LEUPA

Type B(U) Package to Contain Fissile Substances

DOSE RATE CALCULATIONS

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CONTENTS

1 PURPOSE

1. Calculate the dose rates expected for a LEUPA package, both under normal and accidental conditions of transport. From these, calculate the corresponding Transport Index applicable to the LEUPA package.
2. These dose rates are used to demonstrate LEUPA package´s compliance, from the radiological safety point of view, with Argentine Standard AR 10.16.1. Rev. 2 “Transport of Radioactive Materials” [1].

2 SCOPE

1. This document describes the dose rate calculations due to a LEUPA package loaded with a nominal mass of fissile material. It applies to verify compliance of the LEUPA package to the relevant regulations, exclusively from the radiological safety point of view.

3 REFERENCES

[1] ARN. Transport of Radioactive Materials. Standard AR 10.16.1. Rev. 2. Argentine Republic: ARN, 2011.

[2] IAEA. Regulations for the safe transport of radioactive material: safety requirements. 2009 ed. Vienna: International Atomic Energy Agency, 2009.

[3] 0908-LE02-3BEIN-008 "Tests Final Report".

[4] MicroShield 9.06. Grove Software, Inc.

[5] MCNP – A General N-Particle Transport Code, Version 5 – Volume I: Overview and Theory, X-5 Monte Carlo Team, LA-UR-03-1987, Los Alamos National Laboratory (April, 2003, revised February 2008).

[6] 0908-LE01-3BEIN-024 "LEUPA – Criticality Analysis".

[7] ASTM C 1462 – 00 Standard Specification for Uranium Metal Enriched to More than 15% and Less than 20% 235U.

4 ABBREVIATIONS

Abbreviation	Description
ARN	Autoridad Regulatoria Nuclear (Argentinean Nuclear Regulatory Authority)
IAEA	International Atomic Energy Agency
LEUPA	Low Enriched Uranium Package for Transport
TI	Transport Index

5 INTRODUCTION

1. Since the objectives, scope, definitions and requirements in [1] have been taken from the IAEA Standard TS-R-1 “Regulation for the Safe Transport of Radioactive Materials", 2009 Edition [2], the English text given in [2] is used in this document whenever is needed. Paragraph numbering in both standards is exactly the same.
2. The limits imposed on Transport Index and radiation levels are as follows:
   a) The Transport Index shall not exceed 10 (Paragraph 524).
   b) The maximum radiation level at any point on the external surface of the package shall not exceed 2 mSv/h (Paragraph 525).
3. Furthermore, the change in dose rate due to impacts of the normal conditions of transport tests shall be limited. According to Paragraph 646(b), a package shall be so designed that if it were subjected to the tests specified in paras 719–724, it would prevent:
   a) More than a 20% increase in the maximum radiation level at any external surface of the package.
4. Finally, according to Paragraph 657, a package shall be so designed that if it were subjected to the impacts of the accidental conditions of transport tests (paras 726–729), the package would meet the following requirement:
   a) Retain sufficient shielding to ensure that the radiation level at 1 m from the surface of the package would not exceed 10 mSv/h with the maximum radioactive contents which the package is designed to contain (Paragraph 657(b)).
5. Test results are documented in [3]. Both a Type B puncture test (Paragraph 727(b)) and a Type C puncture test (Paragraph 735) were completed and documented. The standard requires that compliance with Paragraph 657 shall be verified considering the cumulative effects of the puncture and thermal tests, in that order. However, no mention of the Type B package thermal test is found in [3]. Instead, a Type C thermal test was done after the Type C puncture test was performed. For this reason, compliance with Paragraph 657 in this document will be verified with the package configuration resulting after the Type C puncture and thermal tests, as reported in [3].

6 CALCULATION MODEL

1. Gamma dose rate calculations, both in contact with the external surface of the package and at 1 m distance where performed using MicroShield [4] and MCNP [5].
2. Since the LEUPA package is designed to transport non-irradiated low enriched uranium, there is no need for neutron dose rate calculations.

6.1 Geometry

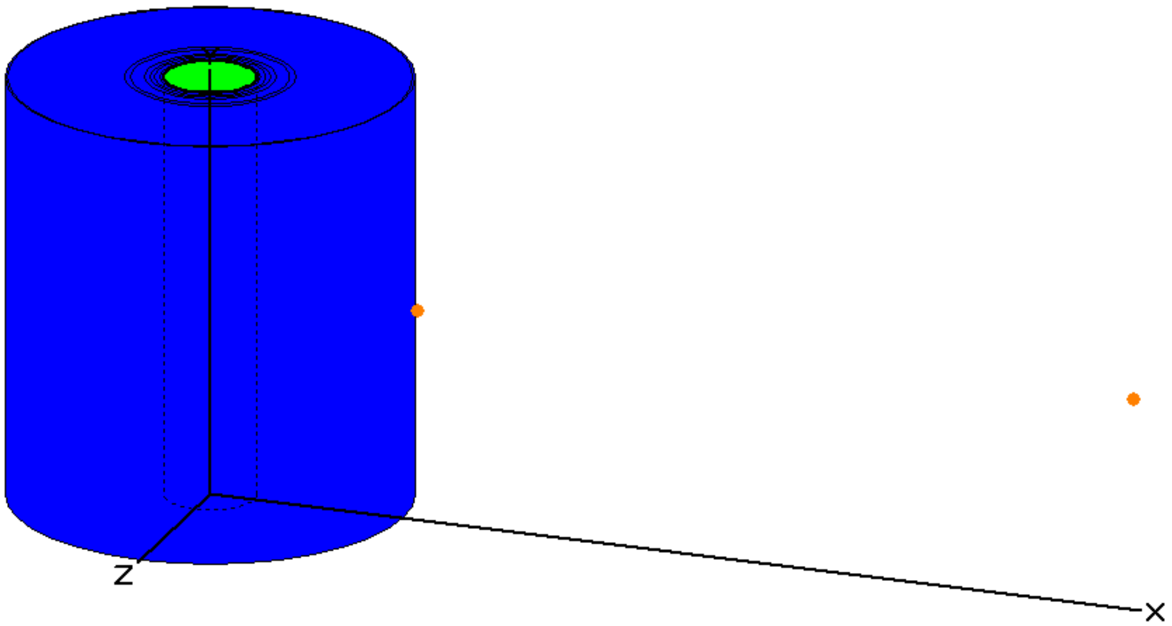
1. The basic dimensions of the LEUPA package can be seen in Figure 1.

Figure 1: LEUPA package, sizes in mm

Security-Related Information Figure Withheld Under 10 CFR 2.390.

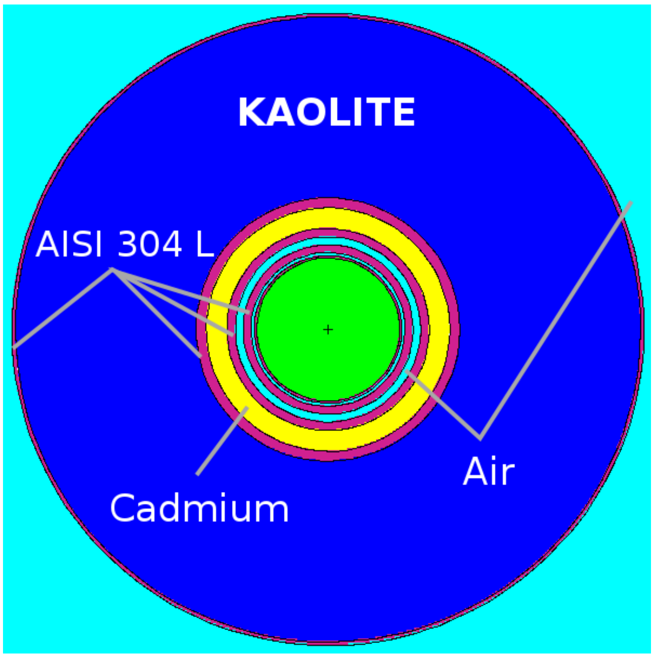
2. The MicroShield model consists of a single cylinder 57 cm high and 6 cm radius surrounded by 9 concentric layers. Dimensions and materials reproduce the ones used in the MCNP models.

Figure 2: LEUPA package, MicroShield model, side view



3. The geometry modeled in MCNP is the same as described in [6] (Figure 4).

Figure 3: LEUPA horizontal section, as modeled in MCNP



6.2 Materials

1. Table 1 describes the materials used in all the calculations to evaluate the external dose rate produced by the nominal load in the LEUPA package.

Table 1: Materials used in external dose rate evaluations

Material	Aprox. Mass [kg]	Density [g/cm^3]	Element	% Weight
Metal Uranium	50.0	18.9	U^{235} U^{238}	19.8 80.1
Kaolite 1600	85	0.405 (without water)	Al_2O_3 SiO_2 Fe_2O_3 TiO_2 CaO MgO Na_2O	11.0 33.0 7.9 1.4 30.0 12.1 4.6
AISI 304 L	244	7.9	Fe Cr Ni Mo Mn Si C	65.47 17.0 12.0 2.5 2.0 1.0 0.03
Cadmium	58	8.65	Cd	100.0

2. The Uranium density in the models is set up as if the material, given a fixed mass, occupies all the interior volume of the containers.
3. In the MicroShield model, the 4 containers are represented as a unique cylinder as described in the previous section. This leads to an interior volume of 6.45×10^3 cm^3 and a density of 7.76 g/cm^3.
4. The shielding is composed by a series of exterior layers. The total thickness of each material in the model is as follows:
   - 2.69 cm of AISI 304 L
   - 1.72 cm of Cadmium
   - 15.05 cm of Kaolite
5. The MCNP model is the same as described in [6]. In this case, a fixed gamma source was modeled, uniformly distributed in the container´s volume. The sum of the interior volumes of the 4 containers is 6.23×10^3 cm^3, and the effective density of the material filling completely the containers (metal uranium) is 8.09 g/ cm^3.

6.3 Photon source

1. The dose rate produced by non-irradiated uranium depends on the isotopic composition of the uranium. The uranium isotopes with a shorter half-life are the ones with the highest impact on the total photon source. In this work, as a conservative assumption, the fraction

of uranium isotopes with shorter half-lives was taken as the maximum allowed by the ASTM standard [7]:

- ^{232}U 0.002 µg/gU
- ^{234}U 0.01 g/gU
- ^{236}U 0.04 g/gU

2. According to these values, the activity for the 50 kg of uranium contained in the LEUPA package is as indicated in Table 2.

Table 2: Isotopic composition of uranium metal and resulting activities (50 kg U, no decay)

U Isotope	Concentration [µg/gU]	Mass [g] in 1 g of U	Half-life [years]	Lambda [1/s]	Activity [Bq]	Total Activity [Bq]
232	2.00E-03	2.00E-09	6.89E+01	3.19E-10	1.65E+03	8.27E+07
234	1.00E+04	1.00E-02	2.46E+05	8.95E-14	2.30E+06	1.15E+11
235		2.00E-01	7.04E+08	3.12E-17	1.60E+04	8.00E+08
236	4.00E+04	4.00E-02	2.34E+07	9.38E-16	9.57E+04	4.79E+09
238		7.50E-01	4.47E+09	4.92E-18	9.33E+03	4.66E+08

3. As a second conservative assumption, the decay of the material whose isotopic composition was given in Table 2 was calculated, and the corresponding photon spectra for different decay times were evaluated. The most intense gamma source was obtained after 10 years decay, and no significant changes were seen if this mixture is further decayed.
4. The resulting photon energy spectrum of the 50 kg uranium mass is used as the input photon source both in MicroShield and MCNP. Table 3 shows this photon energy spectrum.

Table 3: Energy spectrum

Energy [MeV]	Activity [Photons/s]
0.015	4.01E+11
0.02	1.47E+09
0.03	1.13E+08
0.04	1.99E+06
0.05	1.46E+08
0.06	2.69E+07
0.08	1.05E+08
0.1	2.16E+08
0.15	1.34E+08
0.2	5.54E+08
0.3	9.09E+06
0.4	6.27E+05
0.5	6.72E+06
0.6	2.30E+07
0.8	1.27E+07
1	5.31E+06
1.5	1.81E+06

2	3.78E+05
3	2.66E+07
4	6.22E+01
5	1.65E+01
6	3.13E+00
8	3.51E-01
10	3.03E-02
Totals	**4.04E+11**

7 RESULTS AND ANALYSIS

7.1 Normal conditions of transport

1. Paragraph 521(a) of [1] indicates that the Transport Index (TI) must be calculated multiplying by 100 the maximum dose rate (in [mSv/h]) obtained at 1 m distance from the package.
2. As can be seen in Table 4, considering the MicroShield results, the dose rate at 1 m is 8.9E-4 mSv/h, then the Transport Index is TI = 0.09.
3. The contact dose rate for the nominal case is 1.2E-2 mSv/h.
4. These two results demonstrate compliance with paragraphs 524 and 525 of [1].

7.2 Accidental conditions of transport

1. The test results in [3] (section 6.3.2.1 paragraph 2) shows that the Type B puncture test (paragraph 727(b) of [1]) results in a 30 mm depth deformation in the LEUPA package. Taking this case as representative for all Type B tests a 3 cm reduction was introduced in the whole Kaolite layer in the MicroShield model, resulting in a 12 cm thick final layer instead of 15 cm. Table 4 reflects the results of the calculation under Type B puncture tests row.
2. Increment in dose rate due to Type B puncture test is 4% as a maximum, which is within the 20% limit imposed.
3. In order to simulate the damage produced by the Type C puncture test reported in [3] (section 6.3.4.1 paragraph 2) a conical hole of 140 mm in diameter and 180 mm in depth was modeled with MCNP (see Figure 4 and Figure 5). Table 4 reports the results of the calculation in the Type C test row.

Figure 4: LEUPA package, Type C puncture test hole, side view

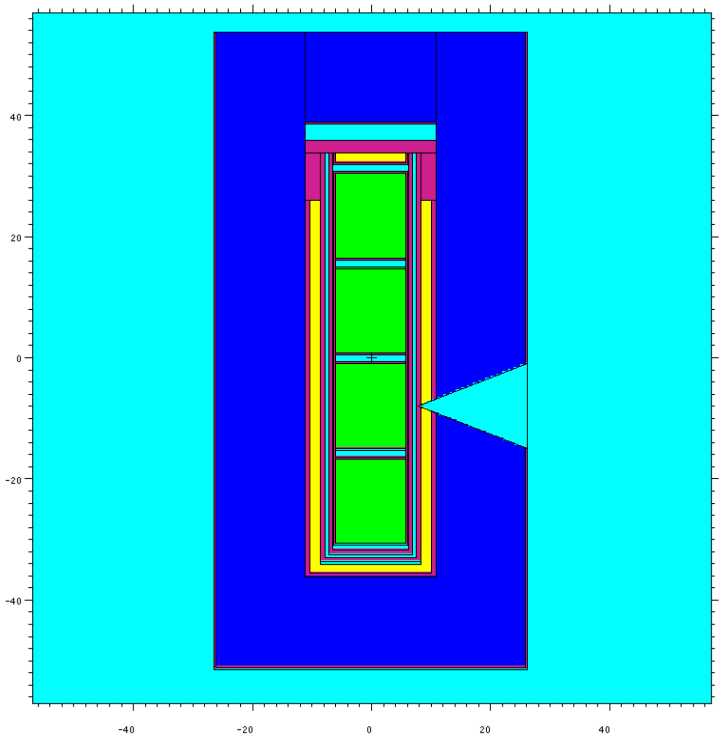
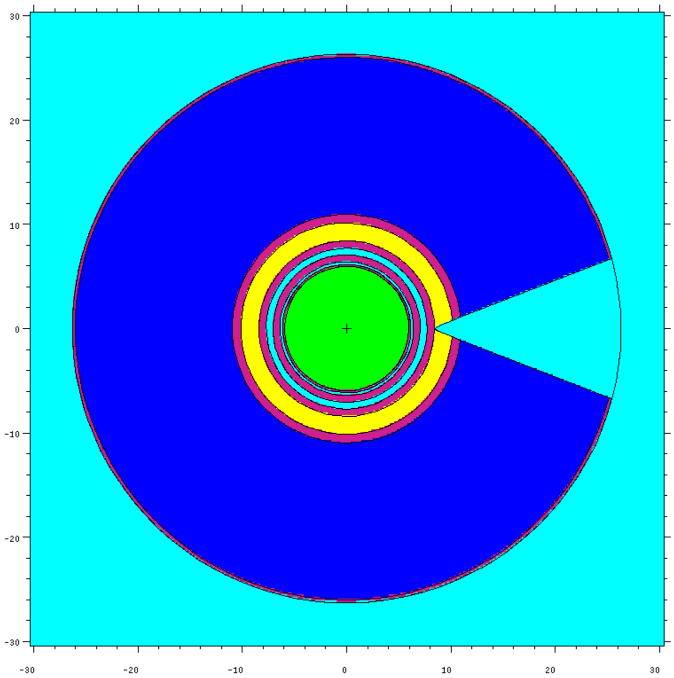


Figure 5: LEUPA package, Type C puncture test, upper view.



4. In order to demonstrate compliance after a Type C puncture test, an extra and conservative calculation was done by removing the exterior shielding except the cadmium layer, both in MCNP and MicroShield. Table 4 shows the results of the calculation in the "Only Cadmium shielding" row.

5. Type C puncture tests (including “Only Cadmium” calculations) described below leads to a dose rate of 0.0017 mSv/h at 1 m, which is well below the limit of 10 mSv/h required.
6. Table 4 reports the calculated dose rates in [mSv/h] for all the cases analyzed.

Table 4: External Dose Rate [mSv/h]

	MicroShield		MCNP	
	Contact	**1 m**	**Contact**	**1 m**
Nominal case, full shielding	1.16E-02	8.9E-04	6.3E-03	6.0E-04
Type B puncture test	1.21E-02	9.2E-04		
Type C puncture test			6.9E-03	6.2E-04
Only cadmium shielding	5.3E-02	1.7E-03	7.2E-03	7.8E-04

7. In Table 4, the MCNP results correspond to tallies averaged in spherical volumes. At 1 m it is a sphere of 20 cm radius with its center at 1 m from the surface of the LEUPA package. In contact, it corresponds to a sphere of 8 cm radius with its center at 4 cm from the surface of the package and directly outside the perforation described in the Type C puncture test.

8 CONCLUSIONS

1. The results shown in the previous section are used to demonstrate compliance with the regulatory requirements of [1] and [2] for a Type B(U) package. In particular, the following requirements are fulfilled:
   - **Paragraph 524.** Except for consignments under exclusive use, the TI of any package or overpack shall not exceed 10.
     - The calculated Transport Index is TI = 0.09
   - **Paragraph 525.** The maximum radiation level at any point on the external surface of a package or overpack shall not exceed 2 mSv/h.
     - The calculations result in 0.012 mSv/h for this value.
   - **Paragraph 646(b).** A package shall be so designed that if it were subjected to the tests specified in paras 719–724, it would prevent more than a 20% increase in the maximum radiation level at any external surface of the package.
     - The calculations result in a 4% increase in the maximum radiation level in contact.
   - **Paragraph 657(b).** A package shall be so designed that if it were subjected to the tests specified in paras 726, 727(b), 728 and 729 and the tests in paras. 727(a) it would retain sufficient shielding to ensure that the radiation level 1 m from the surface of the package would not exceed 10 mSv/h with the maximum radioactive contents which the package is designed to contain.
     - The calculations result in 9.2E-4 mSv/h for this value.
   - **Paragraph 669.** A package shall be so designed that if it were at the maximum normal operating pressure and subjected to the test sequences in paras. 734, it would retain sufficient shielding to ensure that the radiation level 1 m from the surface of the package would not exceed 10 mSv/h with the maximum radioactive contents which the package is designed to contain.
     - The calculations result in 1.7E-3 mSv/h for this value.