transport Worker - Dockworker	0.032	0.020
Transport Worker - Seaman	0.0041	0.0026
Transport Worker - Trainman	0.019	0.012
Public - Living Adjacent to Road	0.00038	0.00024
Public - Living Adjacent to Rail	0.00017	0.00010

- a) Mean annual dose from shipments of tantalum raw materials analyzed in this study.
 b) For perspective, doses to facility workers (who are not considered transport workers) were 0.49 and 0.31 mSv/y from slag and tantalite, respectively.

- Doses to members of the public from the transport of these materials were found to be insignificant, that is, much less than 10 µSv/y (0.01 mSv/y).
- The calculated doses to transport workers were higher than to the members of the public but were well within the internationally accepted dose limit of 1 mSv/y for non-radiation workers. If it were assumed that the tantalum raw materials considered in this study reliably represent the likely range of tantalum raw materials in general, then the expected (mean) dose to the most exposed group during transport would be about 0.24 mSv/y to truck drivers from the transport of slag (Table 6.1). Thus, there is considerable allowance for truck drivers who transport tantalum raw materials to transport other materials containing elevated levels of naturally occurring radioactivity without exceeding a cumulative annual dose of 1 mSv.

The results of this study demonstrate that the radiological doses from the transport of tantalum raw materials are low, and would result in very low doses (a few µSv/y at most) to the public who live or work nearby transportation routes. For workers involved in the transport of tantalum raw materials, doses are expected to be below 0.3 mSv/y and always below 1 mSv/y. Thus, on the basis of the analyses of doses arising from the transport of tantalum raw materials described in this report, there is no apparent reason with regards to radiological dose for an exemption value as restrictive as the current value of 10 Bq/g for these materials. Indeed, even in the absence of an exemption value, no one would be expected to receive a dose above 1 mSv/y arising from the transport of tantalum raw materials.

Notwithstanding the low doses from the transport of tantalum raw materials, specific numerical exemption values are required for the practical, real-world application of the transport regulations. Moreover, the radioactivity levels of future shipments of tantalum raw materials may differ from the distribution of levels from which the conclusions of this study were derived.

One possible method to choose an exemption value that could be associated with a specific dose level (even if not directly derived from it) is conservatively to assume that the radioactivity levels of all future exempted shipments of tantalum raw materials will be at the selected specific exemption value. Under this conservative assumption, the resultant annual doses for actual transport situations would be overestimated. The resulting calculated doses to transport workers are summarized in Table 6.2 (adapted from Table 4.6) for various potential exemption values.

**TABLE 6.2
SUMMARY OF ANNUAL DOSES (mSv/y) TO TRANSPORT WORKERS AT VARIOUS
POTENTIAL EXEMPTION VALUES**

Potential Exemption Value (Bq/g)	Material	Annual Dose (mSv/y) ^a			
		Truck Driver	Dock Worker	Seaman	Trainman
10	Tantalite	0.12	0.016	0.0021	0.0097
	Slag	0.12	0.016	0.0021	0.0098
30	Tantalite	0.31	0.040	0.0052	0.024
	Slag	0.37	0.049	0.0062	0.029
50	Tantalite	0.48	0.062	0.0080	0.038
	Slag	0.54	0.071	0.0090	0.043
70	Tantalite	0.65	0.084	0.011	0.051
	Slag	0.71	0.093	0.012	0.056
90	Tantalite	0.82	0.11	0.014	0.064
	Slag	0.89	0.12	0.015	0.070
100	Tantalite	0.91	0.12	0.015	0.071
	Slag	0.97	0.13	0.016	0.077

a) Assumes maximum load (1.5 tiers) of material and all loads at specific exemption value.

To account for the possibility of other transport-related exposures, 0.3 mSv/y might be considered as a reference dose level or a dose objective.¹⁰ The results in Table 6.2 show that an exemption value of 30 Bq/g for tantalite or slag would result in doses to the most exposed transport worker (truck driver) of 0.31 to 0.37 mSv/y if all the exempt tantalum raw materials were at 30 Bq/g. Considering the conservatism in these calculations, an exemption value of 30 Bq/g (U-238 + Th-232) would result in doses that would be unlikely to exceed 0.3 mSv/y to the most exposed transport workers. On this basis, an exemption value of at least 30 Bq/g is considered appropriate for the transport of tantalum raw materials.

Irrespective of the exemption value selected, the radiological dose assessments described in this report should provide assurance to the tantalum industry and to its shippers that the doses arising from the transport of tantalum raw materials are low and well within international norms for both transport workers and members of the public.

¹⁰ To allow for the possibility of multiple exposures, for example as suggested by the ICRP in the context of radioactive waste disposal, the ICRP (2007) continues to support its previous recommendation (ICRP 1997) of a 0.3 mSv/y dose constraint for members of the public.

7.0 REFERENCES

Grove Software 2005. MicroShield Version 6.20.

Hofmann, J., R. Leicht, H.J. Wingender and J. Wörner 2000. *Natural Radionuclide Concentrations in Materials Processed in the Chemical Industry and the Related Radiological Impact*. European Commission, Nuclear Safety and the Environment, Report EUR 19264.

International Atomic Energy Agency (IAEA) 2005a. *Regulations for the Safe Transport of Radioactive Material - 2005 Edition*. IAEA Safety Standards, Safety Requirements, No. TS-R-1.

International Atomic Energy Agency (IAEA) 2005b. *Coordinated Research Programme Related to Safety of Transport of Naturally-Occurring Radioactive Material*. TRANSSC 10, TM-27203. IAEA Headquarters, Vienna.

International Atomic Energy Agency (IAEA) 2004a. *Regulations for the Safe Transport of Radioactive Material – 1996 Edition (As Amended 2003)*. IAEA Safety Standards Series, Safety Requirements, No. TS-R-1.

International Atomic Energy Agency (IAEA) 2004b. *Applications of the Concepts of Exclusion, Exemption and Clearance*. IAEA Safety Standards Series, Safety Guide, No. RS-G-1.7.

International Atomic Energy Agency (IAEA) 2002. *Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material*. IAEA Safety Standards Series, Safety Guide, No. TS-G-1.1 (ST-2).

International Atomic Energy Agency (IAEA) 2000. *Regulations for the Safe Transport of Radioactive Material – 1996 Edition (Revised)*. IAEA Safety Standards Series, Requirements, No. TS-R-1 (ST-1, Revised).

International Atomic Energy Agency (IAEA) 1996. *International Basic Safety Standards for Protection Against Ionizing Radiation and for the Safety of Radiation Sources*. Safety Series No. 115, IAEA, Vienna.

International Commission on Radiological Protection (ICRP) 2007. *Draft Recommendations of the International Commission on Radiological Protection*. 12 January 2007. Approved 21 March 2007.

International Commission on Radiological Protection (ICRP) 2006. *The Scope of Radiological Protection Regulations*, Draft. 02/258/05, spring 2006 version.

International Commission on Radiological Protection (ICRP) 1997. *Radiological Protection Policy for the Disposal of Radioactive Waste*. ICRP Publication 77, Annals of the ICRP, Vol. 27, Supplement.

International Commission on Radiological Protection (ICRP) 1994. *Dose Coefficients for Intakes of Radionuclides by Workers*. ICRP Publication 68, Annals of the ICRP 24(4).

United States Environmental Protection Agency (U.S. EPA) 1997. *Exposure Factors Handbook Volume 1 - General Factors*. EPA/600/P-95/002Fa, August.

ANNEX A

**GAMMA RADIATION SURVEY
AND SAMPLE COLLECTION PROTOCOL**

ANNEX A GAMMA RADIATION SURVEY AND SAMPLE COLLECTION PROTOCOL

This annex provides the SENES and AHK gamma radiation survey and sample collection protocol. This protocol was distributed as a reference guide to the participating T.I.C. member companies. In addition to the protocol, this annex includes Table A.1 which provides a description of each gamma radiation survey location (used in the survey and sample collection protocol).

**TABLE A.1
GAMMA RADIATION SURVEY LOCATION DESCRIPTION**

Location ID^a	Distance from Container	Location Description
B0 & F0	Contact	1 m above ground in the centre of the short side (back/front)
B1 & F1	1 m	1 m above ground in the centre of the short side (back/front)
B3 & F3	3 m	1 m above ground in the centre of the short side (back/front)
BR0 & BL0 & FR0 & FL0	Contact	1 m above ground, along the right/left side, at 1 m from corner with short side (back/front)
BR1 & BL1 & FR1 & FL1	1 m	1 m above ground, along the right/left side, at 1 m from corner with short side (back/front)
BR3 & BL3 & FR3 & FL3	3 m	1 m above ground, along the right/left side, at 1 m from corner with short side (back/front)
MR0 & ML0	Contact	1 m above ground in centre of the long side (right/left)
MR1 & ML1	1 m	1 m above ground in centre of the long side (right/left)
MR3 & ML3	3 m	1 m above ground in centre of the long side (right/left)

Note:

a) B – Back, F – Front, M – Middle, R – Right, L – Left.

A.1 RADIATION MEASUREMENT PROTOCOL

Background

The Tantalum-Niobium International Study Center (T.I.C.) has commissioned a study and risk assessment of the transport of tantalum raw materials (concentrate and slags) which contain varying levels of naturally occurring radioactivity, namely, uranium and thorium and their radioactive decay products. The study is being done as a research project for the T.I.C. with the results reported to the Transport Safety Standards Committee (TRANSSC) of the International Atomic Energy Agency (IAEA). The Canadian Competent Authority to TRANSSC has agreed to sponsor the T.I.C. study.

The T.I.C. has retained two companies to assist it with the study – Alfred H. Knight International Ltd. (AHK) and SENES Consultants Limited (SENES). Sampling and measuring of the materials will be conducted by AHK of St. Helens, Merseyside, United Kingdom or by member companies in accordance with procedures acceptable to AHK/SENES. AHK have a long established reputation of expertise in the sampling and physical and chemical analysis of tantalum raw materials, as well as carrying out radiation dose rate surveys.

Overall supervision, verification of the results, the risk assessment itself, and completion of the report will be conducted by SENES of Richmond Hill, Ontario, Canada.

The main SENES and AHK contacts for this study are:

<u>SENES</u>	Tel: 001-905-764-9380	
	Dr. Douglas B. Chambers	dchambers@senes.ca
Alternate	Dr. Leo Lowe	llowe@senes.ca
<u>AHK</u>	Tel: 0044-1744-746316	
	David Cross	dave.cross@ahkgroup.com

Purpose of Radiation Measurement Protocol

The consultants will be carrying out this project under the supervision of the T.I.C. Transport Committee.

The purpose of these notes is to establish a common framework for carrying out the radiation surveys and sampling of the various concentrates and slags that will be done for this project.

The radiation survey components of the project involve recording the analysis of the materials (either mineral concentrates or slag) and readings of the radiation exposure rates around the sea-land containers when packed with the raw materials in the state in which they are transported, as well as other properties. An appropriate number and range of material samples will be surveyed.

Confidentiality of individual company issues is a concern of many T.I.C. member companies. Survey information will be collected and kept by SENES and AHK on a confidential basis, and will be used for the report in a collected and anonymous form. No individual company's information will be revealed to another company or to T.I.C.

This study is for the benefit of the tantalum industry as a whole. Your co-operation is greatly appreciated.

General Understanding and Assumptions

Our general understanding how materials are shipped is as follows:

1. Tantalum materials are shipped in standard sea-land containers, with typical dimensions of 20'L x 8'W x 8'6"H (6.1 m x 2.4 m x 2.6 m).
2. The sea-land containers are filled in one of three ways:
 - 2.1 205 L drums, 4 to a pallet with 20 pallets per sea-land container. Each drum weighs approximately 250 kg and hence the weight of tantalum materials is (approximately) 20,000 kg.
 - 2.2 one tonne bags, 20 per sea-land container.
 - 2.3 50 kg bags, 20 per pallet and 20 pallets.
3. Discussions have been held with a number of member companies who have agreed to provide material for surveys and/or to provide material for surveys and actually perform the radiation survey. Every effort is being made to try to ensure that a representative range of concentrates and slags will be surveyed and sampled. In this regard, the consultants are working closely with the T.I.C. Transport Committee.

Physical Sampling

AHK would conduct the physical sampling and carry out any field measurements beyond those done by T.I.C. members themselves. The physical sampling process currently used by AHK is suggested for present purposes.

Sampling Procedure

The procedure used is at the discretion of the company performing the sampling and will depend upon the facilities, manpower and tools available. All sampling procedures used should be forwarded to AHK for approval prior to commencing work, to ensure that all sampling carried out is of an equivalent standard. A sampling procedure that is recommended by AHK is given in Attachment B.

Analysis of Samples

AHK will carry out physical analysis to determine bulk and specific density as well as particle size distribution.

AHK will perform chemical analysis by X-Ray Fluorescence (XRF) or other suitable methods as necessary. Major elements expected are Ta, Nb, Fe, Mn, Si, Sn with minor elements being Th, U and others. It is anticipated that the XRF analysis will reveal any other significant elements present.

SENES will arrange for radiological analysis of a sample of <150 micron material (assumed to be homogeneous) at an ISO certified laboratory.

The radiological analysis of all the samples will be done by ISO-accredited Becquerel Laboratories in Mississauga (near Toronto), Ontario – see <http://www.becquerellabs.com/>. The homogeneous ground samples prepared by AHK will be analyzed by gamma spectroscopy for the uranium and thorium decay series, and for total uranium and thorium by the relatively sensitive (0.2 ppm) neutron activation analysis (NAA).

Radiation Survey

The purpose of the radiation survey protocol as documented in Attachment A is to ensure that measurements made by AHK and the various participating member companies are both reliable and comparable with one another.

Radiation Survey Instruments

Radiation survey instruments used in this study should be capable of measurements down to background levels and have a relatively uniform gamma energy response up to 2.6 MeV or greater (the maximum gamma energy from natural thorium radionuclides). Instrumentation suitable for gamma dose rate measurements around the sea-land containers includes energy compensated Geiger-Müller detectors, instruments using pressurized ion chambers (PICs), or tissue equivalent survey instruments (plastic scintillation detectors). Information on the gamma energy response of the survey device(s) used should be provided to SENES and AHK.

All radiation instruments used in this study should be calibrated using certified traceable protocols. Calibrations with standard Cs-137 or Co-60 sources would be suitable as would calibrations with Ra-226 sources. The surveyor is requested to use gamma survey instruments that have been calibrated within 12 months of the date of measurement, and to send a photocopy of the calibration certificate and all relevant information to SENES Consultants Limited with the results of the survey.

Radiation Survey Protocol

The proposed radiation survey protocol prepared for the study of tantalum material is summarized in Attachment A. In total, there are three elements to the survey protocol:

- **Figure A.1** outlines the protocol for establishing background radiation levels in the area in which sea-land containers will be surveyed;
- **Figure A.2** outlines the requested information concerning the dimensions of contents of the sea-land container; and
- **Figure A.3** provides summary detail sheet for recording the results of the transport container survey in a standard format.

The data sheets in Attachment A are to be photocopied as often as necessary and new sheets are to be used for each sea-land container measured.

In all cases, it is important to specify the type and origin (country/facility) of material (attach copies of shipping documents as available), where it is measured, the instrumentation used for the survey and other key data as indicated on the sheet.

For clarification on any item, contact should be made with AHK or SENES as appropriate. Completed data sheets should be forwarded to AHK with a copy to SENES. The addresses for the two companies are given below.

AHK

Alfred H. Knight International Ltd
Eccleston Grange
Prescot Road, St. Helens
Merseyside WA10 3BQ
United Kingdom
Tel: 0044-1744-746316

SENES

SENES Consultants Limited
121 Granton Drive, Unit 12
Richmond Hill, Ontario
L4B 3N4
Canada
Tel: 001-905-764-9380

Attachment A
Transport Container Gamma Radiation Survey
(Rev. 2A, 6 June 2005)

1.0 Area Background Measurements

- 1.1 Measure the background gamma radiation dose rate (environmental dose equivalent) in the area that will be (or was) used to measure dose rates from the loaded transport container; ensure that the instrument has stabilized (≥ 30 seconds) before recording the measurement;
- 1.2 Background measurements (no transport container present) can be made either before or after the radiation survey of the loaded transport container;
- 1.3 Background measurements will be made using the same instrument that will be (or was) used to measure dose rates around the loaded transport container;
- 1.4 Background measurements will be made within 8 hours of making the radiation survey of the loaded transport container;
- 1.5 Record the following information on **Figure A.1 - Background Data Sheet**:
 - Date of the measurements,
 - Start and end time of the background measurements,
 - Air temperature,
 - Radiation survey instrument – manufacturer, model number and serial number,
 - Certificate number and date for the most recent calibration of the radiation survey instrument (attach a copy with details of calibration),
 - Record time and results of battery and high voltage check before, and after measurements – if applicable;
- 1.6 Record readings at 1 m above the ground at the locations indicated in Figure A.1 (with respect to the (future) location of the transport container).

2.0 Transport Container Measurements - Dimension

[Note: Please attach a copy of shipping certificates associated with the material being shipped in the sea-land container]

- 2.1 Describe the packages (e.g. one tonne bags, 50 kg sacks or drums) used to contain the product within the transport container;
- 2.2 After placing the loaded transport container for measurements, record the following dimensions on **Figure A.2 – Dimensions and Loading Geometry**;
- 2.3 Measure and record the following dimensions in metres:
 - h_1 – height of the base of the transport container above the ground;
 - h_2 – height of the top of the load above the base of the transport container;

- F_1 – distance between the front of the container and the front of the lower tier of the load (the front of the container is the one nearest the driver's seat during road transport);
- F_2 – distance between the front of the container and the front of the upper tier of the load;
- B_1 – distance between the back of the container and the back (door end) of the lower tier of the load;
- B_2 – distance between the back of the container and the back of the upper tier of the load.

3.0 Transport Container Measurements - Radiation

- 3.1 Measure and record the gamma radiation dose rate (environmental dose equivalent) from the loaded transport container; ensure that the instrument has stabilized (≥ 30 seconds) before recording the measurement;
- 3.2 Record the following data on **Figure A.3 – Survey Results**;
 - Date of the measurements,
 - Start and end time of the measurements,
 - Air temperature,
 - Record time and results of battery and high voltage check before, and after measurements – if applicable;
- 3.3 Measure and record the radiation dose rate (environmental dose equivalent) at locations indicated in Figure A.3:
 - at contact with, and 1 and 3 m from the sides of the container (at 1 m above the base of the container – i.e. $(1 + h_1)$ m above the ground); these measurements on each side of the container should be taken at 1 m from each end of the container, and midway between the ends;
 - at contact with, and 1 and 3 m from the ends of the container (at 1 m above the base of the container – i.e. $(1 + h_1)$ m above the ground);
 - at contact on the top of the container, on the centreline of the long dimension, at 1 m from each end and midway between the ends of the container. (If access to the top of the container is difficult, these measurements may be omitted.)

Note: If space restrictions limit the measurements to distances less than 3 m from the container, record the actual distance of the measurement on Figure A.3.

Attachment B
AHK Sampling and Sample Preparation Procedure

The following sampling procedure is recommended for use by AHK or T.I.C. member companies:

1. The sampling tool used is a scoop minimum 50 mm wide and minimum 75 ml capacity. The scoop dimensions should be according to ISO or JIS standards, *e.g.* ISO 3713:1987 or JIS M8100:1973. EXC: for 50 kg bags, a sampling spear (a.k.a. probe/stem/thief) of 50 mm diameter is used instead.
2. Handling one drum (or bag) at a time, open and pour the contents into an empty drum. This may be achieved in various ways, depending on the type of drum (or bag):
 - 2.1. For an old oil drum which is welded shut, tip the closed drum on its side, lift it by fork lift truck (each prong on either side of drum), then using chisel and hammer or a powered rotary grinder, open a small 'air' hole at the top edge of one end, before cutting open the same lid at the bottom edge. Initially only a partial opening is needed, the opening being widened as the material flow slackens; emptying the last part of the drum requires some manual handling.
 - 2.2. For a new drum with fitted lid and ring clip, a clamp device that can grip the drum and rotate it is required. Simply remove the lid and rotate the drum to pour.
 - 2.3. For a big/bulk bag, lift the bag by its straps and open the bottom (if of bottom-opening type) or cut open to allow the material to pour out into a new bag.
3. From the falling stream of material obtained, sample increments are taken by scoop, from the initial, middle and end part of the flow. This is to represent the top, middle and bottom of the original drum/bag. (For 50 kg bags, pouring the material out is impractical and as an exception to the above, samples are taken by spear; this is inserted horizontally through a bag.) The number of increments to be taken are:
 - Drums : 3 per drum
 - one tonne bags: 9 per bag
 - 50 kg bags : 1 per bag
- 3.1. Each increment will be of ~0.5 kg, which is collected in double lined, heavy duty plastic sacks or bag lined drums with lid, then tied/sealed shut with tape.

4. The bulk sample for a sea-land container, i.e. 80 drums or equivalent weight of bagged tantalum material (20 tonnes of tantalum material), will be as shown below. This sample is then taken to sample preparation facilities for the next stage.
 - Drums : 120 kg
 - one tonne bags: 90 kg
 - 50 kg bags : 200 kg
5. Where possible, sample preparation should be conducted locally in order that the bulk of the sample can be returned to the sampled material.
6. The entire bulk sample is first passed through a crusher that reduces the top size to 2 mm. The sample is then piled up as a cone on a clean steel plate or concrete floor and mixed by moving the cone 3 times, by shovel. After the third 'coning', the sample can be divided by riffle splitter three times, to 15 kg. Alternatively, the cone can be flattened and divided into quarters, with two opposite quarters being selected for the sample and the remainder being rejected; this is repeated twice to give 15 kg.
 - 6.1. After the first riffle or quartering division, two 1 kg samples are taken from the reject sample for moisture determination. They are dried in an oven at 105°C to constant weight.
 - 6.2. An additional 5 kg are to be retained from the reject sample, for bulk density testing.
7. The 15 kg quality sample is reduced to 0.5 or 1 mm, possibly in a crusher but probably in a ring or puck mill (a disc mill is not acceptable).
8. The reduced sample is then divided three times to 2 kg.
9. The 2 kg are milled entirely to 150 micron (μm).

FIGURE A.2 - DIMENSIONS AND LOADING GEOMETRY SHEET

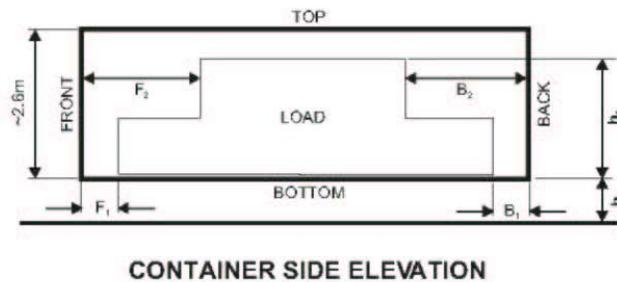
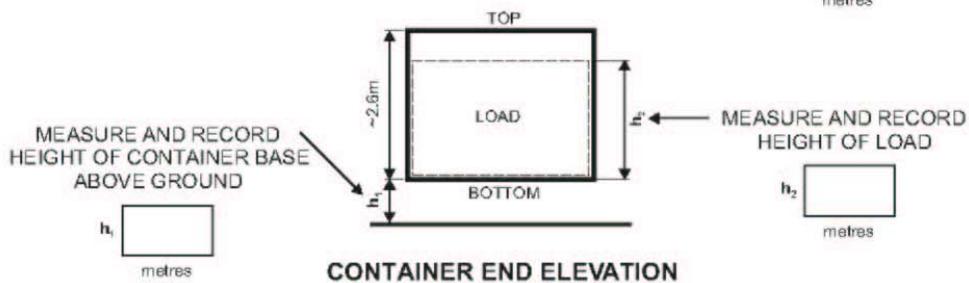
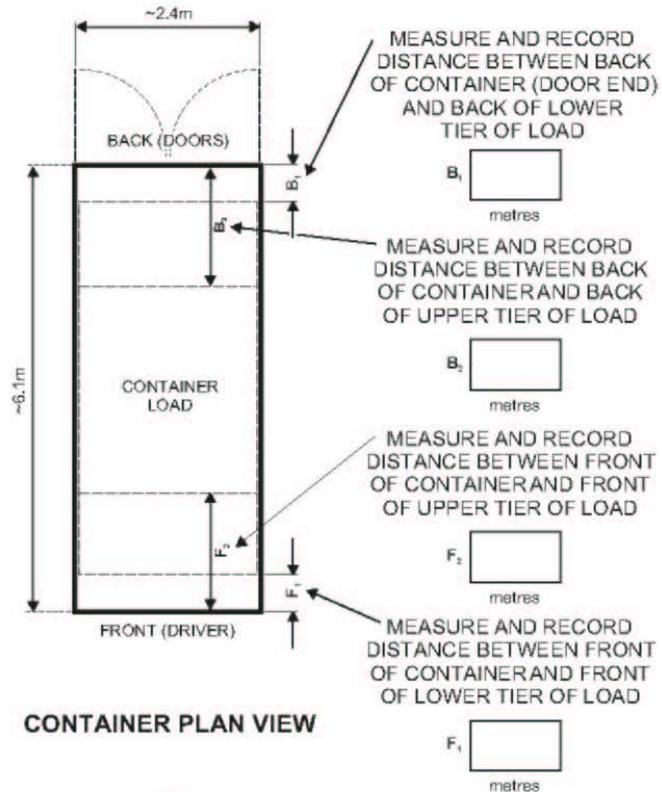
[Rev. 2A, 6 June 2005]

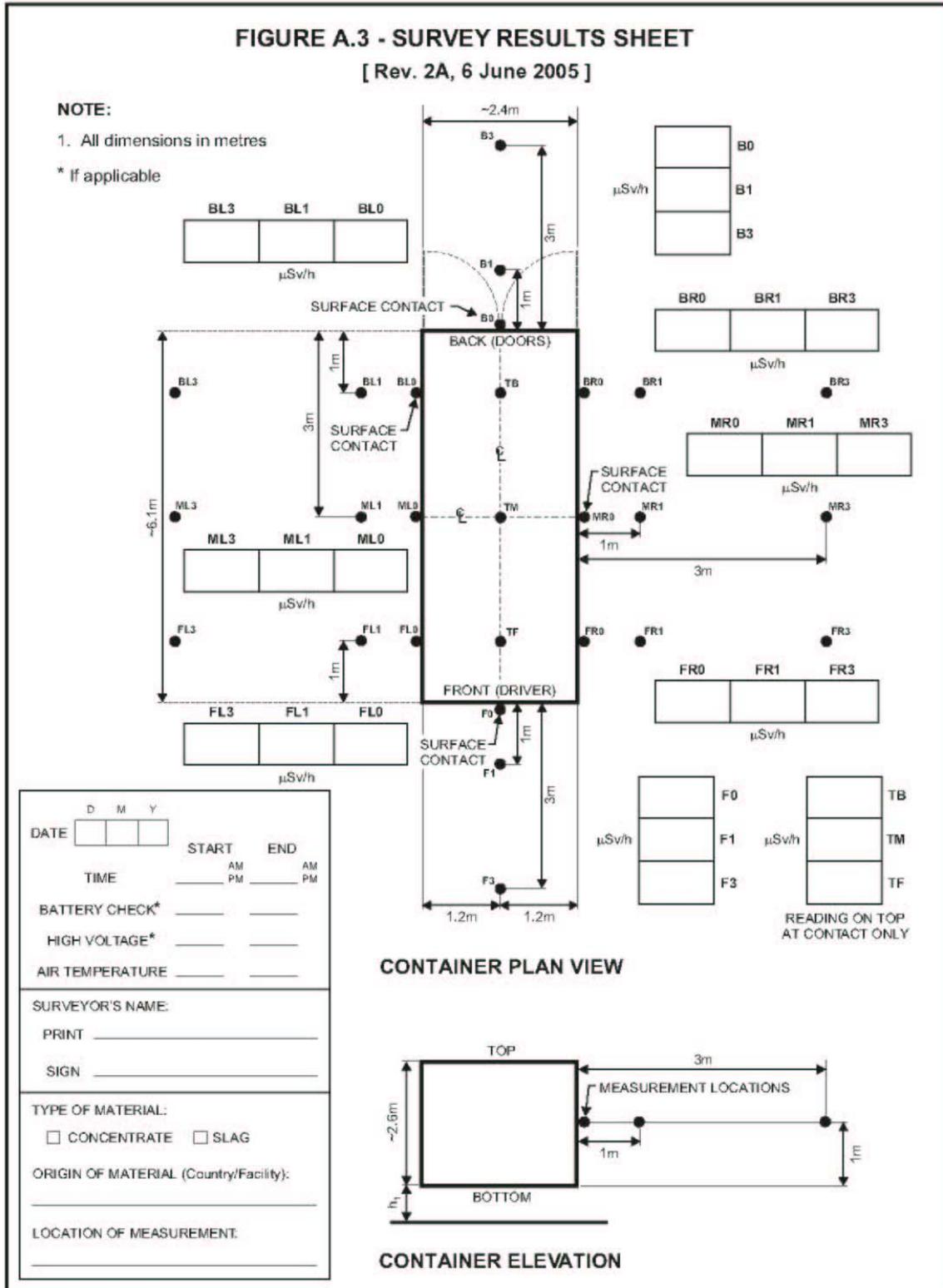
SURVEYOR'S NAME:	
PRINT _____	
SIGN _____	
TYPE OF MATERIAL:	
<input type="checkbox"/> CONCENTRATE	<input type="checkbox"/> SLAG
ORIGIN OF MATERIAL (Country/Facility):	

LOCATION OF MEASUREMENT:	

NOTES:

1. All dimensions in metres
2. Indicate number of tiers/levels of container load
 - ONE TWO





ANNEX B

RESULTS FROM RADIATION SURVEYS

ANNEX B RESULTS FROM RADIATION SURVEYS

B.1 DATA MANAGEMENT

B.1.1 Data Capture

A master table was set up in a spreadsheet with a record for each shipment. The shipment identification was often different between the gamma radiation measurements, the concentration data from the primary laboratory, and the concentration measurements from the independent laboratory. In order to track the data, a unique code was assigned for each shipment. The information in this table included:

- i) ID: a non-informative code for each shipment
- ii) Material Type: e.g. tantalite or slag
- iii) Container Type: e.g. drums or one tonne bags
- iii) Company and source
- iv) Loading configuration
- v) Identification from Each Source of Data (i.e. primary laboratory, independent laboratory, etc.)

Gamma radiation survey measurements for the loaded shipment were entered by hand into a spreadsheet. The unique shipment ID code was assigned and the type of meter used for the survey was recorded. A similar table was created for the background gamma radiation survey measurements (Section B.4).

The independent laboratory provided concentrations for each sample analyzed in electronic documents. This information was converted to spreadsheet format and stored in a spreadsheet along with the unique ID. The analysis results provided by the primary laboratory were entered into another table (Section B.3).

Predicted dose factors from MicroShield were entered into a spreadsheet table along with attributes for the type of material and the load configuration (dose factors are provided in Annex E).

B.1.2 Relational Structure

The spreadsheet data were then electronically transferred into a relational database within SAS software.

B.1.3 Data Completeness

Table B.1 summarizes the collection of data. There were 71 shipments identified; of these there were 59 shipments with gamma radiation survey results and 67 shipments with concentrations of uranium and thorium measured by the primary laboratory. There were 55 shipments with both gamma radiation survey measurements and laboratory analyses of uranium and thorium content.

There were 51 shipments with measurements from both the primary and independent laboratories for intercomparison of measurements.

**TABLE B.1
COMPLETENESS OF DATA**

Number of Shipments	Gamma Radiation Survey	Primary Laboratory Measurements	Independent Laboratory Measurements
12	No	Yes	Yes
4	Yes	No	No
6	Yes	Yes	No
49	Yes	Yes	Yes
Total			
71	59	67	61

B.1.4 Data Processing

The two major data processing activities were calculation of the net gamma radiation dose rates and the conversion of uranium and thorium concentrations reported on a mass basis to the activity concentrations of U-238 and Th-232.

Gamma radiation is present in varying amounts that are independent of the radioactivity present in the tantalum shipments. For each shipment, the gamma radiation dose rates at all locations without the shipment present were averaged to estimate the baseline gamma radiation dose rate for that shipment. This average baseline dose rate was subtracted from the gamma radiation dose rates with the shipment present to estimate the (net) gamma radiation attributable to the radioactivity in the shipment.

Uranium and thorium concentrations were measured by a number of methods with concentrations reported in both mass-based and activity-based units. Since the focus of this study is radioactivity, mass-based measurements of uranium and thorium have been converted to activity-based concentrations of the principal radionuclides of U-238 and Th-232, respectively. Table B.2 shows the conversion factors. For example, 1000 ppm U is equivalent to 12.35 Bq/g U-238 (i.e. $1000 \text{ ppm} \times 0.01235 \text{ Bq/g U-238 per ppm U} = 12.35 \text{ Bq/g}$).

**TABLE B.2
CONVERSION FACTORS**

	Conversion Factor to Activity Concentration	
	U-238 (Bq/g)	Th-232 (Bq/g)
U or Th (ppm)	0.01235	0.004057
U ₃ O ₈ (%)	104.7	n/a
ThO ₂ (%)	n/a	35.66

Note: U₃O₈ is 84.8% uranium by weight; ThO₂ is 87.9% thorium by weight.

B.2 DESCRIPTION OF SHIPMENTS

Information on 71 shipments was tracked and a list of these shipments is provided in Table B.3. Attributes of the shipments included the type of material, the loading configuration within the shipment, the company and the origin (location) of the material were recorded. The company and material origin identities have not been included in this report for reasons of commercial confidentiality. Table B.3 includes an indicator of the status of gamma radiation survey data, concentration measurements from the primary laboratory and the concentration measurements from the independent laboratory.

Most shipments were in sea-land containers with either 1 or partially full 2 tiers (layers) of material; however approximately half of the 71 shipments were considered as Maximum Loads¹¹, the loading configuration was unknown¹² for 12 shipments and there were two shipments in trailers. Additionally, there were two shipments indicated as “On ground”, which did not have the tantalum materials placed inside a transport container, but rather were placed on a flat surface as if they were inside a container and the gamma radiation measurements were taken using the protocol. The majority, 48, of shipments were tantalite, while 23 of the shipments were slag materials. A summary of the different tantalum materials and loading configurations is provided in Table B.4.

**TABLE B.3
SHIPMENT CHARACTERISTICS AND AVAILABLE INFORMATION**

Shipment	Material Type	Loading	Gamma Radiation Survey	Primary Laboratory Measurement	Independent Laboratory Measurement
S1	Tantalite	NF - 1 Tier	Yes	Yes	Yes
S2	Tantalite	NF - 1 Tier	Yes	Yes	Yes
S3	Tantalite	NF - 1 Tier	Yes	Yes	Yes
S4	Tantalite	NF - 1 Tier	Yes	Yes	Yes
S5	Slag	NF - 1 Tier	Yes	Yes	Yes
S6	Tantalite	Maximum	Yes	Yes	Yes
S7	Tantalite	Maximum	Yes	Yes	Yes
S8	Tantalite	NF Trailer	Yes	Yes	Yes
S9	Tantalite	NF Trailer	Yes	Yes	Yes
S10	Tantalite	F - 1 Tier	Yes	Yes	Yes
S11	Slag	Maximum	Yes	Yes	Yes
S12	Slag	Unknown	No	Yes	Yes
S13a	Slag	Maximum	Yes	Yes*	Yes*

¹¹ Maximum Load is a full first layer and a half-full second layer.

¹² The loading configuration of the shipment was not specified on the gamma radiation survey sheets.

TABLE B.3 (Cont'd)
SHIPMENT CHARACTERISTICS AND AVAILABLE INFORMATION

Shipment	Material Type	Loading	Gamma Radiation Survey	Primary Laboratory Measurement	Independent Laboratory Measurement
S13b	Slag	Maximum	Yes	Yes*	Yes*
S13c	Slag	Maximum	Yes	Yes*	Yes*
S13d	Slag	Maximum	Yes	Yes*	Yes*
S13e	Slag	Maximum	Yes	Yes*	Yes*
S14a	Slag	Maximum	Yes	Yes*	Yes*
S14b	Slag	Maximum	Yes	Yes*	Yes*
S14c	Slag	Maximum	Yes	Yes*	Yes*
S14d	Slag	Maximum	Yes	Yes*	Yes*
S14e	Slag	Maximum	Yes	Yes*	Yes*
S15	Slag	Unknown	No	Yes	Yes
S16	Tantalite	Maximum	Yes	Yes	Yes
S17	Tantalite	Maximum	Yes	No	No
S18	Tantalite	Maximum	Yes	Yes	Yes
S19	Tantalite	Maximum	Yes	Yes	Yes
S20	Tantalite	Maximum	Yes	Yes	Yes
S21	Tantalite	Maximum	Yes	Yes	Yes
S22	Tantalite	Maximum	Yes	Yes	Yes
S23	Tantalite	Maximum	Yes	Yes	Yes
S24	Tantalite	Maximum	Yes	Yes	Yes
S25	Tantalite	Maximum	Yes	Yes	Yes
S26	Tantalite	Maximum	Yes	Yes	Yes
S27	Tantalite	F - 1 Tier	Yes	Yes	Yes
S28	Tantalite	F - 1 Tier	Yes	Yes	Yes
S29	Tantalite	F - 1 Tier	Yes	Yes	Yes
S30	Tantalite	F - 1 Tier	Yes	Yes	Yes
S31	Tantalite	Maximum	Yes	Yes	Yes
S32	Slag	F - 1 Tier	Yes	Yes	Yes
S33	Slag	F - 1 Tier	Yes	Yes	Yes
S34	Slag	F - 1 Tier	Yes	No	No
S35	Slag	F - 1 Tier	Yes	Yes	Yes
S36	Tantalite	Maximum	Yes	No	No
S37	Tantalite	Maximum	Yes	Yes	Yes
S38	Tantalite	Maximum	Yes	No	No
S39	Slag	F - 1 Tier	Yes	Yes	Yes
S40	Tantalite	F - 1 Tier	Yes	Yes	Yes
S41	Tantalite	Maximum	Yes	Yes	Yes
S43	Tantalite	Maximum	Yes	Yes	Yes
S45	Tantalite	Unknown	No	Yes	Yes

TABLE B.3 (Cont'd)
SHIPMENT CHARACTERISTICS AND AVAILABLE INFORMATION

Shipment	Material Type	Loading	Gamma Radiation Survey	Primary Laboratory Measurement	Independent Laboratory Measurement
S47	tantalite	Unknown	No	Yes	Yes
S48	tantalite	Unknown	No	Yes	Yes
S49	tantalite	Unknown	No	Yes	Yes
S50	tantalite	Unknown	No	Yes	Yes
S51	tantalite	Unknown	No	Yes	Yes
S52	tantalite	Unknown	No	Yes	Yes
S53	slag	Unknown	No	Yes	Yes
S54	tantalite	Unknown	No	Yes	Yes
S55	tantalite	Unknown	No	Yes	Yes
S56	tantalite	Maximum	Yes	Yes	No
S57	tantalite	Maximum	Yes	Yes	No
S58	tantalite	Maximum	Yes	Yes	No
S59	tantalite	Maximum	Yes	Yes	No
S60	tantalite	Maximum	Yes	Yes	No
S61	tantalite	Maximum	Yes	Yes	No
S62	tantalite	On ground	Yes	Yes	Yes
S63	tantalite	On ground	Yes	Yes	Yes
S64	slag	Maximum	Yes	Yes	Yes
S65	slag	Maximum	Yes	Yes	Yes
S66	slag	Maximum	Yes	Yes	Yes

* A single composite sample was collected for S13a to S13e and a single composite sample was collected for S14a to S14e.

TABLE B.4
MATERIAL TYPES AND LOADING CONFIGURATION

Configuration	Type of Tantalum Raw Material		
	All Shipments	Tantalite	Slag
All	71	48	23
Maximum	39	25	14
F - 1 Tier	11	6	5
NF - 1 Tier	5	4	1
NF Trailer	2	2	0
On ground	2	2	0
Unknown	12	9	3

B.3 CONCENTRATION DATA

Samples were collected from each shipment according to the established protocol and measured for radionuclide and product content.

B.3.1 Sampling of Materials

Composite samples were collected according to the protocol described in Annex A. In two cases, a single composite sample was collected from 5 different containers. A composite sample was collected from shipments S13a through S13e and from shipments S14a through S14e.

Typically, the composite samples were shipped to the primary laboratory for sample preparation and laboratory analyses. In a few cases, samples were not allowed to be shipped from the country: these samples were sent to a local laboratory. Split-samples were sent from the primary laboratory to the independent laboratory for independent analyses of uranium and thorium content as well as measurement of selected radionuclides in the uranium and thorium decay series.

B.3.2 Laboratory Measurements

Table B.5 shows the concentrations reported by the laboratories for uranium, thorium and the corresponding decay series radionuclides. In addition, Table B.5 shows the densities and tantalum and niobium content of each shipment. Laboratory analyses of other constituents were provided by the primary laboratories (e.g. ZrO₂); however, these concentrations were not compiled or used in this study.

Radiological Risk Assessment of the Transport of Tantalum Raw Materials

**TABLE B.5
LABORATORY MEASUREMENTS OF CONCENTRATION**

Shipment	Ta ₂ O ₅	Nb ₂ O ₅	Bulk Density	U ₃ O ₈	ThO ₂	Uranium	Thorium	Th-234	Th-230	Ra-226	Pb-210	U-235	Th-227	Ra-223	Th-228	Ra-228
	%	%	g/cm ³	%	%	ppm	ppm	Bq/g	Bq/g	Bq/g	Bq/g	Bq/g	Bq/g	Bq/g	Bq/g	Bq/g
S1	28	14	3.0	0.093	0.014	830	< 57	8.5		8	7	0.4	0.5	0.4	0.40	0.5
S2	27	12	2.9	0.15	0.012	1280	145	15		15	12	0.6	1.1	0.9	0.80	0.7
S3	27	13	2.9	0.19	0.026	1730	164	20		18	16	0.8	1.4	1.2	1.0	1.4
S4	27	14	2.9	0.19	0.022	1620	166	17		15	14	0.9	1.2	0.9	1.0	0.8
S5	31	8	2.4	0.47	0.050	4300	490	45		42	2.6	2.1	3.2	3.1	1.9	2.0
S6	38	6	2.7	0.13	0.009	1190	106	12		12	13	0.6	0.8	0.8	0.56	0.7
S7	38	5	2.8	0.14	0.008	1200	79	12		10	12	0.7	1.1	0.8	0.54	0.7
S8	22	4	2.8	0.23	0.010	1940	159	22		18	23	1.1	1.3	1.3	0.66	0.3
S9	24	3	2.6	0.21	<0.005	1820	197	19		16	18	1.0	1.0	1.2	0.66	0.7
S10	10	17	2.4	0.054	0.042	500	387	5.3	< 2	6.1	4.2	0.2	0.2	0.2	1.9	2.2
S11	7	10	1.8	0.075	0.78	676	6780	8.5	< 7	5.4	< 0.6	< 0.3	0.7	0.5	23	24
S12	2	3	1.8	0.037	0.17	368	1400	4.5	< 8	4.2	4.4	< 0.7	< 0.4	< 1	5.5	5.7
S13 ^a	3	3	1.8	0.033	0.17	384	1420	4.5	4	5.2	4.0	< 0.6	< 0.3	< 1	5.9	6.1
S14 ^a	2	2	1.7	0.033	0.16	372	1420	4.2	5	4.0	2.8	< 1	0.4	< 0.7	5.5	6.0
S15	2	2	1.8	0.023	0.14	320	1240	3.1	< 6	4.0	2.9	< 0.7	< 0.3	< 0.8	5.1	4.7
S16	35	6	2.9	0.10	<0.005	600	<150	9.7		11	8.7	0.4	0.6	0.6	0.49	0.4
S17																
S18	35	7	2.9	0.10	<0.005	620	<120	10.5		10	8.7	0.5	0.9	<0.6	0.59	0.5
S19	34	6	3.0	0.11	0.006	780	<150	9.3		13	9.2	0.6	0.5	0.4	0.59	0.5
S20	35	7	3.1	0.11	0.008	860	<170	10.2		12	9.0	0.5	0.8	0.6	0.57	0.6
S21	20	3	2.4	0.094	0.011	450	<100	8.0		9.8	7.9	0.4	0.4	0.4	0.39	0.4
S22	19	3	2.5	0.093	0.012	460	<90	8.2		11	7.8	0.5	0.2	<0.7	0.46	0.4
S23	28	5	3.0	0.14	<0.005	840	<110	13.8		16	12.8	0.6	0.7	0.5	0.54	0.4
S24	35	7	3.0	0.11	<0.005	640	<90	11.1		14	10.5	0.4	0.5	0.6	0.61	0.7
S25	29	6	3.0	0.15	<0.005	850	<110	14.3		17	13.0	0.7	0.8	0.6	0.63	0.6
S26	32	6	3.0	0.15	0.012	950	130	13.3		17	12.4	0.4	0.7	0.8	0.64	<0.8
S27	27	22	2.8	0.19	0.022	1110	<120	15		21	13.3	0.7	0.8	0.7	0.85	1.1
S28	28	19	2.6	0.24	0.026	1260	200	19		27	15.7	1.0	1.0	1.1	0.89	1.0
S29	28	17	2.7	0.13	0.016	740	<90	11.7		16	10.4	0.5	0.6	0.5	0.67	0.7

TABLE B.5 (Cont'd)
LABORATORY MEASUREMENTS OF CONCENTRATION

Shipment	Ta ₂ O ₅	Nb ₂ O ₅	Bulk Density	U ₃ O ₈	ThO ₂	Uranium	Thorium	Th-234	Th-230	Ra-226	Pb-210	U-235	Th-227	Ra-223	Th-228	Ra-228
	%	%	g/cm ³	%	%	ppm	ppm	Bq/g	Bq/g	Bq/g	Bq/g	Bq/g	Bq/g	Bq/g	Bq/g	Bq/g
S30	28	21	2.8	0.19	0.024	1130	170	19		25	16.0	0.8	0.8	1.2	0.89	1.1
S31	18	16	2.4	0.15	0.031	880	120	14		20	12.5	0.5	1.1	1.0	0.78	0.9
S32	29	9	2.2	0.83	0.10	5990	850	92		92	1.2	5	3.5	3.9	3.6	3.5
S33	26	10	2.1	0.88	0.13	6440	1060	93		120	2.4	4	5.1	6	4.7	5.8
S34																
S35	25	9	2.1	0.64	0.073	5020	630	67		77	1.8	3.0	3.3	3.7	2.5	2.9
S36																
S37	36	6	2.9	0.11	0.006	700	<150	10.3		14	10.1	0.6	0.6	0.6	0.58	0.6
S38																
S39	28	10	2.2	0.46	0.056	3090	390	46		59	0.9	2.3	2.3	2.6	2.0	2.3
S40	2	2	1.5	0.062	0.030	370	140	5.2		7	6.1	< 0.4	< 0.5	0.3	0.54	0.7
S41	3	2	1.5	0.047	0.052	290	< 25	4.2		6	4.4	< 0.3	< 0.4	0.4	0.28	< 0.5
S43	30	8	3.1	0.13	0.016	730	46	13		13	11	0.8	0.7	0.3	0.49	0.6
S45	31	7	3.0	0.11	0.016	710	53	10		12	8.4	0.7	0.6	0.5	0.56	0.6
S47	30	17	2.9	0.076	0.005	530	71	6.6		8	4.9	0.5	0.5	0.4	0.44	0.5
S48	53.84	16.53		0.31	<0.005	2900	< 260	27		35	22	1.2	2.1	1.6	1.0	1.1
S49	10	14	2.5	0.043	0.031	436	260	4.5	< 4	5.5	4.0	< 2	< 0.4	< 0.6	1.3	0.9
S50	9	12	3.1	0.049	<0.005	460	610	5.1	5	7.0	4.2	< 0.3	0.2	0.4	2.5	2.5
S51	16	2	2.3	0.18	0.26	1670	2201	20	15	18	19	< 2	0.8	< 3	7.2	7.3
S52	15	2	2.2	0.20	0.27	1720	2330	23	< 20	20	21	< 4	1.3	< 3	8.0	7.5
S53	4	0.5	1.9	0.040	0.21	370	1680	5.3	< 7	4.3	4.6	< 1	0.5	< 1	5.6	4.9
S54	16	2	2.2	0.26	0.31	2280	2650	30	28	26	27	< 2	1.3	< 5	8.8	9.2
S55	48.96	17.74		0.65	<0.005	6000	< 210	53		64	39	2.6	4.2	3.0	0.42	0.4
S56	26.53	28.93	2.9	0.07	0.020											
S57	28.71	22.40	3.1	0.05	0.019											
S58	28.38	22.36	2.9	0.07	0.032											
S59	25.74	24.38	2.9	0.05	0.015											
S60	21.39	12.20	2.4	0.11	0.097											
S61	49.05	11.89	2.6	0.27	0.072											
S62	28	15	2.5	0.47	<0.005	4000	180	46		49	45	2.5	2.3	1.9	0.8	0.8
S63	17	6	2.8	0.30	<0.005	2480	< 60	29		34	27	1.4	1.3	1.7	0.19	0.4
S64	4	4	1.7	0.055	0.23	460	2100	5.8		5.5	5.2	0.1	0.3	0.5	7.4	5.7
S65	3	4	1.7	0.051	0.22	390	2100	5.7		5.5	4.5	0.3	0.2	<0.4	7.2	5.8
S66	3	4	1.7	0.053	0.23	420	2000	4.8		5.0	4.2	<0.3	<0.3	0.4	6.5	5.1

a) A single composite was analyzed from five shipments of the same lot.

B.4 DOSE RATE MEASUREMENTS

B.4.1 Instrumentation

Gamma radiation surveys were conducted with a variety of different gamma radiation meters. Table B.6 summarizes the number of shipments measured with each type of meter. In some cases, a shipment was measured with more than one meter. In general, the meters provide measurements of dose rates that are applicable to dose assessment.

**TABLE B.6
METERS USED FOR GAMMA RADIATION SURVEYS**

Instruments	Number of Surveys
Berthold Proportional Counter Tube	1
Bicron MicroRem Dose Rate Meter	15
DKS-96 BDKS-96	6
Ludlum 19	9*
Ludlum 2401-P	18
MiniRad 1000	16
Thermo Eberline Proportional	3
All	68*

Note:

* Nine surveys were conducted with the Ludlum 19 as a second method. As a result, there were 68 gamma radiation surveys from the 59 shipments with gamma radiation surveys.

B.4.2 Measured Dose Rates

Baseline gamma radiation dose rates were collected according to the protocol described in Annex A. These values are shown in Table B.7.

The protocol for gamma radiation surveys specified measurements at the two ends, the two sides and the four corners of the shipment. For each location, measurements were collected on contact, at a distance of 1 m and at a distance of 3 m for a total of 24 measurements per shipment.

Gamma radiation levels with the shipment in place are provided in Table B.8.

**TABLE B.7
BASELINE GAMMA RADIATION DOSE RATES**

SENES ID	Meter	Gamma Dose Rates ($\mu\text{Sv/h}$)							
		B	BR	FR	F	FL	BL	BT	FT
S1	Bicon MicroRem Dose Rate Meter	0.03	0.02	0.04	0.03	0.02	0.02	0.02	0.02
	Ludlum 19	0.06	0.07	0.08	0.06	0.06	0.06	0.06	0.06
S2	Bicon MicroRem Dose Rate Meter	0.03	0.04	0.05	0.04	0.04	0.04	0.03	0.03
	Ludlum 19	0.09	0.08	0.10	0.10	0.09	0.08	0.09	0.09
S3	Bicon MicroRem Dose Rate Meter	0.05	0.05	0.05	0.04	0.03	0.02	0.02	0.03
	Ludlum 19	0.09	0.11	0.11	0.08	0.08	0.08	0.08	0.08
S4	Bicon MicroRem Dose Rate Meter	0.03	0.04	0.05	0.04	0.04	0.04	0.03	0.03
	Ludlum 19	0.09	0.08	0.10	0.10	0.09	0.08	0.09	0.09
S5	Bicon MicroRem Dose Rate Meter	0.05	0.05	0.05	0.04	0.03	0.02	0.02	0.03
	Ludlum 19	0.09	0.11	0.11	0.08	0.08	0.08	0.08	0.08
S6	Bicon MicroRem Dose Rate Meter	0.03	0.02	0.04	0.03	0.02	0.02	0.02	0.02
	Ludlum 19	0.06	0.07	0.08	0.06	0.06	0.06	0.06	0.06
S7	Bicon MicroRem Dose Rate Meter	0.03	0.02	0.04	0.03	0.02	0.02	0.02	0.02
	Ludlum 19	0.06	0.07	0.08	0.06	0.06	0.06	0.06	0.06
S8	Bicon MicroRem Dose Rate Meter	0.03	0.02	0.04	0.03	0.02	0.02	0.02	0.02
	Ludlum 19	0.06	0.07	0.08	0.06	0.06	0.06	0.06	0.06
S9	Bicon MicroRem Dose Rate Meter	0.03	0.04	0.03	0.03	0.03	0.05	0.02	0.03
	Ludlum 19	0.07	0.08	0.07	0.07	0.08	0.10	0.07	0.08
S10	MiniRad 1000	0.1	0.2	0.2	0.2	0.2	0.2	0.2	0.2
S11	Berthold Proportional Counter Tube	0.14	0.14	0.14	0.14	0.14	0.14	0.14	0.14
S13 ^a	MiniRad 1000	0.4	0.3	0.4	0.2	0.2	0.3	0.3	0.2
S14 ^a	MiniRad 1000	0.4	0.3	0.4	0.2	0.2	0.3	0.3	0.2

**TABLE B.7 (Cont'd)
BASELINE GAMMA RADIATION DOSE RATES**

SENES ID	Meter	Gamma Dose Rates (µSv/h)							
		B	BR	FR	F	FL	BL	BT	FT
S16	Ludlum 2401 P Meter	0.1	0.1	0.1	0.1	0.1	0.1	-	-
S17	Ludlum 2401 P Meter	0.1	0.1	0.1	0.1	0.1	0.1	-	-
S18	Ludlum 2401 P Meter	0.1	0.1	0.1	0.1	0.1	0.1	-	-
S19	Ludlum 2401 P Meter	0.1	0.1	0.1	0.1	0.1	0.1	-	-
S20	Ludlum 2401 P Meter	0.1	0.1	0.1	0.1	0.1	0.1	-	-
S21	Ludlum 2401 P Meter	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1
S22	Ludlum 2401 P Meter	0.1	0.1	0.1	0.1	0.2	0.2	-	-
S23	Ludlum 2401 P Meter	0.1	0.1	0.1	0.1	0.1	0.1	-	-
S24	Ludlum 2401 P Meter	0.1	0.1	0.1	0.1	0.1	0.1	-	-
S25	Ludlum 2401 P Meter	0.1	0.1	0.1	0.1	0.1	0.2	-	-
S26	Ludlum 2401 P Meter	0.1	0.1	0.1	0.1	0.1	0.1	-	-
S27	Bicron MicroRem Dose Rate Meter	0.10	0.14	0.16	0.15	0.19	0.20	0.15	0.16
S28	Bicron MicroRem Dose Rate Meter	0.12	0.12	0.15	0.13	0.19	0.20	0.14	0.12
S29	Bicron MicroRem Dose Rate Meter	0.10	0.14	0.16	0.15	0.19	0.20	0.15	0.16
S30	Bicron MicroRem Dose Rate Meter	0.10	0.14	0.16	0.15	0.19	0.20	0.15	0.16
S31	Bicron MicroRem Dose Rate Meter	0.16	0.15	0.12	0.13	0.18	0.17	0.11	0.12
S32	Ludlum 2401 P Meter	0.1	0.1	0.1	0.1	0.1	0.1	-	-
S33	Ludlum 2401 P Meter	0.1	0.1	0.1	0.1	0.1	0.1	-	-

**TABLE B.7 (Cont'd)
BASELINE GAMMA RADIATION DOSE RATES**

SENES ID	Meter	Gamma Dose Rates (µSv/h)							
		B	BR	FR	F	FL	BL	BT	FT
S34	Ludlum 2401 P Meter	0.1	0.1	0.1	0.1	0.1	0.1	-	-
S35	Bicron MicroRem Dose Rate Meter	0.12	0.12	0.15	0.13	0.19	0.20	0.14	0.12
S36	Ludlum 2401 P Meter	0.1	0.1	0.1	0.1	0.1	0.1	-	-
S37	Ludlum 2401 P Meter	0.1	0.1	0.1	0.1	0.1	0.1	-	-
S38	Ludlum 2401 P Meter	0.1	0.1	0.1	-	0.1	0.1	-	-
S39	Ludlum 2401 P Meter	0.1	0.1	0.1	0.1	0.1	0.1	-	-
S40	Thermo Eberline Proportional	0.09	0.10	0.09	0.08	0.09	0.08	0.09	0.08
S41	Thermo Eberline Proportional	0.09	0.10	0.09	0.08	0.09	0.08	0.09	0.08
S43	Thermo Eberline Proportional	0.15	0.16	0.12	0.15	0.16	0.13	0.11	0.12
S56	DKS-96 BDKS-96	0.13	0.12	0.14	0.15	0.12	0.13	0.12	0.12
S57	DKS-96 BDKS-96	0.18	0.18	0.19	0.18	0.19	0.20	0.12	0.13
S58	DKS-96 BDKS-96	0.17	0.20	0.20	0.20	0.20	0.18	0.16	0.14
S59	DKS-96 BDKS-96	0.14	0.13	0.14	0.14	0.12	0.12	0.10	0.10
S60	DKS-96 BDKS-96	0.18	0.17	0.17	0.18	0.19	0.19	0.14	0.14
S61	DKS-96 BDKS-96	0.11	0.12	0.13	0.10	0.12	0.11	0.10	0.10
S62	MiniRad 1000	0.1	0.25	0.1	0.2	0.20	0.25	0.1	0.25
S63	MiniRad 1000	0.2	0.1	0.2	0.1	0.2	0.2	0.1	0.1
S64	MiniRad 1000	0.2	0.3	0.2	0.3	0.2	0.2	0.2	0.2
S65	MiniRad 1000	0.2	0.3	0.2	0.3	0.2	0.2	0.2	0.2
S66	MiniRad 1000	0.2	0.3	0.2	0.3	0.2	0.2	0.2	0.2

a) Five surveys were conducted; however, only one baseline measurement was required since the five shipments came from the same lot.

“-“ Measurement not recorded.

**TABLE B.8
GAMMA RADIATION DOSE RATES FOR EACH SHIPMENT**

SENES ID	Meter	Gamma Dose Rate, $\mu\text{Sv/h}$																							
		B0	B1	B3	BR0	BR1	BR3	MR0	MR1	MR3	FR0	FR1	FR3	F0	F1	F3	FL0	FL1	FL3	ML0	ML1	ML3	BL0	BL1	BL3
S1	Bicron MicroRem Dose Rate Meter	0.70	0.45	0.15	1.40	0.50	0.20	1.40	0.50	0.20	0.90	0.45	0.25	1.15	0.50	0.15	1.10	0.30	0.20	1.40	0.40	0.20	1.40	0.40	0.20
	Ludlum 19	0.80	0.45	0.20	1.15	0.55	0.30	1.15	0.55	0.30	0.85	0.50	0.30	1.15	0.60	0.20	1.15	0.45	0.25	1.25	0.50	0.25	1.25	0.50	0.25
S2	Bicron MicroRem Dose Rate Meter	1.05	0.60	0.30	1.60	0.75	0.30	1.70	0.95	0.40	2.10	1.00	0.35	2.05	0.60	0.30	1.40	0.60	0.40	2.10	0.75	0.45	2.00	0.85	0.40
	Ludlum 19	1.10	0.60	0.30	1.50	0.90	0.45	1.85	1.05	0.55	1.90	0.95	0.45	1.90	0.55	0.35	1.55	0.80	0.50	2.05	0.90	0.55	1.85	0.95	0.45
S3	Bicron MicroRem Dose Rate Meter	1.00	0.60	0.40	1.50	0.90	0.45	1.50	0.90	0.45	2.00	1.00	0.50	1.30	0.45	0.25	2.50	0.85	0.55	3.00	1.20	0.60	3.00	1.20	0.60
	Ludlum 19	1.10	0.65	0.40	1.80	1.00	0.55	1.80	1.00	0.55	2.20	1.20	0.65	1.25	0.80	0.30	2.70	1.00	0.65	2.80	1.25	0.75	2.80	1.25	0.75
S4	Bicron MicroRem Dose Rate Meter	1.55	0.70	0.30	2.30	0.95	0.40	2.00	1.15	0.50	2.40	1.10	0.40	2.10	0.60	0.30	1.80	1.00	0.50	2.50	1.05	0.55	2.50	0.95	0.50
	Ludlum 19	1.65	0.80	0.40	2.20	1.15	0.55	2.30	1.30	0.65	3.20	1.30	0.60	2.25	0.65	0.35	2.00	1.15	0.60	2.40	1.25	0.70	2.50	1.15	0.60
S5	Bicron MicroRem Dose Rate Meter	3.00	1.20	0.10	5.00	2.00	1.00	5.00	2.00	1.00	10.00	3.20	1.10	4.50	1.20	0.50	7.00	2.00	0.90	5.50	2.40	1.00	5.50	2.40	1.00
	Ludlum 19	3.00	1.80	0.80	6.00	2.80	1.60	6.00	2.80	1.60	10.00	3.80	1.90	4.30	1.60	0.80	8.00	2.10	1.60	7.00	2.80	1.60	7.00	2.80	1.60
S6	Bicron MicroRem Dose Rate Meter	1.35	0.65	0.30	1.50	0.90	0.45	1.95	0.95	0.50	2.10	1.00	0.30	2.20	0.65	0.20	1.70	0.80	0.35	1.80	0.65	0.35	1.95	0.75	0.30
	Ludlum 19	1.25	0.65	0.35	1.50	0.95	0.50	1.85	1.10	0.55	2.00	1.05	0.50	1.85	0.75	0.25	1.65	0.90	0.50	1.70	0.80	0.50	1.65	0.80	0.40
S7	Bicron MicroRem Dose Rate Meter	1.20	0.65	0.30	1.90	0.95	0.45	2.10	1.15	0.50	2.30	1.15	0.45	2.15	0.75	0.25	1.70	0.65	0.35	1.75	0.70	0.40	2.10	0.70	0.35
	Ludlum 19	1.30	0.70	0.35	1.75	1.05	0.55	1.95	1.20	0.60	2.30	1.20	0.35	1.90	0.85	0.35	1.65	0.80	0.45	1.70	0.85	0.45	2.05	0.80	0.45
S8	Bicron MicroRem Dose Rate Meter	0.28	0.18	0.14	1.20	0.50	0.35	1.45	0.85	0.40	1.55	0.90	0.50	3.70	1.15	-	1.35	0.80	0.45	2.10	0.80	0.40	1.40	0.65	0.35
	Ludlum 19	0.30	0.22	0.18	1.20	0.65	0.45	1.70	1.10	0.65	1.70	1.15	0.68	3.60	1.10	-	1.65	1.05	0.60	2.25	1.05	0.65	1.70	0.85	0.45
S9	Bicron MicroRem Dose Rate Meter	0.28	0.25	0.10	2.70	0.90	0.40	1.50	0.95	0.50	2.20	0.90	0.50	4.00	1.30	-	1.70	0.95	0.50	1.60	0.80	0.50	3.50	0.95	0.45
	Ludlum 19	0.30	0.23	0.18	2.90	1.05	0.55	1.90	1.15	0.70	2.05	1.10	0.60	3.95	1.50	-	1.80	1.05	0.60	2.00	1.10	0.65	3.80	1.20	0.55

**TABLE B.8 (Cont'd)
GAMMA RADIATION DOSE RATES FOR EACH SHIPMENT**

SENES ID	Meter	Gamma Dose Rate, $\mu\text{Sv/h}$																							
		B0	B1	B3	BR0	BR1	BR3	MR0	MR1	MR3	FR0	FR1	FR3	F0	F1	F3	FL0	FL1	FL3	ML0	ML1	ML3	BL0	BL1	BL3
S10	MiniRad 1000	1.2	0.8	0.4	1.3	0.8	0.5	1.4	0.9	0.5	1.3	0.8	0.5	1.5	0.9	0.5	1.2	0.9	0.5	1.3	0.8	0.5	1.4	0.9	0.5
S11	Berthold Proportional Counter Tube	10	4	0.8	12	4.5	1.4	12	6	1.6	12	4.5	1.4	10	4	0.8	12	4	1.4	12	6	1.6	12	4	1.4
S13a	MiniRad 1000	1.4	0.8	0.6	2.5	1.5	0.7	4.5	2.6	0.8	4.0	1.5	0.7	3.5	1.0	0.4	4.0	1.5	0.6	3.5	1.5	0.7	2.5	1.2	0.6
S13b	MiniRad 1000	1.5	0.8	0.4	3.0	1.2	0.7	3.5	1.7	0.7	4.5	1.5	0.7	4.0	2.5	0.5	4.5	1.7	0.7	3.5	1.7	0.7	2.5	1.0	0.5
S13c	MiniRad 1000	1.5	0.8	0.35	2.5	1.5	0.7	3.5	1.5	0.8	3.5	1.2	0.8	3.5	1.5	0.4	4.0	1.5	0.8	4.5	1.5	1.0	3.0	1.5	0.7
S13d	MiniRad 1000	1.5	1.0	0.7	2.5	1.5	0.7	3.5	2.5	0.8	4.0	2.0	0.9	3.5	1.5	0.5	4.5	2.0	0.9	4.0	1.5	0.8	2.0	1.5	0.7
S13e	MiniRad 1000	1.5	0.7	0.5	2.0	1.5	0.5	3.5	1.5	0.7	3.5	1.2	0.6	3.5	1.0	0.5	4.5	1.8	0.9	4.0	2.0	1.0	2.0	1.0	0.9
S14a	MiniRad 1000	0.5	0.5	1.5	1.5	0.7	0.5	2.0	1.5	1.0	3.0	1.5	0.5	2.5	0.8	0.5	3.0	1.5	0.7	3.0	1.0	0.6	1.5	0.7	0.5
S14b	MiniRad 1000	0.8	0.5	0.5	1.5	1.0	0.6	3.1	1.5	0.6	2.5	1.3	0.5	3.5	1.0	0.5	4.0	1.5	0.5	2.5	1.3	0.6	1.3	0.8	0.5
S14c	MiniRad 1000	0.8	0.5	0.4	1.5	0.9	0.5	3.0	1.5	0.5	3.0	1.5	0.5	3.0	1.5	0.4	4.0	1.8	0.7	3.0	2.0	0.6	1.5	0.8	0.5
S14d	MiniRad 1000	1.0	0.7	0.5	2.0	0.8	0.5	3.5	1.5	0.6	3.0	1.5	0.5	2.8	1.0	0.4	4.0	1.5	0.4	3.5	1.5	0.6	1.8	0.8	0.5
S14e	MiniRad 1000	1.2	0.8	0.5	1.7	0.8	0.5	3.5	1.8	0.6	3.5	1.3	0.6	3.5	1.5	0.4	4.0	1.8	0.6	3.5	1.5	0.7	2.5	1.2	0.5
S16	Ludlum 2401-P	1.1	0.3	0.1	1.2	0.4	0.2	1.9	0.7	0.2	1.2	0.5	0.1	1.3	0.5	0.1	1.9	0.6	0.1	1.9	0.8	0.1	1.8	0.6	0.1
S17	Ludlum 2401-P	1.1	0.6	0.1	1.6	0.6	0.1	1.9	0.9	0.1	1.9	0.8	0.1	2.0	0.8	0.1	2.1	0.8	0.1	1.9	0.9	0.2	1.8	0.5	0.1
S18	Ludlum 2401-P	1.1	0.3	0.1	1.4	0.6	0.2	2.2	0.8	0.3	1.7	0.7	0.2	1.6	0.6	0.2	1.8	0.6	0.2	2.1	1.0	0.3	1.4	0.7	0.2
S19	Ludlum 2401-P	1.2	0.6	0.2	1.9	1.4	0.2	3.0	1.5	0.3	2.0	1.2	0.2	1.8	0.5	0.1	1.2	0.8	0.1	1.9	0.9	0.2	2.1	0.6	0.2
S20	Ludlum 2401-P	1.8	0.8	0.2	1.9	0.9	0.2	2.5	0.9	0.3	1.6	0.6	0.2	1.9	0.7	0.1	2.1	1.3	0.2	2.8	1.4	0.2	2.5	1.2	0.2
S21	Ludlum 2401-P	1.1	0.5	0.3	1.8	0.9	0.2	2.0	0.8	0.3	1.9	0.8	0.3	1.9	0.8	0.1	2.2	0.8	0.2	1.6	0.6	0.2	1.5	0.4	0.1
S22	Ludlum 2401-P	0.9	0.3	0.1	1.6	0.3	0.1	1.6	0.7	0.1	1.9	0.7	0.2	1.8	0.8	0.1	1.4	0.6	0.1	2.0	0.8	0.2	1.8	0.5	0.2
S23	Ludlum 2401-P	1.1	0.6	0.2	2.2	1.0	0.2	3.8	1.1	0.3	3.3	0.9	0.4	3.3	1.2	0.3	4.2	1.2	0.6	4.4	0.6	0.4	1.4	0.6	0.3
S24	Ludlum 2401-P	1.1	0.4	0.1	2.0	0.8	0.1	2.5	0.9	0.1	1.9	0.8	0.1	1.8	0.8	0.1	1.8	0.8	0.1	2.0	0.8	0.1	1.9	0.8	0.1
S25	Ludlum 2401-P	1.6	0.6	0.2	2.8	1.0	0.3	4.4	1.4	0.4	3.5	1.0	0.3	2.1	0.9	0.2	2.2	1.0	0.3	3.2	1.3	0.4	2.1	0.9	0.4
S26	Ludlum 2401-P	2.1	0.8	0.2	2.1	0.9	0.2	3.1	1.2	0.3	3.4	1.3	0.2	4.0	1.5	0.3	3.5	1.3	0.2	4.1	1.8	0.3	2.4	1.0	0.2
S27	Bicron MicroRem Dose Rate Meter	2.00	0.76	0.30	4.00	1.20	0.49	3.10	1.15	0.58	3.20	1.05	0.45	2.70	0.85	0.31	2.30	0.91	0.50	3.70	1.35	0.60	2.10	1.00	0.50

**TABLE B.8 (Cont'd)
GAMMA RADIATION DOSE RATES FOR EACH SHIPMENT**

SENES ID	Meter	Gamma Dose Rate, $\mu\text{Sv/h}$																							
		B0	B1	B3	BR0	BR1	BR3	MR0	MR1	MR3	FR0	FR1	FR3	F0	F1	F3	FL0	FL1	FL3	ML0	ML1	ML3	BL0	BL1	BL3
S28	Bicron MicroRem Dose Rate Meter	3.10	0.82	0.38	4.80	1.35	0.58	5.10	1.51	0.62	4.9	1.41	0.60	4.50	1.10	0.35	4.00	1.25	0.55	5.00	1.51	0.60	4.00	1.10	0.52
S29	Bicron MicroRem Dose Rate Meter	1.50	0.61	0.29	3.10	0.85	0.37	2.60	0.99	0.45	3.40	0.88	0.43	3.00	0.70	0.30	2.30	0.75	0.45	3.00	0.93	0.45	2.70	0.81	0.45
S30	Bicron MicroRem Dose Rate Meter	2.50	0.90	0.30	4.20	1.25	0.49	4.50	1.45	0.60	3.90	1.25	0.51	3.50	0.95	0.37	3.10	1.10	0.55	4.00	1.35	0.61	3.10	1.15	0.55
S31	Bicron MicroRem Dose Rate Meter	2.10	0.75	0.45	4.00	1.43	0.61	6.00	2.00	0.75	3.20	1.38	0.59	3.00	1.05	0.38	4.00	1.37	0.70	4.80	1.75	0.77	3.90	1.33	0.62
S32	Ludlum 2401-P	9.9	2.4	0.8	12.2	4.2	1.2	14.6	5.0	1.4	14.0	4.0	1.2	10.0	3.2	0.7	16.2	4.8	1.2	13.5	4.6	1.2	11.1	4.4	1.2
S33	Ludlum 2401-P	8.8	3.2	0.9	25.0	5.2	1.8	29.0	6.8	2.0	29	5.4	1.9	18.0	4.7	1.2	25.0	5.4	1.8	28.0	6.6	2.0	26.0	5.2	1.8
S34	Ludlum 2401-P	9.9	3.4	0.9	17.2	3.7	1.2	20.0	5.0	1.7	11.2	3.8	1.4	7.6	2.4	0.8	13.6	3.5	1.1	19.8	4.4	1.3	17.4	4.9	1.2
S35	Bicron MicroRem Dose Rate Meter	2.50	1.18	0.51	4.50	1.95	1.02	14.8	4.00	1.30	13.0	3.50	1.20	12.00	2.00	0.65	12.30	3.10	1.24	18.50	4.80	1.40	4.60	2.50	1.10
S36	Ludlum 2401-P	1.9	0.6	0.2	2.1	0.9	0.2	4.4	1.4	0.4	2.0	0.9	0.3	2.2	0.7	0.1	2.1	0.7	0.1	3.8	1.2	0.3	2.0	0.7	0.2
S37	Ludlum 2401-P	1.1	0.6	0.2	2.0	0.8	0.2	2.3	1.1	0.2	1.9	0.8	0.2	2.0	1.0	0.2	1.9	1.0	0.2	3.2	1.4	0.3	2.3	1.1	0.2
S38	Ludlum 2401-P	2.2	0.8	0.1	2.7	1.0	0.3	4.8	1.4	0.5	3.2	1.1	0.2	3.0	1.0	0.2	2.5	0.9	0.1	3.4	1.0	0.2	3.8	1.0	0.2
S39	Ludlum 2401-P	5.8	3.8	1.4	12.0	3.2	1.8	10.5	3.7	1.4	11.2	3.6	1.4	9.0	3.9	1.2	10.1	2.9	1.2	10.2	3.2	1.4	10.4	3.1	1.2
S40	Thermo Eberline Proportional	0.5	0.4	0.2	1.4	0.3	0.2	1.5	0.4	0.2	1.6	0.5	0.2	1.1	0.27	0.14	1.5	0.4	0.2	1.4	0.3	0.2	1.4	0.5	0.2
S41	Thermo Eberline Proportional	0.2	0.13	0.13	0.21	0.16	0.12	0.23	0.16	0.12	0.29	0.17	0.12	0.19	0.17	0.12	0.28	0.13	0.13	0.27	0.16	0.13	0.25	0.16	0.13
S43	Thermo Eberline Proportional	1.4	0.88	0.43	1.8	0.59	0.47	3.0	1.2	0.43	2.8	1.3	0.45	2.0	1.4	0.32	2.5	0.7	0.3	2.5	0.8	0.4	1.80	0.83	0.43
S56	DKS-96 BDKS-96	0.72	0.41	0.32	1.01	0.64	0.45	1.89	1.00	0.51	2.42	0.96	0.49	1.64	0.69	0.32	7.20	2.99	1.10	7.32	3.03	1.08	1.19	0.73	0.44
S57	DKS-96 BDKS-96	0.78	0.38	0.22	0.93	0.58	0.35	1.59	0.82	0.45	1.42	0.71	0.43	1.64	0.72	0.40	1.64	0.82	0.45	2.20	0.88	0.50	0.86	0.62	0.47
S58	DKS-96 BDKS-96	1.34	0.53	0.33	1.42	0.65	0.40	2.26	0.92	0.45	1.54	0.71	0.42	1.54	0.56	0.36	1.64	0.82	0.45	2.20	0.88	0.50	0.86	0.62	0.47
S59	DKS-96 BDKS-96	1.00	0.46	0.26	1.22	0.55	0.35	1.54	0.78	0.39	1.46	0.67	0.32	1.48	0.54	0.29	1.46	0.68	0.34	1.49	0.72	0.32	1.24	0.51	0.29
S60	DKS-96 BDKS-96	3.06	1.31	0.52	4.51	1.86	0.73	4.75	2.05	0.83	3.27	1.45	0.70	3.77	1.38	0.53	4.54	1.90	0.81	5.26	2.24	0.86	3.21	1.62	0.56
S61	DKS-96 BDKS-96	3.42	1.50	0.60	5.81	2.63	1.08	6.15	3.23	1.32	8.26	3.32	1.09	7.63	2.96	0.86	7.20	2.99	1.10	7.32	3.03	1.08	3.97	1.95	0.83
S62	MiniRad 1000	16	3	1.2	16	4	1.2	15	4	1.4	16	4	1.1	16	3	1.4	16	3	1.3	16	4	1.3	16	3	1.3
S63	MiniRad 1000	8	0.9	0.5	9	1.5	0.5	9	1.5	0.5	9	1.5	0.5	9	1.0	0.5	9	1.5	0.6	9	1.5	0.5	9	1.5	0.6
S64	MiniRad 1000	2.5	1.5	0.8	4.0	2.0	0.8	5.0	2.5	1.0	2.0	1.0	0.6	1.5	0.7	0.5	1.5	1.0	0.6	6.0	2.0	1.0	5.0	1.5	0.8
S65	MiniRad 1000	2.0	1.5	0.6	5.0	2.0	0.8	6.0	2.5	1.0	2.5	1.5	0.7	1.5	0.8	0.5	2.5	1.0	0.6	5.0	2.0	0.9	6.5	2.0	0.8
S66	MiniRad 1000	2.5	1.0	0.5	6.0	2.5	0.8	5.0	2.5	1.0	2.5	1.5	0.6	1.5	0.6	0.5	3.0	1.5	0.6	5.0	2.0	0.7	6.0	2.0	0.8

ANNEX C
MICROSHIELD MODELLING

ANNEX C MICROSIELD MODELLING

The purpose of this annex is to provide the modelling assumptions used in the gamma radiation dose rate calculations for transportation of tantalum materials in: i) sea-land container with one full tier; ii) sea-land container with one full tier and half-full second tier (Maximum Load); and iii) one full trailer load. The gamma dose rates were calculated using the MicroShield (Version 6.02) model (Grove Software 2005). This model is designed to incorporate the appropriate shielding with the corresponding geometry to calculate the gamma exposure rate at the specified distances. This exposure rate is used with parameters from ICRP 51 (1987) to calculate various gamma dose rates in MicroShield. For each loading configuration, the deep dose equivalent rate (mSv/h) with the rotational geometry was extracted from MicroShield.

The gamma dose rates from the sea-land container were calculated assuming the material was in standard 200 litre (L) drums or one tonne bags, while the gamma dose rates from the trailer were calculated assuming the material would be in standard 200 litre drums. The modelling assumptions provided in this annex include shielding, source density, source composition, source concentration and source dimensions.

C.1 SHIELDING

The shielding used in the gamma dose rate calculations for the tantalum material in the drums and bags within the sea-land container is different since the bags provide no shielding. The tantalum material in the 200 litre (L) drums was assumed to have 5 mm of iron (1.35 mm from the drums and at least 3 mm from the sea-land container (personal communication with sea-land container manufacturer, MHF Logistics, 2005), while the tantalum material in bags was only shielded by the 3 mm of iron from the sea-land container. Furthermore, the shielding for the full trailer load was assumed to be 3 mm of iron, which would be appropriate for both ‘flat-bed’ trailers and standard trailers. The trailer load would either consist of one sea-land container on a ‘flat-bed’ trailer or 200 litre drums on pallets in a standard trailer (i.e. sides on trailer). The shielding from the load on the ‘flat-bed’ trailer would be 3 mm of iron from the sea-land container, while the standard trailers would have approximately half of the 3 mm iron shielding from the drums and half from the shielding from the trailer.

In order to determine the effect on the gamma dose rate from sea-land containers of different iron thicknesses, the gamma dose rates from tantalum material in a sea-land container were calculated for a variety of different iron shields (0 mm, 3 mm, 5 mm and 8 mm). These results are discussed in Section C.7 at the end of this annex.

C.2 SOURCE DENSITY

The density of the material was dependent upon the type of tantalum material, concentrate or slag. The density of each of these transported tantalum materials was determined by laboratory analyses. The laboratory analyses showed that the density of the concentrates was typically around 3 g/cm³, while the density of the slag materials was typically around 2 g/cm³. Therefore, these densities were used for the concentrate and slag for each loading configuration.

In addition to calculating the gamma radiation dose rates for the three loading configurations listed above, the gamma radiation dose rates were calculated for sources with different densities, in order to see the effect of source density. The results of this sensitivity analysis are discussed in Section C.7 at the end of this annex.

C.3 SOURCE COMPOSITION

The composition of each tantalum material (concentrate and slag) is different, with the tantalum oxide (Ta₂O₅) ranging from 2 to 54% of the sample. Therefore, a sensitivity analysis was completed that calculated the gamma dose rates with a range of Ta₂O₅ compositions (2% to 73%¹³). The gamma dose rates from the range of Ta₂O₅ compositions were within 10%; therefore, the % of Ta₂O₅ can be considered to have an insignificant effect on the gamma dose rate. A further discussion of this sensitivity analysis is provided in Section C.7 at the end of this annex.

The Ta₂O₅ composition of the concentrate and slag were developed by using Material Safety Data Sheets that were received for tantalum glass (another name for slag) and tantalum/niobium concentrate. The source composition used in all of the loading configurations for the concentrate and slag is provided in Table C.1.

**TABLE C.1
CONCENTRATE AND SLAG COMPOSITION**

Source	% Composition									
	Ta ₂ O ₅	Nb ₂ O ₅	U ₃ O ₈	ThO ₂	Al ₂ O ₃	TiO ₂	SiO ₂	SnO ₂	Fe ₂ O ₃	MnO
Concentrate	30	18	0.4	0.05	7	8	15	3.55	18	-
Slag	20	8	1	0.1	10	10	20	3.9	20	7

¹³ For conservative purposes, the Ta₂O₅ compositions ranged up to 73% in order to exceed the % composition in the shipments used for this analysis.

C.4 SOURCE CONCENTRATION

As mentioned in the report, tantalum materials (concentrate and slag) contain varying levels of natural uranium and natural thorium (both with the corresponding decay products). Therefore, in order to account for the difference in the uranium and thorium content between different samples, U-238 (~99% of natural uranium) and Th-232 (100% of natural thorium), the gamma dose rates were calculated separately for uranium, thorium and corresponding decay products in MicroShield (i.e. separate MicroShield runs for U-238 and Th-232).

The U-238 and Th-232 were assumed to be in equilibrium with the corresponding decay products and set equal to 1 Bq/g. Therefore, the resulting gamma dose rate was a unit dose rate (mSv/h per Bq/g) and can be scaled by being multiplied by the desired activity (Bq/g) to calculate specific dose rates.

C.5 SOURCE DIMENSIONS

The source dimensions used for each loading configuration: i) sea-land container with one full tier; ii) sea-land container with one full tier and half-full second tier; and iii) one full trailer load are provided in Sections C.5.1, C.5.2 and C.5.3, respectively.

C.5.1 Sea-land Container with One Full Tier

The gamma dose rates for one full tier of tantalum materials (concentrate or slag) in a sea-land container were calculated assuming that material completely filled the sea-land container. Therefore, the source dimensions were the same as sea-land container dimensions except for the height. The source dimensions were:

- Length of 6.1 m,
- Width of 2.4 m; and
- Height of 1 m.

C.5.2 Sea-land Container with One Full Tier and One Partial Tier

The gamma dose rates for one full tier and one half-full tier of tantalum materials (concentrate and slag) in a sea-land container were modelled separately in MicroShield. The gamma dose rates calculated from the full 1st tier (Section C.5.1) were added to the gamma dose rates from the half-full 2nd tier (described in this section). The 2nd tier of the container was assumed to have one row of pallets that went down the middle of the container; therefore, this tier had half the number of pallets as the full 1st tier. The 2nd tier was assumed to be equal to the width of one pallet (assumed pallet was 2 drums wide) and the height of one tier. For conservative purposes, the 2nd tier was assumed to have no voids within the load and was assumed to be the entire length of the sea-land container. The source dimensions were:

- Length of 6.1 m,
- Width of 1.2 m; and
- Height of 1 m.

C.5.3 Trailer with a Full Load

The gamma dose rates for one full trailer load of tantalite concentrate were calculated assuming that material completely filled the trailer¹⁴. Therefore, the length and width of the source was set equal to the trailer dimensions provided in the gamma survey results received from a T.I.C. member company, while the source height was set equal to the height of the load. The source dimensions were:

- Length of 15.8 m,
- Width of 2.5 m; and
- Height of 1 m.

C.6 MICROSHIELD DOSE FACTORS

The parameter values described in Sections C.1 through to C.5 were used in the MicroShield calculated gamma radiation dose factors. These dose factors were calculated for the three different loading scenarios, i) sea-land container with one full tier; ii) sea-land container with one full tier and half-full second tier (Maximum load); and iii) one full trailer load. The dose factors for each loading scenario are shown in Table C.2.

¹⁴ The tantalite concentrate was assumed to be placed in drums on pallets directly in the trailer, rather than in a sea-land container.

**TABLE C.2
MICROSHIELD DOSE FACTORS FOR EACH LOADING SCENARIO**

Load Configuration	Type of Material	Location of Dose Point with respect to Load	Decay Series	Distance from Load to Dose Point (m)	Deep Dose Equivalent Rate (Rotational Geometry) (mSv/h per Bq/g)	Deep Dose Equivalent Rate (Rotational Geometry) (µSv/h per Bq/g)
F - 1 Tier	Tantalite	1 m above ground, centre of short side (back/front)	U-238	0.02	1.71E-04	1.71E-01
				1	7.31E-05	7.31E-02
				3	1.54E-05	1.54E-02
				10	1.57E-06	1.57E-03
			Th-232	0.02	3.87E-07	3.87E-04
				1	2.54E-04	2.54E-01
				3	1.07E-04	1.07E-01
				10	2.26E-05	2.26E-02
		1 m above ground, centre of long side (right/left)	U-238	0.02	1.71E-04	1.71E-01
				1	9.40E-05	9.40E-02
				3	2.99E-05	2.99E-02
				10	3.85E-06	3.85E-03
			Th-232	0.02	9.73E-07	9.73E-04
				1	2.54E-04	2.54E-01
				3	1.38E-04	1.38E-01
1 m above ground, 1 m along long side (near corner)	U-238	0.02	1.72E-04	1.72E-01		
		1	8.17E-05	8.17E-02		
		3	2.47E-05	2.47E-02		
	Th-232	0.02	2.55E-04	2.55E-01		
		1	1.20E-04	1.20E-01		
		3	3.62E-05	3.62E-02		

TABLE C.2 (Cont'd)
MICROSHIELD DOSE FACTORS FOR EACH LOADING SCENARIO

Load Configuration	Type of Material	Location of Dose Point with respect to Load	Decay Series	Distance from Load to Dose Point (m)	Deep Dose Equivalent Rate (Rotational Geometry) (mSv/h per Bq/g)	Deep Dose Equivalent Rate (Rotational Geometry) (µSv/h per Bq/g)
F - 1 Tier	Slag	1 m above ground, centre of short side (back/front)	U-238	0.02	1.91E-04	1.91E-01
				1	7.40E-05	7.40E-02
				3	1.56E-05	1.56E-02
				10	1.61E-06	1.61E-03
				20	3.98E-07	3.98E-04
			Th-232	0.02	2.81E-04	2.81E-01
				1	1.07E-04	1.07E-01
				3	2.27E-05	2.27E-02
				10	2.35E-06	2.35E-03
				20	5.84E-07	5.84E-04
		1 m above ground, centre of long side (right/left)	U-238	0.02	1.91E-04	1.91E-01
				1	9.72E-05	9.72E-02
				3	3.06E-05	3.06E-02
				10	3.95E-06	3.95E-03
				20	1.00E-06	1.00E-03
			Th-232	0.02	2.81E-04	2.81E-01
				1	1.42E-04	1.42E-01
				3	4.45E-05	4.45E-02
				10	5.77E-06	5.77E-03
				20	1.47E-06	1.47E-03
1 m above ground, 1 m along long side (near corner)	U-238	0.02	1.91E-04	1.91E-01		
		1	8.39E-05	8.39E-02		
		3	2.53E-05	2.53E-02		
	Th-232	0.02	2.81E-04	2.81E-01		
		1	1.22E-04	1.22E-01		
		3	3.68E-05	3.68E-02		

TABLE C.2 (Cont'd)
MICROSHIELD DOSE FACTORS FOR EACH LOADING SCENARIO

Load Configuration	Type of Material	Location of Dose Point with respect to Load	Decay Series	Distance from Load to Dose Point (m)	Deep Dose Equivalent Rate (Rotational Geometry) (mSv/h per Bq/g)	Deep Dose Equivalent Rate (Rotational Geometry) (µSv/h per Bq/g)		
Maximum Load	Tantalite	1 m above ground, centre of short side (back/front)	U-238	0.02	3.42E-04	3.42E-01		
				1	1.20E-04	1.20E-01		
				3	2.35E-05	2.35E-02		
				10	2.36E-06	2.36E-03		
				20	5.80E-07	5.80E-04		
			Th-232	0.02	5.07E-04	5.07E-01		
				1	1.75E-04	1.75E-01		
				3	3.44E-05	3.44E-02		
				10	3.47E-06	3.47E-03		
				20	8.55E-07	8.55E-04		
		1 m above ground, centre of long side (right/left)	U-238	0.02	2.95E-04	2.95E-01		
				1	1.57E-04	1.57E-01		
				3	5.29E-05	5.29E-02		
				10	7.29E-06	7.29E-03		
				20	1.89E-06	1.89E-03		
			Th-232	0.02	4.36E-04	4.36E-01		
				1	2.31E-04	2.31E-01		
				3	7.75E-05	7.75E-02		
				10	1.07E-05	1.07E-02		
				20	2.79E-06	2.79E-03		
				1 m above ground, 1 m along long side (near corner)	U-238	0.02	2.85E-04	2.85E-01
						1	1.34E-04	1.34E-01
					Th-232	3	4.39E-05	4.39E-02
						0.02	4.21E-04	4.21E-01
1	1.96E-04	1.96E-01						
3	6.44E-05	6.44E-02						

TABLE C.2 (Cont'd)
MICROSHIELD DOSE FACTORS FOR EACH LOADING SCENARIO

Load Configuration	Type of Material	Location of Dose Point with respect to Load	Decay Series	Distance from Load to Dose Point (m)	Deep Dose Equivalent Rate (Rotational Geometry) (mSv/h per Bq/g)	Deep Dose Equivalent Rate (Rotational Geometry) (µSv/h per Bq/g)
	Slag	1 m above ground, centre of short side (back/front)	U-238	0.02	3.81E-04	3.81E-01
				1	1.20E-04	1.20E-01
				3	2.38E-05	2.38E-02
				10	2.42E-06	2.42E-03
			Th-232	20	5.97E-07	5.97E-04
				0.02	5.59E-04	5.59E-01
				1	1.74E-04	1.74E-01
				3	3.45E-05	3.45E-02
		1 m above ground, centre of long side (right/left)	U-238	0.02	3.21E-04	3.21E-01
				1	1.62E-04	1.62E-01
				3	5.40E-05	5.40E-02
				10	7.47E-06	7.47E-03
			Th-232	20	1.95E-06	1.95E-03
				0.02	4.70E-04	4.70E-01
				1	2.36E-04	2.36E-01
				3	7.87E-05	7.87E-02
		1 m above ground, 1 m along long side (near corner)	U-238	0.02	3.02E-04	3.02E-01
				1	1.35E-04	1.35E-01
				3	4.40E-05	4.40E-02
				10	4.42E-06	4.42E-03
			Th-232	20	1.96E-06	1.96E-03
				0.02	4.42E-04	4.42E-01
				1	1.96E-04	1.96E-01
				3	6.42E-05	6.42E-02

TABLE C.2 (Cont'd)
MICROSHIELD DOSE FACTORS FOR EACH LOADING SCENARIO

Load Configuration	Type of Material	Location of Dose Point with respect to Load	Decay Series	Distance from Load to Dose Point (m)	Deep Dose Equivalent Rate (Rotational Geometry) (mSv/h per Bq/g)	Deep Dose Equivalent Rate (Rotational Geometry) (µSv/h per Bq/g)
Inside sea-land with F - 1 Tier	Tantalite	1 m above ground, centre of long side (right/left)	U-238	0.5	1.55E-04	1.55E-01
			Th-232	0.5	2.26E-04	2.26E-01
	Slag	1 m above ground, centre of long side (right/left)	U-238	0.5	1.59E-04	1.59E-01
			Th-232	0.5	2.30E-04	2.30E-01
Inside sea-land with Maximum Load	Tantalite	1 m above ground, centre of long side (right/left)	U-238	0.5	3.10E-04	3.10E-01
			Th-232	0.5	4.53E-04	4.53E-01
	Slag	1 m above ground, centre of long side (right/left)	U-238	0.5	3.17E-04	3.17E-01
			Th-232	0.5	4.60E-04	4.60E-01
F - Trailer	Tantalite	1 m above ground, centre of short side (back/front)	U-238	0.02	1.90E-04	1.90E-01
				1	7.85E-05	7.85E-02
			3	1.67E-05	1.67E-02	
			Th-232	0.02	2.82E-04	2.82E-01
				1	1.15E-04	1.15E-01
			3	2.44E-05	2.44E-02	
	1 m above ground, centre of long side (right/left)	U-238	0.02	1.97E-04	1.97E-01	
			1	1.05E-04	1.05E-01	
Th-232	0.02	3	4.14E-05	4.14E-02		
		2.90E-04	2.90E-01			

TABLE C.2 (Cont'd)
MICROSHIELD DOSE FACTORS FOR EACH LOADING SCENARIO

Load Configuration	Type of Material	Location of Dose Point with respect to Load	Decay Series	Distance from Load to Dose Point (m)	Deep Dose Equivalent Rate (Rotational Geometry) (mSv/h per Bq/g)	Deep Dose Equivalent Rate (Rotational Geometry) (µSv/h per Bq/g)
		Th-232		1	1.54E-04	1.54E-01
				3	6.06E-05	6.06E-02
		1 m above ground, 1 m along long side (near corner)	U-238	0.02	1.94E-04	1.94E-01
				1	8.76E-05	8.76E-02
			3	2.85E-05	2.85E-02	
			Th-232	0.02	2.86E-04	2.86E-01
				1	1.28E-04	1.28E-01
				3	4.17E-05	4.17E-02

C.7 SENSITIVITY ANALYSIS

In addition to gamma dose rate calculations for each loading configuration (using the assumptions provided in Sections C.1 to C.5), several MicroShield runs were completed that varied the parameters used in the gamma dose rate calculations. The parameters that were examined in this sensitivity analysis included:

- Shielding Thickness (Section C.7.1);
- Source Density (Section C.7.2);
- Source Composition (C.7.3);
- Location of Load Inside Sea-land Container (C.7.4);
- Contact Distance from Sea-land Container (C.7.5).

C.7.1 Shielding Thickness

The effect of the shielding thickness was examined by calculating gamma dose rates (using MicroShield) from the end of a sea-land container with different thicknesses of iron containing one full tier of tantalite. It should be noted that all parameters (except for the iron thickness of the sea-land container) were set to the appropriate value described in the previous sections (Sections C.1 to C.5). The iron thicknesses used for this sensitivity analysis were 0 cm (i.e. no iron shielding), 0.3 cm, 0.5 cm and 0.8 cm. The resulting gamma dose rates were analyzed using two different comparisons, as shown in Table C.3.

**TABLE C.3
COMPARISON OF GAMMA DOSE RATES WITH DIFFERENT IRON SHIELD THICKNESSES**

Load Config.	Type of Material	Location of Dose Point	Decay Series ^a	Thickness of Iron Shield (cm)	Deep Dose Equivalent (Rotational), mSv/h			1st Comparison % ^b			2nd Comparison % ^c		
					Distance from Sea-land Container			Distance from Sea-land Container			Distance from Sea-land Container		
					0.02 m	1 m	3 m	0.02 m	1 m	3 m	0.02 m	1 m	3 m
1 Full Tier	Tantalite	End of Container	U-238	0	2.4E-04	8.3E-05	1.7E-05	100	100	100	NA	NA	NA
				0.3	1.9E-04	7.7E-05	1.6E-05	81	93	94	113	105	105
				0.5	1.7E-04	7.3E-05	1.5E-05	72	88	90	100	100	100
				0.8	1.5E-04	6.7E-05	1.4E-05	61	81	84	85	92	93
			Th-232	0	3.5E-04	1.2E-04	2.5E-05	100	100	100	NA	NA	NA
				0.3	2.8E-04	1.1E-04	2.4E-05	82	93	94	112	105	104
				0.5	2.5E-04	1.1E-04	2.3E-05	73	88	90	100	100	100
				0.8	2.2E-04	9.9E-05	2.1E-05	63	81	84	86	92	94

Notes:

- a) U-238 and Th-232 at 1 Bq/g. Therefore, the dose rates are in units of mSv/h per Bq/g.
 b) 1st Comparison: % = 100*(Dose Rate for given thickness/Dose Rate at 0 cm thickness).
 c) 2nd Comparison: % = 100*(Dose Rate for given thickness/Dose Rate at 0.5 cm thickness).
 NA – Not Applicable.

The first comparison examined the difference between the gamma dose rates from a sea-land container with the different iron thicknesses with respect to a sea-land container with no shielding (i.e. 0 cm of iron). As shown in Table C.3, the gamma dose rates ranged from approximately 20% (0.3 cm of iron) to 40% (0.8 cm of iron) lower than the sea-land container with no shielding. This confirmed that the presence of iron in the walls of a sea-land container containing tantalum material does have an effect on the gamma dose rate. Therefore, since there is always iron in the walls of a sea-land container a second comparison of the gamma dose rates was completed. It should be noted that this comparison did not include the gamma dose rates with no shielding since these gamma dose rates are not applicable (sea-land containers have at least 0.3 cm of iron, Section C.1).

The second comparison examined the difference between the gamma dose rates from sea-land containers with 0.3 cm and 0.8 cm iron thicknesses with those from sea-land container with 0.5 cm iron thickness (assigned shielding thickness for tantalite, Section C.1). As shown in Table C.3, the gamma dose rates were less than 15% different. Therefore, it was concluded that the shielding thicknesses of 0.5 cm (tantalite) and 0.3 cm (slag) are reasonable in calculating the gamma dose rates from sea-land containers since varying the iron thickness does not have a large effect on the gamma dose rate from a sea-land container containing tantalum material.

C.7.2 Source Density

The effect of density was examined by calculating gamma dose rates (using MicroShield) from the end of a sea-land container with one full tier of tantalite for different source densities. It should be noted that all parameters (except for the source density) were set to the appropriate value described in the previous sections (Sections C.1 to C.5). The densities used for this sensitivity analysis were 3 g/cm³ (density assigned for tantalite, Section C.2) and 1.5 g/cm³ (half of the assigned density). The resulting gamma dose rates from the two densities were no more than 5% different for all three dose points (0.02 m, 1 m and 3 m from the sea-land container), as shown in Table C.4. Therefore, it was concluded that varying the density has an insignificant effect on the gamma dose rate from a transport container containing tantalum material.

**TABLE C.4
COMPARISON OF GAMMA DOSE RATES WITH DIFFERENT DENSITIES**

Load Configuration	Type of Material	Location of Dose Point	Decay Series ^a	Distance from Container (m)	Deep Dose Equivalent (Rotational), mSv/h		Comparison % ^b
					1.5 g/cm ³	3 g/cm ³	
1 Full Tier	Tantalite	End of Container	U-238	0.02	1.7E-04	1.7E-04	100
				1	6.9E-05	7.3E-05	94
				3	1.5E-05	1.5E-05	96
			Th-232	0.02	2.5E-04	2.5E-04	100
				1	1.0E-04	1.1E-04	94
				3	2.1E-05	2.3E-05	95

Notes:

a) U-238 and Th-232 at 1 Bq/g.

b) Comparison % = 100*(Dose Rate at 1.5 g/cm³) / (Dose Rate at 3 g/cm³).

C.7.3 Source Composition

The effect of the source composition was examined by calculating the gamma dose rates (using MicroShield) from the end of a sea-land container with one full tier of tantalite of different Ta₂O₅ compositions (i.e. source compositions). It should be noted that all parameters (except for the source composition) were set to the appropriate value described in the previous sections (Section C.1 to C.5). The Ta₂O₅ compositions used for this sensitivity analysis were 2%, 15%, 24%, 38% and 73%. The resulting dose rates along with a comparison of the gamma dose rates with respect to the 2% Ta₂O₅ are provided in Table C.5. As shown in Table C.5, the gamma dose rates (with respect to the 2% Ta₂O₅) are less than 10% different; therefore, it was concluded that varying the source composition (with respect to Ta₂O₅) has an insignificant effect on the gamma dose rate from a transport container containing tantalum material.

**TABLE C.5
COMPARISON OF GAMMA DOSE RATES WITH DIFFERENT
SOURCE COMPOSITIONS**

Load Config.	Type of Material	Location of Dose Point	Decay Series ^a	Distance from Container (m)	Deep Dose Equivalent (Rotational) (mSv/h)					Comparison % ^b				
					2% Ta ₂ O ₅	15% Ta ₂ O ₅	24% Ta ₂ O ₅	38% Ta ₂ O ₅	73% Ta ₂ O ₅	2% Ta ₂ O ₅	15% Ta ₂ O ₅	24% Ta ₂ O ₅	38% Ta ₂ O ₅	73% Ta ₂ O ₅
1 Full Tier	Tantalite	End of Container	U-238	0.02	1.7E-04	1.7E-04	1.7E-04	1.7E-04	1.6E-04	100	101	98	97	94
				1	7.4E-05	7.4E-05	7.3E-05	7.2E-05	6.9E-05	100	100	98	96	93
				3	1.6E-05	1.6E-05	1.5E-05	1.5E-05	1.5E-05	100	100	98	96	93
			Th-232	0.02	2.6E-04	2.6E-04	2.5E-04	2.5E-04	2.4E-04	100	99	97	95	92
				1	1.1E-04	1.1E-04	1.1E-04	1.0E-04	1.0E-04	100	98	97	95	92
				3	2.3E-05	2.3E-05	2.2E-05	2.2E-05	2.1E-05	100	98	97	95	92

Notes:

a) U-238 and Th-232 at 1 Bq/g.

b) Comparison % = 100*(Dose Rate for given composition/Dose Rate for 2% composition).

C.7.4 Location of Load Inside Sea-land Container

The location of the load inside the sea-land container was examined by calculating gamma dose rates (using MicroShield) with one tier of tantalite at different distances from the inside (end wall) of the container. These distances were varied by shortening the length of the source by the appropriate amount in order for the load to fit within the sea-land container. It should be noted that all parameters (except for the distance from the end of the load to the inside of the container) were set to the appropriate value described in previous sections (Sections C.1 to C.5). The resulting dose rates along with a comparison of the gamma dose rates with respect to the load located directly against the inside wall of the container are provided in Table C.6. As shown in Table C.6, the gamma dose rates (with respect to the load located directly against the inside container wall) were less than 20% lower with the load being located 20 cm away from the inside container wall. Therefore, for the purposes of conservatism, the gamma dose rates were calculated assuming a full load of tantalum material with the load located directly against the inside container wall.

**TABLE C.6
COMPARISON OF GAMMA DOSE RATES WITH LOAD AT DIFFERENT
LOCATIONS IN SEA-LAND CONTAINER**

Load Config.	Type of Material	Location of Dose Point	Decay Series ^a	Distance from Load to Inside Container Wall (cm)	Deep Dose Equivalent (Rotational) (mSv/h)			Comparison % ^b		
					0.02 m from container	1 m from container	3 m from container	0.02 m from container	1 m from container	3 m from container
1 Tier	Tantalite	End of Container	U-238	At End	1.7E-04	7.3E-05	1.5E-05	100	100	100
				5	1.7E-04	6.9E-05	1.5E-05	100	95	97
				10	1.7E-04	6.6E-05	1.5E-05	98	90	94
				15	1.6E-04	6.3E-05	1.4E-05	96	86	92
				20	1.6E-04	6.0E-05	1.4E-05	92	82	89

Notes:

- a) U-238 at 1 Bq/g.
- b) Comparison % = 100 * (Dose Rate at Given Distance/Dose Rate at End).

C.7.5 Contact Distance from Sea-land Container

The distance used to represent the contact dose rate was examined by calculating the gamma dose rates (using MicroShield) at different distances from the side of a sea-land container with one full tier of tantalite. It should be noted that all parameters (except for the distances from the sea-land container) were set to the appropriate value described in previous sections (Sections C.1 to C.5). The distances of 2 cm, 10 cm and 20 cm were selected as representative distances of the contact dose rate from the side of the sea-land container. The resulting dose rates along with a comparison of the gamma dose rates with respect to the 2 cm distance are provided in Table C.7.

As shown in Table C.7, the gamma dose rates (with respect to the 2 cm distance) are no more than 5% different; therefore, it was concluded that a 2 cm distance would be representative (conservative) of the contact dose rate of a sea-land container containing tantalum material.

**TABLE C.7
COMPARISON OF GAMMA DOSE RATES AT DIFFERENT DISTANCES THAT ARE
REPRESENTATIVE OF CONTACT DOSE RATES**

Load Configuration	Type of Material	Location of Dose Point	Decay Series ^a	Distance from Container (cm)	Deep Dose Equivalent (Rotational) (mSv/h)	Comparison % ^b
1 Full Tier	Tantalite	Side of Container	U-238	2	1.7E-04	100
				10	1.7E-04	99
				20	1.6E-04	95
			Th-232	2	2.5E-04	100
				10	2.5E-04	99
				20	2.4E-04	95

Notes:

- a) U-238 and Th-232 at 1 Bq/g.
- b) Comparison % = 100*(Dose Rate at given distance from container / Dose Rate at 2 cm from Container).

C.8 REFERENCES

Grove Software 2005. MicroShield Version 6.20.

International Commission on Radiological Protection (ICRP) 1987. *Data for Use in Protection Against External Radiation*. ICRP Publication 51. Annals of the ICRP, Vol. 17, No. 2/3.

ANNEX D

ANALYSIS OF MEASUREMENT DATA

ANNEX D ANALYSIS OF MEASUREMENT DATA

D.1 CONCENTRATIONS

D.1.1 Summary of Measurements Reported by the Laboratories

Concentrations reported by the laboratories have been summarized in Table D.1(a) for slag materials and Table D.1(b) for tantalite materials. The table includes activity concentrations of U-238 and Th-232 that were converted from the mass concentrations reported by the primary laboratory. The total, U-238 + Th-232, activity concentrations were calculated and have been reported in this table.

The table also includes a summary of the activity concentration for measured radionuclides in the natural uranium and thorium decay series. Concentrations of Ta₂O₅ and Nb₂O₅ are summarized as well as the bulk density.

TABLE D.1(a)
SUMMARY OF CONCENTRATIONS IN SLAG MATERIALS REPORTED BY THE LABORATORIES

Analyte	Units	Num.	< (%)	Median	Mean	Min	Max	>10 Bq/g (%)
PRIMARY LABORATORY								
U ₃ O ₈	%	22	0	0.04	0.18	0.02	0.88	
ThO ₂	%	22	0	0.17	0.18	0.05	0.78	
U-238	Bq/g	22	0	3.67	18.8	2.41	92.2	23
Th-232	Bq/g	22	0	5.88	6.55	1.78	27.8	5
Total	Bq/g	22	0	9.73	25.3	7.4	96.8	45
Ta ₂ O ₅	%	22	0	3	8.59	2	31	
Nb ₂ O ₅	%	22	0	3	4.48	0.5	10	
Bulk Density	g/cm ³	22	0	1.8	1.86	1.7	2.4	
INDEPENDENT LABORATORY								
Uranium	ppm	22	0	384	1437	320	6440	
Thorium	ppm	22	0	1420	1587	390	6780	
U-238	Bq/g	22	0	4.74	17.8	3.95	79.5	23
Th-234	Bq/g	22	0	4.5	19.3	3.1	93	
Th-230	Bq/g	14	29	5	5.21	4	8	
Ra-226	Bq/g	22	0	5.2	21.4	4	120	
Pb-210	Bq/g	22	5	2.85	3.15	0.6	5.2	
U-235	Bq/g	22	68	0.85	1.26	0.1	5	
Th-227	Bq/g	22	36	0.4	1.07	0.2	5.1	
Ra-223	Bq/g	22	64	1	1.47	0.4	6	
Th-232	Bq/g	22	0	5.76	6.44	1.58	27.5	5
Th-228	Bq/g	22	0	5.5	6	1.9	23	
Ra-228	Bq/g	22	0	5.9	6.04	2	24	
Total	Bq/g	22	0	10.5	24.2	8.98	83.8	95

Notes:

- i) The actual number of samples measured is lower than the number shipments because a single composite from five shipments of the same lot was used in two cases for slag materials. The actual number of sample measurements measured by the laboratories was 14 (i.e. 22 - 8).
- ii) Concentrations of U-238 and Th-232 converted from mass concentrations.
- iii) Percentage of measurements reported as less than detection limits for each analyte shown as "< (%)".

TABLE D.1(b)
SUMMARY OF CONCENTRATIONS IN TANTALITE MATERIALS REPORTED BY
THE LABORATORIES

Analyte	Units	Num.	< (%)	Median	Mean	Min	Max	>10 Bq/g (%)
PRIMARY LABORATORY								
Thorium	Bq/g	6	0	0.9	1.5	0.5	3.4	
Uranium	Bq/g	6	0	15	22.2	10	58	
U ₃ O ₈	%	45	0	0.13	0.16	0.04	0.65	
ThO ₂	%	45	24	0.01	0.04	0.01	0.31	
U-238	Bq/g	45	0	13.6	16.4	4.5	68.1	71
Th-232	Bq/g	45	24	0.5	1.28	0.18	11.1	2
Total	Bq/g	45	24	14.2	17.7	5.31	68.3	78
Ta ₂ O ₅	%	45	0	28	26.6	2	53.8	
Nb ₂ O ₅	%	45	0	8	10.9	2	28.9	
Bulk Density	g/cm ³	43	0	2.8	2.71	1.5	3.1	
INDEPENDENT LABORATORY								
Uranium	ppm	39	0	860	1271	290	6000	
Thorium	ppm	39	44	145	324	25	2650	
U-238	Bq/g	39	0	10.6	15.7	3.58	74.1	59
Th-234	Bq/g	39	0	13	15.5	4.2	53	
Th-230	Bq/g	6	50	10	12.3	2	28	
Ra-226	Bq/g	39	0	15	17.3	5.5	64	
Pb-210	Bq/g	39	0	12	13.9	4	45	
U-235	Bq/g	39	18	0.6	0.93	0.2	4	
Th-227	Bq/g	39	8	0.8	0.93	0.2	4.2	
Ra-223	Bq/g	39	15	0.7	1.04	0.2	5	
Th-232	Bq/g	39	44	0.59	1.31	0.1	10.8	3
Th-228	Bq/g	39	0	0.63	1.29	0.19	8.8	
Ra-228	Bq/g	39	5	0.7	1.33	0.3	9.2	

Note:

- i) Concentrations of U-238 and Th-232 converted from mass concentrations.
- ii) Percentage of measurements reported as less than detection limits for each analyte shown as "< (%)".

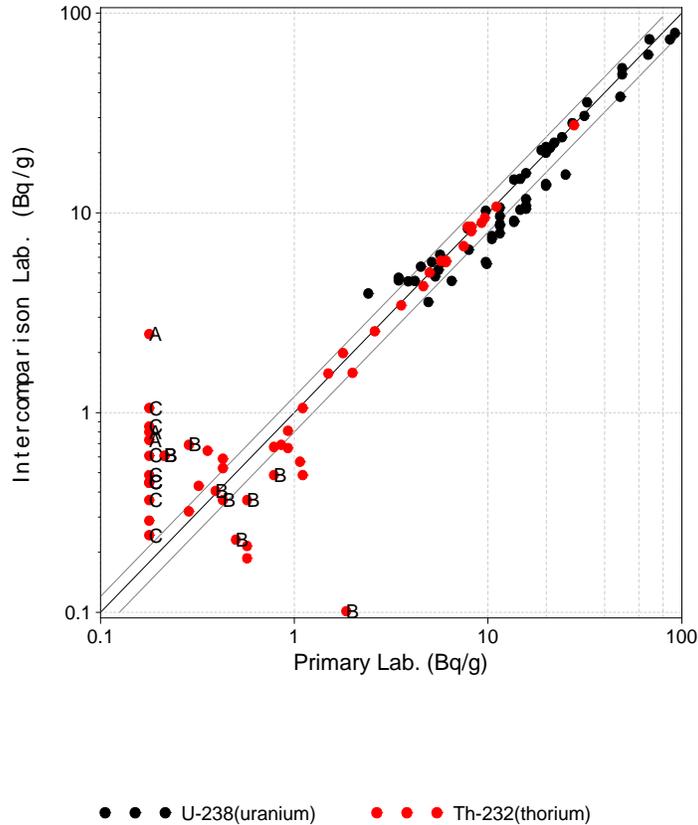
Overall, the frequency of “<” values was low for most radionuclides with the exception being Th-230 from the U-238 decay series and the U-235 decay series. Thorium, and Th-232, concentrations were reported as a “<” value about 20% of the time.

D.1.2 Data Quality Review and Interlaboratory Comparisons

The samples analyzed by the primary and independent laboratories were considered split samples of the same material. A comparison was conducted between the uranium and thorium concentrations measured by primary laboratory and the independent laboratory. Differences between the laboratory measurements could potentially arise from differences in sample digestion, interferences and biases in the analytical method, and the precision of the analytical method.

Figure D.1 shows a scatter plot of U-238 and Th-232 concentrations measured by the two laboratories. Lines of perfect agreement between the two laboratories and bounds of +/- 20% are shown. On a visual basis, most of the independent laboratory concentrations are within 20% of the primary laboratory concentrations for measurements of U-238 and for Th-232 measurements at levels above a value of a few Bq/g. There is less agreement between laboratories for Th-232 concentrations below a few Bq/g.

**FIGURE D.1
INTERLABORATORY COMPARISONS OF
URANIUM AND THORIUM MEASUREMENTS**



Note:

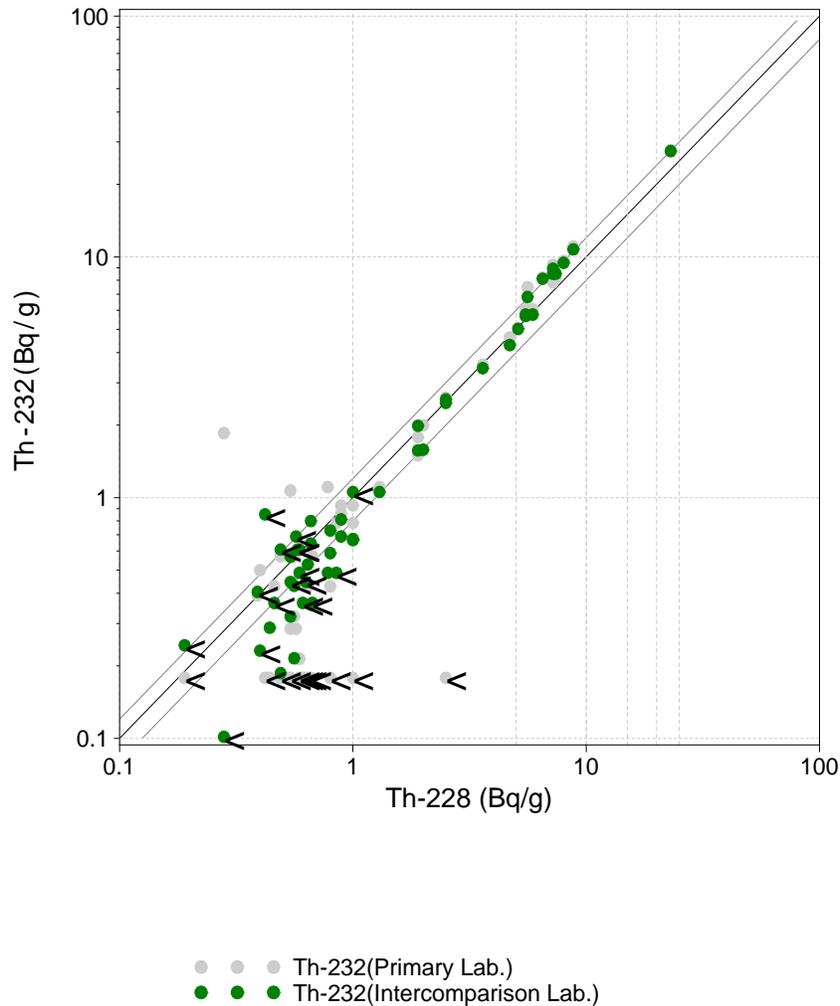
“A” indicates < for primary laboratory; “B” indicates < for independent laboratory; and, “C” indicates < for both laboratories.

Measurements of Low Concentrations of Th-232 (thorium)

There is less agreement in Th-232 concentrations measured by the two laboratories at Th-232 levels below about 3 Bq/g. This has been investigated further by comparing measurements of Th-228, measured by the independent laboratory using a different method, to the concentrations of Th-232 measured by the primary and independent laboratories. Equilibrium conditions are expected between Th-232 and Th-228 for these materials so that the activity concentrations were expected to be the same. Figure D.2 shows a scatter plot of Th-232 concentrations reported by the laboratories against the Th-228 concentration. There is a tendency for the independent laboratory analysis to underestimate Th-232 concentrations relative to the Th-228 concentration at low levels (i.e. there are more points below the line of perfect agreement compared to the number above the line). The figure also shows that the primary laboratory tends to

underestimate Th-232 concentrations at low levels as well: in particular, there are a number of samples where the reporting level of the primary laboratory is much lower than the measured value of Th-228. There is also a small number of measurements where the primary laboratory concentration is much higher than the Th-228 concentration.

FIGURE D.2
COMPARISON BETWEEN Th-232 AND Th-228 MEASUREMENTS



Note:

“<” indicates the value was plotted at the < value.

Concentrations of Th-232 below about 3 Bq/g that are measured by the primary and independent laboratories are somewhat uncertain at these levels of Th-232. Potentially, both methods may tend to underestimate Th-232 concentrations at these levels. This could potentially arise due to matrix interferences where the concentrations of other elements in the samples may affect the measurement of thorium. In general, the NAA measurements (completed by the independent laboratory) of Th-232 are in closer agreement with the measurements of Th-228 than the

measurements by the primary laboratory. Regardless, the uncertainty at these levels has little impact on the estimates of total activity, and resultant dose, as the higher and hence mean concentrations of Th-232, are minimally influenced by the uncertainty in the low concentration samples. Furthermore, the U-238 activity tends to be more than 10 times higher than the thorium content and therefore the thorium typically contributes a small fraction of the dose.

D.1.3 Uranium and Thorium Content

The interlaboratory comparison suggests that characterization of the mean and upper range of uranium and thorium concentrations is reasonably consistent. This provided confidence that the primary laboratory measurements could be used to characterize the uranium (U-238) and thorium (Th-232) concentrations of the materials. Samples from one company's shipments could not be sent outside the country to the primary laboratory; therefore, the concentrations of thorium and uranium in these samples were measured by an alternate laboratory. As a result, an interlaboratory comparison was not available for the concentrations measured by the alternate laboratory. It has been assumed that these measurements are comparable to measurements by the primary laboratory and are also appropriate to characterize the uranium and thorium content.

Table D.2 shows a summary of the activity concentrations of U-238 and Th-232 along with the total (U-238 + Th-232) activity concentration in the shipments using the reported concentrations from the primary laboratory with conversion from mass basis to activity basis. The concentrations were assumed equal to the reporting limit for concentrations reported as "<" by the laboratory. Overall, the concentrations of U-238 are higher than the concentrations of Th-232 with median concentrations of 16.4 and 1.3 Bq/g, respectively, in tantalite materials. The Th-232 concentrations tend to be higher in slag materials compared to tantalite materials with mean concentrations of 6.5 Bq/g for Th-232 and a U-238 mean concentration, 18.8 Bq/g, that was similar to the mean concentration in tantalite materials.

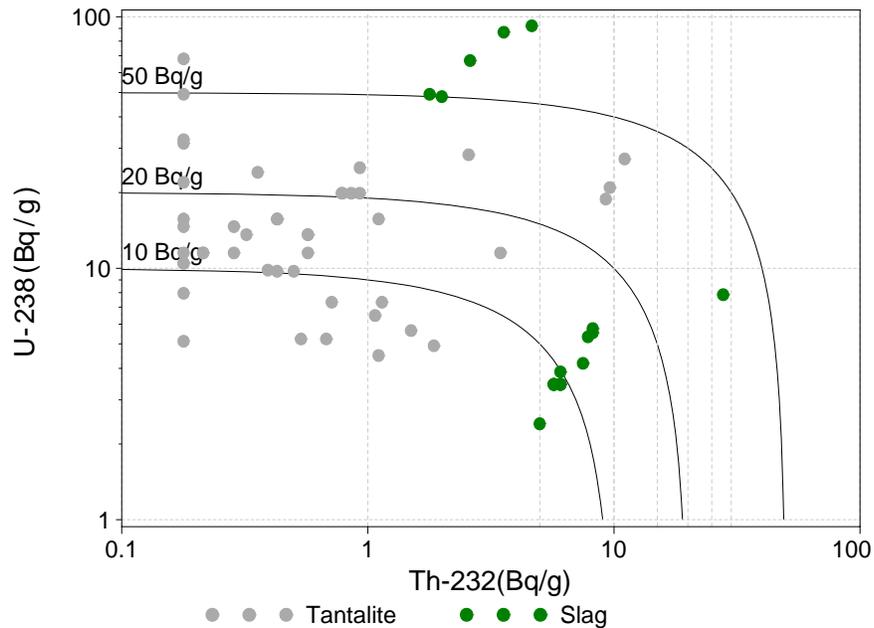
The concentrations were variable, ranging for example, from a minimum of 2.4 to a maximum of 92.2 Bq/g for U-238 in slag materials and from 0.2 to a maximum of 11.1 Bq/g for Th-232 in tantalite materials. The mean total activity concentrations were 17.7 and 25.3 Bq/g for tantalite and slag materials, respectively. The shipments of tantalite were more likely (i.e. 78% vs. 45%) to exceed 10 Bq/g of total activity than the shipments of slag.

**TABLE D.2
SUMMARY OF U-238 AND Th-232 ACTIVITY (Bq/g)**

Radionuclide	Material Type	Number of Shipments	Reported as “<” (%)	Median (Bq/g)	Mean (Bq/g)	Min. (Bq/g)	Max. (Bq/g)	Proportion > 10 Bq/g (%)
Th-232	All	67	16	0.9	3.0	0.2	27.8	3
Th-232	Slag	22	0	5.9	6.5	1.8	27.8	5
Th-232	Tantalite	45	24	0.5	1.3	0.2	11.1	2
U-238	All	67	0	11.5	17.2	2.4	92.2	55
U-238	Slag	22	0	3.7	18.8	2.4	92.2	23
U-238	Tantalite	45	0	13.6	16.4	4.5	68.1	71
Total	All	67		13.2	20.2	5.3	96.8	67
Total	Slag	22		9.7	25.3	7.4	96.8	45
Total	Tantalite	45		14.2	17.7	5.3	68.3	78

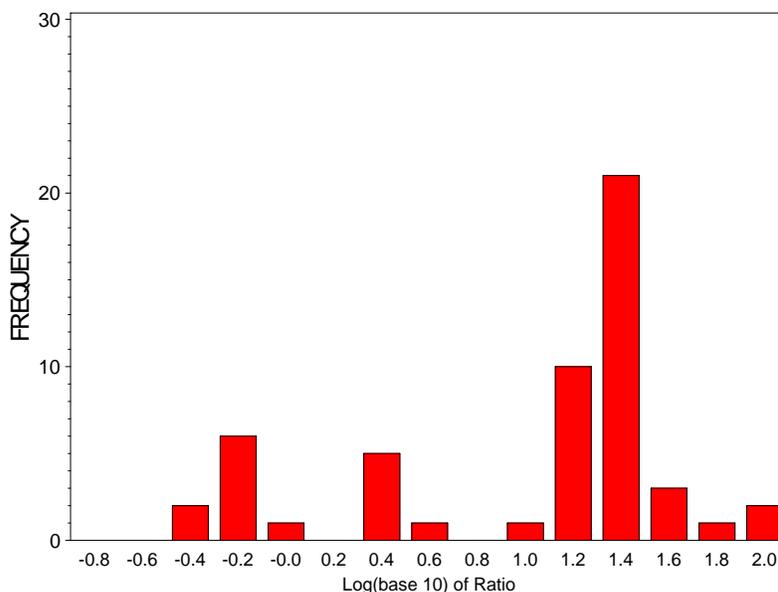
Figure D.3 shows a scatter plot of the concentrations of the U-238 and Th-232 along with lines showing total activities of 10 Bq/g, 20 Bq/g and 50 Bq/g. On an overall basis, there is little overall correlation between the concentrations of U-238 and Th-232 concentrations although there are some patterns in the scatter plot. The slag shipments appear to form two groups. Although Th-232 concentrations tend to be higher in slag shipments than in tantalite shipments, there is one group of slag shipments with high U-238 concentrations and another group of slag shipments with low U-238 concentrations.

FIGURE D.3
COMPARISON BETWEEN URANIUM (U-238)
AND THORIUM (Th-232) CONCENTRATIONS



The ratio between Th-232 concentration and U-238 concentrations varies between shipments. Since the measurement of Th-232 is uncertain at low levels, the more precisely measured concentrations of Th-228 are used as a surrogate for Th-232 to investigate the relationship over the range of Th-232 concentrations. Figure D.4 shows the distribution of the ratio between U-238 and Th-228 (the surrogate for Th-232). It is apparent that there are three modes (or peaks) with ratios between U-238 and Th-232 of about 0.8:1, 3:1 and 25:1 (e.g. on \log_{10} scale from Figure D.4, $10^{-0.2}$, $10^{0.4}$, $10^{1.4}$). These peaks are probably related to the type of minerals associated with the source of tantalum and niobium. For example, zircon minerals tend to have U-238 concentrations that are about 3 times the Th-232 activity and ilmenite tends to have U-238 concentrations that are about half the Th-232 activity (Koperski 1993). The U-238 and Th-232 concentrations in these shipments would be related to the concentration of these minerals in the particular shipment.

FIGURE D.4
RATIO BETWEEN URANIUM (U-238) AND THORIUM (Th-232) CONCENTRATIONS



D.1.4 Equilibrium Conditions

There is potential for disequilibria within the uranium and thorium decay series due to differences in mineralization (e.g. radon emanation) or process activities (e.g. a deficit of Pb isotopes in slag due to volatilization during processing). This means that the activity of the U-238 or Th-232 can be higher or lower than the activity of other radionuclides in the decay series. Equilibrium is an important consideration because the majority of the radiological dose associated with the uranium and thorium decay series comes from other radionuclides in the decay series.

For those measurements without “<” values reported by the independent laboratory, the ratio was calculated between the radionuclide concentration and the parent radionuclide concentration measured using NAA. Table D.3 shows summary statistics on this ratio. The mean ratio was calculated by dividing the average radionuclide concentration by the average parent radionuclide concentration. The activity concentrations in the U-235 decay series (U-235, Th-227 and Ra-223) are approximately 5% (0.05) indicating that the U-235 decay series is in natural equilibrium (e.g. 5% activity relative to U-238). In general, measured radionuclide concentrations in the uranium and thorium decay series are in equilibrium (within measurement precision) with the parent radionuclide. The exception is Pb-210 in slag materials where Pb-210 is on average at 12% equilibrium with U-238; however this varies from 2% to about 100% between shipments.

**TABLE D.3
SUMMARY STATISTICS ON RATIO BETWEEN DECAY SERIES RADIONUCLIDES
AND PARENT RADIONUCLIDE**

Radionuclide	Number	Median Ratio	Minimum Ratio	Maximum Ratio	Overall Mean Ratio
SLAG					
U-238 Series					
Th-234	14	1.02	0.78	1.24	1.1
Th-230	2	0.97	0.84	1.09	0.96
Ra-226	14	0.99	0.65	1.55	1.23
Pb-210	13	0.73	0.02	1.01	0.12
U-235 Series					
U-235	7	0.05	0.02	0.07	0.05
Th-227	10	0.06	0.04	0.11	0.06
Ra-223	8	0.06	0.05	0.09	0.06
Th-232 Series					
Th-228	14	0.96	0.8	1.26	0.9
Ra-228	14	1	0.63	1.45	0.88
TANTALITE					
U-238 Series					
Th-234	39	1.07	0.72	1.44	0.99
Th-230	3	0.88	0.73	0.99	0.88
Ra-226	39	1.23	0.67	1.94	1.11
Pb-210	39	0.96	0.53	1.42	0.88
U-235 Series					
U-235	32	0.05	0.03	0.09	0.05
Th-227	36	0.06	0.03	0.12	0.06
Ra-223	33	0.06	0.03	0.11	0.05
Th-232 Series					
Th-228	22	1.22	0.81	2.63	0.96
Ra-228	21	1.23	0.47	3.22	0.97

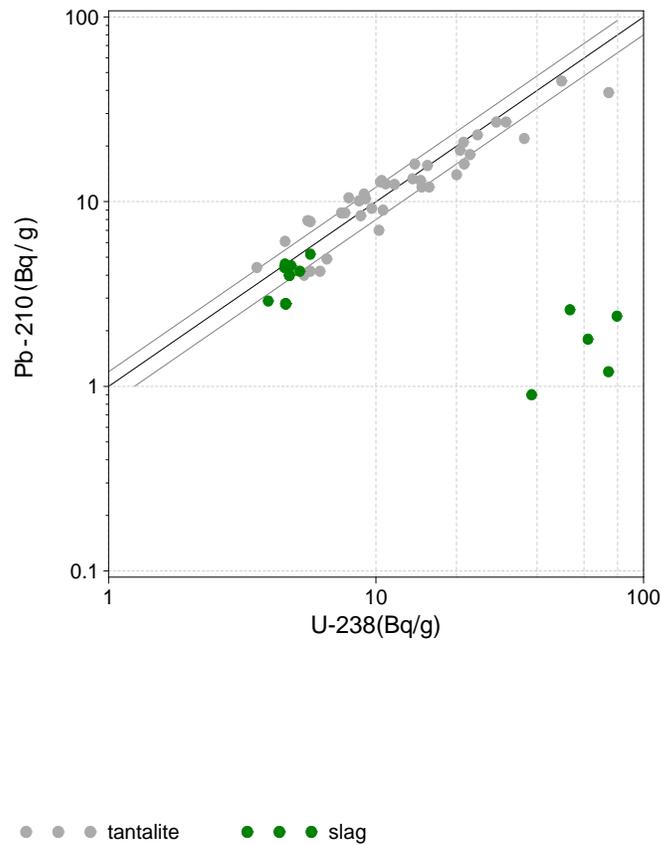
Note:

Table shows summary statistics of the ratio between radionuclides and the parent radionuclide in the decay series (i.e. U-238 or Th-232).

The ratios for U-235, Th-227 and Ra-223 relative to U-238 are about 5% as is expected for natural uranium. For most of the other radionuclides, there is close agreement between the radionuclide concentration and the parent radionuclide concentration indicating close to equilibrium conditions in the decay series. The ratio for Pb-210 tends to be lower than for other radionuclides. Figure D.5 shows the Pb-210 concentration plotted against the U-238 concentration. Most shipments have close agreement between Pb-210 and U-238 concentration

as would be expected if the radionuclides are in equilibrium; however, there were five shipments with a much lower Pb-210 concentration relative to U-238 concentration. These shipments were slag materials and the pattern may reflect high temperature processes that created these particular slags.

FIGURE D.5
COMPARISON BETWEEN Pb-210 AND U-238 CONCENTRATIONS (Bq/g)



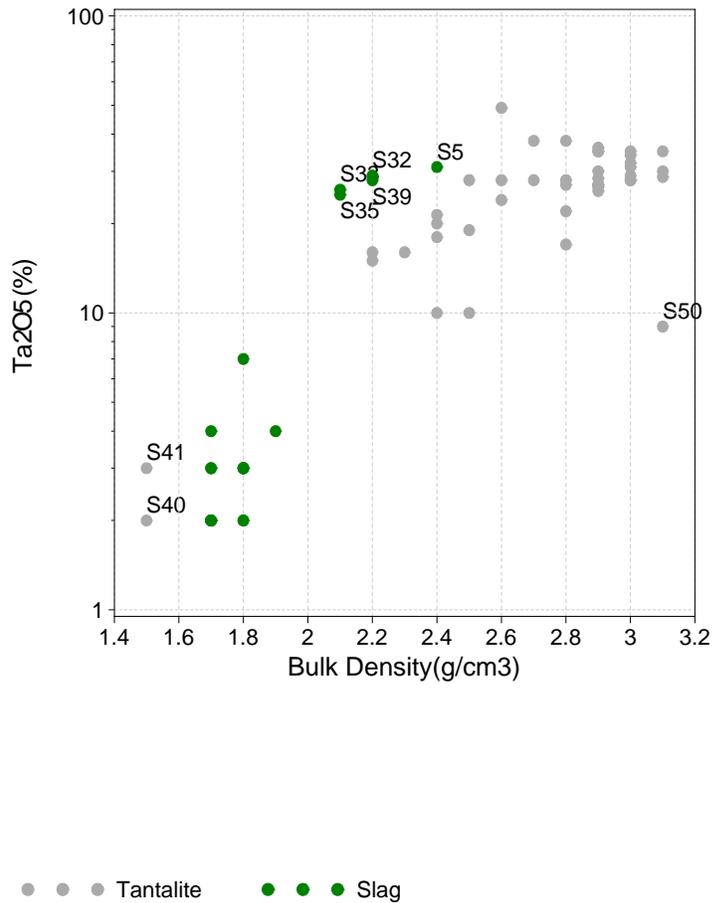
D.1.5 Tantalum Content and Density

There was variation in the tantalum content and density measured in the samples. Figure D.6 shows a scatter plot of Ta₂O₅ concentrations and bulk density. Typically, the slag materials had lower Ta₂O₅ content and lower bulk density than the tantalite materials.

There was a number of shipments that departed from this pattern. Shipments S40 and S41 were tantalite shipments with low bulk density and low tantalum content. Shipments S5, S32, S33, S35 and S39 were slag shipments with much higher Ta₂O₅ content than the other slags and a

somewhat higher density. Shipment S50 was a somewhat unusual tantalite shipment with a high bulk density but a relatively low Ta_2O_5 content, < 10%, for tantalite materials.

FIGURE D.6
TANTALUM CONCENTRATION (%) AND BULK DENSITY IN SHIPMENTS



D.2 MODELLED AND MEASURED GAMMA RADIATION EXPOSURES

D.2.1 Predicted Dose Rates

MicroShield modelling was used to estimate the gamma radiation dose rates at locations around the shipment. The overall approach was to use information on the loading configuration, the elemental composition of the material, the density and the shielding present to calculate dose factors (i.e. $\mu\text{Sv/h}$ (dose rate) per Bq/g (concentration) in the materials). These factors were multiplied by the concentration in the tantalum materials to estimate the gamma dose rate. A detailed discussion is provided in Annex C.

Table D.4 summarizes the modelled dose factors for three loading configurations and for two types of material. For the same location, loading configuration and material type, the factors for Th-232 are about 50% higher than the factors for U-238 because of the higher energy release. This is due to a combination of photon energy and abundance, from the thorium series compared to the uranium series. The factors decrease with distance from the container. Factors for the Maximum Load shipments are about twice as large as the factors for the full 1 tier shipments since there is more material in the Maximum load shipments. The factors for slag are slightly larger (about 10%) than the corresponding factors for tantalite due to differences in the mixtures of other materials, shielding and density.

TABLE D.4
SUMMARY OF MICROSIELD DOSE FACTORS ($\mu\text{Sv/h}$ per Bq/g)

Material	Analyte	Long Side (Right/Left)			Near Corner on Long Side			Short Side (Front/Back)		
		contact	1 m	3 m	contact	1 m	3 m	contact	1 m	3 m
F - 1 Tier Shipment										
Slag	Th-232	0.281	0.142	0.045	0.281	0.122	0.037	0.281	0.107	0.023
Slag	U-238	0.191	0.097	0.031	0.191	0.084	0.025	0.191	0.074	0.016
Tantalite										
Tantalite	Th-232	0.254	0.138	0.044	0.255	0.120	0.036	0.254	0.107	0.023
Tantalite	U-238	0.171	0.094	0.030	0.172	0.082	0.025	0.171	0.073	0.015
Maximum Load Shipment										
Slag	Th-232	0.470	0.236	0.079	0.442	0.196	0.064	0.559	0.174	0.035
Slag	U-238	0.321	0.162	0.054	0.302	0.135	0.044	0.381	0.120	0.024
Tantalite										
Tantalite	Th-232	0.436	0.231	0.078	0.421	0.196	0.064	0.507	0.175	0.034
Tantalite	U-238	0.295	0.157	0.053	0.285	0.134	0.044	0.342	0.120	0.024
F - Trailer Shipment										
Tantalite	Th-232	0.290	0.154	0.061	0.286	0.128	0.042	0.282	0.115	0.024
Tantalite	U-238	0.197	0.105	0.041	0.194	0.088	0.028	0.190	0.078	0.017

Predicted dose rates based on the MicroShield modelling were calculated by multiplying the dose factors ($\mu\text{Sv/h}$ per Bq/g) by the corresponding concentrations of uranium and thorium measured in the shipment. Uranium and thorium concentrations were available for 67 shipments and have been summarized in an earlier section of this report. The concentration data for each shipment were merged with the MicroShield factors for the corresponding loading configuration and type of material.

Information on the material type was complete for all shipments; however, the loading configuration was missing or incomplete for several of the shipments. If the loading configuration was not reported, it was assumed that the shipment was full 1 tier. Many of the loads were partially full 1 tier or partially full 2 tier; however, information on what proportion of the container was full and where the material was located was not available. In these cases, the factors for a Maximum Load were used recognizing that this would tend to overestimate exposures.

Exposure rates were calculated for uranium and thorium separately and these rates were totalled to get the predicted exposure rate from the material. These calculated exposure rates are presented in the main text.

D.2.2 Measured Gamma Radiation Dose Rates

Gamma radiation dose rates around the shipment were surveyed as described in Annex A and Annex B.

The measurement protocol typically provides multiple measurements for the same geometry assuming that the shipment was full. For example, the gamma radiation level at B0 (centre of the back on contact) is expected to be the same as the gamma radiation level at the F0 location (centre of front on contact) if the shipment were full. In a similar manner, there are two measurements on contact at the middle of the side of the shipment (R0, L0) and four measurements at the corners. For each shipment, the mean, maximum and variability of the measured gamma radiation was summarized for each geometry and distance.

For some containers, variation in gamma dose rates was observed at locations which would be expected to have the same dose rate if the shipment were uniformly loaded. It was assumed that this variation was more likely to arise due to variations in the loading configurations (e.g. partial loading with void spaces) than due to variation of radioactivity in the shipment.

Table D.5 summarizes the median variability by geometry. It can be seen that least variation occurs at the “side” of the loads (i.e. long side) and is highest at the “end” of the loads (i.e. short side). The relative variation decreases with distance as would be expected since the radiation observed by the detector comes from a wider field of view and therefore the results are less subject to variation in loading than measurements on contact. The highest amount of variation was at the ends of the container.

**TABLE D.5
MEDIAN VARIABILITY (%) OF GAMMA RADIATION
MEASUREMENTS FOR SHIPMENTS**

Geometry	Contact	1 m	3 m
Corner	20	19	15
End	31	26	21
Side	10	11	9

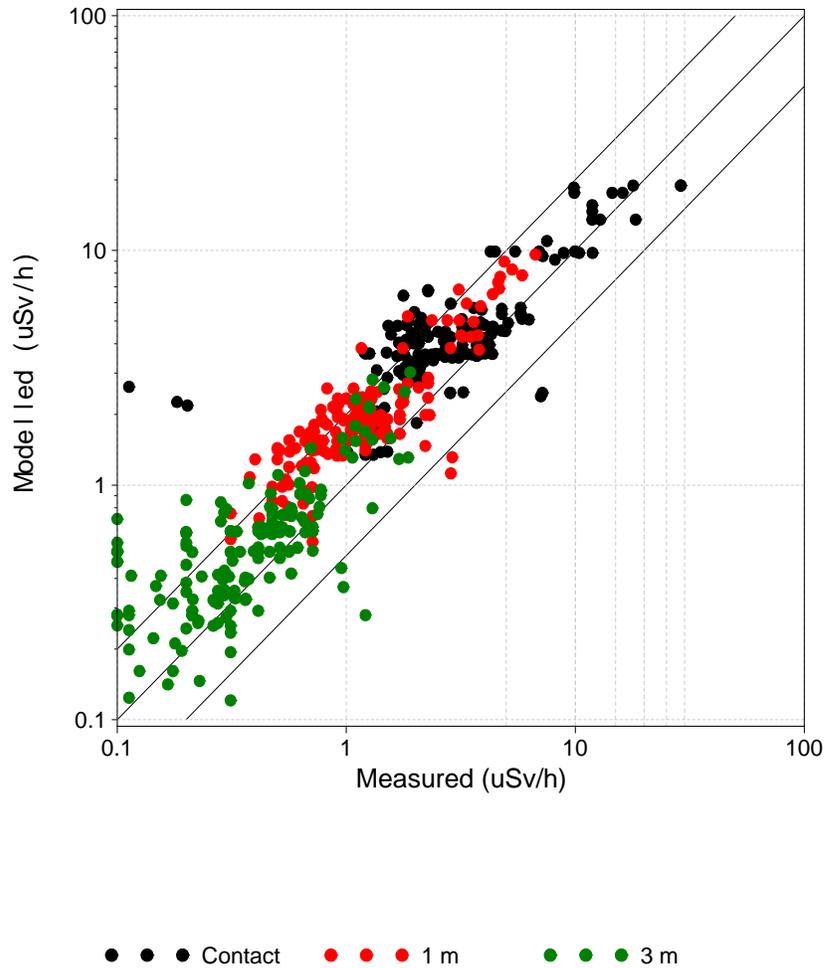
The measured gamma radiation levels were summarized for the nine combinations of distance and geometry. For each of the nine similar locations, the maximum and mean gamma dose rates were determined for each shipment. A summary of these dose rates is provided in Chapter 3 of this report.

D.2.3 Comparison between Modelled and Measured Dose Rates

An exploratory comparison between measured and modelled gamma exposure rate was conducted. Only full 1 tier or Maximum Load shipments were modelled; however, many of the shipments were partial and contained void areas and this would be expected to result in measured gamma radiation levels that would be lower than predicted if there were no voids. In order to account for this effect, both the mean and the maximum measured exposure rate for a given distance and geometry were used for comparison with the predicted exposure rates using MicroShield. The assumption is that the modelled configuration may be appropriate in some locations (areas without voids in material) and the highest gamma radiation levels may occur there.

Figure D.7 shows a scatter plot of predicted exposure rates using MicroShield modelling and the measured concentrations of uranium and thorium in the individual shipments plotted against the maximum exposure rates measured with the gamma radiation survey for the same shipment. The graph shows lines of perfect agreement plus bounds showing agreement within a factor of two. Overall, there is correlation between modelled and measured gamma radiation exposure rates; however, there is scatter about the line of perfect agreement. In general, the predicted exposure rates are higher than the measured exposure rates and, in many cases more than twice as high. This pattern may arise in part due to the partial loading of the numerous 2-tier loads but also partially filled 1-tier loads and trailers.

FIGURE D.7
COMPARISON BETWEEN MODELLLED AND MEASURED EXPOSURE RATES



D.2.4 Summary

Both the modelled and measured exposure rates indicate that there is variation in exposure rates, with the rates ranging by about a factor of 50 between shipments. Much of this variation is probably due to the range of radioactivity concentrations in the material as the total (U-238 + Th-232) concentrations range from about 5 Bq/g to about 100 Bq/g.

Exposure rates decrease rapidly with increasing distance from the container. The rates at a distance of 1 m are about ½ the exposure rates on contact and the rates at a 3 m distance are about 5 to 10 times lower than the exposure rates on contact. Other sources of variation include different loading configurations (e.g. 1 – tier versus 2-tier), differences in type of material (e.g.

slag vs. concentrate) and, for measurement data, potential for different response by different meters.

The modelled exposure rates tended to be higher than the measured exposure rates. An important factor in the difference was probably the incomplete loading of the shipments which was not modelled. From the comparison, the theoretical modelling approach provided conservative predictions of exposure rate, which were correlated with measured gamma radiation exposure rates. This provides confidence that the theoretical model can be used to predict (conservative estimates of) exposure rates at other locations considered in the dose assessment for transport of the tantalum raw materials.

D.3 REFERENCES

Koperski, J. 1993. *Radiation Protection in the Mining and Milling of Mineral Sands* Radiation Protection in Australia, Vol. 11, No. 2, pg. 46-52.

ANNEX E

DOSE SCENARIO FACTORS AND DOSE CALCULATIONS

ANNEX E DOSE SCENARIO FACTORS AND CALCULATIONS

E.1 ANNUAL DOSE FACTORS

A variety of transport dose scenarios was developed for transport workers (e.g. truck drivers) and for members of the public. These scenarios defined the locations of the individual relative to the shipment, the gamma radiation exposure rate at the location, the duration of time spent at this location for each shipment and how many tantalum raw material shipments that the person would be exposed to during a year.

For example, a truck driver was assumed to spend 10 hours driving each shipment of tantalum material at a location 3 m in front of the end of shipment. MicroShield modelling predicts dose factors of 0.035 $\mu\text{Sv/h}$ per Bq/g for Th-232 and 0.024 $\mu\text{Sv/h}$ per Bq/g. Based on 10 hours per shipment at this location and 36 shipments per year, the annual dose factors are about 0.013 mSv/y per Bq/g and 0.009 $\mu\text{Sv/y}$ per Bq/g for Th-232 and U-238, respectively.

The parameters and resulting calculations for each dose scenario are provided in Table E.1.

E.2 PROBABILISTIC SIMULATION OF ANNUAL DOSE

The annual dose for each scenario is based on multiple exposures (i.e. several shipments) during the year; therefore, it is unreasonable to expect that the transport worker, or member of the public, will be exposed to shipments containing the maximum, or minimum, measured concentration. The concentrations in the shipments and the exposures arising from the material in the shipments will tend to “average-out”. However, since there is variation in the concentrations between shipments, the average concentration a person is exposed to from the group of shipments encountered during a year can be higher than the average concentration in all tantalum raw material shipments. A reasonable upper-bound on the average concentrations for the exposure scenarios has been developed using probabilistic simulation (i.e. Monte Carlo methods) and the procedure has been outlined in Figure E.1. It was assumed that for each probabilistic trial, the concentrations in the shipments would be a random selection from the sample of measured concentrations. For example, if the dose scenario was based on 36 shipments, then a probabilistic trial would comprise a random selection of 36 sets of concentrations from the measured values. These concentrations were then averaged to get the “average” concentrations of Th-232 and U-238 for that trial. These concentrations were multiplied by the corresponding annual dose factor for the receptor scenario to get the annual dose for the trial. After many probabilistic trials, distributions of “average” concentrations and doses arising from those average concentrations are developed. These distributions can be summarized to get the mean value and the likely upper bound for concentrations and for dose.

**TABLE E.1
CALCULATION OF DOSE FACTORS FOR EACH EXPOSURE PATHWAY WITH MAXIMUM LOAD**

Parameter	Unit	ID	Value	Reference/Notes
<u>Truck Driver - Maximum Load of Tantalite:</u>				
U-238 Gamma Dose Rate at 3 m from short side of sea-land	mSv/h per Bq/g	udr1.5t3	2.35E-05	MicroShield Calculations
Th-232 Gamma Dose Rate at 3 m from short side of sea-land	mSv/h per Bq/g	thdr1.5t3	3.44E-05	MicroShield Calculations
Trips per month	trips/month	trimon	3	Discussions with T.I.C. Transport Committee (Dec. 2005)
Time loaded per trip	hours/trip	trihour	10	Discussions with T.I.C. Transport Committee (Dec. 2005)
Conversion – months/y	months/year	moncon	12	
U-238 Dose Factor for Truck Driver - Maximum Load of Tantalite	mSv/y per Bq/g	-	8.47E-03	=udr1.5t3*trimon*moncon*trihour
Th-232 Dose Factor for Truck Driver - Maximum Load of Tantalite	mSv/y per Bq/g	-	1.24E-02	=thdr1.5t3*trimon*moncon*trihour
<u>Truck Driver - Maximum Load of Slag:</u>				
U-238 Gamma Dose Rate at 3 m from short side of sea-land	mSv/h per Bq/g	udr1.5s3	2.38E-05	MicroShield Calculations
Th-232 Gamma Dose Rate at 3 m from short side of sea-land	mSv/h per Bq/g	thdr1.5s3	3.45E-05	MicroShield Calculations
Trips per month	trips/month	Same as Truck driver with tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
Time loaded per trip	hours/trip	Same as Truck driver with tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
U-238 Dose Factor for Truck Driver - Maximum Load of Slag	mSv/y per Bq/g	-	8.55E-03	=udr1.5s3*trimon*moncon*trihour
Th-232 Dose Factor for Truck Driver - Maximum Load of Slag	mSv/y per Bq/g	-	1.24E-02	=thdr1.5s3*trimon*moncon*trihour
<u>Dockworker - Maximum Load of Tantalite:</u>				
U-238 Gamma Dose Rate at 3 m from short side of sea-land	mSv/h per Bq/g	Same as truck driver with tantalite		MicroShield Calculations
U-238 Gamma Dose Rate at 1 m from long side of sea-land	mSv/h per Bq/g	urdl.5tl	1.57E-04	MicroShield Calculations

Radiological Risk Assessment of the Transport of Tantalum Raw Materials

TABLE E.1 (Cont'd)
CALCULATION OF DOSE FACTORS FOR EACH EXPOSURE PATHWAY WITH MAXIMUM LOAD

Parameter	Unit	ID	Value	Reference/Notes
Th-232 Gamma Dose Rate at 3 m from short side of sea-land	mSv/h per Bq/g	Same as truck driver with tantalite		MicroShield Calculations
Th-232 Gamma Dose Rate at 1 m from long side of sea-land	mSv/h per Bq/g	thdr1.5t1	2.31E-04	MicroShield Calculations
Time to Load each container at 3 m	minutes/container	mindock3	5	Discussions with T.I.C. Transport Committee (Dec. 2005)
Time within 1 m from sea-land for inspection	minutes/container	mindock1	5	Discussions with T.I.C. Transport Committee (Dec. 2005)
Conversion - h/min	h/min	conmin	1.67E-02	-
# of sea-land containers per month	# of containers/month	mocont	6	Discussions with T.I.C. Transport Committee (Dec. 2005)
U-238 Dose Factor for Dockworker - Maximum Load of Tantalite	mSv/y per Bq/g	-	1.08E-03	$=(udr1.5t3*mindock3*conmin+udr1.5t1*mindock1*conmin)*mocont*moncon$
Th-232 Dose Factor for Dockworker - Maximum Load of Tantalite	mSv/y per Bq/g	-	1.59E-03	$=(thdr1.5t3*mindock3*conmin+thdr1.5t1*mindock1*conmin)*mocont*moncon$
<u>Dockworker - Maximum Load of Slag:</u>				
U-238 Gamma Dose Rate at 3 m from short side of sea-land	mSv/h per Bq/g	Same as truck driver with slag		MicroShield Calculations
U-238 Gamma Dose Rate at 1 m from long side of sea-land	mSv/h per Bq/g	udr1.5s1	1.62E-04	MicroShield Calculations
Th-232 Gamma Dose Rate at 3 m from short side of sea-land	mSv/h per Bq/g	Same as truck driver with slag		MicroShield Calculations
Th-232 Gamma Dose Rate at 1 m from long side of sea-land	mSv/h per Bq/g	thdr1.5s1	2.36E-04	MicroShield Calculations
Time to Load each container at 3 m	minutes/container	Same as dockworker for tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
Time within 1 m from sea-land for inspection	minutes/container	Same as dockworker for tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
# of sea-land containers per month	# of containers/month	Same as dockworker for tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
U-238 Dose Factor for Dockworker - Maximum Load of Slag	mSv/y per Bq/g	-	1.12E-03	$=(udr1.5s3*mindock3*conmin+udr1.5s1*mindock1*conmin)*mocont*moncon$
Th-232 Dose Factor for Dockworker - Maximum Load of Slag	mSv/y per Bq/g	-	1.62E-03	$=(thdr1.5s3*mindock3*conmin+thdr1.5s1*mindock1*conmin)*mocont*moncon$

TABLE E.1 (Cont'd)
CALCULATION OF DOSE FACTORS FOR EACH EXPOSURE PATHWAY WITH MAXIMUM LOAD

Parameter	Unit	ID	Value	Reference/Notes
<u>Seaman - Maximum Load of Tantalite:</u>				
U-238 Gamma Dose Rate at 3 m from sea-land	mSv/h per Bq/g	Same as truck driver with tantalite		MicroShield Calculations
Th-232 Gamma Dose Rate at 3 m from sea-land	mSv/h per Bq/g	Same as truck driver with tantalite		MicroShield Calculations
Time within 3 m of sea-land container	minutes/container	minsea	5	Discussions with T.I.C. Transport Committee (Dec. 2005)
# of sea-land containers per month	# of containers/month	Same as dockworker for tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
U-238 Dose Factor for Seaman - Maximum Load of Tantalite	mSv/y per Bq/g	-	1.41E-04	= $udr1.5t3 * minsea * conmin * mocont * moncon$
Th-232 Dose Factor for Seaman - Maximum Load of Tantalite	mSv/y per Bq/g	-	2.06E-04	= $thdr1.5t3 * minsea * conmin * mocont * moncon$
<u>Seaman - Maximum Load of Slag</u>				
U-238 Gamma Dose Rate at 3 m from short side of sea-land	mSv/h per Bq/g	Same as truck driver with slag		MicroShield Calculations
Th-232 Gamma Dose Rate at 3 m from short side of sea-land	mSv/h per Bq/g	Same as truck driver with slag		MicroShield Calculations
Time within 3 m of sea-land container	minutes/container	Same as seaman with tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
# of sea-land containers per month	# of containers/month	Same as dockworker for tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
U-238 Dose Factor for Seaman - Maximum Load of Slag	mSv/y per Bq/g	-	1.43E-04	= $udr1.5s3 * minsea * conmin * mocont * moncon$
Th-232 Dose Factor for Seaman - Maximum Load of Slag	mSv/y per Bq/g	-	2.07E-04	= $thdr1.5s3 * minsea * conmin * mocont * moncon$
<u>Trainman - Maximum Load of Tantalite:</u>				
U-238 Gamma Dose Rate at 3 m from short side of sea-land	mSv/h per Bq/g	Same as truck driver with tantalite		MicroShield Calculations
U-238 Gamma Dose Rate at 1 m from long side of sea-land	mSv/h per Bq/g	Same as Dockworker with tantalite		MicroShield Calculations

Radiological Risk Assessment of the Transport of Tantalum Raw Materials

TABLE E.1 (Cont'd)
CALCULATION OF DOSE FACTORS FOR EACH EXPOSURE PATHWAY WITH MAXIMUM LOAD

Parameter	Unit	ID	Value	Reference/Notes
Th-232 Gamma Dose Rate at 3 m from short side of sea-land	mSv/h per Bq/g	Same as truck driver with tantalite		MicroShield Calculations
Th-232 Gamma Dose Rate at 1 m from long side of sea-land	mSv/h per Bq/g	Same as Dockworker with tantalite		MicroShield Calculations
Time to Load & Unload each container at 3 m	min/container	trainload3	10	Discussions with T.I.C. Transport Committee (Dec. 2005)
Time for inspection within 1 m from sea-land	min/container	trainload1	2	Discussions with T.I.C. Transport Committee (Dec. 2005)
# of sea-land containers per month	# of containers/month	Same as dockworker for tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
U-238 Dose Factor for Trainman - Maximum Load of Tantalite	mSv/y per Bq/g	-	6.60E-04	$=(\text{udr}1.5\text{t}3*\text{trainload}3*\text{conmin}+\text{udr}1.5\text{t}1*\text{trainload}1*\text{conmin})*\text{mocont}*\text{moncon}$
Th-232 Dose Factor for Trainman - Maximum Load of Tantalite	mSv/y per Bq/g	-	9.66E-04	$=(\text{thdr}1.5\text{t}3*\text{trainload}3*\text{conmin}+\text{thdr}1.5\text{t}1*\text{trainload}1*\text{conmin})*\text{mocont}*\text{moncon}$
<u>Trainman - Maximum Load of Slag:</u>				
U-238 Gamma Dose Rate at 3 m from short side of sea-land	mSv/h per Bq/g	Same as truck driver with slag		MicroShield Calculations
U-238 Gamma Dose Rate at 1 m from long side of sea-land	mSv/h per Bq/g	Same as dockworker with slag		MicroShield Calculations
Th-232 Gamma Dose Rate at 3 m from short side of sea-land	mSv/h per Bq/g	Same as truck driver with slag		MicroShield Calculations
Th-232 Gamma Dose Rate at 1 m from long side of sea-land	mSv/h per Bq/g	Same as dockworker with slag		MicroShield Calculations
Time to Load & Unload each container at 3 m	min/container	Same as trainman with tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
Time for inspection within 1 m from sea-land	min/container	Same as trainman with tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
# of sea-land containers per month	# of containers/month	Same as dockworker with slag		Discussions with T.I.C. Transport Committee (Dec. 2005)
U-238 Dose Factor for Trainman - Maximum Load of Slag	mSv/y per Bq/g	-	6.74E-04	$=(\text{udr}1.5\text{s}3*\text{trainload}3*\text{conmin}+\text{udr}1.5\text{s}1*\text{trainload}1*\text{conmin})*\text{mocont}*\text{moncon}$
Th-232 Dose Factor for Trainman - Maximum Load of Slag	mSv/y per Bq/g	-	9.81E-04	$=(\text{thdr}1.5\text{s}3*\text{trainload}3*\text{conmin}+\text{thdr}1.5\text{s}1*\text{trainload}1*\text{conmin})*\text{mocont}*\text{moncon}$

Radiological Risk Assessment of the Transport of Tantalum Raw Materials

TABLE E.1 (Cont'd)
CALCULATION OF DOSE FACTORS FOR EACH EXPOSURE PATHWAY WITH MAXIMUM LOAD

Parameter	Unit	ID	Value	Reference/Notes
Public Living Adjacent to Road - Maximum Load of Tantalite:				
U-238 Gamma Dose Rate at 10 m from long side of sea-land	mSv/h per Bq/g	udr1.5t10	7.29E-06	MicroShield Calculations
Th-232 Gamma Dose Rate at 10 m from long side of sea-land	mSv/h per Bq/g	thdr1.5t10	1.07E-05	MicroShield Calculations
Time Stopped at Traffic light	minute/truck	minlight	3	Discussions with T.I.C. Transport Committee (Dec. 2005)
Conversion - h/min	h/min	chmin	1.67E-02	-
# of trucks per month	# of trucks/month	trucstop	6	Discussions with T.I.C. Transport Committee (Dec. 2005)
Fraction of Trucks Stopped at Traffic Light	-	fractrst	0.5	Discussions with T.I.C. Transport Committee (Dec. 2005)
Conversion - months/y	months/y	cmyear	12	-
U-238 Dose Factor for Public (Road) - Maximum Load of Tantalite	mSv/y per Bq/g	-	1.31E-05	=udr1.5t10*minlight*chmin*trucstop*fractrst*cmyear
Th-232 Dose Factor for Public (Road) - Maximum Load of Tantalite	mSv/y per Bq/g	-	1.93E-05	=thdr1.5t10*minlight*chmin*trucstop*fractrst*cmyear
Public Living Adjacent to Road - Maximum Load of Slag:				
U-238 Gamma Dose Rate at 10 m from long side of sea-land	mSv/h per Bq/g	udr1.5s10	7.47E-06	MicroShield Calculations
Th-232 Gamma Dose Rate at 10 m from long side of sea-land	mSv/h per Bq/g	thdr1.5s10	1.09E-05	MicroShield Calculations
Time Stopped at Traffic light	minute/truck	Same as Public (Road) with tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
Conversion - h/min	h/min	Same as Public (Road) with tantalite		-
# of trucks per month	# of trucks/month	Same as Public (Road) with tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
Fraction of Trucks Stopped at Traffic Light	-	Same as Public (Road) with tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
Conversion - months/y	months/y	Same as Public (Road) with tantalite		-
U-238 Dose Factor for Public (Road) - Maximum Load of Slag	mSv/y per Bq/g	-	1.35E-05	=udr1.5s10*minlight*chmin*trucstop*fractrst*cmyear

Radiological Risk Assessment of the Transport of Tantalum Raw Materials

TABLE E.1 (Cont'd)
CALCULATION OF DOSE FACTORS FOR EACH EXPOSURE PATHWAY WITH MAXIMUM LOAD

Parameter	Unit	ID	Value	Reference/Notes
Th-232 Dose Factor for Public (Road) - Maximum Load of Slag	mSv/y per Bq/g	-	1.97E-05	=thdr1.5s10*minlight*chmin*trucstop*fractrst*cmyear
<u>Public Living Adjacent to Rail - Maximum Load of Tantalite:</u>				
U-238 Gamma Dose Rate at 20 m from sea-land	mSv/h per Bq/g	udr1.5t20	1.89E-06	MicroShield Calculations
Th-232 Gamma Dose Rate at 20 m from sea-land	mSv/h per Bq/g	thdr1.5t20	2.79E-06	MicroShield Calculations
Time Train Stopped on Rail	minutes/train	minrail	5	Discussions with T.I.C. Transport Committee (Dec. 2005)
Conversion - h/min	h/min	Same as Public (Road) with tantalite		-
# of trains per month	# of trains/month	traistop	6	Discussions with T.I.C. Transport Committee (Dec. 2005)
Fraction of trains stopped on Rail	-	fractrast	0.5	Discussions with T.I.C. Transport Committee (Dec. 2005)
Conversion - months/y	months/y	Same as Public (Road) with tantalite		-
U-238 Dose Factor for Public (Rail) - Maximum Load of Tantalite	mSv/y per Bq/g	-	5.67E-06	=udr1.5t20*minrail*chmin*traistop*fractrast*cmyear
Th-232 Dose Factor for Public (Rail) - Maximum Load of Tantalite	mSv/y per Bq/g	-	8.36E-06	=thdr1.5t20*minrail*chmin*traistop*fractrast*cmyear
<u>Public Living Adjacent to Rail - Maximum Load of Slag:</u>				
U-238 Gamma Dose Rate at 20 m from long side of sea-land	mSv/h per Bq/g	udr1.5s20	1.95E-06	MicroShield Calculations
Th-232 Gamma Dose Rate at 20 m from long side of sea-land	mSv/h per Bq/g	thdr1.5s20	2.85E-06	MicroShield Calculations
Time Train Stopped on Rail	minutes/train	Same as Public (Rail) with tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
Conversion - h/min	h/min	Same as Public (Road) with tantalite		-
# of trains per month	# of trains/month	Same as Public (Rail) with tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
Fraction of trains stopped on Rail	-	Same as Public (Rail) with tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
Conversion - months/y	months/y	Same as Public (Road) with tantalite		-

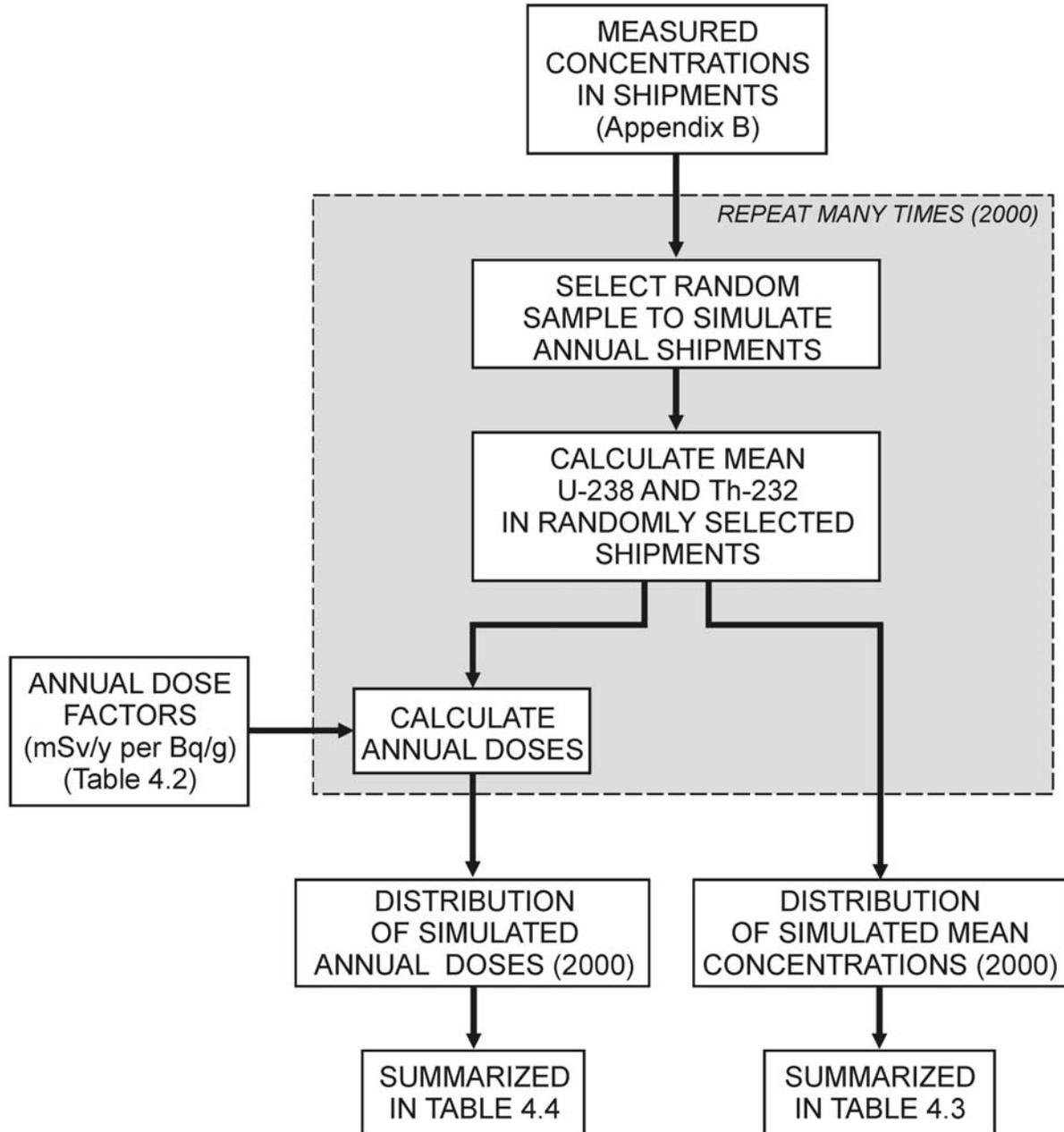
TABLE E.1 (Cont'd)
CALCULATION OF DOSE FACTORS FOR EACH EXPOSURE PATHWAY WITH MAXIMUM LOAD

Parameter	Unit	ID	Value	Reference/Notes
U-238 Dose Factor for Public (Rail) - Maximum Load of Slag	mSv/y per Bq/g	-	5.84E-06	= $udr1.5s20 * minrail * chmin * traistop * fractrast * cmyear$
Th-232 Dose Factor for Public (Rail) - Maximum Load of Slag	mSv/y per Bq/g	-	8.56E-06	= $thdr1.5s20 * minrail * chmin * traistop * fractrast * cmyear$
Facility Worker (Shipping & Receiving) - Tantalite				
U-238 Gamma Dose Rate at 1 m from long side of sea-land	mSv/h per Bq/g	udr1.5t1	1.57E-04	MicroShield Calculations
U-238 Gamma Dose Rate in sea-land at 0.5 m from Tantalite (long side)	mSv/h per Bq/g	udr1.5t0.5	3.10E-04	MicroShield Calculations
Th-232 Gamma Dose Rate at 1 m from long side sea-land	mSv/h per Bq/g	thdr1.5t1	2.31E-04	MicroShield Calculations
Th-232 Gamma Dose Rate in sea-land at 0.5 m from Tantalite (long side)	mSv/h per Bq/g	thdr1.5t0.5	4.53E-04	MicroShield Calculations
# of sea-land containers per month	# of containers/month	mocont	6	Discussions with T.I.C. Transport Committee (Dec. 2005)
Conversion - months/y	months/y	moncon	12	
Time spent per sea-land container	h/container	hcont	1	Discussions with T.I.C. Transport Committee (Dec. 2005)
Fraction of time spent at 1 m from sea-land container	-	frac1	0.5	Discussions with T.I.C. Transport Committee (Dec. 2005)
Fraction of time spent inside sea-land at 0.5 m from Tantalite	-	frac1	0.5	Discussions with T.I.C. Transport Committee (Dec. 2005)
U-238 Dose Factor for Facility Worker - Maximum Load of Tantalite	mSv/y per Bq/g		0.016816	=$(udr1.5t1 * frac1 + udr1.5t0.5 * frac1) * mocont * moncon * hcont$
Th-232 Dose Factor for Facility Worker - Maximum Load of Tantalite	mSv/y per Bq/g		0.024602	=$(thdr1.5t1 * frac1 + thdr1.5t0.5 * frac1) * mocont * moncon * hcont$

TABLE E.1 (Cont'd)
CALCULATION OF DOSE FACTORS FOR EACH EXPOSURE PATHWAY WITH MAXIMUM LOAD

Parameter	Unit	ID	Value	Reference
Facility Worker (Shipping & Receiving) - Slag				
U-238 Gamma Dose Rate at 1 m from long side of sea-land	mSv/h per Bq/g	udr1.5s1	1.62E-04	MicroShield Calculations
U-238 Gamma Dose Rate in sea-land at 0.5 m from Slag (long side)	mSv/h per Bq/g	udr1.5s0.5	3.17E-04	MicroShield Calculations
Th-232 Gamma Dose Rate at 1 m from long side of sea-land	mSv/h per Bq/g	thdr1.5s1	2.36E-04	MicroShield Calculations
Th-232 Gamma Dose Rate in sea-land at 0.5 m from Slag (long side)	mSv/h per Bq/g	thdr1.5s0.5	4.60E-04	MicroShield Calculations
# of sea-land containers per month	# of containers/month	Same as L&U for tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
Conversion - months/y	months/y	Same as L&U for tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
Time spent per sea-land container	h/container	Same as L&U for tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
Fraction of time spent at 1 m from sea-land container	-	Same as L&U for tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
Fraction of time spent inside sea-land at 0.5 m from Slag	-	Same as L&U for tantalite		Discussions with T.I.C. Transport Committee (Dec. 2005)
U-238 Dose Factor for Facility Worker - Maximum Load of Slag	mSv/y per Bq/g		0.017244	=(udr1.5s1*frac1+udr1.5s0.5*frac1)*mocont*moncon*hcont
Th-232 Dose Factor for Facility Worker - Maximum Load of Slag	mSv/y per Bq/g		0.02506	=(thdr1.5s1*frac1+thdr1.5s0.5*frac1)*mocont*moncon*hcont

FIGURE E.1
PROCEDURE FOR THE PROBABILISTIC SIMULATION OF ANNUAL DOSE AND CONCENTRATION



Two thousand (2000) probabilistic trials were completed for each scenario and the doses from each trial were analyzed. The upper bound chosen for this study was the 95th percentile: there is only one chance in 20 (i.e. 5%) that the actual dose would be higher. The results of the probabilistic simulation are summarized in Chapter 4.

A second probabilistic assessment was conducted to estimate the dose to truck drivers that would be received below potential alternate exemption values. For example, if an alternate exemption value of 30 Bq/g (Th-232 + U-238) were being considered, the dose from shipments with less than 30 Bq/g in each probabilistic trial was determined. These results are summarized elsewhere.