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Purpose of the document:

This chapter constitutes the radiological protection analysis of the TN-BGC 1 package model

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Field of application and summary:

APPENDICES (included in this document and therefore in global page numbers)			ATTACHMENTS (separate page numbers, identification and formal procedures)		
No.	TITLES	N° of pages	No.	TITLES	N° of pages
			3.5-1	Calculation of photon and neutron flow rates around a TN-BGC 1 package containing americium ref. CEA DCE-S-UGSP-SPR-SRI-T-99-1172 of 21/10/99	7
			3.5-2	Study of the shielding of the TN-BGC 1 packaging ref. CAL-07-00091006-002 Rév. 1 of 07/11/07	27

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	S. CLAVERIE-FORGUES (CEA/DEN/DPIE/SET)	

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1 INTRODUCTION.

The purpose of this chapter is to:

- assess the effectiveness of the package made up of the TN-BGC 1 packaging fitted with its internal arrangements and loaded with different contents detailed in Chapter 1,
- show that this type of package complies with regulations in terms of dose flow rates in contact with and 1 m from the packaging [1].

Several scenarios for various types of source are calculated:

- plutonium as oxide power PuO_2.
- source of cobalt-60,
- americium in random form (attachment 3.5-1).

The special case of contents 1b and 3b is dealt with in attachment 3.5-2.

2 CRITERIA TO BE RESPECTED

The regulatory boundaries for the total dose equivalent rate (DED) (neutrons and gamma) when transporting radioactive material are:

- in routine transport conditions:
  - 2 $mSv.h^{-1}$ on contact with the package walls,
  - 0.1 $mSv.h^{-1}$ 1 metre from the package walls,
- in accident conditions in transport (TAC):
  - 10 $mSv.h^{-1}$ 1 metre from the package walls.

In NTC, the dose equivalent rates in contact with the package walls should not exceed the routine condition criteria by more than 20%.

3 REFERENCES

[1] International Atomic Energy Agency Regulations for the transport of radioactive materials, Safety standards collection, no. TS-R-1 - 1996 edition (amended in 2005).

[2] MERCURE-IV Code
DEMT report 86/255 of 15/07/86
SERMA/LEPF/86/808
C. DUPONT - JC. NIMAL

[3] Nuclear databank on IBM-PC and compatible, NT-SPR/SRI/RPL/ 86-07 B. DESBRIERE, M. CHAIX

[4] TN-NEUTRON neutron dose rate computer code

[5] User manual for a gamma shielding calculation program PHMB1. 9571-P-51 Rev.2 of 26 September 1990

[6] Reactor Shielding Design Manual. First edition. T. ROCKWELL

4 DESCRIPTION OF SHIELDING

We remind in this paragraph the composition of neutron and gamma shielding (deduced from the detailed description in Chapter 2).

4.1 SHIELDING IN ROUTINE CONDITIONS AND NORMAL TRANSPORT CONDITIONS

4.1.1.1 Lateral shielding

This is formed by the stainless steel of the internal (6 mm) and external (1.5 mm) shells, for the most part the shielding from gamma radiation, and by the resin (50 mm, 48 mm min) for the shielding from the neutron radiation.

4.1.1.2 Shielding of the bottom

This is formed by the stainless steel at the bottom of the cavity (8 mm) and two closing metal plates (2 x 1.5 mm) as well as by the carbon steel of the diffuser plate (25 mm), for the most part shielding from gamma radiation, and by the resin (25 mm, 24 mm min.) and the wood (65 mm) for the shielding from the neutron radiation.

4.1.1.3 Shielding of the head

It is formed by the stainless steel of the plug (59 mm) and the metal plates of the cover (2 x 1.5 mm) for the gamma protection and by the resin (24 mm) and the wood (70 mm) contained in the cover for the neutron protection.

4.2 SHIELDING IN ACCIDENT CONDITIONS IN TRANSPORT

4.2.1.1 Radial shielding

It is assumed that the resin has totally disappeared after combustion (although it has been shown elsewhere that this layer is only partially burned in the regulatory fire conditions).

4.2.1.2 Axial shielding

It is assume that the wood and the resin in the covers have totally disappeared after combustion. The cover thickness is therefore reduced to a thickness equal to the thickness of wood before combustion.

5 CALCULATING DOSE RATES AROUND THE PACKAGE: PLUTONIUM AS OXIDE POWER PUO_2

5.1 CALCULATION HYPOTHESES

5.1.1 Characteristics of the content

The possible contents of the TN-BGC 1 packaging and the various related internal arrangements are detailed in Chapter 1.

Several calculation scenarios can be envisaged according to the permitted content; in this chapter, only the content inducing the most intense radiological source will be studied. The characteristics of this content are listed in the table below. It is made up of 17 kg of plutonium as PuO_2 powder packed in five boxes stacked on top of each other then inserted inside the internal arrangements.

Internal arrangement type	Materials	Thickness (mm)	External diameter (mm)	Matter contained	Maximum weight (kg)
Boxes	Stainless steel	0.7	100	PuO_2 powder	19.27
AA236	Stainless steel	4.0	121		
AA227	Stainless steel	5.0	138		

5.1.2 Isotopic composition of the plutonium

We have performed two types of calculation in this chapter:

- the first with a type of plutonium resulting from reprocessing PWR-type fuel assemblies irradiated to 33,000 MW.d/tU,
- the second taking into account a type of plutonium resulting from reprocessing type UNGG fuel assemblies.

Plutonium resulting from reprocessing PWR-type fuel assemblies		**Plutonium resulting from reprocessing UNGG-type fuel assemblies**	
Isotopic composition (% by mass)			
Pu 236	7.10-6 %	Pu 236	7.10^{-6} %
Pu 238	1.91 %	Pu 238	0.27 %
Pu 239	59.49 %	Pu 239	70.7 %
Pu 240	24.49 %	Pu 240	24.3 %
Pu 241	10.11 %	Pu 241	3.95 %
Pu 242	4.0 %	Pu 242	0.78 %
Residual fission products			
4µCi of ^{137}Cs per gram of plutonium			
Cooling time			
5 years		5 years and 10 years	

The atomic concentrations of various isotopes formed figure in Tables 3.5-1 (PWR plutonium) and 3.5-2 (UNGG plutonium).

5.2 ESTIMATION OF GAMMA DOSE RATES

5.2.1 Calculation method

The MERCURE-IV code [2] is used for these calculations. This deals with gamma propagation using the straight line attenuation method with a dose accumulation factor to take account of diffusions. The selective attenuation nucleus is integrated, in space and in energy, with the source volume by the Monte Carlo method to find the optimum importance. The MERCURE-IV code uses a gamma library of fifteen groups from 105 KeV to 8.65 MeV.

The gamma dose rates caused by neutron captures by shielding materials have been taken conservatively equal to 50% of the dose rate caused by the neutrons.

5.2.2 Gamma source

The gamma source considered in this chapter basically comes from:

- gamma radiation emitted by residual fission products present in the PuO_2 powder,
- gamma radiation accompanying the reactions α and β of the various decay chains of the plutonium isotopes,
- gamma radiation emitted following neutron captures in the source medium and in the shielding materials.

The gamma sources (apart from the share due to neutron captures) are given in Tables 3.5-3 and 3.5-4 for the PWR source and Tables 3.5-5 and 3.5-6 for the UNGG source; they have been calculated for the source using the ACTIV-PA code [3].

5.2.2.1 Modelling the package

The geometric calculation model presented in this paragraph consists of a simplified description of the package, which nevertheless retains the essential characteristics for the envisaged calculations; it is illustrated schematically in Figure 3.5-1. The TN-BGC 1 packaging is modelled by a set of meshes representing the various shielding areas (Figure 3.5-2). As the cover and the damping cover are more extensively shielded than the bottom, only the lower part of the packaging will be considered.

Table 3.5-7 gives the cross-references between physical media and geometric meshes. The composition of the various regions of the calculation model is specified below.

5.2.2.2 Composition of the source region (active part, meshes 1 to 5)

The source region is formed by the content of five boxes each loaded with 3.854 kg of PuO_2. Below are given the atomic concentrations of the PuO_2 after homogenisation in boxes with an internal radius of 4.93 m and a useful height of 23.76 cm.

Element	Atomic concentration 10^{24} at/cm^3
O	$9.39.10^{-3}$
Pu	$4.64.10^{-3}$

5.2.2.3 Shielding region no. 1 (meshes 6 to 20, 31 to 33, 37, 38, 41 to 43)

It is formed mainly by the stainless steel of the body; it represents the main shielding from the gamma rays.

The atomic concentrations considered for the calculations are listed below:

Element	Atomic concentration 10^{24} at/cm^3
Fe	$6.1341.10^{-2}$
Ni	$8.107.10^{-3}$
Cr	$1.6467.10^{-2}$

The same atomic concentrations are adopted for the bottom reinforcement in carbon steel.

5.2.2.4 Shielding region no. 2 (meshes 39 and 40)

It is formed of loaded resin; the atomic concentrations are defined as follows:

Element	**Atomic concentration 10^{24} at/cm^3**
O	$2.359.10^{-2}$
C	$2.383.10^{-2}$
Al	$4.3356.10^{-3}$
H	$4.0956.10^{-2}$
B	$9.308.10^{-4}$

5.2.2.5 Shielding region no. 3 (mesh 60)

Formed of wood incorporated into the bottom of the packaging, it has been replaced by the water with a density of 0.25 g/cm^3 and the following composition:

Element	**Atomic concentration 10^{24} at/cm^3**
H	$1.67.10^{-2}$
O	$8.37.10^{-3}$

5.2.3 Results

The dose equivalent rates calculated in contact with the packaging (cage) and 1 metre away (see Figure 3.5-3) due to the gamma rays emitted by the fission products and the decay gamma rays are summarised in Table 3.5-8 for the PWR plutonium and Table 3.5-9 for the UNGG plutonium.

The dose equivalent rates for the UNGG plutonium have been obtained from the PWR plutonium calculation by applying the ratio of UNNG/PWR gamma sources (for each energy group) to the dose rates.

5.3 ESTIMATION OF NEUTRON DOSE RATES

5.3.1 Calculation method

The neutron dose equivalent rates in the radial direction are calculated using the TN-NEUTRON code [4].

5.3.2 Neutron source

The neutron source taken into account in this chapter mainly comes from the following three methods:

- spontaneous fission of transuranium elements formed during the irradiation of fuel before reprocessing,
- reactions (α,n) of these transuranium elements on the oxygen-18 present in the powder PuO_2,
- the multiplication of these neutrons in the fissile medium of the package.

The detail of the neutron source is represented in Table 3.5-10 for the PWR plutonium and Table 3.5-11 for the UNGG plutonium.

A k_{eff} of 0.6 has been taken into account for the PWR plutonium for the multiplication of neutrons. The neutron emission corresponding to a weight of 17 kg of plutonium is broken down as follows:

FS = $8.34.10^6$ n/s: neutron source of path 1,

α,n = $6.471.10^6$ n/s: neutron source of path 2,

i.e. a total neutron emission of:

$$S = \frac{S_{FS} + S_{\alpha,n}}{1 - keff} = 3{,}705.10^7 \; n/s$$

The total source at five years and at ten years for the UNGG origin plutonium is calculated in the same way.

5.3.3 Modelling the package

To estimate the radial dose equivalent rates, the five boxes have been homogenised in a cylinder with a radius of 5 cm and 128 cm high.

To calculated the axial dose equivalent rates, we have considered a point source affected from the total neutron source located in the middle of the cavity (self-absorption is therefore ignored). The calculation produces very conservative results.

The calculation models are represented in Figure 3.5-4.

5.3.4 Results

The dose equivalent rates from captured neutrons and gamma rays in contact with and 1 metre from the packaging are summarised in Table 3.5-8 for the PWR plutonium and Table 3.5-9 for the UNGG plutonium.

The dose equivalent rates for the UNGG plutonium have been obtained from the PWR plutonium calculation by applying the ratio of UNGG/PWR neutron sources to the dose rates.

5.4 CONCLUSION

The maximum dose equivalent rates for 17 kg of Pu in oxide form are listed below.

		Dose equivalent rate in contact (mSv/h)	Dose equivalent rate 1 m away from the package (mSv/h)
Routine conditions	Radial	1.976	0.227
	Axial	0.694	0.079
Accident conditions	Radial		0.676
	Axial		< 9

The dose equivalent rates calculated around the package comply with the regulatory criteria except in routine conditions 1 m from the package, radially, where the calculation reveals a dose equivalent rate of 0.23 mSv/h greater than the criterion of 0.1 mSv/h.

However, Chapter 4 provides for measuring dose rates and checking their admissibility with respect to regulatory boundaries, which makes this overrunning of criterion acceptable.

The radiological protections of the package model are not degraded following the tests representative of normal transport conditions; they therefore comply with the dose equivalent rate increase criterion in contact with the package restricted to 20% of the dose equivalent rate determined in routine conditions.

6 CALCULATING DOSE RATES AROUND THE PACKAGE: SOURCE OF COBALT-60

The aim of this paragraph is to assess the effectiveness of the shielding of TN-BGC 1 packaging containing a 450 MBq source of cobalt-60 loaded in a TN90 internal arrangement.

6.1 CALCULATION HYPOTHESES

The calculations are performed using the PHMB1 code [5]. It is based on the gamma attenuation formalism presented in [6] and uses a homogenised cylindrical source system shielded by infinite media.

It is assumed that the source is a cylinder 2 cm high and 2 cm in diameter. The source is in contact with the internal shell of the TN90.

6.2 ASSESSING SOURCES

The dose equivalent rate calculations take into account characteristics of a cobalt-60 source emitting 450.10^{6} ph/s in each energy ray 1.17 MeV and 1.33 MeV.

6.3 ASSESSING DOSE RATES IN NORMAL TRANSPORT CONDITIONS

6.3.1 Radial calculation

The calculation takes place half way up the source, which is made up of 100% air of density 1.

6.3.2 Facing source

MEDIUM	DENSITY	THICKNESS (cm)
Source	1	R = 1
TN90 arrangement	7.85	0.2
Air	$1.29.10^{-3}$	2.7
Inner shell	7.85	0.6
Resin	1.6	4.8
Outer shell	7.85	0.15

The results of the dose equivalent rates are grouped in Table 3.5-12.

Extremities: this case is covered by the previous one as the shielding is constant over the entire useful length.

6.3.3 Axial calculation

Given that the steel parts making up the gamma shielding are extremely thick, the axial dose rate at the plug and the bottom is lower than the radial rate.

The radial gamma shielding is formed of 0.95 cm of steel and 4.8 cm of resin density 1.6 (the resin thickness is equivalent to 0.98 cm of steel). The total radial thickness of gamma shielding is 1.93 cm in steel equivalent. Axially, the gamma shielding is provided by steel plates at least 3.2 cm thick.

The axial study is therefore covered by the previous radial study.

6.4 ASSESSING DOSE RATES IN ACCIDENT CONDITIONS IN TRANSPORT

The gamma shielding is not altered. On the other hand, the resin has totally disappeared.

Given the dose equivalent rate values 1 m away from the packaging in routine conditions (see Table 3.5-12) and no loss in integrity of the steel biological shield following the tests representative of normal and accident conditions in transport, the dose equivalent rate will not increase by 20% in normal transport conditions and the boundary of 10 mSv/h 1 m away from the packaging in accident conditions will never be reached.

6.5 CONCLUSION

For a 450 MBq source of cobalt-60 loaded in the TN90 internal arrangement, the values of gamma dose equivalent rates remain less than the regulatory boundaries [1], regardless of whether in routine conditions, normal conditions or accident conditions in transport.

7 CALCULATING DOSE RATES AROUND THE PACKAGE: 50 G OF AMERICIUM IN RANDOM FORM

Attachment 3.5-1 reveals compliance with the regulatory criteria for the dose equivalent rate in routine conditions and accident conditions in transport for the package model made up of TN-BGC 1 packaging loaded with 50 g of americium in random form.

8 CONCLUSION

In routine conditions, the dose equivalent rate in contact with the packaging is less than 2 $mSv.h^{-1}$. It is lower than 0.1 $mSv.h^{-1}$ 1 m from the packaging except radially, for a load of 17 kg of plutonium in oxide form. Operating feedback on transport operations already taken place reveals that the regulatory criteria, measured before transport, have never been exceeded. The model used seems therefore widely conservative. Chapter 4 of this safety analysis report reaffirms the need for dose equivalent rate measurements before despatch, which makes the results presented as a rough guide acceptable.

In normal transport conditions, the dose equivalent rates in contact with the package walls do not exceed the routine condition criteria by more than 20%.

In accident conditions in transport, the dose equivalent rate 1 m away from the packaging remains less than 10 $mSv.h^{-1}$.

TABLE 3.5-1: ATOMIC CONCENTRATION OF ISOTOPES (PWR PLUTONIUM)

NUCLEIDE	N ATOME	ACTIVITE (Ci)
Tl208	7.66E+9	7.79E-4
Pb208	5.38E+15	-
Tl210	2.24E-1	5.38E-14
Pb210	3.63E+8	9.65E-12
Bi210	2.20E+5	9.50E-12
Pb211	1.79E+2	1.55E-12
Bi211	1.59E-2	2.33E-15
Pb212	4.44E+12	2.17E-3
Bi212	4.21E+11	2.17E-3
Po212	22.13	1.39E-3
Bi213	23.12	1.58E-13
Pb214	2.20E+4	2.56E-10
Bi214	1.63E+4	2.56E-10
Po214	2.25E-3	2.56E-10
Po215	1.47E-4	1.55E-12
Po216	1.68E+7	2.17E-3
At217	2.73E-4	1.58E-13
Po218	2.50E+3	2.56E-10
At218	5.47E-3	5.13E-14
Rn219	3.28E-1	1.55E-12
Rn220	6.44E+9	2.17E-3
Fr221	2.48	1.58E-13
Rn222	4.52E+6	2.56E-10
Fr223	1.60	2.30E-14
Ra223	8.18E+4	1.55E-12
Ra224	3.66E+13	2.17E-3
Ra225	1.12E+4	1.63E-13
Ac225	7.30E+3	1.58E-13
Ra226	6.97E+11	2.59E-10
Ac227	6.10E+7	1.66E-12
Th227	1.36E+5	1.57E-12
Ra228	5.82E+3	6.01E-16
Ac228	7.07E-1	6.01E-16
Th228	7.02E+15	2.18E-3
Th229	2.12E+9	1.71E-13
Th230	4.54E+16	3.57E-7
Th231	3.03E+9	6.17E-7
Pa231	1.80E+12	3.26E-11
Th232	8.18E+13	3.45E-15
U232	4.15E+17	3.52E-3
Pa233	2.50E+13	2.01E-4
U233	3.91E+14	1.46E-9
Th234	4.52E+7	4.06E-10
Pa234M	1.52E+3	4.06E-10
Pa234F	6.80E+2	5.28E-13
U234	6.36E+21	1.54E-2
U235M	9.90E+15	1.25E+2
U235	7.33E+20	6.18E-7
U236F	1.10E+21	2.80E-5
Pu236	1.80E+17	3.75E-2
U237	2.09E+16	6.72E-1
Np237	7.53E+20	2.09E-4
U238	3.12E+18	4.14E-10
Pu238	1.58E+23	1.07E+3
Pu239	5.09E+24	1.25E+2
Pu240	2.09E+24	1.89E+2
Pu241	6.78E+23	2.74E+4
Am241	1.80E+23	2.46E+2

N atome	N atom
Nucléide	Nuclide
Activité	Activity

TABLE 3.5-2: ATOMIC CONCENTRATION OF ISOTOPES (UNGG PLUTONIUM)

NUCLEIDE	DERNIERE DECROISSANCE			
	5 ANS		10 ANS	
	N ATOME	ACTIVITE (Ci)	N ATOME	ACTIVITE (Ci)
Pb206	2.64E+6	-	4.22E+7	-
Tl207	28.07	1.84E-12	2.24E+2	1.47E-11
Pb207	0	-	0	-
Tl208	7.66E+9	7.79E-4	1.36E+10	1.38E-3
Pb208	5.38E+15	-	2.37E+16	-
Tl209	0	1.34E-15	0	2.20E-14
Pb209	38.63	6.18E-14	6.36E+2	1.02E-12
Bi209	1.05E+6	-	1.08E+6	-
Tl210	0	7.61E-15	0	6.05E-14
Pb210	5.13E+7	1.36E-12	7.94E+8	2.11E-11
Bi210	3.11E+4	1.34E-12	4.85E+5	2.10E-11
Po210	5.82E+5	9.12E-13	1.09E+7	1.71E-11
Pb211	2.13E+2	1.84E-12	1.70E+3	1.47E-11
Bi211	12.63	1.84E-12	1.01E+2	1.47E-11
Pb212	4.44E+12	2.17E-3	7.87E+12	3.85E-3
Bi212	4.21E+11	2.17E-3	7.46E+11	3.85E-3
Po212	22.13	1.39E-3	39.24	2.47E-3
Bi213	9.03	6.19E-14	1.49E+2	1.02E-12
Po213	0	6.05E-14	0	9.95E-13
Pb214	3.11E+3	3.62E-11	2.47E+4	2.88E-10
Bi214	2.31E+3	3.62E-11	1.84E+4	2.88E-10
Po214	0	3.62E-11	0	2.88E-10
Po215	0	1.84E-12	0	1.47E-11
Po216	1.68E+7	2.17E-3	2.98E+7	3.85E-3
At217	0	6.19E-14	0	1.02E-12
Po218	3.54E+2	3.62E-11	2.82E+3	2.88E-10
At218	0	7.25E-15	0	5.77E-14
Rn219	0	1.84E-12	3.11	1.47E-11
Rn220	6.44E+9	2.17E-3	1.14E+10	3.85E-3
Fr221	0	6.18E-14	15.97	1.02E-12
Rn222	6.39E+5	3.62E-11	5.08E+6	2.88E-10
Fr223	1.90	2.73E-14	14.68	2.10E-13
Ra223	9.72E+4	1.84E-12	7.76E+5	1.47E-11
Ra224	3.66E+13	2.17E-3	6.50E+13	3.85E-3
Ra225	4.36E+3	6.39E-14	7.06E+4	1.03E-12
Ac225	2.85E+3	6.19E-14	4.69E+4	1.02E-12
Ra226	9.86E+10	3.66E-11	7.80E+11	2.90E-10
Ac227	7.25E+7	1.98E-12	5.59E+8	1.52E-11
Th227	1.61E+5	1.87E-12	1.27E+6	1.47E-11
Ra228	5.78E+3	5.96E-16	4.05E+4	4.18E-15
Ac228	0	5.96E-16	4.92	4.18E-15
Th228	7.02E+15	2.18E-3	1.24E+16	3.85E-3
Th229	8.27E+8	6.69E-14	1.31E+10	1.06E-12
Th230	6.42E+15	5.05E-8	2.53E+16	1.99E-7
Th231	3.60E+9	7.34E-7	7.20E+9	1.47E-6
Pa231	2.14E+12	3.88E-11	8.57E+12	1.55E-10
Th232	8.11E+13	3.43E-15	3.24E+14	1.37E-14
U232	4.15E+17	3.52E-3	5.18E+17	4.39E-3
Pa233	9.76E+12	7.84E-5	3.68E+13	2.96E-4
U233	1.53E+14	5.70E-10	1.18E+15	4.42E-9
Th234	8.81E+6	7.93E-11	1.78E+7	1.60E-10
Pa234M	2.97E+2	7.93E-11	6.00E+2	1.60E-10
Pa234F	1.33E+2	1.03E-13	2.68E+2	2.08E-13
U234	8.99E+20	2.18E-3	1.76E+21	4.28E-3
U235M	1.18E+16	1.49E+2	1.18E+16	1.49E+2
U235	8.71E+20	7.34E-7	1.74E+21	1.47E-6
U236F	1.10E+21	2.78E-5	2.19E+21	5.56E-5
Pu236	1.80E+17	3.75E-2	5.34E+16	1.11E-2
U237	8.17E+15	2.63E-1	6.46E+15	2.07E-1
Np237	2.94E+20	8.16E-5	1.09E+21	3.01E-4
U238	6.08E+17	8.08E-11	1.22E+18	1.62E-10
Pu238	2.23E+22	1.51E+2	2.15E+22	1.45E+2
Pu239	6.05E+24	1.49E+2	6.05E+24	1.49E+2
Pu240	2.07E+24	1.88E+2	2.07E+24	1.88E+2

Nucléide	Nuclide
N atome	N atom
Activité	Activity
Dernière décroissance	Last decay

TABLE 3.5-3: DECAY GAMMA SOURCE (PWR PLUTONIUM)

Groupe Mercure	Energie Moyenne (Mev)	γ/s (*)	γ/s.cm³ (*)
7	2,5	$2,88.10^{7}$	$1,58.10^{4}$
8	2,0	0	0
9	1,5	$1,46.10^{6}$	$8,00.10^{2}$
10	1,0	$5,22.10^{7}$	$2,86.10^{4}$
11	0,628	$4,048.10^{7}$	$2,22.10^{4}$
12	0,424	$1,44.10^{8}$	$7,89.10^{4}$
13	0,287	$3,44.10^{8}$	$1,88.10^{5}$
14	0,194	$6,19.10^{9}$	$3,39.10^{6}$

TABLE 3.5-4: GAMMA SOURCE FROM FISSION PRODUCTS (PWR PLUTONIUM)

Groupe Mercure	Energie Moyenne (Mev)	γ/s (*)	γ/s.cm³ (*)
11	0,628	$3,38.10^{8}$	$1,852.10^{5}$
15	0,1308	$3,14.10^{7}$	$1,72.10^{4}$

* Ces sources sont calculées pour une masse de 3,4 kg de Pu, masse maximum admissible par boîte.

Groupe mercure	Mercure group
Energie moyenne	Average energy
Ces sources sont calculees pour une masse de 3,5 kg de pu, masse maximum admissible par boite	These sources are calculated for a weight of 3.5 of pu, the maximum permitted weight per box

TABLE 3.5-5: DECAY GAMMA SOURCE (UNGG PLUTONIUM)

Energie Mercure	Energie moyenne (MeV)	γ/s		γ/s.cm³	
		5 ans	10 ans	5 ans	10 ans
7	2,5	$2,87.10^7$	$5,096.10^7$	$1,56.10^4$	$2,77.10^4$
8	2,0	0	0	0	0
9	1,5	$1,464.10^6$	$2,612.10^6$	$7,96.10^2$	$1,42.10^3$
10	1,0	$1,214.10^7$	$1,610.10^7$	$6,60.10^3$	$8,75.10^3$
11	0,628	$3,80\ .10^7$	$6,60\ .10^7$	$2,065.10^4$	$3,58.10^4$
12	0,424	$1,711.10^8$	$1,716.10^8$	$9,30\ .10^4$	$9,33.10^4$
13	0,287	$1,608.10^8$	$1,729.10^8$	$8,74\ .10^4$	$9,40.10^4$
14	0,194	$2,373.10^9$	$1,919.10^9$	$1,29\ .10^6$	$1,043.10^6$

TABLE 3.5-6: GAMMA SOURCE FROM FISSION PRODUCTS (UNGG plutonium)

Energie Mercure	Energie moyenne (MeV)	γ/s		γ/s.cm³	
		5 ans	10 ans	5 ans	10 ans
11	0,628	$3,38.10^8$	$3,0.10^8$	$1,85.10^5$	$1,63.10^5$
15	0,131	$3,14.10^7$	$2,8.10^7$	$1,72.10^4$	$1,52.10^4$

Ces sources sont calculées pour une masse de 3,4 kg de Pu, masse maximum admissible par boîte.

Energie mercure	Mercure energy
Energie moyenne	Average energy
5 [10] ans	5 [10] years
Ces sources sont calculees pour une masse de 3,5 kg de pu, masse maximum admissible par boite	These sources are calculated for a weight of 3.5 of pu, the maximum permitted weight per box

TABLE 3.5-7: CROSS-REFERENCES BETWEEN PHYSICAL MEDIA AND GEOMETRIC MESHES

Maille n°	Milieu physique correspondant
1 à 5	Milieu source
6 à 20	Acier inox des boîtes
31 à 33	Acier des aménagements internes
37 et 38	Viroles en acier
39 et 40	Résine, protection neutronique radiale et axiale
60	Bois du fond de l'emballage
42 et 43	Tôles extérieures en acier inox
41	Tôle en acier inox séparant le bois de la résine
21 à 29	Air autour des boîtes
63, 64 et 35	Air autour du colis et entre les aménagements internes
36 et 66	Air dans la cavité de l'emballage

Maille n°	Mesh no.
Milieu physique correspondant	Corresponding physical medium
Milieu source	Source medium
Acier inox des boîtes	Stainless steel of the boxes
Acier des aménagements internes	Steel of the internal arrangements
Viroles en acier	Steel shells
Résine, protection neutronique radiale et axiale	Resin, radial and axial neutron protection
Bois du fond de l’emballage	Wood at bottom of the packaging
Tôles extérieures en acier inox	External stainless steel plates
Tôle en acier inox séparant le bois de la résine	Stainless steel plate separating the wood from the resin
Air autour des boîtes	Air around boxes
Air autour du colis et entre les aménagements internes	Air around the package and between the internal arrangements
Air dans la cavité de l’emballage	Air in the packaging cavity:

TABLE 3.5-8: SUMMARY OF MAXIMUM DOSE RATES AROUND THE PACKAGE FOR THE PWR PLUTONIUM

	DEBIT DE DOSE MAXIMUM (mRem/h)		au contact de la cage		à 1 m de la cage		à 2 m de la cage	
			Radial	Axial (*)	Radial	Axial (*)	Radial	Axial (*)
Conditions normales de transport	γ	Décroissance	37,0	< 14,7	4,3	< 0,5	1,3	< 0,14
	γ	Dus aux produits de fission	26,7	< 6,7	3,3	< 0,2	1,1	< 0,05
	γ	Capture	44,6	< 16	5,0	< 2,4	1,7	0,9
	neutrons		89,3	< 32	10,1	< 4,8	3,4	1,8
	total		197,6	69,4	22,7	7,9	7,5	2,9
Conditions accidentelles de transport	γ	Décroissance	--	--	6,5	< 1,5	--	--
	γ	Dus aux produits de fission	--	--	5,6	< 0,6	--	--
	γ	Capture	--	--	18,5	< 300	--	--
	neutrons		--	--	37,0	< 600	--	--
	total		--	--	67,6	< 902	--	--

(*) Seuls les débits de dose axiaux côté fond sont donnés dans ce tableau, ils couvrent ceux du côté couvercle qui est surblindé par rapport au fond.

Débit de dose maximum	Maximum dose rate
au contact de la cage	in contact with the cage
à 1 [2] m de la cage	1 [2] m away from the cage
Radial	Radial
Axial	Axial
Décroissance	Decay
Dues au produits de fission	Due to the fission products
Capture	Capture
neutrons	neutrons
total	total
Conditions normales de transport	Normal transport conditions
Conditions accidentelles de transport	Accident conditions in transport
Seuls les débits de dose axiaux côté fond sont donnés dans ce tableau, ils couvrent ceux du côté couvercle qui est surblindé par rapport au fond.	Only the axial dose rates on the bottom side are given in this table. They cover those on the cover side which is shielded more than the bottom.

TABLE 3.5-9: SUMMARY OF MAXIMUM DOSE RATES AROUND THE PACKAGE FOR THE UNGG PLUTONIUM

	DEBIT DE DOSE MAXIMUM (mRem/h)		au contact de la cage (*)		à 1 m de la cage (*)		à 2 m de la cage (*)	
			5 ans	10 ans	5 ans	10 ans	5 ans	10 ans
Conditions normales de transport	γ	Décroissance	23,1	32	2,61	3,48	1,0	1,2
		Dus aux produits de fission	26,7	25,6	3,25	3,13	1,1	1,05
		Capture	25,3	26	2,86	2,96	1,0	1,0
	neutrons		50,7	52,2	5,73	5,91	1,93	2,0
	total		126	135,6	14,45	15,48	5,1	5,3

(*) Les points de calculs sont situés à mi-hauteur de l'emballage

Débit de dose maximum	Maximum dose rate
au contact de la cage	in contact with the cage
à 1 [2] m de la cage	1 [2] m away from the cage
Décroissance	Decay
Dues au produits de fission	Due to the fission products
Capture	Capture
neutrons	neutrons
total	total
Conditions normales de transport	Normal transport conditions
Les points de calculs sont situés à mi-hauteur de l'emballage	The calculation points are half way up the packaging

Note: the axial dose equivalent rates in normal transport conditions and the dose rates in accident conditions are lower than those given in Table 3.5-8 for PWR plutonium.

TABLE 3.5-10: TOTAL NEUTRON SOURCE (PWR PLUTONIUM)

Isotope	Source (α,n) n/s	Source FS n/s	Source totale n/s <1>
Pu 238	$4{,}365.10^{6}$	$1{,}0663.10^{6}$	$1{,}36.10^{7}$
Pu 239	$4{,}55.10^{5}$	0	$1{,}14.10^{5}$
Pu 240	$7{,}045.10^{5}$	$5{,}712.10^{6}$	$1{,}60.10^{6}$
Am 241	$9{,}45.10^{5}$	0	$2{,}36.10^{5}$
Pu 242	1833	$1{,}562.10^{6}$	$3{,}91.10^{6}$
Total	$6{,}471.10^{6}$	$8{,}340.10^{6}$	$3{,}70.10^{7}$

<1> La source totale tient compte du facteur de multiplication des neutrons dans le milieu fissile imposé par un keff de 0,6. Cette source a été calculée pour 17 kg de Pu (masse maximum contenue par l'ensemble des 5 boîtes).

TABLE 3.5-11: Total neutron source (UNGG plutonium)

Isotope	Source (α,n) n/s		Source FS n/s		Source totale * n/s	
	5 ans	10 ans	5 ans	10 ans	5 ans	10 ans
Pu 238	$6{,}19.10^{5}$	$5{,}92.10^{5}$	$1{,}51.10^{5}$	$1{,}446.10^{5}$	$1{,}92.10^{6}$	$1{,}84.10^{6}$
Pu 239	$5{,}41.10^{5}$	$5{,}41.10^{5}$	0	0	$1{,}35.10^{6}$	$1{,}35.10^{6}$
Pu 240	$7{,}04.10^{5}$	$7{,}04.10^{5}$	$5{,}82.10^{6}$	$5{,}719.10^{6}$	$1{,}61.10^{7}$	$1{,}61.10^{7}$
Am 241	$3{,}69.10^{5}$	$6{,}57.10^{5}$	0	0	$9{,}22.10^{5}$	$1{,}64.10^{6}$
Pu 242	358	358	$3{,}05.10^{5}$	$3{,}05.10^{5}$	$7{,}63.10^{5}$	$7{,}63.10^{5}$
Total	$2{,}23.10^{6}$	$2{,}49.10^{6}$	$6{,}17.10^{6}$	$6{,}17.10^{6}$	$2{,}10.10^{7}$	$2{,}17.10^{7}$

* La source totale tient compte du facteur de multiplication des neutrons dans le milieu fissile imposé par un keff de 0,6.

Isotope	Isotope
Source totale	Total source
La source totale tient compte du facteur de multiplication des neutrons dans le milieu fissile imposé par un keff de 0,6. Cette source a été calculée pour 17 kg du Pu (masse maximum contenue par l'ensemble des 5 boîtes).	The total source takes into account the neutron multiplication factor in the fissile medium imposed by a keff of 0.6. This source has been calculated for 17 kg of Pu (maximum weight contained by the five boxes together).

TABLE 3.5-12: SOURCE OF COBALT-60: DOSE RATES IN NTC OUTSIDE THE PACKAGING CALCULATED RADIALLY

External surface distance of the packaging - calculation point		Source of cobalt-60 (mSv/h)
Contact	γ	1,95 < 2
1 m	γ	0,075 < 0,1
2 m	γ	0,023 < 0,1

FIGURE 3.51: GEOMETRIC CALCULATION MODEL FOR MERCURE IV

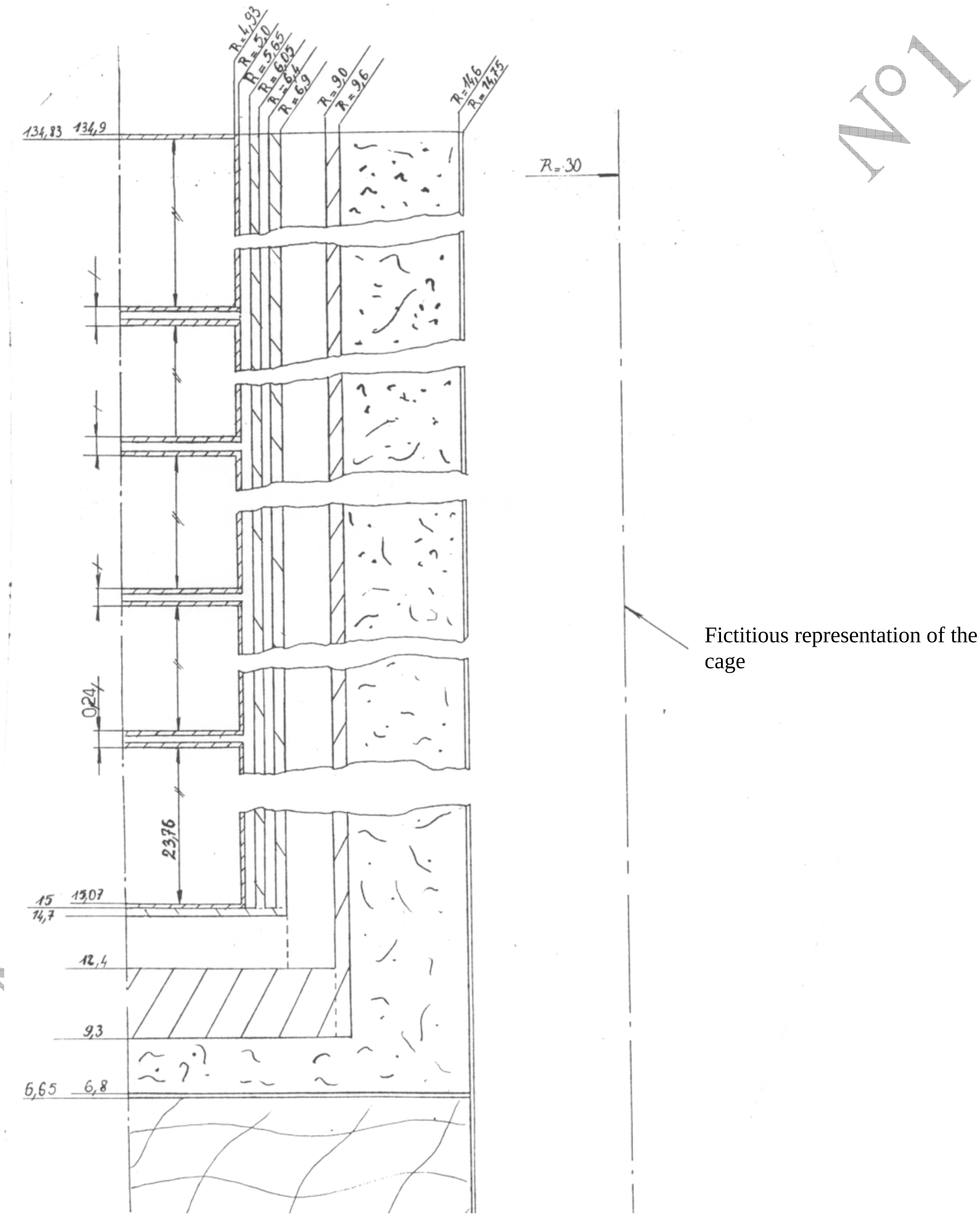


FIGURE 3.52: CALCULATION MODEL MESHING FOR MERCURE IV

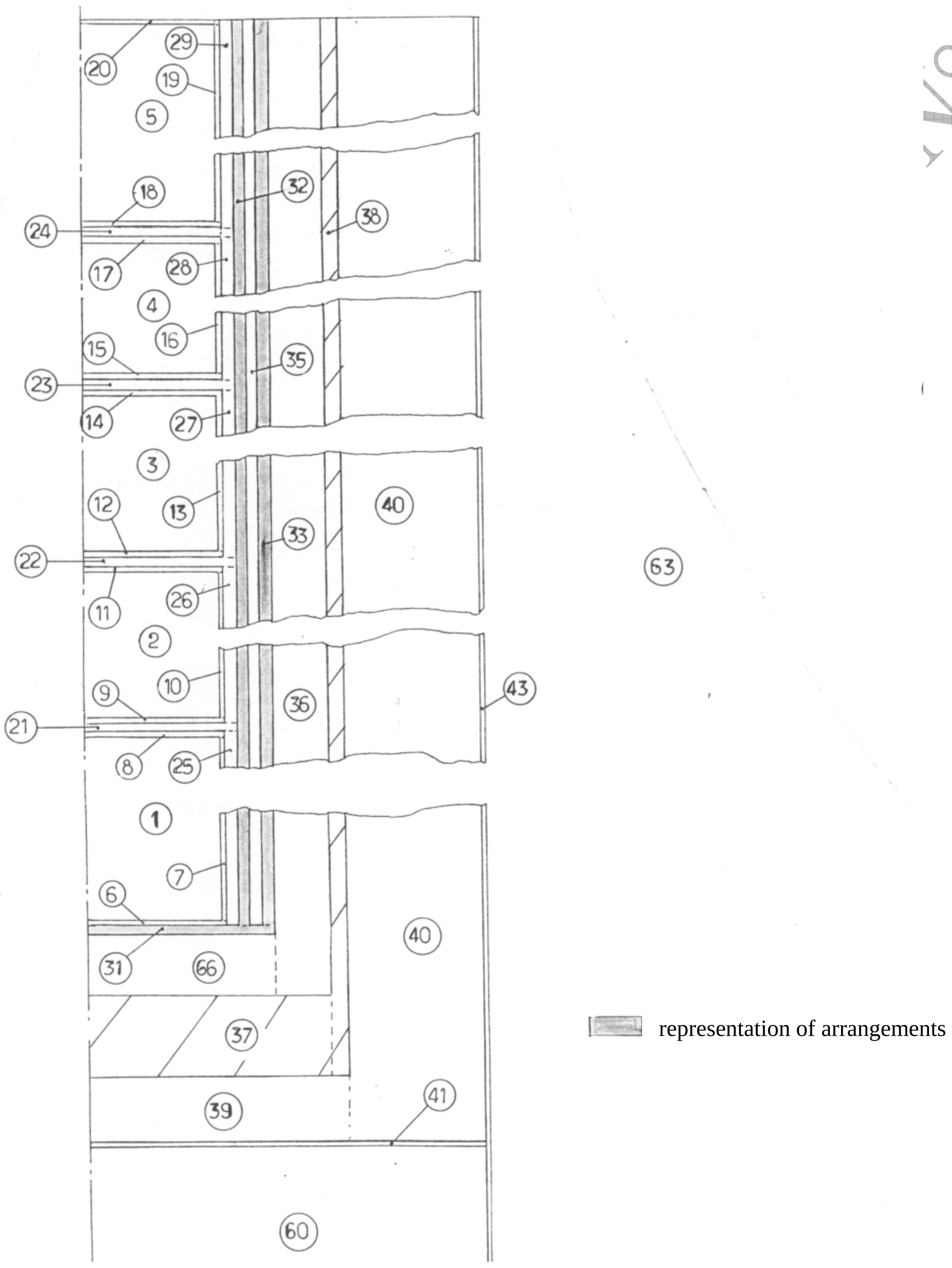


FIGURE 3.53: POSITION OF CALCULATION POINTS AND CORRESPONDING FLOW RATES

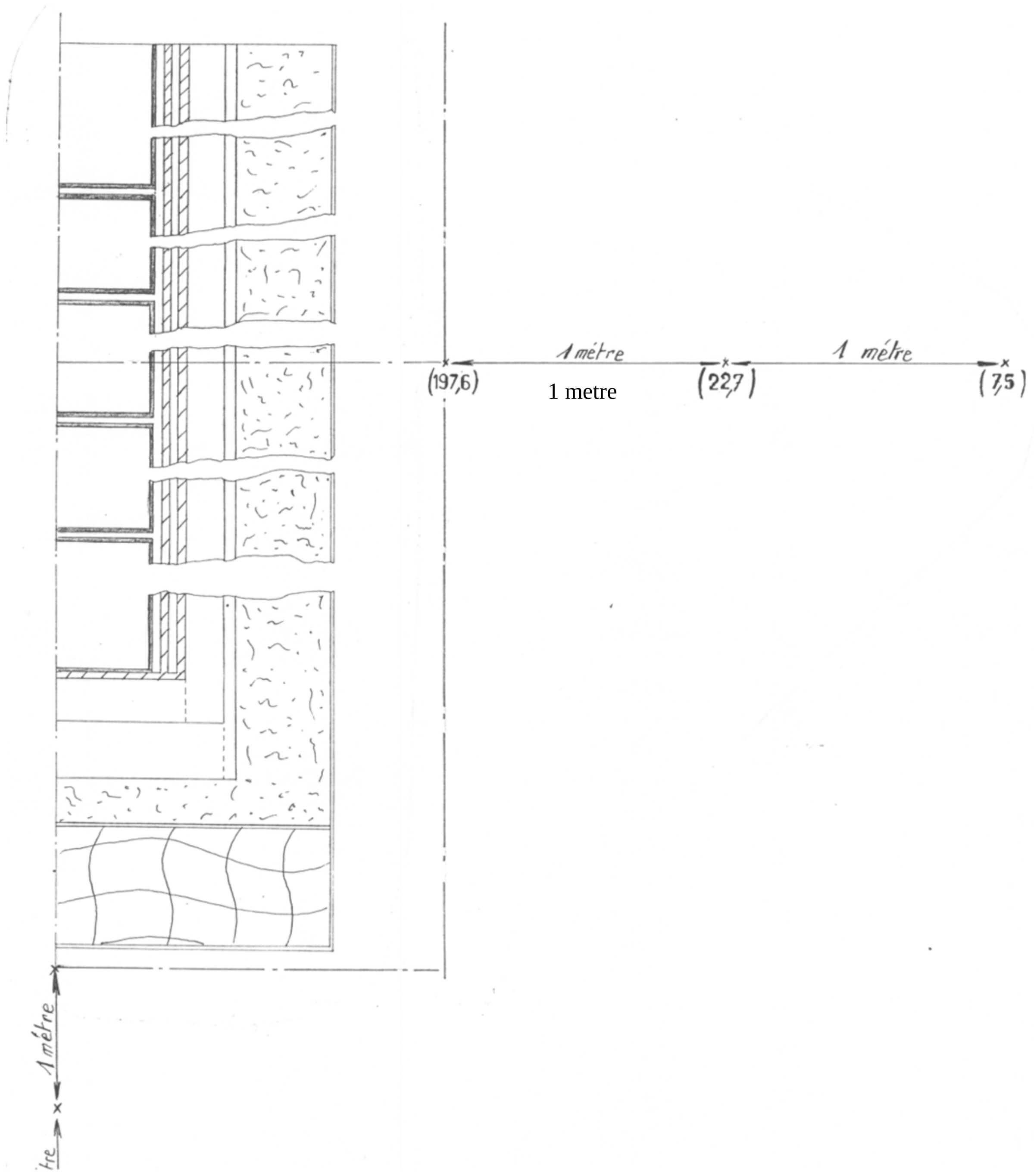
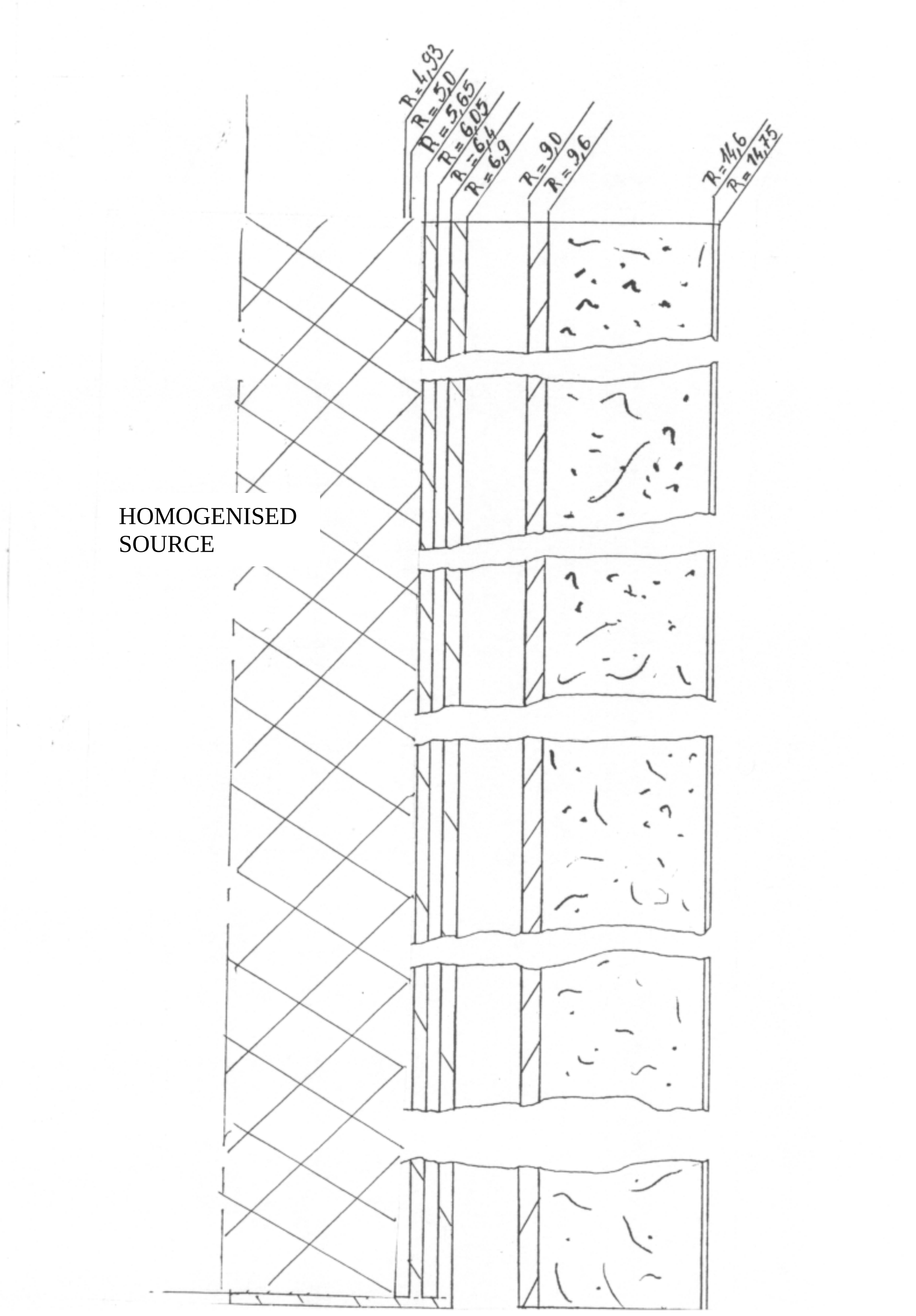


FIGURE 3.54: GEOMETRIC MODEL FOR THE TN-NEUTRON NEUTRON CALCULATION



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APPENDICES(included in this document and therefore in global page numbers)			ATTACHMENTS(separate page numbers, identification and formal procedures)		
No.	TITLES	N° of pages	No.	TITLES	N° of pages
			3.6-1	Additional criticality calculations ref. EMB TNBGC PBC DJS CA 000481 A of 05/01/04	51
			3.6-2	Criticality-safety of the TN-BGC 1 packaging - Search for permitted weight - content 8 ref. NT.12862.00.008. Rev. A of 19-Nov-04	47
			3.6-3	Criticality-safety of the TN-BGC 1 packaging - Search for permitted number of packages - content 8 ref. NT.12862.00.009. Rev. A of 19-Nov-04	41
			3.6-4	Criticality calculations for the TN-BGC 1 packaging ref. CEA/DEN/CAD/DTAP/SET DO 247 of 14/05/04	3
			3.6-5	Criticality study of the TN-BGC 1 packaging loaded with plutonium oxide powders ref. 160 EMBAL PFM NOT 06001678 A of 07/11/06	36
			3.6-6	Criticality study of the TN-BGC 1 packaging loaded with miscellaneous contents: influence of the type of wedges ref. 160 EMBAL PFM NOT 06001677 A of 07/11/06	27
			3.6-7	Criticality study of the TN-BGC 1 packaging loaded with content 6: uranyl nitrate solution ref. 160 EMBAL PFM NOT 06001679 A of 07/11/06	21
			3.6-8	Criticality study of the TN-BGC 1 packaging loaded with content 11: 100% U metal medium in ^{235}U ref. 160 EMBAL PFM NOT 06001680 A of 07/11/06	31
			3.6-9	Criticality study of the TN-BGC 1 packaging loaded with content 26: $UZrH_2$ medium ref. 160 EMBAL PFM NOT 06001518 A 06/10/06	33
			3.6-10	Criticality study of the TN-BGC 1 packaging loaded with miscellaneous MCMF contents (contents from the central fissile material warehouse) in normal and accident conditions in transport ref. 160 EMBAL PFM NOT 06001310 C of 24/07/07	56
			3.6-11	Criticality-safety of the TN-BGC 1 package model loaded with content 26: TRIGA fuel ref. EMB TNBGC PBC DJS CA 000386 A of 07/08/03	69
			3.6-12	Study of the criticality-safety of the TN-BGC 1 packaging ref. CAL-07-00090811-002 Rev. 00 of 01/10/07	26

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TABLE OF CONTENTS

1 INTRODUCTION

In this chapter, we show that the package formed by the TN-BGC 1 packaging loaded with contents described in Chapter 1 satisfies the regulatory stipulations regarding packages containing fissile materials.

2 REFERENCES

[1] International Atomic Energy Agency Regulations for the transport of radioactive materials, Safety standards collection, no. TS-R-1 - 1996 edition (amended in 2005).

3 METHODOLOGY ADOPTED TO ASSESS NUCLEAR SAFETY

We describe in this paragraph the general methodology adopted to assess the nuclear safety of the TN-BGC 1 package model loaded with the various contents described in Chapter 1.

The general assumptions made in the various studies attached to this chapter are also presented.

3.1 GENERAL ASSUMPTIONS AND CALCULATION METHOD

The nuclear safety of the TN-BGC 1 package model is based on:

- the characteristics of the fissile material transported,
- the constitution and structural resistance of the packaging and internal arrangements (containers type AA 226, AA 227, AA 236, AA 303, AA 204, AA 203, AA 41, TN 90 as well as certain wedges) which maintain their geometry under the regulatory tests representative of normal and accident conditions in transport,
- the composition and thickness of constituent packaging materials.

The nuclear safety analysis is based on the multiplication coefficient calculation (K_{eff}) of an isolated package and a package array, under regulatory moderation and reflection conditions.

The admissibility criteria adopted are as follows:

- $k_{eff} + 3\sigma \leq 0.95$ for an isolated package,
- $k_{eff} + 3\sigma \leq 0.98$ for a package array.

Additional margins over these criteria can be applied for certain media where there are still uncertainties over the nuclear data (slightly moderate plutonium-bearing media, etc. . .).

3.2 DEFINING THE ISOLATION SYSTEM

The following components make up the isolation system to guarantee:

- the packaging: geometry (maximum diameter 181 mm), materials (the internal and external packaging shells are in stainless steel), materials used, composition and thickness of the neutron-absorbing

borated resin (hydrogen and boron content, thickness of burnt resin),

- the diameter and thickness of the internal arrangement in stainless steel or aluminium alloy,
- the wedging system which delimits the radial position occupied by the fissile material,
- in some cases, the packaging cage (60 cm x 60 cm) which spaces the array in package configuration 5N in NTC,
- the fissile material: checking the criticality by restricting the weight and, in some cases, the composition of the fissile medium.

3.3 ASSESSING PACKAGES CONSIDERED IN ISOLATION

3.3.1 Reminder of the assessment conditions

The regulations specify for the packages considered in isolation:

- that it should be assumed that the water can penetrate all the voids in the package, especially those inside the containment system, or escape from them,
- for the isolation system, total reflection should be assumed through at least 20 cm of water or any other greater reflection as a supplement by the materials in the neighbouring packaging. However, where it can be demonstrated that the isolation system remains inside the packaging following tests specified in sub-paragraph 682b) of reference [1], total reflection of the package can be assumed through at least 20 cm of water.

The packages must be sub-critical in the package conditions producing the maximum multiplication of neutrons compatible following:

- routine transport conditions,
- normal transport conditions,
- accident conditions in transport.

3.3.2 Assumptions made

The studies on the packages considered in isolation adopted the following common assumptions:

- water penetrates in all the voids in the package, including in the internal arrangements which are in the packaging cavity,
- the covers are ignored,
- the package is isolated by a ring of water 20 cm thick,
- the package is damaged, with its state that of a package after the following tests:
  - **mechanical test** comprising a drop test (dynamic flattening of the package by a 500 kg plate falling from a height of 9 m) combined with a free fall test of the package from a height of 1 m on a punch.
    The analyses in Chapters 3.1 and 3.2 of this safety dossier show that the mechanical tests above have no effect on the geometry of the isolation system. However, certain studies attached to this chapter consider conservatively local deformations of internal arrangements for certain types of packing. These deformations result from the free fall test of the package from a height of 9 m and should therefore no longer be considered in strict logic as the plate drop is the test required by [1].
  - **fire test**.
    The analysis performed in Chapter 3.3 shows that following the fire test, the maximum thickness of the burnt resin is, radially, 10 mm; the resin is not damaged at the packaging extremities and maintains a thickness of 24 mm.

For the criticality-safety evaluation, the ring of burnt resin is replaced by air, water or a water mist of optimum density to maximise reactivity. Conservatively, certain calculations consider a burnt resin thickness of 15 mm instead of the 10 mm noted after tests representative of accident conditions in transport. The composition of the resin is identical to the nominal composition in the non-burnt part.

3.4 ASSESSING PACKAGES IN NUMBER IN NORMAL TRANSPORT CONDITIONS

3.4.1 Reminder of the assessment conditions

A number N is determined such that 5N is sub-critical for the package layout and conditions producing the maximum multiplication of neutrons compatible with the following conditions:

- There is nothing between the packages and the package layout is surrounded on all sides by a layer of water at least 20 cm acting as a reflector,
- the package state is that observed after the tests representative of normal transport conditions.

3.4.2 Assumptions made

It is demonstrated in Chapter 3.1 that, following tests representative of normal transport conditions, the package isolation system has not been damaged.

The hypotheses can vary from one content to the next. Their broad outlines are summarised below:

- the covers are ignored,
- water penetrates in all the voids in the packages, including in the internal arrangements which are in the packaging cavity,
- the penalising hypothesis of 15 mm of burnt resin is normally kept; this is replaced by air so as to maximise the neutron interactions.

3.5 ASSESSING PACKAGES IN NUMBER IN ACCIDENT CONDITIONS IN TRANSPORT

3.5.1 Reminder of the assessment conditions

A number N is determined such that 2N complies with the criteria fixed for the package layout and conditions producing the maximum multiplication of neutrons compatible with the following conditions:

- moderation is optimum and the assembly formed by the 2N packages is surrounded by a 20 cm ring of water,
- the package is in the state noted after tests in normal transport conditions followed by tests in accident conditions in transport (burnt resin and no cage).

3.5.2 Assumptions made

The studies of an array of 2N packages makes the same assumptions as described above for the normal transport conditions, except for the following points:

- the cage is ignored,
- in some cases, internal arrangement deformation is considered.

Note: it is important to state that the hypotheses can differ from those listed above. In this case, they are justified and lead to optimum reactivity.

3.6 PRESENCE OF MATERIALS MORE HYDROGENATED THAN WATER

For certain contents, the fissile medium is moderated by a random quantity of polyethylene.

3.7 AIR TRANSPORT (CONTENTS 11 AND 26)

In this case, the studies take into account the following elements:

- possible moderation due to hydrogenated materials in the packaging (residual water in the wood),
- taking into account carbonated elements in the wood,
- reflection from the steel making up the packaging.

4 RESULTS

4.1 INFLUENCE OF THE TYPE OF WEDGES

Note 160 EMBAL PFM NOT 06001677 A of 07/11/06 (attachment 3.6-6) reveals that no conclusion can be drawn on significant changes in reactivity (as it is lower than the calculation uncertainty) due to the variation in composition and/or density of the aluminium alloy.

The criteria defined in § 3.1 are therefore not influenced significantly by the existing uncertainties in the cross sections of elements making up the aluminium alloy.

4.2 CONTENT 1: PLUTONIUM OXIDE POWDER

A minimum margin of 2500 pcm over the criteria defined in § 3.1 is considered for this content, to take into account the uncertainties over the nuclear data of this fissile medium when it is only slightly moderated.

The calculations presented in attachment 3.6-5 consider a fissile material (PuO_2) with a density no greater than 3.5, in homogenous form and moderated by a **variable amount of water**.

The results are listed below:

- for N=25 package, the permitted weight of Pu is:

Fissile media	Φ Arrangement (mm)	Arrangement thickness	Permitted weight of Pu (kg)
PuO_2 - 100% ^{239}Pu	≤ 120	≥ 2 mm	5
PuO_2 - $^{240}Pu \geq 5\%$	≤ 120	≥ 2 mm	13,5
	≤ 130	≥ 5 mm	10

- the search for the permitted number of packages leads to:

Fissile medium	Φ Arrangement (mm)	Weight of Pu per package (kg)	Arrangement thickness	Permitted number of packages
PuO_2 - $^{240}Pu \geq 5\%$	≤ 120	17	≥ 2 mm	16
	≤ 130	13	≥ 5 mm	9

The calculations presented in attachment 3.6-10 consider the presence of a **random amount of polyethylene** and a fissile material with random density:

- in homogeneous form for the PuO_2 with 100% ^{239}Pu,

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- in heterogeneous form for the PuO_2 with 5% ^{240}Pu.

The results are listed below:

- for the PuO_2 with 100% ^{239}Pu and for N=1 package, the permitted weight of Pu is:

Fissile media	Φ Arrangement (mm)	Arrangement thickness	Permitted weight of Pu (kg)
PuO_2 - 100% ^{239}Pu	≤ 120	≥ 2 mm	4

- for the PuO_2 with 5% ^{240}Pu and for N=10 packages, the permitted weight of Pu is:

Fissile media	Φ Arrangement (mm)	Arrangement thickness	Permitted weight of Pu (kg)
PuO_2 - ^{240}Pu ≥ 5%	≤ 120	≥ 2 mm	5

The special case of content 1b is dealt with in attachment 3.6-12.

4.3 CONTENT 2: URANIUM OXIDE POWDER

The calculations presented in attachment 3.6-1 consider a fissile material (UO_2), of random density, in homogeneous form and moderated by a variable amount of water. N = 25.

Fissile media	Φ Arrangement (mm)	Arrangement thickness	Permitted weight of Pu (kg)
UO_2 - $^{235}U/U$ ≥ 20%	≤ 120	≥ 2 mm	20
UO_2 - $^{235}U/U$ 20%	≤ 120	≥ 2 mm	40

4.4 CONTENT 3: MIXED URANIUM-PLUTONIUM OXIDE POWDER

A minimum margin of 2500 pcm over the criteria defined in § 3.1 is considered for this content, to take into account the little experience in only slightly moderated plutonium-bearing media.

The calculations presented in attachments 3.6-1 and 3.6-4 consider a **homogenous mix of water** and UO_2+PuO_2.

Two types of MOX contents are studied:

- natural UO_2 (enriched with 0.71% ^{235}U) combined with the PuO_2 of isotopic vector 95% ^{239}Pu and 5%

^{240}Pu,

- UO2 enriched 100% with ^{235}U combined with the PuO_2 of isotopic vector 95% ^{239}Pu and 5% ^{240}Pu.

The Pu/U+Pu ratio will be fixed at 30% and N=25 in both cases.

The density of the fissile medium is random.

Fissile media	Φ Arrangement (mm)	Arrangement thickness	Permitted weight of U-Pu (kg)
UO_2 - Unat	≤ 120	≥ 2 mm	40
	≤ 130	≥ 5 mm	40
UO_2 – $^{235}U/U \leq 100\%$	≤ 120	≥ 2 mm	20
	≤ 130	≥ 5 mm	18

Attachment 3.6-10 reveals that if the uranium is natural uranium and if:

- the PuO_2 is 100% isotopic vector ^{239}Pu,
- the Pu/U+Pu ratio is fixed at 10%,
- the **density** of the fissile medium is **random**.
- **polyethylene is present**,

then the number of packages relating to the permitted weight of U,Pu is as listed in the table below.

Permitted number of packages	Φ Arrangement (mm)	Arrangement thickness	Permitted weight of Pu (kg)
10	≤ 120	≥ 2 mm	6,6

The special case of content 3b is dealt with in attachment 3.6-12.

4.5 CONTENT 4: METALLIC URANIUM INGOTS

Document EMB TNBGC PBC DJS CA 0000357 provided in Appendix to Chapter 9 of the safety dossier EMB TNBGC PBC DS- CA 000001 B of 20/08/03 states that the criteria defined in § 3.1 are complied with when, for N=50 packages, the TN-BGC 1 is loaded with nine 5-kg uranium ingots of random density and enrichment packed with E4 wedges ensuring a minimum distance of 90 mm between ingots.

4.6 CONTENT 5: COMPACT STACKS OF ZEBRA PLATES

The document provided in Appendix 4 to Chapter 9 of the safety dossier EMB TNBGC PBC DS- CA 01000001 B of 20/08/03 states that the criteria defined in § 3.1 are complied with when the TN-BGC 1 is loaded with seven compact stacks of ZEBRA cladding plates, 100 mm high or less and packed with wedges E5 ensuring a minimum distance of 90 mm between ingots. N = 50 packages. The plutonium is of random density and isotopic composition such as

$^{240}Pu \geq 15\%$.

4.7 CONTENT 6: URANYL NITRATE

A minimum margin of 3000 pcm over the criteria defined in § 3.1 is considered for this content, to take into account the little experience in the isotopic compositions studied associated with steel and aluminium alloy reflectors.

The calculations presented in attachment 3.6-7, which taken into account local deformations of the internal arrangement following the drop test, consider a fissile material ($UO_2(NO_3)_2$) in homogenous form moderated by a random quantity of water. They reveal the compliance with criteria for a load of a random weight of uranium contained in a solution of uranyl nitrated enriched 95% with ^{235}U and contained in a cylinder with a maximum diameter of 120 mm and a minimum thickness of 2 mm.

4.8 CONTENT 7: UO2 URANIUM OXIDE AS PELLETS, ROD SECTIONS OR RODS

The results presented in § 4.3 cover the potential configurations for this content.

4.9 CONTENT 8: MIXED URANIUM-PLUTONIUM OXIDE AS PELLETS, ROD SECTIONS OR RODS

A minimum margin of 3000 pcm over the criteria defined in § 3.1 is considered for this content, to take into account the little experience in only slightly moderated plutonium-bearing media.

The calculations presented in attachments 3.6-2 and 3.6-3 consider a **heterogeneous mix of water** and UO_2+PuO_2.

The PuO_2 is 95% isotopic vector ^{239}Pu and 5% ^{240}Pu.

The Pu/U+Pu ratio is fixed at 30%.

The density of the fissile medium is random.

If the ^{235}U enrichment is random, the number of packages associated with the permitted weight of U,Pu is listed in the table below.

Permitted number of packages	Φ Arrangement (mm)	Arrangement thickness	Permitted weight of U+Pu (kg)
25	≤ 120	≥ 2 mm	15,6
25	≤ 130	≥ 5 mm	12,2
7	≤ 120	≥ 2 mm	20
7	≤ 130	≥ 5 mm	18

If the uranium is natural or depleted uranium for N=25, the maximum permitted weight regardless of the type of arrangement is 40 kg.

Attachment 3.6-10 reveals that if the uranium is natural or depleted uranium and if:

- the PuO_2 is 100% isotopic vector ^{239}Pu,
- the Pu/U+Pu ratio is fixed at 10%,
- the density of the fissile medium is random,

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- **polyethylene is present**,

then the number of packages relating to the permitted weight of U,Pu is as listed in the table below.

Permitted number of packages	Φ Arrangement (mm)	Arrangement thickness	Permitted weight of U+Pu (kg)
10	≤ 120	≥ 2 mm	6,6

4.10 CONTENT 9: HETEROGENEOUS PLUTONIUM OXIDE MIX

When the Pu isotopic composition is 100% ^{239}Pu, the envelope fissile form is the homogeneous medium. Otherwise the heterogeneous form is envelope.

The calculations presented in attachment 3.6-10 consider a fissile material of random density:

- in homogeneous form for the PuO_2 with 100% ^{239}Pu,
- in heterogeneous form for the PuO_2 with 5% ^{240}Pu,

and moderated by a **variable amount of polyethylene**.

They therefore cover the case of a heterogeneous mix.

The results are listed below:

- for the PuO_2 with 100% ^{239}Pu and for N=1 package, the permitted weight of Pu is:

Fissile media	Φ Arrangement (mm)	Arrangement thickness	Permitted weight of Pu (kg)
PuO_2 - 100% ^{239}Pu	≤ 120	≥ 2 mm	4

- for the PuO_2 with 5% ^{240}Pu and for N=10 packages, the permitted weight of Pu is:

Fissile media	Φ Arrangement (mm)	Arrangement thickness	Permitted weight of Pu (kg)
PuO_2 - ^{240}Pu ≥ 5%	≤ 120	≥ 2 mm	5

4.11 CONTENT 10: HETEROGENEOUS BLEND OF MIXED URANIUM AND PLUTONIUM OXIDE

The results presented in § 4.9 cover the potential configurations for this content.

4.12 CONTENT 11: SOLID URANIUM-BEARING MATERIALS

4.12.1 Non-air transport

The calculations presented in attachment 3.6-1 which involve transport other than by air consider a fissile material (U metal), of random density, in homogeneous form and moderated by a variable amount of water. N = 50. The isotopic composition is unique.

The results are listed below:

Fissile media	Φ Arrangement (mm)	Arrangement thickness	Permitted weight of U (kg)
UO_2 – $^{235}U/U \leq 100$ %	≤ 120	≥ 2 mm	7

Attachment 3.6-8 states that for the U metal with unique isotopic composition and random density moderated by a variable amount of water and packed in an internal arrangement with an internal diameter of less than 100 mm and a minimum thickness of 2 mm.

The results are listed below:

Fissile media	Φ Arrangement (mm)	Arrangement thickness	Permitted weight of U (kg)
UO_2 – $^{235}U/U \leq 100$ %	Φ ≤ 100	≥ 2 mm	15

Note: the limitations are as follows when it is impossible to guarantee a unique isotopic composition for each package.

Fissile media	Permitted weight of U (kg)
UO_2 > 20 % ^{235U}	7
UO_2 ≤ 20% ^{235U}	40

4.12.2 Air transport

Attachment 3.6-8 reveals compliance with criteria defined in § 3.1 for the transport by air of 7 kg of metallic uranium with random enrichment and unique isotopic composition; this takes into account the carbonated elements in the wood in the moderation of the fissile medium.

4.13 CONTENT 15: SPECIAL FORMS OF RADIOACTIVE SOURCES

This content is covered for criticality-safety by content 20 studied in § 4.16.

4.14 CONTENT 18: PLUTONIUM FLUORIDE

A minimum margin of 3000 pcm over the criteria defined in § 3.1 is considered for this content, to take into account the little experience in only slightly moderated plutonium-bearing media.

It is acknowledged that the PuO_2, PuOF and PuF_4 behaviour in equivalent fashion in terms of criticality-safety.

The results presented in § 4.2 therefore apply to this content.

They are listed below.

In the **absence of materials with greater hydrogen content than water**:

- for N=25 package, the permitted weight of Pu is:

Fissile media	Φ Arrangement (mm)	Arrangement thickness	Permitted weight of Pu (kg)
PuO_2 - 100% ^{239}Pu	≤ 120	≥ 2 mm	7
PuO_2 - $^{240}Pu \geq 5\%$	≤ 120	≥ 2 mm	13,5
	≤ 130	≥ 5 mm	10

- the search for the permitted number of packages leads to:

Fissile medium	Φ Arrangement (mm)	Weight of Pu per package (kg)	Arrangement thickness	Permitted number of packages
PuO_2 - $^{240}Pu \geq 5\%$	≤ 120	17	≥ 2 mm	16
	≤ 130	13	≥ 5 mm	9

In the **presence of materials with greater hydrogen content than water**:

- for the PuO_2 with 100% ^{239}Pu and for N=1 package, the permitted weight of Pu is:

Fissile media	Φ Arrangement (mm)	Arrangement thickness	Permitted weight of Pu (kg)
PuO_2 - 100% ^{239}Pu	≤ 120	≥ 2 mm	4

- for the PuO_2 with 5% ^{240}Pu and for N=10 packages, the permitted weight of Pu is:

Fissile media	Φ Arrangement (mm)	Arrangement thickness	Permitted weight of Pu (kg)
PuO_2 - $^{240}Pu \geq 5\%$	≤ 120	≥ 2 mm	5

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4.15 CONTENT 19: MIXED URANIUM-PLUTONIUM NITRIDE AS A NON-POWDERED SOLID

This content is covered for criticality-safety by content 20 studied in § 4.16.

4.16 CONTENT 20: PLUTONIUM IN METALLIC, NON-POWDERED FORM

A minimum margin of 3000 pcm over the criteria defined in § 3.1 is considered for this content, to take into account the little experience in only slightly moderated plutonium-bearing media.

The calculations presented in attachments 3.6-1 and 3.6-4 consider for content 20 a plutonium-bearing medium with random density and isotopic composition moderated homogeneously by a variable amount of water. They reveal that the permitted weight of Pu is 4 kg for N = 10 in an internal arrangement with an internal diameter ≤ 120 mm and thickness ≥2 mm.

4.17 CONTENT 23: PLUTONIUM, URANIUM, NEPTUNIUM OR AMERICIUM OXIDE POWDER OR A MIX OF THESE POWDERS

This content is covered in terms of criticality-safety by the content formed of 7 kg of ^{239}Pu with random isotopic composition studied in § 4.2.

4.18 CONTENT 26: TRIGA FUEL

The calculations presented in attachment 3.6-11 supplemented by the calculations presented in attachment 3.6-9 reveal compliance with criteria defined in § 3.1, whether or not transported by air, for the fine or standard TRIGA elements in the weight boundaries defined in Chapter 1, packed in an internal arrangement of internal diameter ≤ 120 mm and thickness ≥2 mm and moderated by a random amount of water.

As for content 11, in air transport, the carbonated elements in the wood in the packaging are taken into account in the moderation of the fissile material.

Note also that attachment 3.6-9 proves that modelling aluminium alloy spaces leads to a drop in reactivity.

PROGRAMME: REPLACEMENT OF CEA PACKAGING

TITLE:

PACKAGING TN-BGC 1

Additional criticality calculations

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Summary:
These additional calculations are in response to an IRSN opinion on taking aluminium fillers into account in the criticality studies.
These studies determine the new permitted weights for certain TN-BGC 1 contents where taking aluminium fillers into account proves penalising.

Visa			
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E	M	B	T	N	B	G	C	P	B	C	N	T	T	C	A	0	0	0	4	8	1	A
1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23

INDEX OF CHANGES

EDITION	DATE	Type of change	Pages modified
A		Issue of document	

CONTENTS

REFERENCES

Reference documents are as follows:

[1] SSTR opinion of 10/04/03, reference SSTR/03.335

[2] IAEA Safety standards collections - Regulations for the safe transport of radioactive materials TS-R-1 - 1996 edition (revised).

[3] Extension of the qualification base for the CRISTAL form: Metallic experiments ref. SEC/T/03.013.

[4] TN-BGC 1 packaging safety analysis report ref. EMB TNBGC PBC DS CA000001 rev. B

1. SCOPE

TN-BGC 1 is type BF packaging normally used to transport non-irradiated materials, the most frequently fissile. Each content has been the subject of specific criticality studies. However, looking towards a future agreement, additional studies are required or some studies must be reviewed. Following a comment from the safety authority [1], the AUG4 fillers had not been modelled and had been replaced by water. Taking them into account could result in an increased Keff. The purpose of this study is to measure the impact of taking these fillers into account in the weights of transported fissile material and to demonstrate the lack of criticality risk for the TN-BGC 1 packaging in conformity with the regulatory stipulations for the safe transport of radioactive materials on the public highway from the IAEA [2].

2. CALCULATION MEANS

The calculations are made using the APOLLO2-MORET IV calculation method in the CRISTAL V0 criticality form (standard method), used according to the recommended scheme.

The APOLLO2 code (version 2.4.3) used with the CEA93-V4 library of cross sections calculates the neutron characteristics ($k\infty$, B^2_m and the slowdown factor Qr) of fissile materials and generates for all media (whether or not fissile) macroscopic cross sections with 172 energy groups for use by the MORET code.

The APOLLO2 calculations sheets are generated using the CIGALES code (version 2.0).

The MORET IV code (version 4.A.4) calculates the effective coefficient of multiplication of neutrons (k_{eff}) in a three-dimensional geometry. It uses macroscopic cross sections with 172 energy groups created by the APOLLO2 code for all the media (fissile or otherwise) used in the modelling.

3. BASIC DATA

3.1 Description of the packaging

The packaging consists of a parallelepiped cage inside which an overall cylindrical body equipped with a closure system and a cover is fixed.

The packaging dimensions are as follows:

- cage cross-section: 600x600 mm,
- overall height of the cage: 1,821 mm
- diameter of main cover part: 295 mm
- cover diameter: 466 mm
- overall body length equipped with the cover: 1,808 mm.

3.1.1 Cage

The cage is a tubular aluminium structure of 30x30 mm and a thickness of 2 mm.

3.1.2 Body

The cavity with useful diameter 181 mm and useful length 1475 mm is formed by a 6 mm-thick stainless steel shell (providing the majority of radial gamma shielding) and an 8 mm-thick bottom also in stainless steel.

The space between this shell and a second stainless steel shell 1.5 mm thick and with an inside diameter of 292 mm is filled with resin loaded with hydrogen and boron (minimum thickness 48 mm) which acts as a neutron absorber and active heat insulation.

The bottom is supplemented, from the inside outwards, by a resin diffuser plate in 25 mm steel with high elastic limit, a 24 mm layer of resin, a false bottom, a wooden shock-absorbing disc and a stainless steel sheet.

In the upper part, a machined stainless steel flange is welded to the two shells to accommodate the closing system.

The plug is machined from a stainless steel disc 92 mm thick. It has a 20 mm shoulder on its edge which is supported by the body flange. This plug is held in place by a bronze tightening ring which screws into the stainless steel bayonet ring.

The plug shoulder is fitted with two machined concentric trapezoidal grooves. Two O-rings inserted into these grooves maintain the leaktightness of the plug in the body.

3.1.3 Cover

A shock-absorbing cover is placed over the top of the body and the closing system.

It is formed of two steel plate compartments; the one closest to the body is filled with resin and the other with wood.

3.2 Description of contents

3.2.1 Description of internal arrangements

The packaging is loaded respectively with an internal packing container in stainless steel loaded with the radioactive material and internal packing devices. Different types of packing container exist:

- thin stainless steel walls 2 mm thick and with an inside diameter of 120 mm (TN 90, AA 204, AA 203, AA41),
- thick stainless steel walls 5 mm thick and with an inside diameter of 130 mm (AA226, AA227).

The following fillers are used to filler the packing container in the packaging cavity (AUG4 aluminium fillers):

- With TN 90: filler E1 + filler E2,
- With AA 203: filler E1 + filler E8,
- With AA 204: filler E1 + filler E10,
- With 1 AA 41: filler E1 + filler E11,
- With 2 AA 41: filler E1 + filler E12 + filler E13,
- With 3 AA 41: filler E1 + filler E12 + 2 fillers E13.

The diagrams summarising these different packings are given in the safety analysis report in ref. [4].

3.2.2 Description of the fissile materials to be studied

3.2.2.1 Plutonium oxide as a powder

This content has the following properties:

- Isotopic composition: random,
- Maximum density: 3.5.

3.2.2.2 (U, Pu)O_2 in random form (powder, pellets, rods)

This content has the following properties:

- Isotopic composition: random,
- Plutonium content such that Pu/U+Pu < 30%
- Random density.

3.2.2.3 Uranium oxide in random form (powder, pellets, rods)

This content has the following properties:

- Isotopic composition: random,
- Random density.

3.2.2.4 Uranium metal in random form

Isotopic composition: random.

3.2.2.5 Plutonium metal in random form

Isotopic composition: random.

3.2.3 Description of structural media

The structural media are the following:

- Stainless steel with density 7.9:

Chemical element	Atomic concentrations (10^{24} at/cm^3)
Fe	$6.1341.10^{-2}$
Ni	$1.6467.10^{-2}$
Cr	$8.1070.10^{-3}$

- Water of density 1,

- Air,

- Neutron-absorbing resin of density 1.186:

Chemical element	Atomic concentrations (10^{24} at/cm^3)
H	$4.0616.10^{-2}$
C	$2.3803.10^{-2}$
O	$2.3580.10^{-2}$
Bnat	$9.4597.10^{-4}$

- Aluminium AUG4 of density 2.67:

Chemical element	Proportions by mass (%)
Si	0.5
Fe	0.7
Cu	4
Mn	0.7
Mg	0.7
Cr	0.1
Zn	0.25
Al	93.05

4. CALCULATION HYPOTHESES

4.1 Configurations studied

The different configurations studied are:

- isolated package in ACT,

- 5N package array in ACT.

This latter configuration covers the two types of package stacking provided for in the regulations, namely:

- 5N package in NTC,
- 2N package in ACT.

The approach taken by this study is to fix as input data a number N of packages representative of plausible transport scenarios and to determine the maximum permitted weight per packaging for each content. The package array changes according to this number. The following table summarised the configurations studied:

Contents	N	Array (XxYxZ)
PuO_2	25	8x8x2
UO_2-PuO_2	25	8x8x2
UO_2	25	8x8x2
U metal	50	12x12x2
Pu metal	10	7x8x1

These stacking configurations relate to the most conservative configurations found in the previous studies of the TN-BGC 1 (see [4]).

For stacking two packagings according to Z, the fissile material will be modelled head-to-tail. For a row at height it will be centred. The isolated cases will take up the same fissile material positions. These are the most conservative configurations (see § 7.3).

The arrays studied will be triangular step.

4.2 Modelling the packaging

4.2.1 Modelling fillers

The fillers will be modelled in two different ways:

- 1st case: the fillers will not be modelled and replaced by water,
- 2nd case: the fillers are modelled with AUG4 aluminium and occupy all the free space left by the container in the internal packaging cavity.

Through inappropriate language, we shall subsequently name the case where the space between the container and the internal packaging shell is filled with water "water filler".

4.2.2 Geometric modelling

The cage and the covers will not be modelled for this packaging.

The upper and lower packaging parts will be modelled by 33 mm of steel and 24 mm of resin as in the studies included in reference [4].

The cavity of useful diameter 181 mm and useful length 1475 mm is formed by a 6 mm-thick stainless steel shell.

A second stainless steel shell 1.5 mm thick and 292 mm inside diameter creates with the first shell a space filled with resin loaded with hydrogen and boron (minimum thickness 48 mm).

In TAC, the resin will be burned over a thickness of 15 mm (conservative hypothesis).

4.2.3 Modelling internal arrangements

The steel container containing the fissile material will be modelled and will have a variable diameter and thickness depending on the contents being transported.

The two types of container studied are:

- Internal diameter 130 mm, thickness 5 mm (transport of PuO_2 and Pu metal)
- Internal diameter 120 mm, thickness 2 mm (transport of all the contents studied).

In addition, this second container could be deformed in ACT as in the previous study of the TN-BGC 1 (see Figure 4). This corresponds to cases where the type E3 fillers could be used.

There are four deformations. They involve increasing the internal diameter of the container from 120 to 130 mm over a thickness of 3 cm.

These deformations are located every 32.6 cm from the base on the containment.

The following table summarises the types of container used in the calculations:

Contents	Diameter/thickness in mm	Deformation of the container
PuO_2	130/5 or 120/2	No
UO_2-PuO_2	130/5 or 120/2	No
UO_2	120/2	Yes
U metal	120/2	Yes
Pu metal	130/5	No

4.2.4 *Reflection condition*

All parts of the isolated package will be reflected through 20 cm of water.

The 2N and 5N stacking configurations will be covered in ACT by the 5N package configuration. In addition, as for the isolated package, this stack will be reflected through 20 cm of water. The space between the packages will be modelled by air so as to encourage the coupling between the packages as in the previous studies [4].

Figures 2, 3 and 4 show sample geometric models used (for the isolated package and in stacking).

4.3 Modelling the contents

4.3.1 *Reactivity control methods*

The control method adopted corresponds to restricting the geometry associated with restricting the fissile weight and with the presence of neutron-absorbing resin.

The moderation of the fissile material is random in the volume offered by the internal packing container. The fissile material is modelled by a cylinder of equal diameter to the internal diameter of the container (120 or 130 mm), deformed or otherwise and of variable height depending on the moderation.

4.3.2 *Modelling the fuel*

4.3.2.1 Plutonium oxide as a powder

This content will be modelled by a homogeneous mix of water and PuO_2 of maximum density 3.5.

Two isotopic vectors are studied:

- 95% ^{239}Pu and 5% ^{240}Pu,
- 100% ^{239}Pu.

4.3.2.2 Mixed uranium and plutonium oxide in random form (powder, pellets, rods)

This content will be modelled by a homogeneous mix of water and UO_2+PuO_2.

Two types of MOX contents will be studied:

- natural UO_2 (enriched with 0.71% ^{235}U) combined with the PuO_2 of isotopic vector 95% ^{239}Pu and 5% ^{240}Pu,
- $^{235}UO_2$ combined with the PuO_2 of isotopic vector 95% ^{239}Pu and 5% ^{240}Pu,

The Pu/U+Pu ratio will be fixed at 30% in both cases.

The density is random.

4.3.2.3 Uranium oxide as a powder

This content will be modelled by a homogeneous mix of water and UO_2.

Two isotopic vectors are studied:

- 20% ^{235}U,
- 100% ^{235}U.

The density is random.

4.3.2.4 Uranium metal in random form

This content will be modelled by a homogeneous mix of water and uranium in metallic form.

Two isotopic vectors are studied:

- 20% ^{235}U,
- 100% ^{235}U.

4.3.2.5 Plutonium metal in random form

This content will be modelled by a homogeneous mix of water and Pu metal.

The isotopic vector of the plutonium will be 100% ^{239}Pu.

4.4 Admissibility criterion

The admissibility criterion, all uncertainties included (uncertainties linked to the method and its qualification) adopted for this study is:

- $k_{eff} \leq 0.95$, for the isolated case
- $k_{eff} \leq 0.98$, for the stacking case

The value of σ is taken as equal to 200 pcm for the method-related uncertainties.

5. RESULTS

5.1 PuO_2 content

Two types of container have been studied for this content. The steel container is modelled with an internal diameter of 130 mm and a thickness of 5 mm or with an internal diameter of 120 mm and a thickness of 2 mm.

5.1.1 Diameter 130 mm/thickness 5 mm

5.1.1.1 5N stacking (N=25)

The two types of filler (water or aluminium) are simulated for this calculation. The table below therefore presents the results obtained in both cases for a weight of 13 kg of Pu in the form of PuO_2:

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
Hmin	0.97470	0.95435
35	0.96825	0.95840
40	0.95915	0.93637
45	0.95313	0.93083
70	0.93286	0.91017
Hmax	0.90478	0.88463

5.1.1.2 Isolated Package

The two types of filler (water or aluminium) are simulated for this calculation. The following results are obtained for a weight of 13 kg of Pu in the PuO_2 form:

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
Hmin	0.84311	0.86439
35	0.83265	0.86064
40	0.82255	0.84732
45	0.81549	0.83857
70	0.78301	0.81337
Hmax	0.75289	0.78277

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5.1.2 *Diameter 120 mm/thickness 2 mm*

5.1.2.1 5N stacking (N=25)

The two types of filler (water or aluminium) are simulated for this calculation. The table below therefore presents the results obtained in both cases for a weight of 17 kg of Pu in the form of PuO_2:

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
Hmin	0.94336	0.93413
50	0.96816	0.95235
55	0.96000	0.94253
60	0.94121	0.93681
70	0.92732	0.92022
100	0.89891	0.89793
120	0.88473	0.88137
Hmax	0.87454	0.87580

5.1.2.2 Isolated Package

The two types of filler (water or aluminium) are simulated for this calculation. The following results are obtained for a weight of 17 kg of Pu in the PuO_2 form:

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
Hmin	0.79885	0.84939
50	0.81625	0.86368
55	0.80362	0.85266
60	0.79520	0.84538
70	0.77753	0.82703
100	0.74713	0.80590
120	0.72601	0.79121
Hmax	0.71577	0.77940

5.1.3 Diameter 120 mm/thickness 2 mm for plutonium 100% enriched with ^{239}Pu

5.1.3.1 5N stacking (N=25)

The two types of filler (water or aluminium) are simulated for this calculation. The table below therefore presents the results obtained in both cases for a weight of 5 kg of Pu in the form of PuO_2:

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
Hmin	0.83318	0.85149
30	0.88961	0.89934
50	0.90918	0.91422
70	0.90779	0.91605
100	0.90847	0.91427
120	0.90820	0.91380
Hmax	0.89982	0.90156

5.1.3.2 Isolated Package

The two types of filler (water or aluminium) are simulated for this calculation. The following results are obtained for a weight of 5 kg of Pu in the PuO_2 form:

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
Hmin	0.71849	0.78197
30	0.75197	0.82377
50	0.76132	0.83036
70	0.75828	0.82612
100	0.74406	0.82073
120	0.74656	0.81722
Hmax	0.73700	0.81083

5.1.4 Results

For the PuO_2 content, the maximum permitted weights obtained are as follows:

Contents	Type:	N	Filler type	Weight
PuO2	D130/E5 (5% ^{240}Pu)	25	Alu	13 kg
	D120/E2 (5% ^{240}Pu)	25	Alu	17 kg
	D120/E2 (100% ^{239}Pu)	25	Water - Alu (equivalent)	5 kg

The water fillers are always the most penalising for the isolated case. The fillers considered like water reflect the neutrons better towards the fissile material and screen the neutron absorption.

For the stacks, the aluminium fillers encourage the interactions between packages (the boron only captures thermalised neutrons). The aluminium fillers therefore become more penalising for the case with 95% ^{239}Pu.

The aluminium and water fillers are equivalent in this case.

5.2 Mixed UO_2/PuO_2 content

Two types of media "moxified" with 30% plutonium will be studied for this content. The difference between these two media will be the isotopic vector of the uranium. In the first case it will be considered as 100% ^{235}U, and in the second as natural uranium with 0.71% of ^{235}U.

5.2.1 Mixed oxide with natural uranium

5.2.1.1 *Container diameter of 130 mm (5N stacking (N=25))*

The two types of filler (water or aluminium) are simulated for this calculation. The table below therefore presents the results obtained in both cases for a weight of 40 kg of U+Pu as an oxide:

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
50	0.74300	0.74909
70	0.75302	0.75289
120	0.76511	0.76375
140	0.76864	0.76847
Hmax (40 kg)	0.77826	0.77376
Hmax (30 kg)	0.78137	0.78063
Hmax (20 kg)	0.79573	0.78636
Hmax (15 kg)	0.79827	0.78894
Hmax (10 kg)	0.79491	0.78751

5.2.1.2 *Container diameter of 130 mm (isolated package)*

The two types of filler (water or aluminium) are simulated for this calculation. The following results are obtained for a weight of 40 kg of U+Pu as an oxide:

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
50	0.61154	0.65832
70	0.62668	0.66644
120	0.62710	0.67181
140	0.63632	0.67361
Hmax (40 kg)	0.64179	0.67936
Hmax (30 kg)	0.64728	0.68512
Hmax (20 kg)	0.65636	0.69416
Hmax (15 kg)	0.66692	0.70125
Hmax (10 kg)	0.66436	0.70037

5.2.1.3 Container diameter of 120 mm

The weight of 40 kg obtained for a 130 mm diameter is considered sufficient for the transport needs of the TN-BGC 1.

To check that this study (internal arrangement diameter of 130 mm and thickness of 5 mm) covers the use of a container 120 mm in diameter and 2 mm thick, two calculations were performed for these two types of container with a 40 kg load, cell entirely filled, in an array, with aluminium fillers.

The results are given below. They show the envelope nature of the study with the 130 mm-diameter container.

Container	D130/E5	D120/E2
Keff+3σ	0.77826	0.72740

5.2.2 *Mixed oxide with uranium 100% enriched with ^{235}U*

5.2.2.1 Container diameter of 130 mm (5N stacking (N=25))

The two types of filler (water or aluminium) are simulated for this calculation. The table below therefore presents the results obtained in both cases for a weight of 18 kg of U+Pu as an oxide:

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
Hmin	0.95022	0.93784
20	0.95673	0.93793
30	0.96951	0.94692
40	0.96946	0.94085
50	0.96670	0.94632
60	0.96297	0.93647
70	0.9624	0.93519
100	0.96323	0.93329
120	0.95168	0.92918
Hmax	0.94826	0.92603

5.2.2.2 Container diameter of 130 mm (isolated package)

The two types of filler (water or aluminium) are simulated for this calculation. The following results are obtained for a weight of 18 kg of U+Pu as an oxide:

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
Hmin	0.83525	0.85439
20	0.83455	0.85374
30	0.84507	0.85948
40	0.82683	0.85737
50	0.82249	0.85156
60	0.81547	0.84184
700	0.80872	0.84224
100	0.80516	0.83365
120	0.79715	0.82748
Hmax	0.79359	0.82128

5.2.2.3 Container diameter of 120 mm (5N stacking (N=25))

The two types of filler (water or aluminium) are simulated for this calculation. The table below therefore presents the results obtained in both cases for a weight of 20 kg of U+Pu as an oxide:

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
Hmin	0.96985	0.95586
20	0.96363	0.95027
30	0.95465	0.94463
40	0.95142	0.93806
50	0.94616	0.91686
60	0.93937	0.91468
70	0.91057	0.90918
100	0.90485	0.90392
120	0.89950	0.89975
Hmax	0.89667	0.89940

5.2.2.4 Container diameter of 120 mm (isolated package)

The two types of filler (water or aluminium) are simulated for this calculation. The following results are obtained for a weight of 20 kg of U+Pu as an oxide:

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
Hmin	0.84290	0.87555
20	0.83379	0.87103
30	0.81555	0.85877
40	0.80606	0.85288
50	0.79612	0.84369
60	0.78658	0.83666
70	0.77615	0.83632
100	0.75656	0.81946
120	0.74860	0.81462
Hmax	0.74430	0.80428

5.2.3 *Results*

For the mixed oxide content, the maximum permitted weights obtained are as follows:

Contents	Type:	N	Filler type	Weight
PuO2+UO2	Natural uranium	25	Alu	40 kg
	^{235}U (D130/E5)	25	Alu	18 kg
	^{235}U (D120/E2)	25	Alu	20 kg

The water fillers are always the most penalising for the isolated case. The water fillers reflect the neutrons better towards the fissile material.

For the package stacks, the aluminium fillers encourage the interactions between packages (the boron only captures thermalised neutrons) and are therefore more conservative.

5.3 UO2 content

For this content, the UO_2 will be studied in powder form with miscellaneous enrichments (20 and 100% ^{235}U) with deformations of the containment caused by the E3 fillers.

5.3.1 UO_2 100% enriched with ^{235}U

5.3.1.1 5N stacking (N=25)

The two types of filler (water or aluminium) are simulated for this calculation. The table below therefore presents the results obtained in both cases for a weight of 20 kg of U as an oxide:

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
N2 (51.9)	0.95501	0.94445
N3 (68.2)	0.96604	0.95991
N4 (84.5)	0.96952	0.96234
N5 (100.8)	0.97432	0.96696
120	0.96979	0.96578
140	0.97356	0.96192
Hmax (20 kg)	0.97723	0.96686
Hmax (18 kg)	0.96934	0.96880
Hmax (16 kg)	0.96670	0.96281

The Ni heights are defined in Figure 1.

5.3.1.2 Isolated Package

The two types of filler (water or aluminium) are simulated for this calculation. The following results are obtained for a weight of 20 kg of U as an oxide:

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
N2 (51.9)	0.78913	0.85237
N3 (68.2)	0.79562	0.86154
N4 (84.5)	0.79718	0.86338
N5 (100.8)	0.79841	0.86502
120	0.79885	0.86882
140	0.79640	0.86799
Hmax (20 kg)	0.79677	0.86708
Hmax (18 kg)	0.79187	0.86687
Hmax (16 kg)	0.79122	0.86437

5.3.2 UO_2 20% enriched with ^{235}U

5.3.2.1 5N stacking (N=25)

The effect of water replacing the aluminium fillers has been studied for this calculation. The table below therefore presents the results obtained in both cases for a weight of 40 kg of U as an oxide:

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
120	0.72893	0.77298
130	0.73790	0.77788
140	0.74461	0.78568
Hmax (40 kg)	0.74518	0.78763
Hmax (35 kg)	0.75422	0.79202
Hmax (30 kg)	0.75999	0.79575
Hmax (25 kg)	0.75672	0.79977
Hmax (20 kg)	Calculation not performed	0.79251

5.3.2.2 Isolated Package

The two types of filler (water or aluminium) are simulated for this calculation. The following results are obtained for a weight of 40 kg of U as an oxide:

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
120	0.58772	0.68397
130	0.59868	0.68904
140	0.59836	0.69455
Hmax (40 kg)	0.60559	0.69944
Hmax (35 kg)	0.61477	0.70543
Hmax (30 kg)	0.62537	0.70965
Hmax (25 kg)	0.62492	0.71075
Hmax (20 kg)	0.62232	0.70427

5.3.3 *Results*

For the UO_2 oxide content, the maximum permitted weights obtained are as follows:

Contents	Type:	N	Filler type	Weight
UO2	20% enr.	25	Water	40 kg
	100% enr.	25	Alu	20 kg

The water fillers are always the most penalising for the isolated case. The water fillers reflect the neutrons better towards the fissile material.

For the package stacks, the difference in favour of the water fillers is offset by the greater neutron coupling between the packages and lower absorption by the resin.

Cases with 20% enrichment with ^{235}U show more thermalised spectra than the 100% cases. The resin is therefore going to be more effective, for the aluminium fillers, with a 20% enrichment and will cause the Keff value to plummet. With 20% ^{235}U, the boron is just as effective with the water as with the aluminium.

The fillers replaced by the water therefore become more penalising.

For the 100% cases, the aluminium fillers are always the most penalising.

5.4 Uranium content in metallic form

For this content, the uranium is going to be studied in metallic form with 100% or 20% ^{235}U and with deformations of the containment caused by the fillers.

5.4.1 U metal 100% enriched with ^{235}U

5.4.1.1 5N stacking (N=50)

The two types of filler (water or aluminium) are simulated for this calculation. The table below therefore presents the results obtained in both cases for a weight of 7 kg and, as a rough guide, for a weight of 5.5 kg of U in metallic form:

Weight 7 kg

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
N1 (35.6)	0.91877	0.93306
N2 (51.9)	0.94449	0.95130
N3 (68.2)	0.94925	0.95523
N4 (84.5)	0.95255	0.95728
N5 (100.8)	0.94839	0.95409
120	0.94451	0.94923
140	0.93822	0.94313

Weight 5.5 kg

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
N1 (35.6)	0.90436	0.91951
N2 (51.9)	0.93000	0.94014
N3 (68.2)	0.94366	0.94876
N4 (84.5)	0.93594	0.94373
N5 (100.8)	0.93292	0.94111
120	0.92608	0.93485
140	0.91669	0.92621

5.4.1.2 Isolated Package

The two types of filler (water or aluminium) are simulated for this calculation. The following results are obtained for a weight of 7 kg and, as a rough guide, for a weight of 5.5 kg of U in metallic form:

Weight 7 kg

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
N1 (35.6)	0.77643	0.84791
N2 (51.9)	0.79802	0.85832
N3 (68.2)	0.78987	0.86054
N4 (84.5)	0.78910	0.86136
N5 (100.8)	0.77770	0.85694
120	0.77828	0.85134
140	0.77067	0.84590

Weight 5.5 kg

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
N1 (35.6)	0.77029	0.83981
N2 (51.9)	0.78012	0.85088
N3 (68.2)	0.77886	0.85107
N4 (84.5)	0.77995	0.85071
N5 (100.8)	0.77495	0.84461
120	0.76629	0.83687
140	0.75854	0.83123

5.4.2 U metal 20% enriched with ^{235}U

5.4.2.1 5N stacking (N=50)

The two types of filler (water or aluminium) are simulated for this calculation. The table below therefore presents the results obtained in both cases for a weight of 40 kg of U in metallic form:

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
N3 (68.2)	0.77463	0.81008
N4 (84.5)	0.78778	0.8176
N5 (100.8)	0.79841	0.8273
120	0.80283	0.83202
140	0.80905	0.83465
Hmax (40 kg)	0.80939	0.83397
Hmax (35 kg)	0.80772	0.83332
Hmax (30 kg)	0.80258	0.82983

5.4.2.2 Isolated Package

The two types of filler (water or aluminium) are simulated for this calculation. The following results are obtained for a weight of 40 kg of U in metallic form:

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
N3 (68.2)	0.64387	0.72790
N4 (84.5)	0.65470	0.73560
N5 (100.8)	0.65686	0.74105
120	0.65736	0.74830
140	0.66157	0.74728
Hmax (40 kg)	0.66321	0.75279
Hmax (35 kg)	0.66605	0.74561
Hmax (30 kg)	0.66420	0.74705

5.4.3 Results

For the metallic uranium content, the maximum permitted weights obtained are as follows:

Contents	Type:	N	Filler type	Weight
U metal	100% enr.	50	Water	7 kg
	20% enr.	50	Water	40 kg

The water fillers are always the most penalising for the isolated case. The water fillers reflect the neutrons better towards the fissile material.

For the package stacks, note that the water fillers are the most penalising. In the cases dealt with previously, it was found that a low weight or a 20% enrichment gave a more thermalised spectrum.

5.5 Plutonium content in metallic form

5.5.1 5N stacking (N=10)

The two types of filler (water or aluminium) are simulated for this calculation. The table below therefore presents the results obtained in both cases for a weight of 4 kg of Pu in metallic form:

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
30	0.91277	0.92543
50	0.94702	0.94742
70	0.95356	0.95215
80	0.95486	0.95542
90	0.95191	0.95400
100	0.95137	0.95540
120	0.95127	0.95258
Hmax	0.94429	0.94293

5.5.2 *Isolated Package*

The two types of filler (water or aluminium) are simulated for this calculation. The following results are obtained for a weight of 4 kg of Pu in metallic form:

Fissile height	Keff+3σ (Aluminium filler)	Keff+3σ (Water filler)
Non-moderated sphere (radius 3.64 cm)*	0.91100	0.91557
30	0.79730	0.85080
50	0.80243	0.86024
70	0,80381	0.85990
80	0.79981	0.85899
90	0.79923	0.85503
100	0.79398	0.84939
120	0.78591	0.84570
Hmax	0.78061	0.83847

** in accordance with the study figuring in Chapter 5 of <2>*

5.5.3 *Results*

For the metallic plutonium content, the maximum permitted weights obtained are as follows:

Contents	Type:	N	Filler type	Weight
Pu met.	100% 239Pu	10	Water - Alu (equivalent)	4 kg

The water fillers are always the most penalising for the isolated case. The water fillers reflect the neutrons better towards the fissile material.

The two types of filler are equivalent for the package stacks.

6. ADDITIONAL STUDIES

6.1 Effect of steel fillers

Following the calculations by IRSN (see [1]) with steel fillers, several cases were tested for the effect of these steel fillers compared with water or aluminium modelling.

The cases studied are:

- 4 kg of Pu metal with a 130 mm-diameter container in 8x7x1 array,
- 40 kg of UO_2 100% enriched with ^{235}U with a non-deformed 120 mm-diameter container in 8x8x2 array,
- 16 kg of PuO_2 with a 130 mm container in 8x2x2 array,
- 7 kg of U metal in a 130 mm-diameter container in 8x8x2 array.

These measurements showed that:

Contents	Water filler	Steel filler	Alu. filler
Pu metal	0.95592	0.98147	0.95486
UO_2	0.97784	0.99577	0.99481
PuO_2	0.98495	1.01115	1.0040
U metal	1.00022	1.02528	0.99583

The steel fillers are the most penalising. They reflect the neutrons clearly towards the fissile material and do not slow down the neutrons to encourage the capture of neutrons by the resin. Note, however, that for the heavy weights (UO_2 and PuO_2) where the fissile material is not as well moderated, the aluminium and steel fillers produce similar Keff values.

6.2 Positioning the fissile medium in the cell

The position of the fissile material was defined in § 7.1.

For the PuO_2, MOX and Pu metal contents, various calculations have been performed with the fissile material placed in the middle for the array (Pu metal) and isolated cases. The results are given below. These cases correspond to the maximum Keff obtained in the study.

Contents	Centred	Off-centre
PuO_2 (isolated)	0.84304	0.84311
Mox (isolated)	0.83593	0.84057
Pu metal (isolated)	0.86024	0.85428
Pu metal (array)	0.95592	0.95452

Off-centering the fissile material has no significant effect on package reactivity.

For the cases of arrays stacked on two heights, the fissile material has been position head-to-tail so as to maximise the interactions between the packages.

6.3 Study on filler modelling

The space between the container and the internal space of the packaging was previously totally filled with either water or aluminium. In the following study, a check is made to confirm that air between the internal arrangement shell and the aluminium filler is not likely to raise doubts over the previous results. The presence of air encourages the interactions between packages.

To achieve this, the fillers are replaced by air in the first case and in the second, it was assumed that a fictitious filler occupied half the space left free by the container (half the space is left free and filled with air). These calculations were performed on a stack of packages for 20 kg of UO2 100% enriched with ^{235}U, with the cell totally filled with fissile material.

Filler type	Keff+3σ
Filler not modelled and replaced by air	0.92583
Intermediate case: (Air + Aluminium)	0.92947
Filler modelled by the aluminium	0.97723

For air-aluminium case, the most penalising case is obtained for an all-aluminium configuration.

6.4 Justification for choosing the homogeneous medium

The fissile material is assumed homogeneous for all the TN-BGC 1 calculations. However, this choice can be open to debate for the MOX medium in natural uranium. To justify this, we have carried out in this case (40 kg of mixed oxide with natural uranium in a 130 mm-diameter container, in package stack) a study on taking a heterogeneous fuel medium into account.

For this case, the fissile medium was represented as a heterogeneous medium made up of spherical particles of variable diameter surrounded by water.

The results are presented in the table below. They have been obtained at optimum moderation, for a package stack, with aluminium fillers (checks were made beforehand that this case covered the case of the filler modelled in water).

Weight of (U,Pu)O2	Spherical radius (producing the maximum reactivity)	Keff + 3 σ
40 kg	0.4	0.82788
35 kg	0.4	0.82635
30 kg	0.4	0.82244

The results for the isolated package are given in the table below. This case corresponds to the case where the filler is replaced by water. It had been checked that this case is the most conservative one:

Weight of (U,Pu)O2	Spherical radius (producing the maximum reactivity)	Keff + 3 σ
40 kg	0.2	0.71340
35 kg	0.2	0.71586
30 kg	0.2	0.71571
25 kg	0.2	0.72045
20 kg	0.2	0.72309

15 kg	0.1	0.71855

Note a gain in reactivity in the order of 3000 pcm for the package array and 2000 pcm for the isolated package compared with the results obtained for a homogeneous medium. Given the margins noted compared with the normal admissibility criteria, these results raise no doubts over the weights determined previously.

7. CRISTAL QUALIFICATION STATE FOR THE MEDIA STUDIED

7.1 Oxide content

For this type of content, given the lack of qualification for the oxide powder fissile media extensively enriched with ^{235}U or ^{239}Pu, using the APOLLO2-MORET4 calculation scheme could under-estimate the Keff. It is therefore wise to factor an additional margin into the calculations performed.

7.1.1 Oxide content (UO_2/PuO_2)

For an isolated package, there is a fairly large margin in terms of the acceptability criterion adopted to ensure the criticality-safety of the package.

For package stacks, 5N array calculations were performed in ACT, which constitutes a conservative configuration with respect to the requirements of the regulations (2N in ACT and 5N in NTC). Calculations in 2N in ACT and 5N in NTC were performed for the most penalising cases of each content.

The 2N array will be modelled by an 8x7x1 array for the ACT modelling.

For the NTC modelling, the cages whose resistance under NTC has been demonstrated in <4> will be modelled and the resin will be considered an integral part.

Aluminium fillers - the most penalising - will be used in both cases.

These measurements showed that:

Contents	5N in NTC	2N in ACT
UO_2	0.84418	0.96476
PuO_2	0.87935	0.95539

Although the qualification basis for the CRISTAL calculation scheme is insufficient, margins do exist with respect to the criteria in the light of the penalisation injected by the 5N array study in ACT .

7.1.2 Mixed UO_2/PuO_2 oxide content

Given the Keff values obtained in the order of 0.97 in package array and 0.85 in isolated package for the most penalising cases and the envelope nature of hypotheses for the modelling (homogeneous medium throughout the holster, modelling of fillers, choice of a 5N stack in ACT , etc.), this possible under-estimation is not likely to raise doubts over the criticality-safety of the packaging.

7.2 Metallic content

For the highly-enriched metallic media in homogeneous form, we have a dozen qualification points from the CRISTAL form for the cases with high enrichment (ref. [3]).

7.2.1 U metal content

It is generally noted that the standard method (APOLLO2 - MORET IV) under-estimates the k_{eff} for the bare configurations or those reflected by the water or the CH_2 of 800 pcm on average for the uranium-based media.

Although these qualification points do not correspond exactly to the situation studied (spherical rather than cylindrical mass in our case), it shows that the calculations made with fillers in the form of water do not require additional margins.

For the calculations with aluminium fillers, a benchmark has been created with a sphere of Umetal surrounded by aluminium. This case produces an over-estimated of K_{eff} in the order of 1300 pcm. Even by adding this value to the results obtained, this under-estimation does not raise doubts over the criticality-safety of the packaging.

7.2.2 Pu metal content

For this type of content, we only have a benchmark with a sphere of Pu metal surrounded by water showing that the Cristal scheme over-estimates the Keff value.

Given the Keff values obtained in the order of 0.955 in package array and 0.85 in isolated package for the most penalising cases and the conservative nature of hypotheses for the modelling (homogeneous medium throughout the holster, choice of a 5N stack in ACT, etc.), this possible over-estimation is not likely to raise doubts over the criticality-safety of the packaging.

8. ISOLATION SYSTEM

The isolation system represents all the package components required to maintain the fissile material in the configuration adopted in justifying criticality-safety.

The components included in the isolation system under the TN-BGC 1 study are:

- the packaging: geometry (maximum diameter of the packaging to encourage the interactions, cage), materials used, composition and thickness of neutron-absorbing resin (hydrogen and boron content, thickness of burnt resin),
- internal arrangements: filler geometry, materials making up the fillers (aluminium AUG4), geometry and material used for the container (diameter, thickness, material),
- fissile material: control of the criticality by restricting the geometry combined with restricting the fissile mass, compositions of various fissile media, type of content (UO_2, PuO_2, MOX, U metal and Pu metal).

9. CONCLUSION

This study demonstrates the lack of criticality risk for the TN-BGC 1 packaging loaded with standard contents in compliance with the regulatory stipulations for the safe transport of radioactive materials on the public highway [2].

For isolated packages, the fillers replaced by a water medium are the most penalising. The aluminium fillers reflect the neutrons less than their water counterparts.

For the stacks, the main differences noted between the aluminium fillers and those replaced by water are due to:

- improved water reflection,
- improved neutron moderation by the fillers replaced by the water which renders the resin more effective, whilst at the same time reducing neutron leaks and the interactions between the packages for the stacks.

The maximum weights which can be loaded in the TN-BGC 1 for the various contents are as follows:

- PuO_2 content:
  - ^{239}Pu content < 95% and ^{240}Pu > 5%:
    - 13 kg of Pu as an oxide (powder of density < 3.5) in a container with an internal diameter of 130 mm and thickness of 5 mm,
    - 17 kg of Pu as an oxide (powder of density < 3.5) in a container with an internal diameter of 120 mm and thickness of 2 mm,
  - ^{239}Pu content > 95%: 5 kg of Pu as an oxide (powder of density < 3.5) in a container with an internal diameter of 120 mm and thickness of 2 mm,
- mixed oxide content with Pu/U+Pu < 30%:
  - the plutonium complies with the following limitations (^{239}Pu < 95% and ^{240}Pu > 5%),
  - case with natural uranium: 40 kg U+Pu as mixed oxide (random physical form) regardless of the container used,
  - case with a random U235 enrichment:
    - 18 kg U+Pu as mixed oxide (random physical form) in a container with an internal diameter of 130 mm and thickness of 5 mm,

        - 20 kg U+Pu as mixed oxide (random physical form) in a container with an internal diameter of 120 mm and thickness of 2 mm,

- UO_2 content:
    - ^{235}U enrichment < 20% : 40 kg U as oxide (random physical form),
    - ^{235}U enrichment > 20%: 20 kg U as oxide (random physical form),
- Pu metal content: 4 kg Pu metal regardless of the composition or the physical form,
- U metal content:
    - ^{235}U enrichment < 20% : 40 kg U metal in random form,
    - ^{235}U enrichment > 20%: 7 kg U metal in random form,

FIGURE 1: MODELLING OF DEFORMATIONS

1: fissile medium

2: internal steel container

3: aluminium or water filler

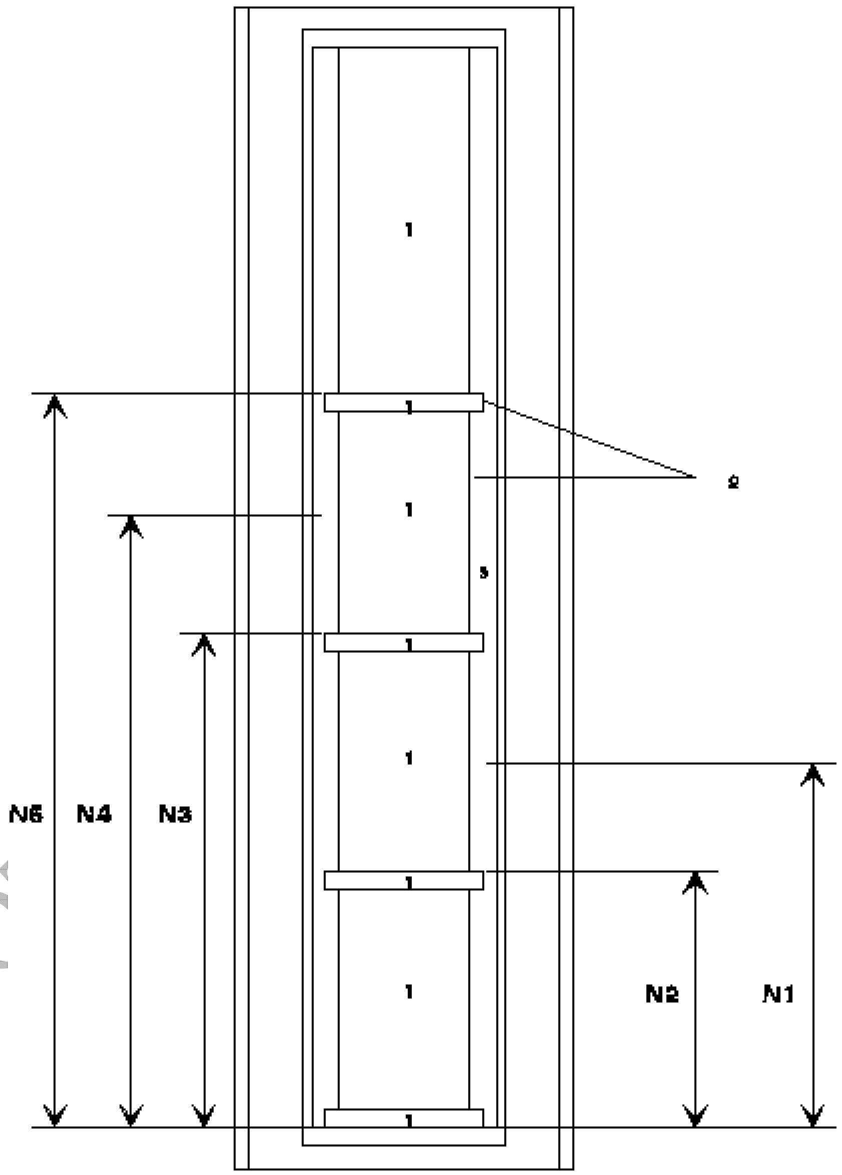
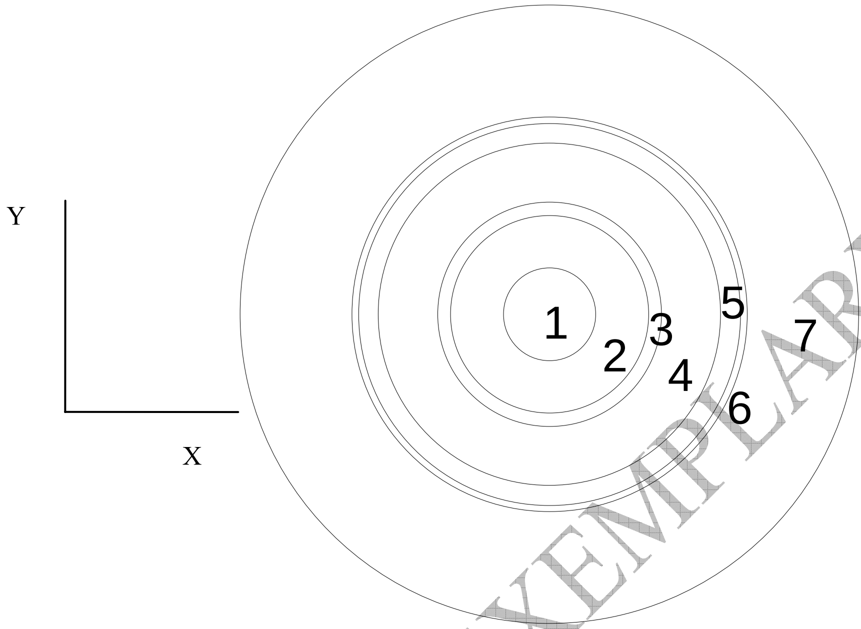
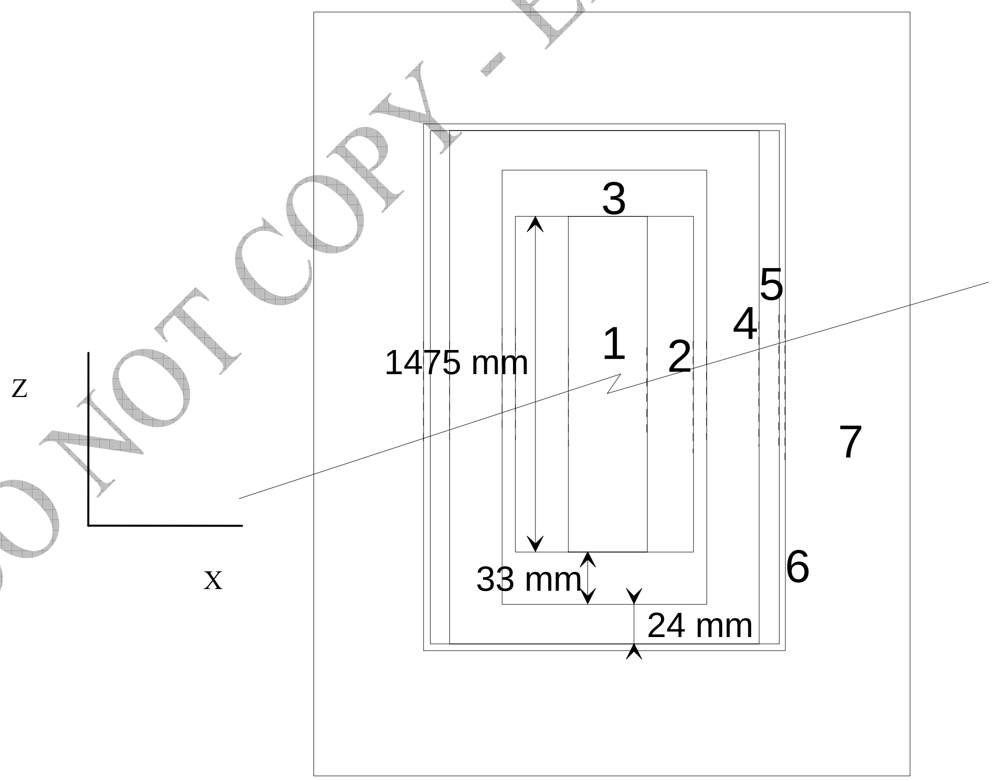


FIGURE 2: DAMAGED ISOLATED PACKAGE





1 fissile medium distributed in a diameter of 130 mm

2 internal packaging cavity 181 mm in diameter filled with water or aluminium

3 6 mm-thick stainless steel for the internal cavity

4 33 mm-thick neutron-absorbing resin

5 burnt resin assimilated with the water over a thickness of 15 mm

6 1.5 mm-thick external stainless steel cavity

7 reflection through 20 cm of water

FIGURE 3: EXAMPLE OF DAMAGED PACKAGE STACKS

ARRAY 8 X 7 X 1 VIEW FROM FRONT

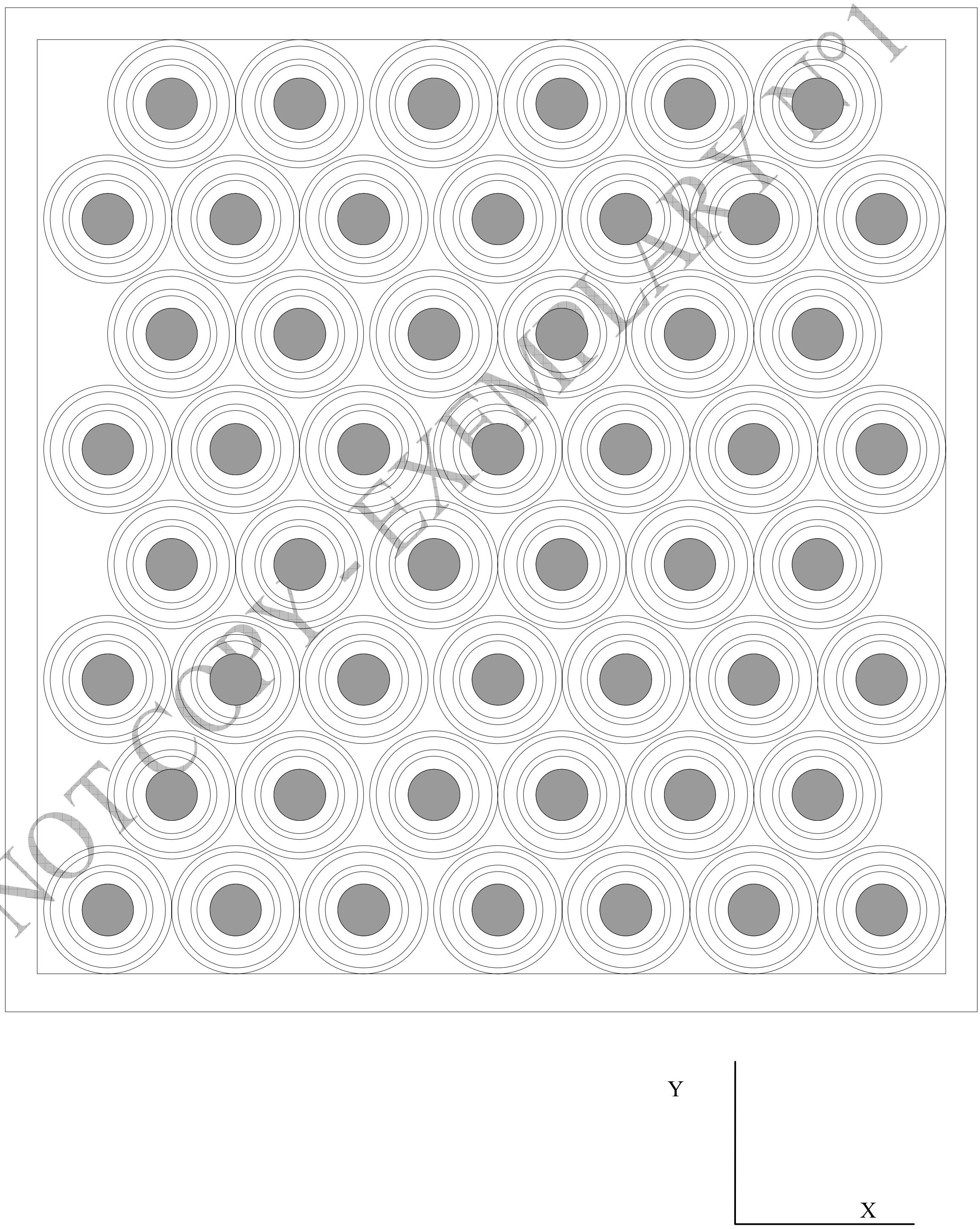
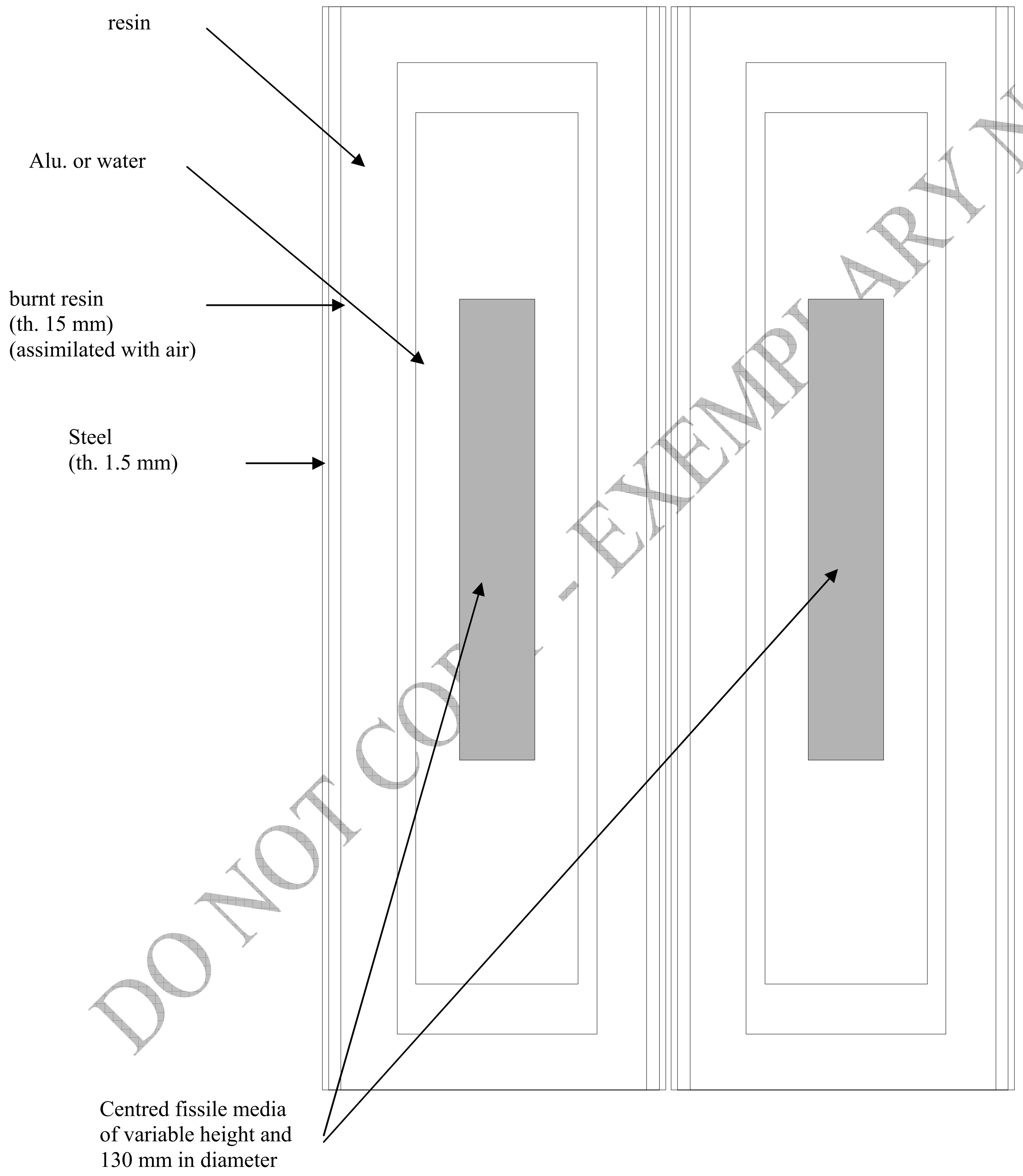


FIGURE 4: EXAMPLE OF DAMAGED PACKAGE STACKS

ARRAY 8 X 7 X 1 VIEW FROM ABOVE OF TWO ADJACENT PACKAGES



CEA
NUCLEAR ENERGY MANAGEMENT
Technical and Project Assistance Department
CEA transport packaging section

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Document purpose:	
Criticality study of the TN-BGC 1 packaging loaded with content no.11: uranium-bearing metal 100% enriched with U235	CEA/DEN/CAD/DTAP/SET DO 873 07/11/06 06PPFM001680

Application scope and summary

The purpose of this study is to demonstrate the criticality-safety during non-air transport of the package model constituted by the TN-BGC 1 packaging loaded with 15 kg of uranium-bearing metal 100% enriched with U235 with an arrangement of 100 mm; the influence of the aluminium alloy (AU4G, AG3) strut modelling is studied. The purpose of this study as regards air transport is to revise the [DA04] reference document so as to take the presence of carbon elements from wood into account in the moderation of the fissile medium (7 kg of uranium-bearing metal 100% enriched with U235).

The general conclusions of the study are as follows:

- the criticality-safety of the TNBGC 1 package model loaded with 15 kg of uranium-bearing metal 100% enriched with U235 is guaranteed during non-air transport provided that the number N=16 of packages transported is observed.
- the taking of carbon into account in the moderation of the fissile medium does not bring the criticality-safety of the TNBGC 1 package model loaded with 7 kg of uranium-bearing metal 100% enriched with U235 into question during air transport.

APPENDICES (included in the document and, therefore part of its overall paging)			ATTACHMENTS (formalism, identification and independent paging)		
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Summary

The purpose of this study is to demonstrate the criticality-safety during non-air and air transport of the package model constituted by the TN-BGC 1 packaging loaded with content no.11 (15 kg of uranium-bearing metal 100% enriched with U235 with an arrangement of 100 mm for non-air transport and 7 kg of uranium-bearing metal 100% enriched with U235 for air transport).

Normal and accident conditions for non-air transport shall be studied for a number of packages N=50. The number of packages permissible for transport shall be determined in the event that the permissibility criteria are not observed.

The sub-criticality of the packaging is checked for air transport, when the carbon elements from wood are taken into account as the moderating medium of the fissile medium.

This study is carried out using the hypotheses and the results of the study for both non-air [DA5] and air [DA4] transport.

The general conclusions of the study of the TNBGC 1 package model loaded with content no.11 (uranium-bearing metal 100% enriched with U235 with an arrangement of 100 mm) are as follows:

- the criticality-safety of the TNBGC 1 package model loaded with 15 kg of uranium-bearing metal 100% enriched with U235 is guaranteed during non-air transport provided that the number N=16 of packages transported is observed,
- the taking of carbon into account in the moderation of the fissile medium does not bring the criticality-safety of the TNBGC 1 package model loaded with 7 kg of uranium-bearing metal 100% enriched with U235 into question during air transport.

1 PURPOSE

The purpose of this study is to demonstrate the criticality-safety during non-air and air transport of the package model constituted by the TN-BGC 1 packaging loaded with content no.11 (15 kg of uranium-bearing metal 100% enriched with U235 with an arrangement of 100 mm for non-air transport and 7 kg of uranium-bearing metal 100% enriched with U235 for air transport).

Normal and accident conditions for non-air transport shall be studied for a number of packages N=50. The number of packages permissible for transport shall be determined in the event that the permissibility criteria are not observed.

The sub-criticality of the packaging is checked for air transport, when the carbon elements from wood are taken into account as the moderating medium of the fissile medium.

This study is carried out using the hypotheses and the results of the [DA4] and [DA5] reference studies relating to air and non-air transport of the TNBGC 1 packaging.

The form of the fissile medium is deemed to be homogeneous, moderated by an undefined quantity of water.

2 APPLICABLE DOCUMENTS

- [DA1] File justifying the safety of the TN-BGC-1 packaging – Analysis of the package criticality-safety – EMB TN-BGC PBC DS CA000001B.
- [DA2] CRISTAL-V1/NdI-V1.0/A – DSU/SEC/T/2004-280 Index A – SERMA/LENR/RT/04 3441/A – SPRC/LECy/004-325/0 – CRISTAL form version 1.0 identification manual.
- [DA3] Note SEC/T/02.086 – CRISTAL form (version V0) standard APPOLLO2-MORET 4 process qualification results.
- [DA4] Criticality-safety of package model TN-BGC 1 loaded with content no.11, air transport – EMB TNBGC PBC DJS CA000387 A.
- [DA5] Criticality-safety of package model TN-BGC 1 loaded with content nos.7, 8 and 11 – EMB TNBGC PBC DJS CA000355 A

3 REFERENCE DOCUMENTS

- [DR1] Radioactive material transport regulations – 1996 revised issue, Safety standard collection no.TS-R1
- [DR 2] Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material – Safety Guide No.TS-G1.1 (ST2).

4 TN-BGC 1 PACKAGING CHARACTERISTICS

The characteristics of the packaging come from reference [DA1].

The packaging comprises a parallelepiped cage, inside of which is attached a cylindrical body fitted with a closure system and a cover.

The dimensions of the packaging are as follows:

- cross-section of the cage: 600 x 600 mm²,
- overall height of the cage: 1,821 mm,
- diameter of the working section of the body: 295 mm,
- diameter of the cover: 466 mm,
- overall height of the body fitted with the cover: 1,808 mm.

The maximum weights of the packaging are:

- empty: 280 kg
- full: 396 kg

4.1 CAGE

The 30 x 30 mm², 2 mm thick structure of the cage is made of aluminium tubes.

4.2 BODY

The cavity with an effective diameter of 178 mm and effective length of 1,475 mm is made from a 6 mm thick stainless steel shell (guaranteeing the essential radial gamma shielding) and an 8 mm thick base also made of stainless steel.

A second, 1.5 mm thick, stainless steel shell with a 292 mm internal diameter forms a space around the first shell which is filled with hydrogen and boron-loaded resin (minimum thickness of 48 mm). Said resin acts as a neutron absorber and active thermal insulation.

The base is completed, from the inside to the outside, with a 25 mm high yield strength steel diffuser plate, a 24 mm layer of resin, an intermediate base, a wooden shock absorbing disc and a sheet of stainless steel.

The plug is machined in a 92 mm thick stainless steel disc. Its periphery comprises a 20 mm ledge which rests on the body flange. Said plug is held in position by a bronze tightening ring which is tightened in the stainless steel bayonet locking collar.

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4.3 COVER

There is a shock absorbing cover over the top of the body head-end piece and the closure system. It comprises two stainless steel sheet metal units. The one nearest to the body is filled with resin, the other one is filled with wood.

4.4 INTERNAL ARRANGEMENT

The packaging is loaded with both an internal stainless steel packing container loaded with radioactive material and internal containment devices. There are different types of packing containers:

- TN 90,
- AA 203,
- AA 204,
- AA 41,
- AA 226 or AA227.

The following chocks are used to attach the packing container in the packaging cavity:

- with TN 90: chock E1 + chock E2,
- with AA 203: chock E1 + chock E8,
- with AA 204: chock E1 + chock E10,
- with 1 AA 41: chock E1 + chock E11,
- with 2 AA 41: chock E1 + chock E12 + chock E13,
- with 3 AA 41: chock E1 + chock E12 + 2 x chock E13
- with AA 226 or AA 227: chock E1.

Chocks E1 and E2 are made of AU4G aluminium, the other types of chocks are made of AG3 aluminium.

An aluminium alloy E7 strut (see [DA1]) is used for the radial chocking inside the TN90 in the configuration studied as part of this study, namely the packing of the material in a 100 mm diameter stainless steel container.

5 CONTENT CHARACTERISTICS

The content studied is content no.11 on the certificate of approval, namely uranium-bearing metal 100% enriched with U235 in a 100 mm internal diameter, 2 mm thick container. The weights transported are:

- 15 kg for non-air transport (number of packages N=50).
- 7 kg for air transport.

6 CALCULATION METHODS

6.1 CALCULATION CODES

keff is calculated in two stages using the CRISTAL criticality form [DA2] standard process:

- study of the fissile medium with APOLLO2,
- calculation of keff using the MORET 4 code.

6.1.1 Studying fissile media using the APOLLO2 code

When modelling the fuel, the fissile medium (U235-water) is deemed homogeneous and the moderation ratio varied.

The Cigales version V 3.0 MMI is used for generating the set of APOLLO2 data.

The APOLLO2 code provides:

- k_∞, the infinite multiplication factor, and B2m, the material buckling,
- a set of tamper-proof, homogenised cross-sections with 172 energy groups for the fissile media,
- a set of cross-sections with 172 energy groups for the structure environments.

The APOLLO2 code inputs are shown in table 1 (structure materials) and in table 2 (fissile medium).

6.1.2 Calculating keff using the MORET 4 MONTE-CARLO code

The keff effective multiplication factor is obtained using the 3D multi-group MORET 4 Monte-Carlo code. Said code uses the cross-sections generated by the APOLLO2 code:

- for homogenised fissile media
- for structure environments.

Calculation uncertainty is taken as being less than or equal to 200 pcm for all the calculations made in this study.

6.2 PERMISSIBILITY CRITERIA

The permissibility criterion adopted is as follows:

- for non-air transport

  → keff + 3σ ± 0.950, whatever the uncertainty, for an isolated package under accident conditions in transport accumulated with normal transport conditions,

  → keff + 3σ ± 0.980, whatever the uncertainty, for an infinite package network under accident conditions in transport, the moderation between the packages is not defined.

A bias of an additional 1,000 pcm linked to the uncertainties for the aluminium cross-sections is taken into account regarding the usual permissibility criteria when the importance of the aluminium alloy chocks is demonstrated.

- for air transport

  → keff + 3σ ± 0.950, whatever the uncertainty.

6.3 METHODS

The search for optimum reactivity for each environment is carried out in accordance with the following parameters:

- moderation ratio,
- density of the interstitial water spray,
- position of the fissile medium in the cavity.

7 NON-AIR TRANSPORT

7.1 CALCULATION HYPOTHESES

The calculation hypotheses for this study are based on the [DA1] reference safety file. They are repeated later in this document.

7.1.1 Package damage during ACT

The damage to packages following the regulatory drop tests and the thermal test during ACT is reiterated in reference [DA1]. A 15 mm reduction in the thickness of the neutron-absorbing resin biological shielding is considered (the radial thickness of the biological protection varies between 48 mm (NTC) and 33 mm (ACT) following tests during ACT).

Penetration of water into all the free spaces in both the cavity and the packaging and between the packages in the event of a stacked packaging configuration is also considered.

The packaging cage is not taken into account during accident conditions for a worst case scenario.

7.1.2 Package modelling

The package calculation model is shown below:

Radial section of the packaging body

- 6 mm thick, stainless steel shell with an internal diameter of 178 mm;
- 48 mm thick, boron-loaded resin (burnt to a depth of 15 mm in an accident situation);
- 1.5 mm thick, stainless steel shell.

Plug

- 92 mm thick stainless steel.

Base (from the inside to the outside)

- 33 mm thick, stainless steel shell and diffuser plate;
- 24 mm thick resin;
- 1.5 mm thick, stainless steel shell.

The internal height of the packaging cavity is considered equal to 1,475 mm. The internal diameter of the packing container is considered equal to 100 mm. The thickness of the steel packing container shell is taken to equal 2 mm.

A study was carried out to check whether it is better to assume that the free space between the internal container and the inner shell of the packaging is totally comparable to AG3, AU4G, water or air in order to determine the worst case scenario modelling of the chocks.

The volume of the fissile medium for a given weight of fissile material, is modelled by a 100 mm cylinder of varying height as it varies in accordance with the moderation ratio.

The fissile material is placed against the top stop in the internal container, thus, encouraging the reflection by the steel of the upper plug.

The chemical composition of the materials used are given in table 1.

7.1.3 Isolated package

The isolated package is surrounded (both radially and axially) by a 20 cm ring of water to guarantee neutron reflection.

The internal cavity of the packaging in which the fissile material is modelled as well as the space created by the burnt resin are filled with density 1 water for a worst case scenario.

The MORET 4 model of the TN-BGC 1 package is shown in figures 1 and 2.

7.1.4 Finished package networks (2N package during ACT, 5N during NTC)

During NTC, normal transport conditions, the study is concerned with 5N packages in the package conditions from which result the maximum multiplication of neutrons with the following hypotheses:

- the unit formed by the 5N packages is surrounded by a 20 cm ring of water,
- the state of the packages is that noted after the tests under normal transport conditions (undamaged boron-loaded resin, presence of the cage guaranteeing a regular 60 cm gap between packages).

During ACT, accident conditions in transport, the study is concerned with 2N packages in the package conditions from which result the maximum multiplication of neutrons with the following hypotheses:

- moderation is optimal and the unit formed by the 2N packages is surrounded by a 20 cm ring of water,
- the state of the packages is that noted after the tests under normal transport conditions followed by tests under accident conditions in transport (burnt resin and no cage).

The study is carried out by considering a unit of 5N (with N=50) damaged packages. If this hypothesis is too conservative, the number N of packages is limited. The possible determination of the number N of permissible packages is carried out with a 2N package configuration during ACT. The sub-criticality is checked for the 5N package configuration during NTC and the 2 N package configuration in ACT.

The internal cavity of the packaging in which the fissile material is modelled is filled with density 1 water for a worst case scenario. In addition, the space between packages and in place of the burnt resin is taken as being filled with air for package stacking configurations in order to maximise neutron interactions.

The MORET 4 model of a finished 5N network (with N=50) for a TN-BGC 1 package is shown in figures 3 and 6.

7.2 CRITICALITY-SAFETY STUDIES

7.2.1 Criticality-safety analysis in an isolated package configuration

This involves estimating the influence on the reactivity of a variation of the nature of the aluminium alloy chocks on the isolated package configuration during ACT. The influence of the nature of the chocks was carried out with the following hypotheses:

- the weight is 15 kg of U235 in metal form,
- the fissile material is off-centred towards the plug and the packages are stacked alternate ways up,
- also, density 1 water is taken as being in the fissile cavity (cavity filled with water),
- lastly, the internal arrangement is of 2 mm thick stainless steel with a 100 mm diameter.

The search for optimum reactivity is carried out in accordance with the following parameters: moderation ratio (H/U) and aluminium alloy chock composition. The table below shows the values of keff + 3σ (σ≤ 200 pcm) obtained for 4 types of chocks; chocks of AG3, AU4G, air and water.

Moderation ratio H/U	Cylinder height (U + water) (in cm)	Nature of the chock							
		AG3		AU4G		Water		Air	
		keff	keff+3σ	keff	keff+3σ	keff	keff+3σ	keff	keff+3σ
0	10.09	**0.872**	**0.878**	**0.870**	**0.876**	0.900	0.906	**0.803**	**0.809**
0.01	10.17	0.864	0.870	0.865	0.871	**0.901**	**0.907**	0.792	0.798
0.05	10.46	0.862	0.868	0.861	0.867	0.897	0.903	0.792	0.798
0.1	10.83	0.857	0.863	0.857	0.863	0.895	0.900	0.787	0.793
0.5	13.76	0.835	0.841	0.834	0.840	0.876	0.882	0.754	0.760
1	17.43	0.816	0.822	0.813	0.819	0.863	0.869	0.732	0.738
1.2	18.90	0.807	0.813	0.811	0.817	0.864	0.870	0.722	0.728

For the configuration of the isolated package during ACT, the enclosure case corresponds to a space between the container and the inner shell of the water-filled packaging. The criticality-safety of the TN-BGC 1 package model loaded with 15 kg of uranium-bearing metal 100% enriched with U235 is observed in the isolated package configuration during ACT.

7.2.2 Criticality-safety analysis in a 5N (with N=50) package configuration during ACT

The container is maintained in the inner space of the packaging by different types of aluminium chocks (AU4G or AG3). The permissibility of the configuration is checked with the following hypotheses:

- the weight is 15 kg of U235 in metal form,
- the fissile material is off-centred towards the plug and the packages are stacked head-to-tail,
- the space between the packages is filled with air so as to encourage interaction between packages,
- also, density 1 water is taken as being in the fissile cavity (cavity filled with water),
- lastly, the internal arrangement is of 2 mm thick stainless steel with a 100 mm diameter.

The search for the optimum reactivity for each container is carried out in accordance with the following parameters: moderation ratio (H/U) and aluminium alloy chock composition.

The table below shows the values of keff + 3σ (σ≤ 200 pcm) obtained for 4 types of chocks; chocks of AG3, AU4G, air and water.

Moderation ratio H/U	Cylinder height (U + water) (in cm)	Nature of the chock							
		AG3		AU4G		Water		Air	
		keff	keff+3σ	keff	keff+3σ	keff	keff+3σ	keff	keff+3σ
0	10.09	**0.986**	**0.989**	0.981	0.985	**0.966**	**0.970**	**0.948**	**0.951**
0.01	10.17	0.983	0.987	**0.982**	**0.985**	0.964	0.968	0.946	0.950
0.05	10.46	0.983	0.988	0.980	0.983	0.963	0.967	0.943	0.947
0.1	10.83	0.978	0.981	0.976	0.980	0.959	0.963	0.941	0.944
0.5	13.76	0.964	0.968	0.963	0.967	0.946	0.950	0.925	0.929
1	17.43	0.955	0.959	0.955	0.959	0.939	0.943	0.921	0.925
1.2	18.90	0.951	0.955	0.953	0.957	0.934	0.938	0.914	0.918

For the 5N (N=50) package configuration during ACT, maximum reactivity is obtained with AG3 chocks for a dry fissile medium. The results bring the sub-criticality of the TNBGC1 packaging into question for non-air transport of content no.11 (for a number N=50 of packages). A permissible number N of packages is determined in order to observe the permissibility criteria.

7.2.3 Determination of the number of permissible packages

The number of permissible packages is determined in the 2N package configuration during ACT with the following hypotheses:

- the weight is 15 kg of U235 in metal form,
- the fissile material is off-centred towards the plug and the packages are stacked alternate ways up,
- density 1 water is taken as being in the fissile cavity (cavity filled with water),
- the spaces left by the burnt resin (following the transport tests) and between the packages are filled with air in order to encourage interaction between packages.

We also consider (see conclusion 7.2.2):

- a dry fissile medium
- the space between the container and the inner shell where the chocks are set out is filled with AG3 aluminium.

The results are shown in the following table:

Number of packages N	Network structure	Moderation ratio H/U	Cylinder height (U235 + water) (in cm)	keff+3σ
50	7x8x2	0	10.09	0.992
25	5x5x2			0.981
20	5x4x2			0.973
16	**4x4x2**			**0.965**

The number N of permissible packages (keff+3σ ≤ 0.970) for content no.11 (uranium-bearing metal 100% enriched with U235) is N=16.

7.2.4 CRITICALITY-SAFETY ANALYSIS IN NETWORK CONFIGURATIONS (N=16 PACKAGES)

7.2.4.1 2N (with N=16) package network during ACT

This involves estimating the influence on the reactivity of a variation of the nature of the aluminium alloy chocks on the 2N (with N=16) package configuration during ACT. The influence of the nature of the chocks was carried out with the following hypotheses:

- the weight is 15 kg of U235 in metal form,
- the fissile material is off-centred towards the plug and the packages are stacked head-to-tail,
- the space between the packages and the burnt resin were replaced with air in order to encourage interaction between packages,
- also, density 1 water is taken as being in the fissile cavity (cavity filled with water),
- lastly, the internal arrangement is of 2 mm thick stainless steel with a 100 mm diameter.

The search for optimum reactivity is carried out in accordance with the following parameters: moderation ratio (H/U) and aluminium alloy chock composition. The table below shows the values of keff + 3σ (σ≤ 200 pcm) obtained for 4 types of chocks; chocks of AG3, AU4G, air and water.

Moderation ratio H/U	Cylinder height (U + water) (in cm)	Nature of the chock							
		AG3		AU4G		Water		Air	
		keff	keff+3σ	keff	keff+3σ	keff	keff+3σ	keff	keff+3σ
0	10.09	**0.959**	**0.965**	**0.963**	**0.969**	**0.954**	**0.960**	**0.894**	**0.900**
0.01	10.17	0.960	0.966	0.958	0.964	0.950	0.956	0.890	0.896
0.05	10.46	0.957	0.963	0.957	0.963	0.947	0.953	0.887	0.893
0.1	10.83	0.952	0.958	0.955	0.961	0.947	0.953	0.886	0.892
0.5	13.76	0.936	0.942	0.937	0.943	0.931	0.937	0.862	0.868
1	17.43	0.926	0.932	0.921	0.927	0.923	0.929	0.846	0.852
1.2	18.90	0.922	0.928	0.918	0.924	0.923	0.929	0.840	0.846

For the 2N (N=16) package configuration during ACT, the conservative case corresponds to a space between the container and the inner shell of the packaging filled with AU4G aluminium. Both types of aluminium AU4G and AG3 give statistically equivalent results. The criticality-safety of the TN-BGC 1 package model loaded with 15 kg of uranium-bearing metal 100% enriched with U235 is observed in the 2N (with N=16) package network configuration during ACT.

7.2.4.2 5N (with N=16) package network during NTC

This involves estimating the influence on the reactivity of a variation of the nature of the aluminium alloy chocks on the 5N (with N=16) package configuration during NTC. The influence of the nature of the chocks was carried out with the following hypotheses:

- the weight is 15 kg of U235 in metal form,
- the fissile material is off-centred towards the plug and the packages are stacked head-to-tail,
- the space between the packages is filled with air so as to encourage interaction between packages,
- also, density 1 water is taken as being in the fissile cavity (cavity filled with water),
- lastly, the internal arrangement is of 2 mm thick stainless steel with a 100 mm diameter.

The search for optimum reactivity is carried out in accordance with the following parameters: moderation ratio (H/U) and aluminium alloy chock composition. The table below shows the values of keff + 3σ (σ≤ 200 pcm) obtained for 4 types of chocks; chocks of AG3, AU4G, air and water.

Moderation ratio H/U	Cylinder height (U + water) (in cm)	Nature of the chock							
		AG3		AU4G		Water		Air	
		keff	keff+3σ	keff	keff+3σ	keff	keff+3σ	keff	keff+3σ
0	10.09	**0.897**	**0.902**	0.889	0.895	0.914	0.920	0.821	0.826
0.01	10.17	0.890	0.897	**0.894**	**0.900**	0.916	0.922	**0.822**	**0.828**
0.05	10.46	0.890	0.896	0.882	0.898	**0.917**	**0.922**	0.817	0.822
0.1	10.83	0.884	0.889	0.884	0.889	0.910	0.916	0.813	0.819
0.5	13.76	0.858	0.864	0.857	0.863	0.895	0.901	0.783	0.789
1	17.43	0.845	0.851	0.842	0.848	0.884	0.890	0.760	0.766
1.2	18.90	0.838	0.844	0.835	0.841	0.877	0.883	0.747	0.753

For the 5N (N=16) package configuration during NTC, the enclosure case corresponds to a space between the container and the inner shell of the packaging filled with water. The criticality-safety of the TN-BGC 1 package model loaded with 15 kg of uranium-bearing metal 100% enriched with U235 is observed in the 5N (with N=16) package network configuration during NTC.

7.3 INSULATION SYSTEM

The following elements constitute the insulation system to be guaranteed:

- packaging: geometry (maximum diameter of 178 mm), materials (the inner and outer packaging shells are made of stainless steel), materials used, composition and thickness of the neutron-absorbing boron-loaded resin (boron and hydrogen content, thickness of burnt resin),
- diameter and thickness of the stainless steel internal arrangement,
- chock system defining the radial position taken up by the fissile material,
- packaging cage (60 cm x 60 cm) guaranteeing the network gap in the 5N package configuration during NTC,
- fissile material: checking the criticality by limiting the weight and composition of the fissile medium (15 kg of uranium-bearing metal 100% enriched with U235).

7.4 CONCLUSION

The criticality-safety of the TNBGC 1 package model loaded with content no. 11 (uranium-bearing metal 100% enriched with U235 in a 2 mm thick, stainless steel arrangement with a 100 mm diameter) is guaranteed provided that the following are observed:

- the number of packages transported N=16,
- the 15 kg weight of uranium-bearing metal 100% enriched with U235.

8 AIR TRANSPORT

8.1 CALCULATION HYPOTHESES

The taking into account of the carbon elements from the wood as the moderating medium is studied using the conservative configuration of the [D4] reference study. The fissile material is moderated by the hydrogen and carbon elements present in the wood of the contents in the cover:

- the weight of water contained in the wood is less than 2,000 g,
- the maximum weight of carbon contained in the wood is 2,500 g.

The fuel is treated by a heterogeneous moderation:

- a proportion of the total weight of fissile material is moderated by a mixture of water (with a maximum weight of 2,000 g) and carbon (maximum weight 2,500 g).
- the rest of the total weight is modelled around the moderated sphere in the shape of a dry shell, playing the role of a reflector.

The configuration has 4 concentric spherical areas:

- the fuel moderated by a mixture of water and carbon,
- the dry fuel,
- the steel (340 kg corresponding to the steel structure of the packaging),
- a 20 cm ring of water.

The MORET 4 model of the isolated package model for air transport is shown in figure 7.

8.2 AIR TRANSPORT CRITICALITY-SAFETY STUDIES

The carbon elements from the wood participate in the moderation of the fissile medium. The study takes this parameter into account using the most reactive configuration of the [DA4] reference air transport study (section 4.5: configuration 5), the hypotheses of which are listed in section 8.1. The influence of the nature of the chocks was carried out with the following hypotheses:

- the weight is 7 kg of U235 in metal form in the shape of a moderated sphere not exceeding 2 kg of water and a maximum of 2.5 kg of carbon,
- the steel in the packaging structure is taken into account (not exceeding 340 kg).

The maximum reactivities obtained are as follows:

Weight of carbon (g)	keff	keff+3σ
0	**0.977**	**0.983**
400	0.964	0.970
750	0.954	0.960
2,500	0.909	0.915

Details of the calculations are shown in table 3.

Taking the carbon elements from the wood into account leads to a lower reactivity.

The maximum reactivity keff+3σ = 0.983 exceeds the permissibility criteria. However, the [DA4] reference study (section 4.5) shows the over-estimation of the results provided by the APOLLO2 – MORET IV calculation diagram in the presence of a significant thickness of steel when it is used as a reflector. The deviation observed is due to the treatment of the steel and notably Fe56. We would reiterate that a study using the ad hoc energy TRIPOLI4 calculation code makes it possible to assess the over-estimation of the results provided by the APOLLO2 – MORET IV calculation diagram. Said over-estimation for the worst case scenario reaches 6,000 pcm and the maximum value of keff+3σ (σ=50 pcm) = 0.915.

<u>The presence of carbon from the wood does not bring the criticality-safety of the TNBGC 1 packaging into question during air transport</u>.

8.3 INSULATION SYSTEM

The criticality-safety is controlled by limiting the weight, the fissile medium composition, the moderation and the nature of the moderator.

8.4 CONCLUSION

The study in reference [DA4] has proved the criticality-safety of the TNBGC 1 packaging during air transport. However, it does not consider the carbon elements from the wood in the fissile medium moderation. Taking the carbon into account in the moderation of the fissile medium does not bring the criticality-safety of the TNBGC 1 package loaded with 7 kg of uranium-bearing metal 100% enriched with U235 into question. The results of the [study in reference [DA4] are not brought into question.

9 QUALIFICATION

For a homogeneous environment of heavily enriched uranium-bearing metal (H/U=0), we have 9 qualification points on the CRISTAL form using the APOLLO2-MORET4. Calculation-feedback deviations have been identified and show that the standard process under-estimates the keff for basic configurations or those reflected using water by an average of 800 pcm.

As far as experiments implementing steel as a reflector are concerned, significant deviations from the standard process are noted, deviations that can be attributed to the multi-group treatment of cross-sections of iron (predominantly isotope 56). Said over-estimation increases with the thickness of the reflector.

Lastly, as far as the presence of aluminium alloy struts is concerned, the APOLLO – MORET 4 code is qualified when the aluminium is used as a reflector and for a rapid spectrum. The fact that the current qualification of the APOLLO – MORET 4 process is missing for the aluminium in a thermal spectrum means that a margin will have to be taken in order to cover the uncertainties concerning the cross-sections of the aluminium. An additional bias of 1,000 pcm is taken into account as regards the usual permissibility criteria when the significance of the aluminium alloy chocks is demonstrated.

10 GENERAL CONCLUSION

The general conclusions of the study of the TNBGC 1 package model loaded with content no.11 (uranium-bearing metal 100% enriched with U235 in a 2 mm thick, stainless steel arrangement with a 100 mm diameter) are as follows:

- the criticality-safety of the TNBGC 1 package model loaded with 15 kg of uranium-bearing metal 100% enriched with U235 is guaranteed for non-air transport provided that the number of packages transported N=16 is observed,
- the taking into account of the carbon in the fissile medium moderation for air transport does not bring the criticality-safety of TNBGC 1 package model loaded with 7 kg of uranium-bearing metal 100% enriched with U235 into question.

TABLES

TABLE 1: CHEMICAL COMPOSITION OF STRUCTURE MATERIALS

Environment	Elements	Concentration (atoms/cm³ x 1024)
Stainless steel d=7.87	Fe Cr Ni	6.134.10-2 1.647.10-2 8.107.10-3
Neutron-absorbing resin d=1.186 Boron weight content 1.43%	H C O B_{nat}	4.0616.10-2 2.3803.10-2 2.3580.10-2 9.4597.10-4
AG3 d=2.65	Al Mg Cr Zn Ti	5.6780.10-2 2.1011.10-3 1.2277.10-4 4.8810.10-5 6.6660.10-5
AU4G d=2.67	Si Fe Cu Mn Mg Cr Zn Al	2.8625.10-4 2.0154.10-4 1.021.10-3 2.0487.10-4 4.6309.10-4 3.0923.10-5 6.2878-10-5 5.5451.10-2
Air	N O	4.1985.10-5 1.1263.10-5
Water	H O	6.68558.10-2 3.34279.10-2
Carbon (graphite) d=2.3	C	1.1532.10-1

TABLE 2: FISSILE MEDIUM – APOLLO2 CODE INPUTS

	Fissile medium
Chemical form	Uranium-bearing metal
U235 enrichment (weight %)	100
Density	18.91966

TABLE 3: AIR TRANSPORT – INFLUENCE OF THE CARBON ELEMENTS FROM WOOD

Weight of carbon under consideration: 0 kg						
Weight of moderated uranium (g)	Uranium concentration in the mixture [U235-H2O] (g/l)	Moderated fissile sphere shell radius (cm)	Dry shell uranium weight (g)	Dry fissile shell radius (cm)	Steel shell radius (cm)	Keff + 3σ (σ≤200 pcm)
200	99.276	7.835	6,800			0.948
400	197.516	7.849	6,600			0.973
600	**294.736**	**7.862**	**6,400**	**8.276**	**25.626**	**0.983**
2,000	947.994	7.956	5,000			0.975
3,000	1,387.236	8.022	4,000			0.968

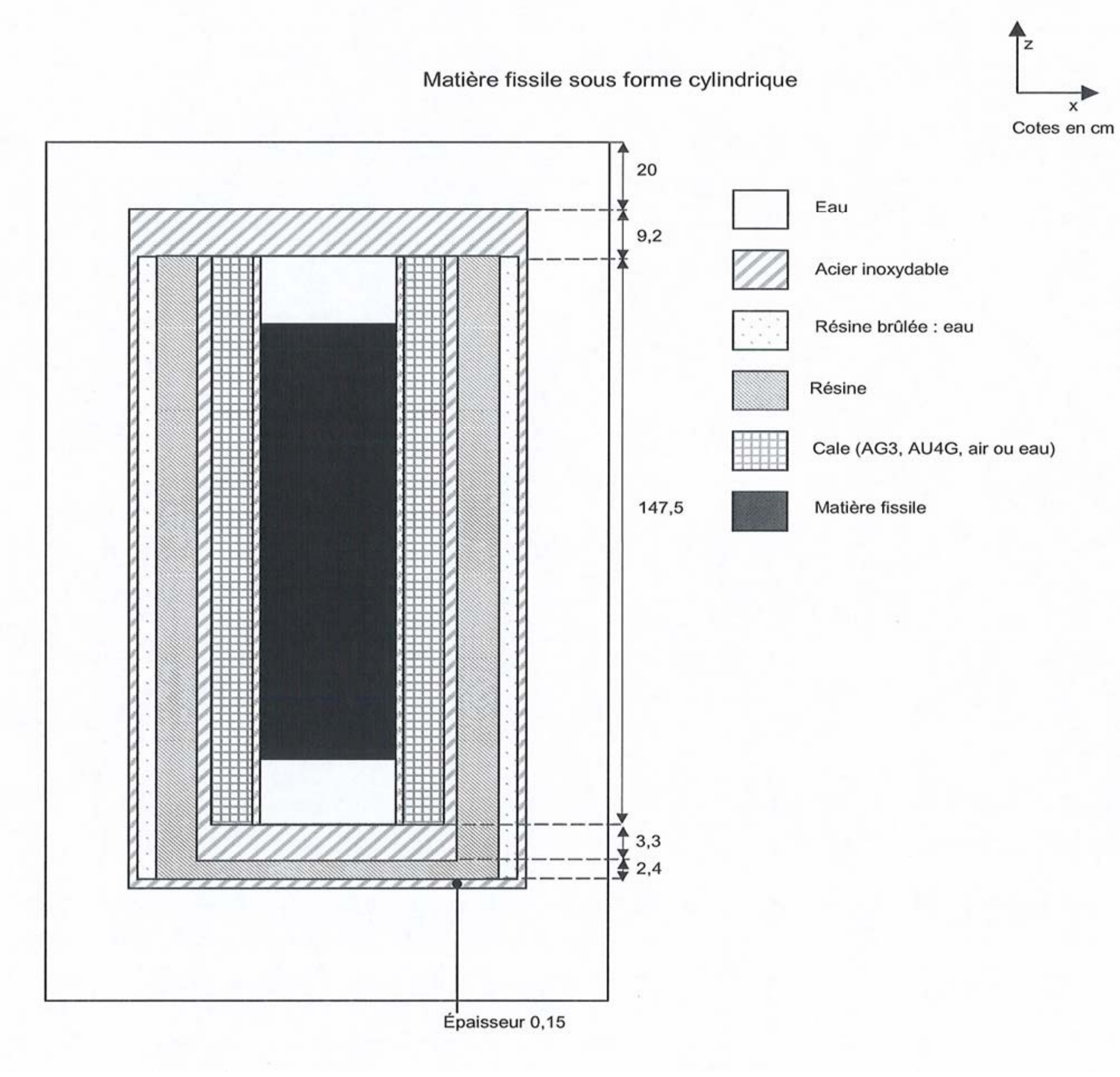
Weight of carbon under consideration: 0.4 kg						
Weight of moderated uranium (g)	Uranium concentration in the mixture [U235-(H2O-GRAPH)] (g/l)	Moderated fissile sphere shell radius (cm)	Dry shell uranium weight (g)	Dry fissile shell radius (cm)	Steel shell radius (cm)	Keff + 3σ (σ≤200 pcm)
400	181.895	8.067	6,600			0.960
600	271.537	8.080	6,400			0.964
1,000	448.272	8.106	**6,000**	**8.473**	**25.823**	**0.970**
2,000	875.794	8.169	5,000			0.963
3,000	1.283.973	8.232	4,000			0.959

Weight of carbon under consideration: 0.75 kg						
Weight of moderated uranium (g)	Uranium concentration in the mixture [U235-(H2O-GRAPH)] (g/l)	Moderated fissile sphere shell radius (cm)	Dry shell uranium weight (g)	Dry fissile shell radius (cm)	Steel shell radius (cm)	Keff + 3σ (σ≤200 pcm)
400	170.122	8.249	6,600			0.946
600	254.042	8.261	6,400			0.954
1,000	419.646	8.286	**6,000**	**8.638**	**25.988**	**0.960**
2,000	821.080	8.347	5,000			0.951
3,000	1,205.463	8.407	4,000			0.945

Weight of carbon under consideration: 2.5 kg						
Weight of moderated uranium (g)	Uranium concentration in the mixture [U235-(H2O-GRAPH)] (g/l)	Moderated fissile sphere shell radius (cm)	Dry shell uranium weight (g)	Dry fissile shell radius (cm)	Steel shell radius (cm)	Keff + 3σ (σ≤200 pcm)
400	128.530	9.057	6,600			0.900
600	192.142	9.067	6,400			0.909
1,000	**318.083**	**9.088**	**6,000**	**9.384**	**26.734**	**0.915**
2,000	625.648	9.138	5,000			0.910
3,000	923.207	9.188	4,000			0.905

FIGURES

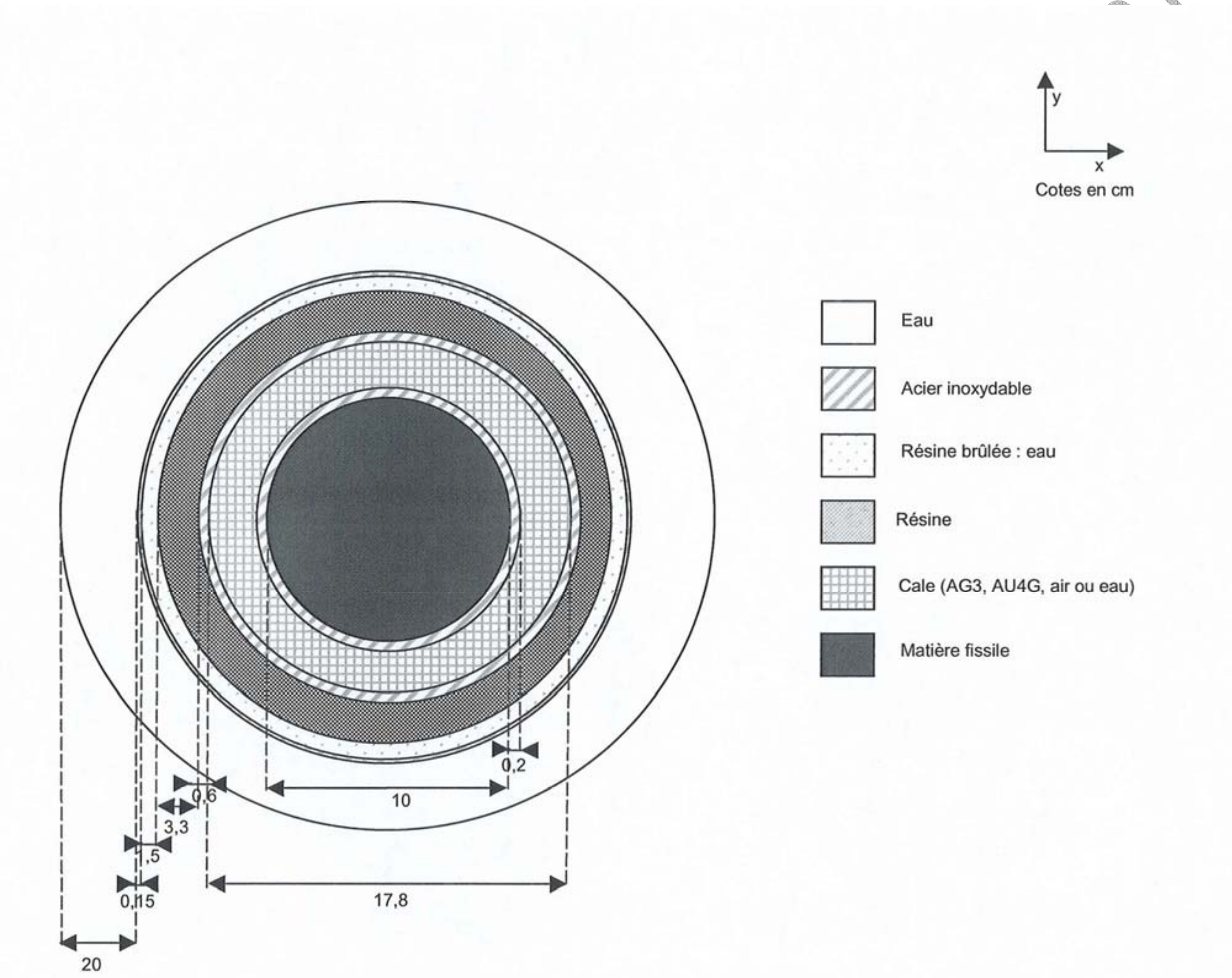
FIGURE 1: NON-AIR TRANSPORT – ISOLATED PACKAGES DURING ACT – AXIAL SECTION



Matière fissile sous forme cylindrique	Fissile material in cylindrical form
Cotes en cm	Dimensions in cm
Eau	Water
Acier inoxydable	Stainless steel
Résine brûlée: eau	Burnt resin: water
Résine	Resin
Cale (AG3, AU4G, air ou eau)	Chock (AG3, AU4G, air or water)
Matière fissile	Fissile material

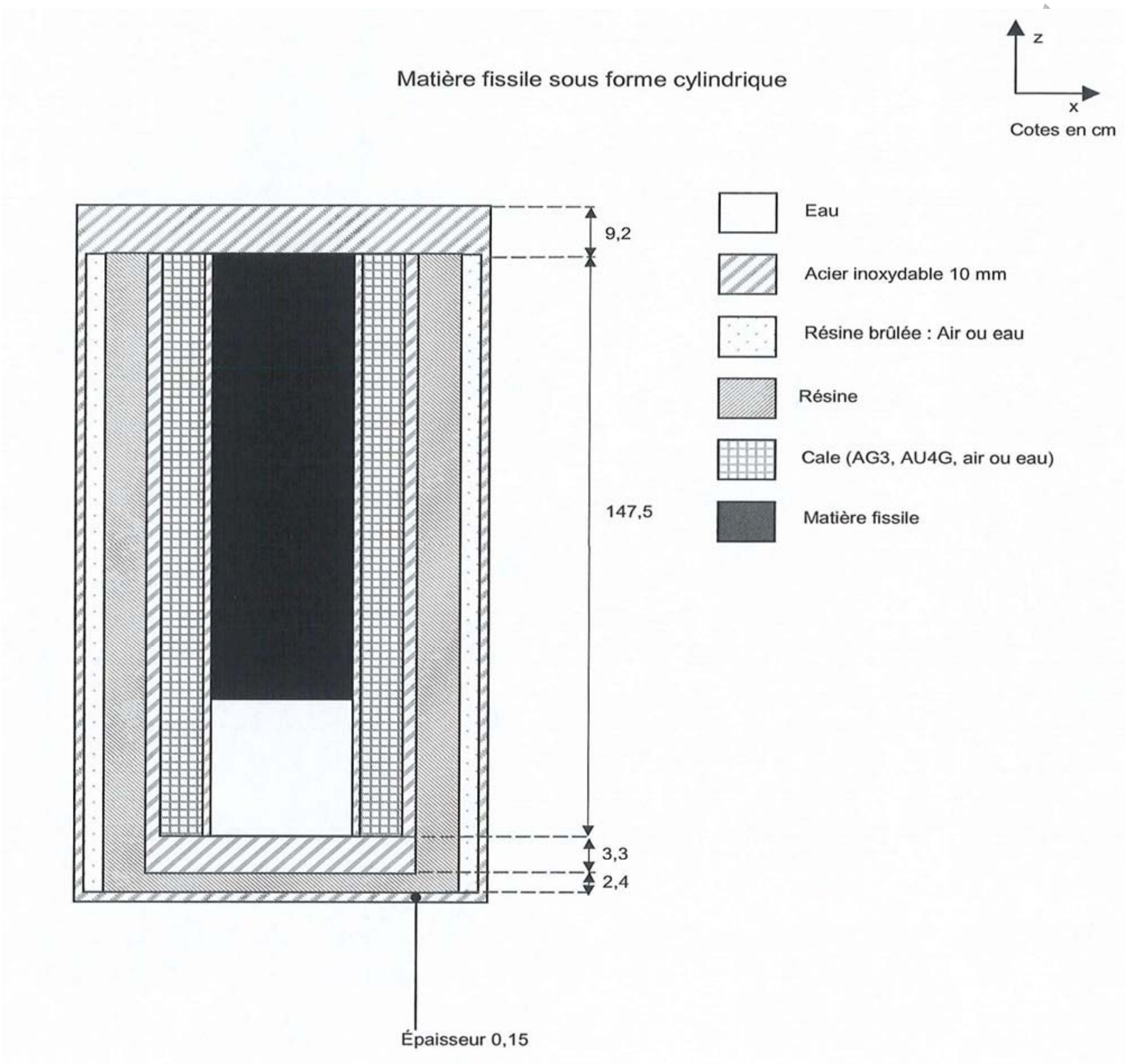
Epaisseur 0,15	0.15 thick

FIGURE 2: NON-AIR TRANSPORT – ISOLATED PACKAGES DURING ACT – RADIAL SECTION



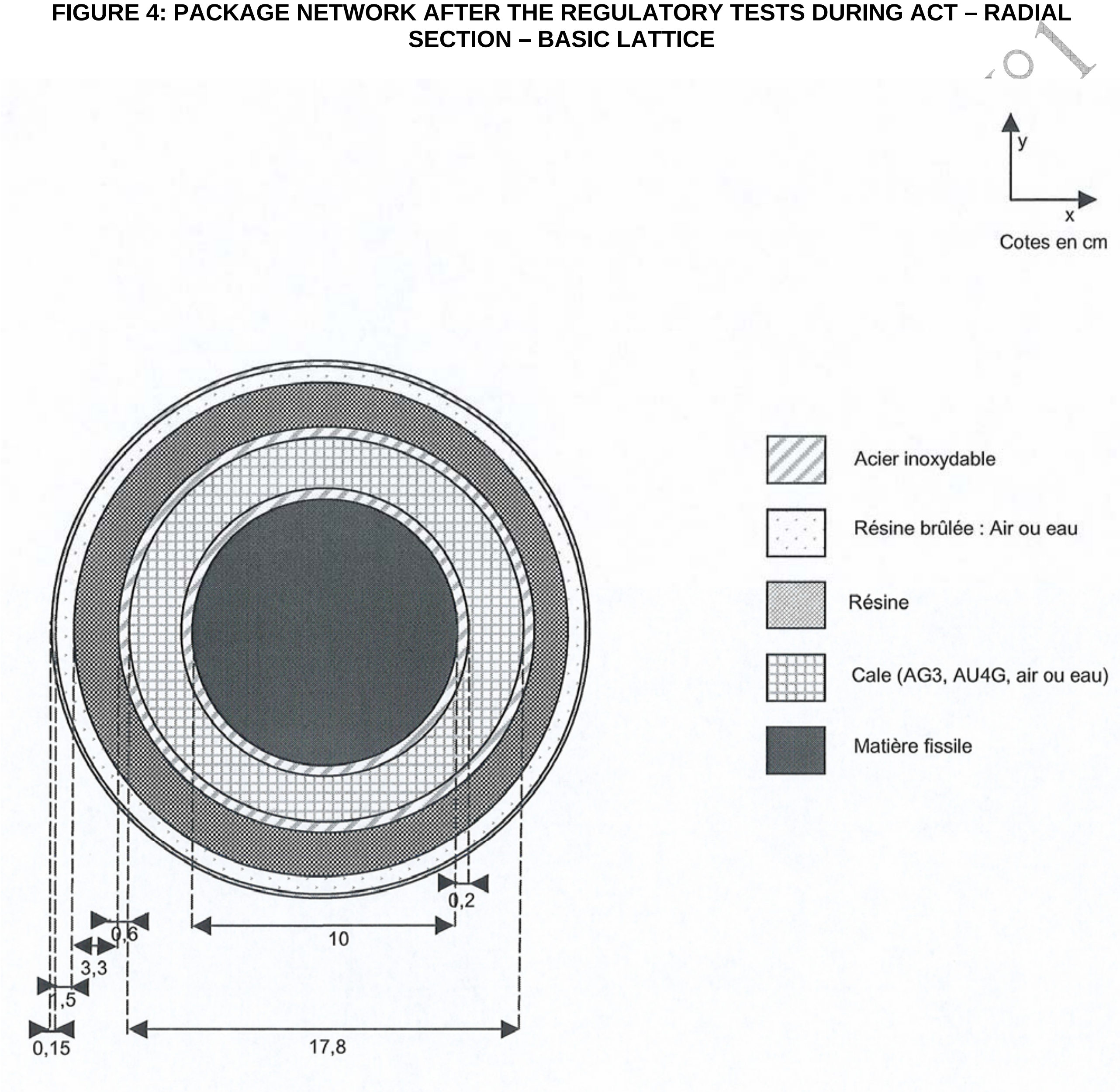
Cotes en cm	Dimensions in cm
Eau	Water
Acier inoxydable	Stainless steel
Résine brûlée: eau	Burnt resin: water
Résine	Resin
Cale (AG3, AU4G, air ou eau)	Chock (AG3, AU4G, air or water)
Matière fissile	Fissile material

FIGURE 3: PACKAGE NETWORK AFTER THE REGULATORY TESTS DURING ACT – AXIAL SECTION – BASIC LATTICE



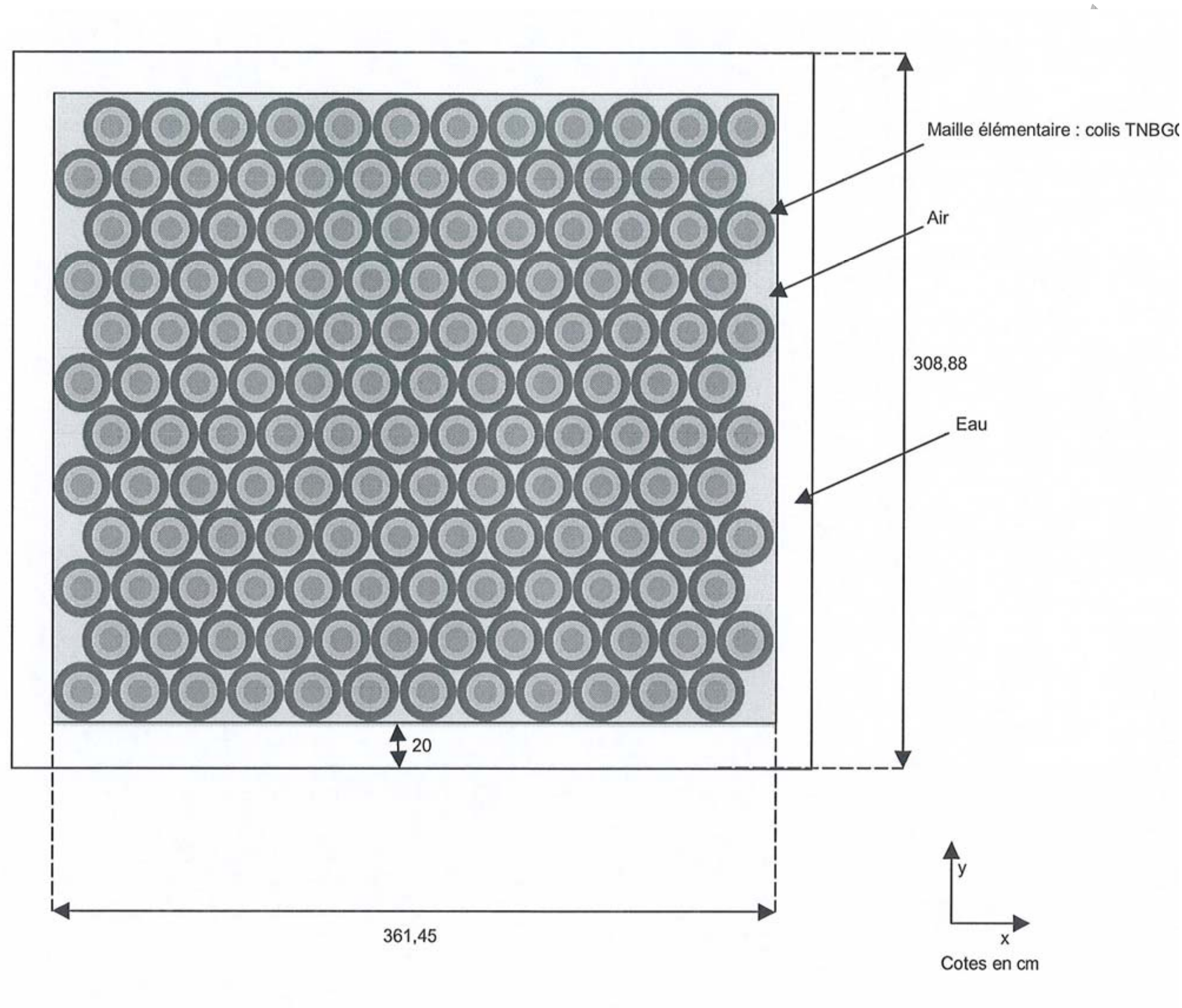
Matière fissile sous forme cylindrique	Fissile material in cylindrical form
Cotes en cm	Dimensions in cm
Eau	Water
Acier inoxydable 10 mm	10 mm stainless steel
Résine brûlée: Air ou eau	Burnt resin: air or water
Résine	Resin
Cale (AG3, AU4G, air ou eau)	Chock (AG3, AU4G, air or water)
Matière fissile	Fissile material
Epaisseur 0,15	0.15 thick

FIGURE 4: PACKAGE NETWORK AFTER THE REGULATORY TESTS DURING ACT – RADIAL SECTION – BASIC LATTICE



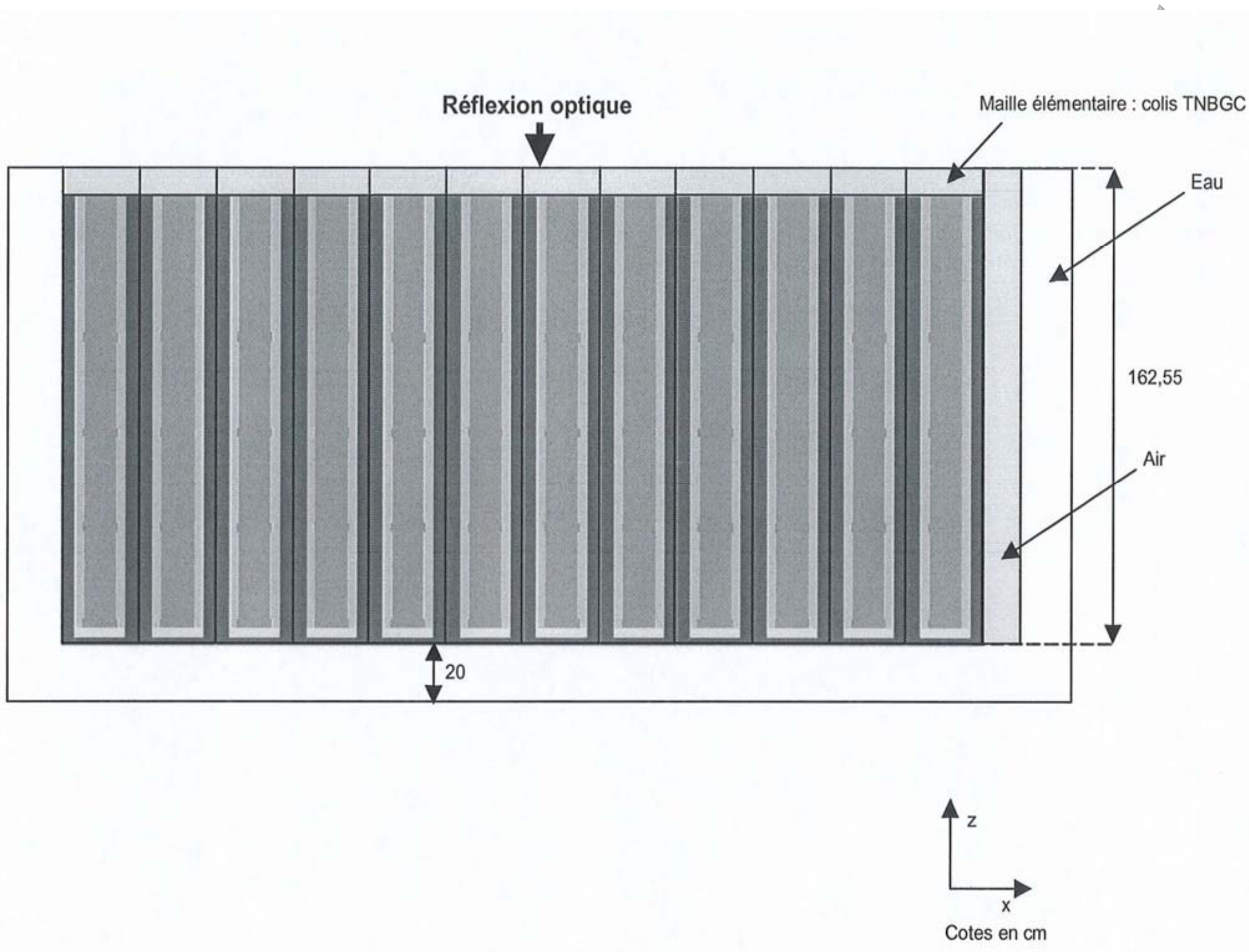
Cotes en cm	Dimensions in cm
Acier inoxydable	Stainless steel
Résine brûlée: Air ou eau	Burnt resin: air or water
Résine	Resin
Cale (AG3, AU4G, air ou eau)	Chock (AG3, AU4G, air or water)
Matière fissile	Fissile material

FIGURE 5: FINISHED 5N PACKAGE NETWORK WITH N=50 DURING ACT – RADIAL SECTION



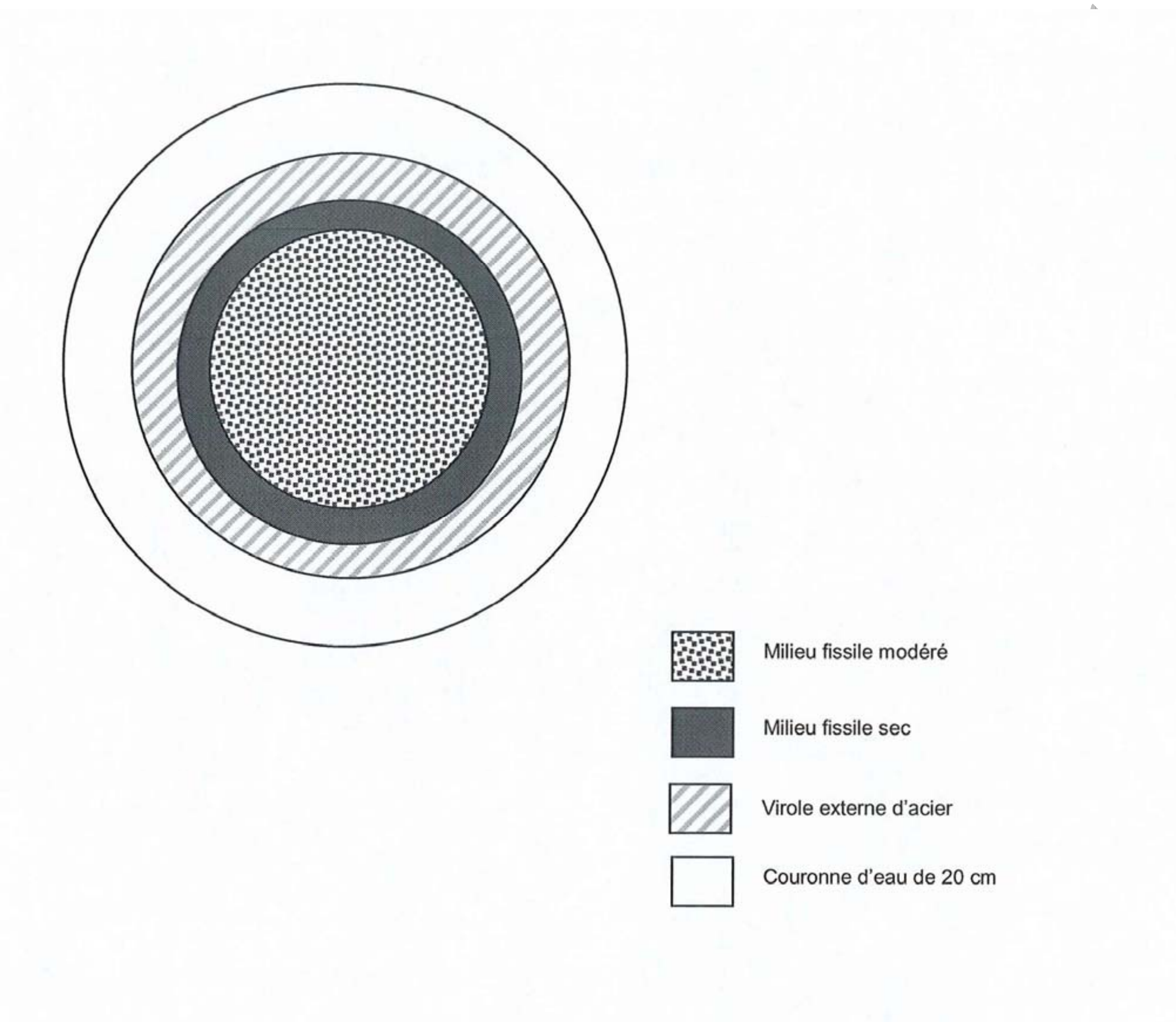
Maille élémentaire: colis TNBGC	Basic lattice: TNBGC package
Air	Air
Eau	Water
Cotes en cm	Dimensions in cm

FIGURE 6: FINISHED 5N PACKAGE NETWORK WITH N=50 DURING ACT – AXIAL SECTION



Réflexion optique	Optical reflectivity
Maille élémentaire: colis TNBGC	Basic lattice: TNBGC package
Eau	Water
Air	Air
Cotes en cm	Dimensions in cm

FIGURE 7: AIR TRANSPORT – RADIAL SECTION



Milieu fissile modéré	Moderated fissile medium
Milieu fissile sec	Dry fissile medium
Virole externe d'acier	Outer steel shell
Couronne d'eau de 20 cm	20 cm ring of water

CEA

NUCLEAR ENERGY MANAGEMENT
Technical and Project Assistance Department
CEA transport packaging section

Classification: 7.4.1	Page 1/33
Reference: 160 EMBAL PFM NOT 06001518	Index A
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Document purpose: Criticality study of the TN-BGC 1 packaging loaded with content no.26: U-ZrH_2 medium.	CEA/DEN/CAD/DTAP/SET DO 785 06/10/06 06PPFM001518

Application scope and summary

The purpose of this study is to revise the [DA4] reference document in order to:

- justify the criticality-safety of the package loaded with fuel element rods 103, 105, 107 and 117,
- take the presence of carbon elements from the wood into account in the moderation of the fissile medium,

during air transport and to

- study the influence of the aluminium alloy (AU4G, AG3) strut modelling during non-air transport.

APPENDICES (included in the document and, therefore part of its overall paging)			ATTACHMENTS (formalism, identification and independent paging)		
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Summary

The purpose of this study is to check that the criticality-safety of the TN-BGC 1 packaging loaded with content no.26 (U-ZrH2) is not brought into question when:

- aluminium alloy (AU4G, AG3, water and air) strut modelling during non-air transport,
- fuel elements 103, 105, 107 and 117 during air transport,
- and carbon elements from wood as the moderating medium of the fissile medium during air transport, are taken into account.

This study is carried out using the results of the [DA4] reference study relating to non-air and air transport.

- The form of the fissile medium is deemed to be either a heterogeneous U-ZrH2 rod network or homogeneous, moderated by an undefined quantity of water.

The general conclusions of the study are as follows:

- the study of fuel element rods 103, 105, 107 and 117 during air transport does not bring the sub-criticality of the TNBGC 1 packing into question,
- the carbon elements from wood lead to a drop in reactivity during air transport when they are taken into account in the moderation of the fissile medium,
- aluminium alloy (AU4G or AG3) strut modelling was studied. Modelling chocks using water leads to maximum reactivity and does not bring the criticality-safety of the packaging into question during non-air transport.

1 PURPOSE

The purpose of this study is to check that the sub-criticality of the packaging for content no.26 (U-ZrH2) (See [DA5]) is not brought into question when:

- aluminium alloy (AU4G, AG3) strut modelling during non-air transport,
- fuel elements 103, 105, 107 and 117 during air transport,
- and carbon elements from wood as the moderating medium of the fissile medium during air transport, are taken into account.

This study is carried out using the results of the [DA4] reference study relating to non-air and air transport.

The form of the fissile medium is deemed to be either a heterogeneous U-ZrH_2 rod network or homogeneous, moderated by an undefined quantity of water.

2 APPLICABLE DOCUMENTS

- [DA1] File justifying the safety of the TN-BGC-1 packaging - Analysis of the package criticality-safety - EMB TN-BGC PBC DS CA000001B.
- [DA2] CRISTAL_V0/ND12/A - SEC/T/02.242 Index A - SERMA/LEPP/RT/02.3116/A - SPRC/LECy/02.320/0 - CRISTAL form version VO.2 identification manual - 01/04/2003.
- [DA3] Note SEC/T/02.086 - CRISTAL form (version V0) standard APPOLLO2-MORET 4 process qualification results.
- [DA4] Criticality-safety of package model TN-BGC 1 loaded with content no.26: TRIGA fuel - EMB TNBGC PBC DJS CA000386 A.
- [DA5] TN-BGC 1 packaging - certificate of approval F/313/B(U)F-96 T (Haa)

3 REFERENCE DOCUMENTS

- [DR1] Radioactive material transport regulations - 1996 revised issue, Safety standard collection no.TS-R1
- [DR 2] Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material - Safety Guide No.TS-G1.1 (ST2).

4 TN-BGC 1 PACKAGING CHARACTERISTICS

The characteristics of the packaging come from reference [DA1].

The packaging comprises a parallelepiped cage, inside of which is attached a cylindrical body fitted with a locking system and a cover.

The dimensions of the packaging are as follows:

- cross-section of the cage: 600 x 600 mm²,
- overall height of the cage: 1,821 mm,
- diameter of the working section of the body: 295 mm,
- diameter of the cover: 466 mm,
- overall height of the body fitted with the cover: 1,808 mm.

The maximum weights of the packaging are:

- empty: 280 kg
- full: 396 kg

4.1 CAGE

The 30 x 30 mm², 2 mm thick structure of the cage is made of aluminium tubes.

4.2 BODY

The cavity with an effective diameter of 178 mm and effective length of 1,475 mm is made from a 6 mm thick stainless steel shell (guaranteeing the essential radial gamma shielding) and an 8 mm thick base also made of stainless steel.

A second, 1.5 mm thick, stainless steel shell with a 292 mm internal diameter forms a space around the first shell which is filled with hydrogen and boron-loaded resin (minimum thickness of 48 mm). Said resin acts as a neutron absorber and active thermal insulation.

The base is completed, from the inside to the outside, with a 25 mm high yield strength steel diffuser plate, a 24 mm layer of resin, an intermediate base, a wooden shock absorbing disc and a sheet of stainless steel.

The plug is machined in a 92 mm thick stainless steel disc. Its periphery comprises a 20 mm ledge which rests on the body flange. Said plug is held in position by a bronze compression ring which is tightened in the stainless steel bayonet locking collar.

4.3 COVER

There is a shock absorbing cover over the top of the body head-end piece and the locking system.

It comprises two stainless steel sheet metal units. The one nearest to the body is filled with resin, the other one is filled with wood.

4.4 INTERNAL ARRANGEMENT

The packaging is loaded with both an internal stainless steel packing container loaded with radioactive material and internal protection devices. The container is a TN 90.

E1 and E2 chocks are used to attach the packing container in the packaging cavity. Said chocks are made of AU4G aluminium.

5 CONTENT CHARACTERISTICS

The content studied is content no.26 on the certificate of approval, namely a U-ZrH_2 medium in a 130 mm internal diameter, 2 mm thick container. The physical quantities that can be transported are:

- 10 standard fuel elements,
- 73 thin fuel elements.

The characteristics of TRIGA fuel are as follows:

TYPE	**Weight %**				**U-ZrH_2 density**	**Maximum U weight during transport**	
	Uranium*	**ZrH**	**Zr**	**H**		**Air (kg)**	**Non-air (kg)**
Standard fuel elements							
103	8	92	90.0110	1.9890	6.04	1.1	9
105	12	88	86.0975	1.9025	6.22	1.7	14
107	12	88	86.0975	1.9025	6.22	1.7	14
117	21	79	77.2921	1.7079	6.64	3.3	27
119	31	69	67.5083	1.4917	7.24	5.3	43
Thin fuel elements							
424	47	53	51.8542	1.1458	8.4	6.6	76

*U^{235} enrichment = 20%

TRIGA rods are made of U-ZrH_x (x varying between 0 and 2). There are two types of fuel:

- Standard TRIGA fuel elements: 3.63 cm in diameter and 12.7 cm in height,
- Thin TRIGA fuel elements: 1.29 cm in diameter and 18.6 cm in height.

A 6.35 mm diameter hole is made in the centre of the standard fuel elements.

The fuel elements have stainless steel cladding.

6 NON-AIR TRANSPORT

6.1 CALCULATION HYPOTHESES

The calculation hypotheses for this study are based on the safety analysis report in reference [DA1]. They are repeated later in this document.

6.1.1 Package damage during ACT

The damage to packages following the regulatory drop tests and the thermal test during ACT is reiterated in reference [DA1]. A 15 mm reduction in the thickness of the neutron-absorbing resin biological shielding is considered (the radial thickness of the biological protection varies between 48 mm (NTC) and 33 mm (ACT) following tests during ACT).

Penetration of water into all the free spaces in both the cavity and the packaging and between the packages in the event of a stacked packaging configuration is also considered.

The packaging cage is not taken into account during accident conditions for a worst case scenario.

6.1.2 Package modelling

The package calculation model is shown below:

Radial section of the packaging body (from the inside to the outside)

- 6 mm thick, stainless steel shell with an internal diameter of 178 mm;
- 48 mm thick, boron-loaded resin (burnt to a depth of 15 mm in an accident situation);
- 1.5 mm thick, stainless steel shell.

Plug

- 92 mm thick stainless steel.

Base (from the inside to the outside)

- 33 mm thick, stainless steel shell and diffuser plate;
- 24 mm thick resin;
- 1.5 mm thick, stainless steel shell.

In accordance with the hypotheses of the note in reference [DA4], the internal height of the packaging cavity is taken to equal to 1,475 mm and the diameter of the fissile material (U-ZrH_2 - water) is taken to equal to 130 mm whereas the actual diameter of the container is 120 mm. The thickness of the steel packing container shell is taken to equal to 2 mm.

A study was carried out to check whether it is better to assume that the free space between the internal container and the inner shell of the packaging is totally comparable to AG3, AU4G, water or air in order to determine the worst case scenario modelling of the chocks.

The fissile material is axially centred in the internal container.

The chemical composition of the materials used is given in table 1.

6.1.3 Isolated package

The isolated package is surrounded (both radially and axially) by a 20 cm ring of water to guarantee neutron reflection. The internal cavity of the packaging in which the fissile material is modelled and the burnt resin are filled with density 1 water for a worst case scenario.

The MORET 4 model of the TN-BGC 1 package is shown in figures 1 and 2.

6.1.4. Package networks

During NTC, normal transport conditions, the study is concerned with 5N packages in the package conditions from which result the maximum multiplication of neutrons with the following hypotheses:

- the unit formed by the 5N packages is surrounded by a 20 cm ring of water,
- the state of the packages is that noted after the tests under normal transport conditions (undamaged boron-loaded resin, presence of the cage guaranteeing a regular 60 cm gap between packages).

During ACT, accident conditions in transport, the study is concerned with 2N packages in the package conditions from which result the maximum multiplication of neutrons with the following hypotheses:

- moderation is optimal and the unit formed by the 2N packages is surrounded by a 20 cm ring of water,
- the state of the packages is that noted after the tests under normal transport conditions followed by tests under accident conditions in transport (burnt resin and no cage).

Both studies are covered by a single hypothesis by considering an infinite network of packages after the tests simulating normal transport conditions followed by tests simulating accident conditions in transport. The reflection conditions are deemed total at the radial and axial limits of the package.

The internal cavity of the packaging in which the fissile material is modelled is filled with density 1 water for a worst case scenario so as to maximise the fissile medium reflection. In addition, the space left by the burnt resin and that between the packages is filled with air.

The MORET 4 model of an infinite TN-BGC 1 package network is shown in figures 3 and 4.

6.2 CALCULATION METHODS

6.2.1 Calculation codes

The calculation codes used in this study are the codes developed jointly by the IRSN and the CEA. This calculation process has been adopted by French industry.

k_{eff} is calculated in two stages using the CRISTAL criticality form [DA2] standard process:

- study of the fissile medium with APOLLO2,
- calculation of k_{eff} using the MORET 4 code.

6.2.1.1 Studying fissile media using the APOLLO2 code

When modelling the fuel, the fissile medium is deemed heterogeneous. The fissile medium is modelled in the form of U-ZrH_2 rods and the moderation ratio $V_{mod}/V_{U\text{-}ZrH2}$ varied from 0 to the value corresponding to the fissile medium filling the entire cell cavity (130 mm fissile diameter).

When modelling U-ZrH_2 fuel, the fissile medium diameter is taken as being equal to the diameter of a fuel element:

- 37.3 mm (standard fuel element),
- 14 mm (thin fuel element),

The height of the fissile medium is taken as being equal to the total height of the fuel element:

- 752 mm (standard fuel element),
- 770 mm (thin fuel element).

The APOLLO2 code inputs are shown in table 1 (structure materials) and in section 5 (fissile media).

6.2.1.2 Calculating k_{eff} using the MORET 4 MONTE-CARLO code

The k_{eff} effective multiplication factor is obtained using the 3D multi-group MORET 4 Monte-Carlo code. Said code uses the cross-sections generated by the APOLLO2 code:

- for homogenised fissile media
- for structure environments.

Calculation uncertainty is taken as being less than or equal to 200 pcm for all the calculations made in this study.

6.2 Permissibility criteria

The permissibility criteria adopted are as follows:

- keff + 3σ ≤ 0.950, whatever the uncertainty, for an isolated package under accident conditions in transport accumulated with normal transport conditions,

- keff + 3σ ≤ 0.980, whatever the uncertainty, for an infinite package network under accident conditions in transport, the moderation between the packages is not defined.

A bias of an additional 1,000 pcm linked to the uncertainties for the aluminium cross-sections is taken into account regarding the usual permissibility criteria when the importance of the aluminium alloy chocks is demonstrated.

6.3 CRITICALITY-SAFETY STUDIES

This involves calculating the influence on the reactivity of a variation of the nature of the aluminium alloy chocks for fuel rods 117, 119 and 424 (enclosure fissile media defined in section 6.1 of the [DA4] reference study). Hence, 4 cases of chock modelling were considered: AG3, AU4G, WATER and AIR.

Different studies were carried out using the hypotheses (nature of the network, fissile medium modelling) of the studies inherent to each of the media. The hypotheses adopted for the packaging modelling are from the study in reference [DA4].

The search for optimum reactivity for each content is carried out in accordance with the following parameters: moderation ratio ($V_{mod}/V_{U\text{-}ZrH2}$) and aluminium alloy chock composition. The heterogeneous fissile material is contained in a 130 mm fissile diameter (2 mm thick internal arrangement).

The maximum reactivities of k_{eff} + 3σ (σ≤ 200 pcm) obtained are as follows:

		Nature of the chocks			
Reference contents	**Configuration**	**AG3**	**AU4G**	**WATER**	**AIR**
117 Table 2	Isolated package during ACT	0.749	0.752	**0.822**	0.704
119 Table 3		0.769	0.768	**0.831**	0.719
424 Table 4		0.754	0.753	**0.822**	0.704
117 Table 2	Package network during ACT	0.880	0.879	**0.917**	0.860
119 Table 3		0.905	0.905	**0.932**	0.881
424 Table 4		0.902	0.902	**0.921**	0.872

Maximum reactivity is obtained for “WATER” struts in an isolated package and a package network.

Modelling aluminium alloy struts leads to a drop in reactivity.

The results do not bring the sub-criticality of the TN-BGC 1 packaging into question for non-air transport of content no.26 ([DA5]).

6.4 INSULATION SYSTEM

The following elements constitute the insulation system to be guaranteed:

- packaging: geometry (maximum diameter of 178 mm), materials (the inner and outer packaging shells are made of stainless steel), materials used, composition and thickness of the neutron-absorbing boron-loaded resin (boron weight content: 1.43%, and thickness of burnt resin: 48 mm during NTC and 33 mm during ACT),
- stainless steel internal arrangement: 130 mm diameter and 2 mm thick,
- chock system defining the radial position taken up by the fissile material,
- fissile material: checking the criticality by limiting the weight and composition of the fissile medium.

6.5 CONCLUSION

The [DA4] reference study demonstrates the TN-BGC1 packaging criticality-safety by considering:

- a number of elements greater than that which the container can hold,
- a container diameter equal to 130 mm throughout its height whereas its actual diameter is 120 mm.
- a fissile material weight greater than reality as the study deals with a height of 1,475 mm, whereas the TN90 has an effective height of 1,397 mm.

However, the study in reference [DA4] is carried out by considering water strut modelling as a hypothesis. The justification for this hypothesis is provided by this study. Modelling the struts in aluminium alloy (AU4G and AG3) or air does not bring the criticality-safety of the TN-BGC 1 package model loaded with content no.26: U-ZrH_2 into question. Maximum reactivity is obtained for water strut modelling. The results of the [DA4] reference study are not brought into question.

The criticality-safety of the packaging is observed for content no.26 ([DA5]) under both normal and accident conditions in transport for an infinite number N of packages.

7 AIR TRANSPORT

7.1 CALCULATION HYPOTHESES

The demonstration of the sub-criticality of the package loaded with fuel element rods 103, 105, 107 and 117 as well as the taking into account of the carbon elements from wood as the moderating medium is studied using the conservative configuration of the [DA4] reference study (configuration c). In this configuration, moderation of the fissile material comes from:

- wood: the weight of water contained in the wood is less than 1,700 g,
- boxes: the boxes used for protecting the fuel are compared to water with a weight of 200 g per box for standard fuel elements and a weight of 84.2 g for thin fuel elements for a worst case scenario.

The fuel is treated by a heterogeneous moderation:

- a proportion of the total weight of fissile material is moderated by water or a mixture of water and carbon (maximum weight of carbon: 2,500 g) when studying the influence of wood in fissile medium moderation.
- the rest of the total weight is modelled around the moderated sphere in the shape of a dry shell, playing the role of a reflector.

The configuration has 5 concentric spherical areas:

- the fuel moderated by a mixture of water and carbon,
- the dry fuel,
- the steel (340 kg corresponding to the steel structure of the packaging),
- the resin,
- a 20 cm ring of water.

The [DA4] reference study shows that the resin has no influence. The model is simplified by not including the corresponding area.

The MORET 4 model for air transport is shown in figure 5.

7.2 CALCULATION METHODS

7.2.1 Calculation codes

The calculation codes used in this study are the codes developed jointly by the IRSN and the CEA. This calculation process has been adopted by French industry.

k_{eff} is calculated in two stages using the CRISTAL criticality form [DA2] standard process:

- study of the fissile medium with APOLLO2,
- calculation of k_{eff} using the MORET 4 code.

7.2.1.1 Studying the fissile medium using the APOLLO2 code

When modelling the fuel, the fissile medium is deemed homogeneous.

The fissile medium is modelled by a mixture (U-ZrH_2-water) and the moderation ratio is varied.

The GUI Cigales version V 3.0 is used for generating the set of APOLLO2 data.

The APOLLO2 code provides:

- k_∞, the infinite multiplication factor, and B^2_m, the material buckling,
- a set of tamper-proof, homogenised cross-sections with 172 energy groups for the fissile media,
- a set of cross-sections with 172 energy groups for the structure environments.

The APOLLO2 code inputs are shown in table 1 (structure materials) and in section 5 (fissile media).

7.2.1.2 Calculating k_{eff} using the MORET 4 MONTE-CARLO code

The k_{eff} effective multiplication factor is obtained using the 3D multi-group MORET 4 Monte-Carlo code. Said code uses the cross-sections generated by the APOLLO2 code:

- for homogenised fissile media
- for structure environments.

Calculation uncertainty is taken as being less than or equal to 200 pcm for all the calculations made in this study.

7.2.2 Permissibility criteria

The permissibility criteria adopted is keff + 3σ ≤ 0.950, whatever the uncertainty.

7.3 AIR TRANSPORT

7.3.1 Study of fuel elements 103, 105, 107 and 117

The [DA4] air transport reference study is limited to fuel elements 119 and 424. We would reiterate that these fuel elements were selected following a comparison of B^2_m, this demonstration does not apply outside of a check using geometry as is the case during air transport. The revising of this study shall consist of proving the criticality-safety of fuel elements 103, 105, 107 and 117 using the conservative configuration that can be envisaged during air transport (configuration c in [DA4] - hypotheses reiterated in section 7.1).

The weight of water under consideration is 3,700 g and corresponds to the sum of the weight of 10 boxes (200 g for standard fuel elements) compared to water and the weight of 1,700 g of water equivalent to the wood.

The maximum reactivities (k_{eff} + 3σ) obtained are as follows:

Fuel element	Weight of uranium (kg)	Weight of water (kg)	Calculation detail reference	k_{eff} + 3σ (σ≤200 pcm)
103	1.1		Table 5	0.698
105-107	1.7	3.7	Table 6	0.802
117	3.3		Table 7	0.913

The criticality-safety of the TN-BGC 1 package loaded with fuel elements 103, 105, 107 and 117 is not brought into question during air transport.

7.3.2 Taking the carbon elements from wood into consideration

The carbon elements from wood participate in the moderation of the fissile medium. This study takes this parameter into account using the configuration adopted in the [DA4] reference air transport study in section 6.5.2. The calculation hypotheses are reiterated in section 7.1. We would reiterate that [DA4]

highlights the fact that in order to be able to observe the criticality-safety criterion, the package cannot transport more than:

- 6 standard fuel elements
- or 23 thin fuel elements.

The maximum reactivities (k_{eff} + 3σ) obtained are as follows:

Fuel element	**Calculation detail reference**	**Weight of water under consideration (g)**	**Weight of carbon (g)**	**k_{eff} + 3σ**
119	Table 8	2,900 (1,700+6x200)	**0** 500 2,000 2,500	**0.939** 0.931 0.908 0.898
424	Table 9	3,640 (1,700+23x84.2)	**0** 500 1,500 2,500	**0.942** 0.932 0.917 0.901

Taking the carbon elements from wood into consideration leads to a lower reactivity. The presence of the carbon from wood does not bring the criticality-safety of the TN-BGC1 packaging into question during air transport.

7.4 INSULATION SYSTEM

The criticality-safety is controlled by:

- limiting the weight;
  → 6 x standard 119 fuel elements,
  → 10 x 103, 105, 107 or 117 fuel elements,
  → 23 x thin 424 fuel elements
- the fissile medium composition,
- moderation and the nature of the moderator.

7.5 CONCLUSION

The [DA4] reference study proves the criticality-safety of the TN-BGC1 packaging during air transport by considering a reflector weight greater than that of reality as it is equal to the weight of the empty packaging whereas said packaging contains resin and wood amongst other things.

However, the [DA4] reference study is limited to fuel elements 119 as regards standard fuel elements and fuel elements 424 as regards thin fuel elements during air transport. On the other hand, it does not take the carbon elements from wood into account in the moderation of the fissile medium. This study provides the necessary complementary justification.

After verification, the hypothesis of the [DA4] reference study as regards the choice of reference fuel (fuel element 119) is conservative. In fact, the air transport study of fuel element rods 103, 105, 107 and 117 does not bring the criticality-safety of the TN-BGC 1 packaging into question and maximum reactivity is obtained for fuel element 119. On the other hand, the taking into account of the carbon in the moderation of the fissile medium does not bring the criticality-safety of the TN-BGC 1 package model loaded with content no.26: U-ZrH_2 into question. It even has a tendency to reduce reactivity. The [DA4] reference study results are not brought into question.

8. QUALIFICATION

The purpose of this section is to set out a CRISTAL form qualification state for the content studied, namely U-ZrH_2 media 20% U^{235} (the description of the content is repeated in section 5) reflected by different materials such as aluminium (chocks) or steel (upper plug).

The APOLLO2-MORET4 calculation diagram does not lead to a significant underestimation of the reactivity for a homogeneous environment or one that takes the form of uranium-bearing rods.

As far as the presence of aluminium alloy struts is concerned, the APOLLO 2 - MORET 4 code, is qualified when the aluminium is used as a reflector and for a rapid spectrum. The fact that the current qualification of the APOLLO - MORET 4 process is missing for the aluminium in a thermal spectrum means that a margin will have to be taken in order to cover the uncertainties concerning the cross-sections of the aluminium. An additional bias of 1,000 pcm is taken into account as regards the usual permissibility criteria when the significance of the aluminium alloy chocks is demonstrated.

Lastly, for the experiments using steel as a reflector, significant deviations between the calibration process and the standard process are noted. Said deviations can be attributed to the multi-group treatment of the cross sections of iron (predominantly isotope 56). This overestimation increases with the thickness of the reflector.

In conclusion, the minimum 1,000 pcm margin (in the presence of aluminium alloy struts) of this study can be deemed as safe.

9 GENERAL CONCLUSION

The general conclusions of the study are as follows:

- the study of fuel element rods 103, 103, 107 and 117 does not bring the criticality-safety of the TNBGC 1 packaging during air transport into question,
- carbon elements from wood lead to a drop in reactivity during air transport when they are taken into account in the moderation of the fissile medium.
- modelling of the aluminium alloy (AG3 and AU4G) struts has been studied and does not bring the criticality-safety of the packaging into question during air transport, reactivity is at its maximum when the chocks are replaced by water (configuration studied in the [DA4] reference study).

TABLES

TABLE 1: CHEMICAL COMPOSITION OF STRUCTURE MATERIALS

Environment	Elements	Concentration (atoms/cm³ x 1024)
Stainless steel d=7.87	Fe Cr Ni	$6.134.10^{-2}$ $1.647.10^{-2}$ $8.107.10^{-3}$
Neutron-absorbing resin d=1.186	H C O B_{nat}	$4.0616.10^{-2}$ $2.3803.10^{-2}$ $2.3580.10^{-2}$ $9.4597.10^{-4}$
AG3 d=2.65	Al Mg Cr Zn Ti	$5.6780.10^{-2}$ $2.1011.10^{-3}$ $1.2277.10^{-4}$ $4.8810.10^{-5}$ $6.6660.10^{-5}$
AU4G d=2.67	Si Fe Cu Mn Mg Cr Zn Al	$2.8625.10^{-4}$ $2.0154.10^{-4}$ $1.0121.10^{-3}$ $2.0487.10^{-4}$ $4.6309.10^{-4}$ $3.0923.10^{-5}$ $6.2878\text{-}10^{-5}$ $5.5451.10^{-2}$
Air	N O	$4.1985.10^{-5}$ $1.1263.10^{-5}$
Water	H O	$6.6721.10^{-2}$ $3.34279.10^{-2}$
Carbon (graphite) d=2.3	C	$1.1532.10^{-1}$

TABLE 2: NON-AIR TRANSPORT - FUEL ELEMENT 117 - INFLUENCE OF THE NATURE OF THE CHOCKS

The study configuration sets out:

- a 130 mm packing container corresponding to the fissile section,
- a fissile medium in the form of a 3.73 cm diameter rod network moderated by water.

→ In an isolated package: the burnt resin is replaced by water

Number of elements	**Moderation ratio V_{mod} / V_{U-ZrH2}**	**Height of the heterogeneous fissile+water mixture (in cm)**	**K_{eff} + 3σ (σ≤200 pcm)**			
			AG3	**AU4G**	**WATER**	**AIR**
10	0.2147		**0.749**	**0.752**	**<u>0.822</u>**	**0.704**
9	0.3497	75.20	0.735	0.734	0.803	0.688
8	0.5184		0.723	0.724	0.786	0.674

→ In an infinite package network: the burnt resin is replaced by air

Number of elements	**Moderation ratio V_{mod} / V_{U-ZrH2}**	**Height of the heterogeneous fissile+water mixture (in cm)**	**K_{eff} + 3σ (σ≤200 pcm)**			
			AG3	**AU4G**	**WATER**	**AIR**
10	0.2147		**0.880**	**0.879**	**<u>0.917</u>**	**0.860**
9	0.3497	75.20	0.862	0.862	0.900	0.840
8	0.5184		0.850	0.849	0.880	0.821

TABLE 3: NON-AIR TRANSPORT - FUEL ELEMENT 119 - INFLUENCE OF THE NATURE OF THE CHOCKS

The study configuration sets out:

- a 130 mm packing container corresponding to the fissile section,
- a fissile medium in the form of a 3.73 cm diameter rod network moderated by water.

→ In an isolated package: the burnt resin is replaced by water

Number of elements	**Moderation ratio V_{mod} / V_{U-ZrH2}**	**Height of the heterogeneous fissile+water mixture (in cm)**	**K_{eff} + 3σ (σ≤200 pcm)**			
			AG3	**AU4G**	**WATER**	**AIR**
10	0.2147		**0.769**	**0.768**	**<u>0.831</u>**	**0.719**
9	0.3497	75.20	0.755	0.752	0.821	0.703
8	0.5184		0.745	0.747	0.814	0.680

→ In an infinite package network: the burnt resin is replaced by air

Number of elements	**Moderation ratio V_{mod} / V_{U-ZrH2}**	**Height of the heterogeneous fissile+water mixture (in cm)**	**K_{eff} + 3σ (σ≤200 pcm)**			
			AG3	**AU4G**	**WATER**	**AIR**
10	0.2147		**0.905**	**0.905**	**<u>0.932</u>**	**0.881**
9	0.3497	75.20	0.894	0.896	0.914	0.867
8	0.5184		0.886	0.881	0.909	0.859

TABLE 4: NON-AIR TRANSPORT - FUEL ELEMENT 424 - INFLUENCE OF THE NATURE OF THE CHOCKS

The study configuration sets out:

- a 130 mm packing container corresponding to the fissile section,
- a fissile medium in the form of a 1.4 cm diameter rod network moderated by water.

→ In an isolated package: the burnt resin is replaced by water

Number of elements	**Moderation ratio V_{mod} / V_{U-ZrH2}**	**Height of the heterogeneous fissile+water mixture (in cm)**	**K_{eff} + 3σ (σ≤200 pcm)**			
			AG3	**AU4G**	**WATER**	**AIR**
73	0.1812		**0.754**	0.752	0.820	**0.704**
65	0.3265	77	0.753	**0.753**	**0.822**	0.702
55	0.5677		0.750	0.751	0.814	0.698

→ In an infinite package network: the burnt resin is replaced by air

Number of elements	**Moderation ratio V_{mod} / V_{U-ZrH2}**	**Height of the heterogeneous fissile+water mixture (in cm)**	**K_{eff} + 3σ (σ≤200 pcm)**			
			AG3	**AU4G**	**WATER**	**AIR**
73	0.1812		0.896	0.900	**0.921**	**0.872**
65	0.3265	77	**0.902**	**0.902**	0.919	0.872
55	0.5677		0.896	0.894	0.916	0.868

TABLE 5: AIR TRANSPORT - FUEL ELEMENT 103

The weight of water under consideration is 3,700 g corresponding to the sum of 10 boxes (200 g) compared to water and the weight of 1,700 g of water equivalent to the wood.

Weight of moderated uranium (g)	Uranium concentration in the mixture [(U-ZrH_2)-H2O] (g/l)	Moderated fissile sphere shell radius (cm)	Dry shell uranium weight (g)	Dry fissile shell radius (cm)	Steel shell radius (cm)	$K_{eff} + 3\sigma$ ($\sigma \leq 200$ pcm)
750 1,000 **1,100**	142.5973 173.1017 **183.8265**	10.7883 11.1311 **11.2624**	350 100 **0**	**11.2624**	**21.8937**	0.656 0.687 **0.698**

TABLE 6: AIR TRANSPORT - FUEL ELEMENT 105-107

The weight of water under consideration is 3,700 g corresponding to the sum of 10 boxes (200 g) compared to water and the weight of 1,700 g of water equivalent to the wood.

Weight of moderated uranium (g)	Uranium concentration in the mixture [(U-ZrH_2)-H2O] (g/l)	Moderated fissile sphere shell radius (cm)	Dry shell uranium weight (g)	Dry fissile shell radius (cm)	Steel shell radius (cm)	$K_{eff} + 3\sigma$ ($\sigma \leq 200$ pcm)
1,000 1,500 **1,700**	198.1305 262.3726 **284.0428**	10.6411 11.0925 **11.2631**	700 200 **0**	**11.2631**	**21.8938**	0.754 0.784 **0.802**

TABLE 7: AIR TRANSPORT - FUEL ELEMENT 117

The weight of water under consideration is 3,700 g corresponding to the sum of 10 boxes (200 g) compared to water and the weight of 1,700 g of water equivalent to the wood.

Weight of moderated uranium (g)	Uranium concentration in the mixture [(U-ZrH_2)-H2O] (g/l)	Moderated fissile sphere shell radius (cm)	Dry shell uranium weight (g)	Dry fissile shell radius (cm)	Steel shell radius (cm)	$K_{eff} + 3\sigma$ ($\sigma \leq 200$ pcm)
2,000 2,500 **3,300**	388.9746 454.5206 **543.2972**	10.7071 10.9505 **11.3187**	1,300 800 **0**	**11.3187**	**21.9086**	0.897 0.903 **0.913**

TABLE 8: AIR TRANSPORT - FUEL ELEMENT 119 - INFLUENCE OF THE CARBON ELEMENTS FROM WOOD

The weight of water under consideration is 2,900 g corresponding to the sum of 6 boxes (200 g) compared to water and the weight of 1,700 g of water equivalent to the wood.

Weight of carbon under consideration: 0 kg

Weight of moderated uranium (g)	Uranium concentration in the mixture [(U-ZrH2)-H2O] (g/l)	Moderated fissile sphere shell radius (cm)	Dry shell uranium weight (g)	Dry fissile shell radius (cm)	Steel shell radius (cm)	$K_{eff} + 3\sigma$ ($\sigma \leq 200$ pcm)
800	245.2292	9.2005	4,500			0.884
1,000	298.3859	9.2835	4,300			0.892
2,000	526.7430	9.6779	3,300			0.925
3,000	**707.1349**	**10.0425**	**2,300**	**10.7936**	**21.7740**	**0.939**
4,000	853.2377	10.3825	1,300			0.938
4,500	916.3473	10.5444	800			0.937
5,300	1,006.2187	10.7936	0			0.936

Weight of carbon under consideration: 0.5 kg

Weight of moderated uranium (g)	Uranium concentration in the mixture [(U-ZrH2)-(H2O-GRAPH)] (g/l)	Moderated fissile sphere shell radius (cm)	Dry shell uranium weight (g)	Dry fissile shell radius (cm)	Steel shell radius (cm)	$K_{eff} + 3\sigma$ ($\sigma \leq 200$ pcm)
1,000	280.2097	9.4800	4,300			0.881
1,500	395.6183	9.6733	3,800			0.910
2,000	498.2177	9.8592	3,300			0.914
2,500	590.0281	10.0383	2,800			0.920
3,000	672.6664	10.2112	2,300	**10.9410**	**21.8505**	0.924
3,500	747.4417	10.3785	1,800			0.924
4,000	815.4252	10.5405	1,300			0.929
4,500	877.5021	10.6977	800			0.926
5,000	934.4101	10.8505	300			0.930
5,300	**966.3358**	**10.9401**	**0**			**0.931**

Weight of carbon under consideration: 2 kg						
Weight of moderated uranium (g)	**Uranium concentration in the mixture** [(U-ZrH2)-(H2O-GRAPH)] **(g/l)**	**Moderated fissile sphere shell radius (cm)**	**Dry shell uranium weight (g)**	**Dry fissile shell radius (cm)**	**Steel shell radius (cm)**	**$K_{eff} + 3\sigma$ ($\sigma \leq 200$ pcm)**
1,000	236.9146	10.0255	4,300	**11.3576**	**21.9130**	0.855
1,500	337.5560	10.1989	3,800			0.873
2,000	428.5882	10.3665	3,300			0.886
2,500	511.3248	10.5289	2,800			0.894
3,000	586.8502	10.6865	2,300			0.894
3,500	656.0679	10.8395	1,800			0.905
4,000	719.7364	10.9884	1,300			0.907
4,500	778.4973	11.1333	800			0.902
5,000	832.8969	11.2745	300			0.906
5,300	**863.6409**	**11.3576**	**0**			**0.908**

Weight of carbon under consideration: 2.5 kg						
Weight of moderated uranium (g)	**Uranium concentration in the mixture** [(U-ZrH2)-(H2O-GRAPH)] **(g/l)**	**Moderated fissile sphere shell radius (cm)**	**Dry shell uranium weight (g)**	**Dry fissile shell radius (cm)**	**Steel shell radius (cm)**	**$K_{eff} + 3\sigma$ ($\sigma \leq 200$ pcm)**
1,000	225.3104	10.1948	4,300	**10.8232**	**21.7813**	0.843
1,500	321.8126	10.3625	3,800			0.865
2,000	409.5109	10.5251	3,300			0.875
2,500	489.5576	10.6827	2,800			0.880
4,000	692.6429	11.1298	1,300			0.890
3,000	562.9122	10.8359	2,300			0.896
3,500	630.3802	10.9848	1,800			0.895
4,500	750.2803	11.2711	800			0.893
5,000	803.7894	11.4090	300			0.897
5,300	**834.0938**	**11.4901**	**0**			**0.898**

TABLE 9: AIR TRANSPORT - FUEL ELEMENT 424 - INFLUENCE OF THE CARBON ELEMENTS FROM WOOD

The weight of water under consideration is 3,640 g corresponding to the sum of 23 boxes (84.2 g) compared to water and the weight of 1,700 g of water equivalent to the wood.

Weight of carbon under consideration: 0 kg

Weight of moderated uranium (g)	Uranium concentration in the mixture [(U-ZrH2)-H2O] (g/l)	Moderated fissile sphere shell radius (cm)	Dry shell uranium weight (g)	Dry fissile shell radius (cm)	Steel shell radius (cm)	$K_{eff} + 3\sigma$ ($\sigma \leq 200$ pcm)
3,000	681.2351	10.1682	3,600			0.939
3,500	772.5566	10.2648	3,100			0.940
4,000	858.9112	10.3596	2,600			0.940
4,500	940.6934	10.4526	2,100	**10.8265**	**21.7821**	0.942
5,000	**1,018.2570**	**10.5441**	**1,600**			**0.942**
5,500	1,091.9202	10.6339	1,100			0.939
6,600	1,241.6237	10.8265	0			0.937

Weight of carbon under consideration: 0.5 kg

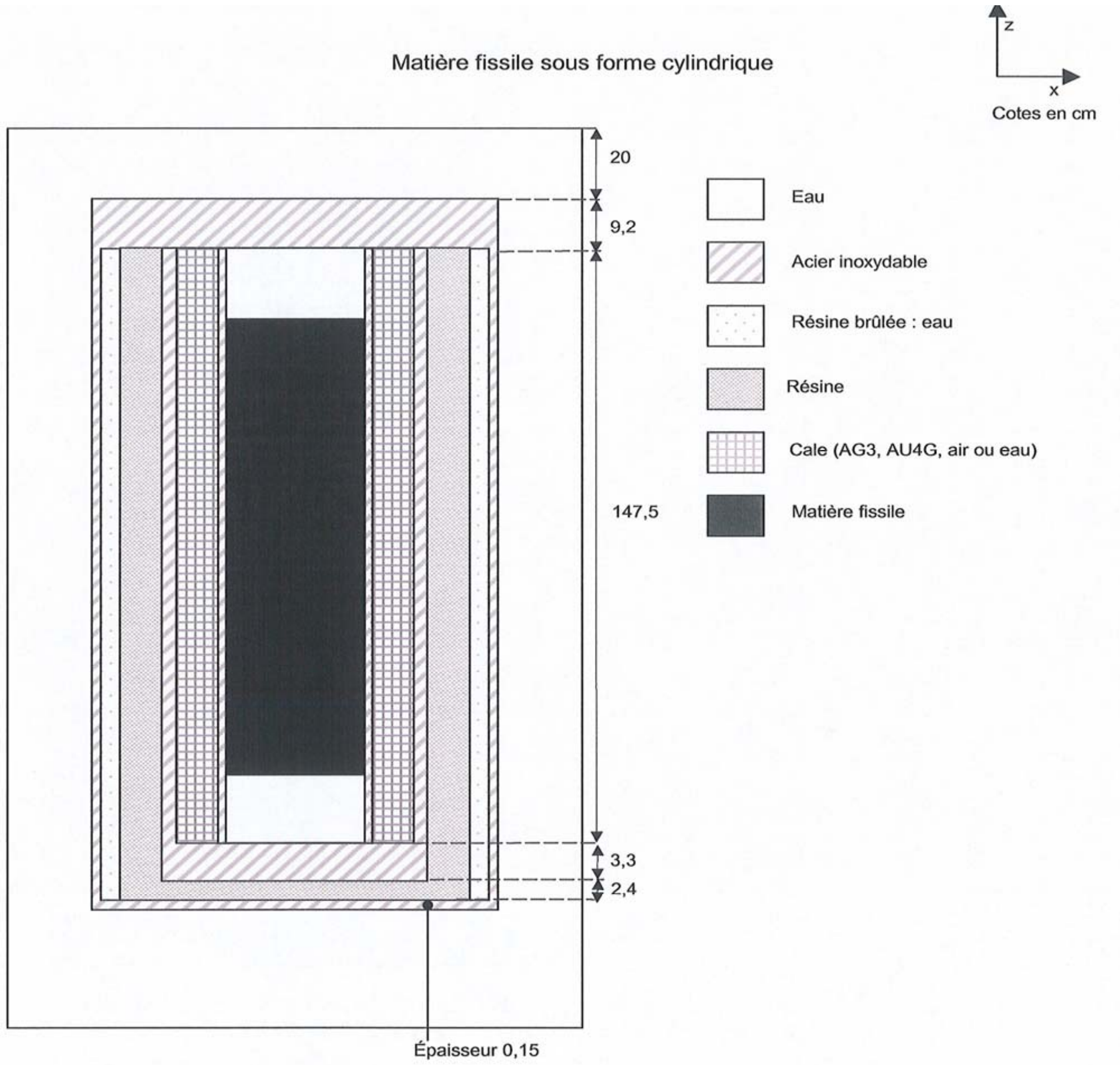
Weight of moderated uranium (g)	Uranium concentration in the mixture [(U-ZrH2)-(H2O-GRAPH)] (g/l)	Moderated fissile sphere shell radius (cm)	Dry shell uranium weight (g)	Dry fissile shell radius (cm)	Steel shell radius (cm)	$K_{eff} + 3\sigma$ ($\sigma \leq 200$ pcm)
3,000	649,1880	10.3329	3,600			0.930
3,500	737,1829	10.4264	3,100			0.930
4,000	820,6054	10.5183	2,600			0.931
4,500	**899,8026**	**10.6086**	**2,100**	**10.9721**	**21.8185**	**0.934**
5,000	975,0878	10.6974	1,600			0.931
5,500	1,046,7439	10.7848	1,100			0.932
6,600	1,192,8404	10.9721	0			0.923

Weight of carbon under consideration: 1.5 kg						
Weight of moderated uranium (g)	**Uranium concentration in the mixture** [(U-ZrH2)-(H2O-GRAPH)] **(g/l)**	**Moderated fissile sphere shell radius (cm)**	**Dry shell uranium weight (g)**	**Dry fissile shell radius (cm)**	**Steel shell radius (cm)**	**K_{eff} + 3σ** (σ≤200 pcm)
3,000	593.3615	10.6473	3,600			0.912
3,500	675.3385	10.7354	3,100			0.910
4,000	753.4045	10.8222	2,600			0.917
4,500	827.8329	10.9075	2,100	**11.2523**	**21.8910**	0.917
5,000	898.8722	10.9916	1,600			0.916
5,500	**966.7487**	**11.0744**	**1,100**			**0.917**
6,600	1,105.9363	11.2523	0			0.911

Weight of carbon under consideration: 2.5 kg						
Weight of moderated uranium (g)	**Uranium concentration in the mixture** [(U-ZrH2)-(H2O-GRAPH)] **(g/l)**	**Moderated fissile sphere shell radius (cm)**	**Dry shell uranium weight (g)**	**Dry fissile shell radius (cm)**	**Steel shell radius (cm)**	**K_{eff} + 3σ** (σ≤200 pcm)
3,000	546.3762	10.9441	3,600			0.897
3,500	623.0675	11.0276	3,100			0.898
4,000	696.3769	11.1099	2,600			0.896
4,500	766.5234	11.1909	2,100	**11.5192**	**21.9629**	0.900
5,000	833.7073	11.2708	**1,600**			**0.901**
5,500	898.1125	11.3496	1,100			0.901
6,600	1,030.8350	11.5192	0			0.899

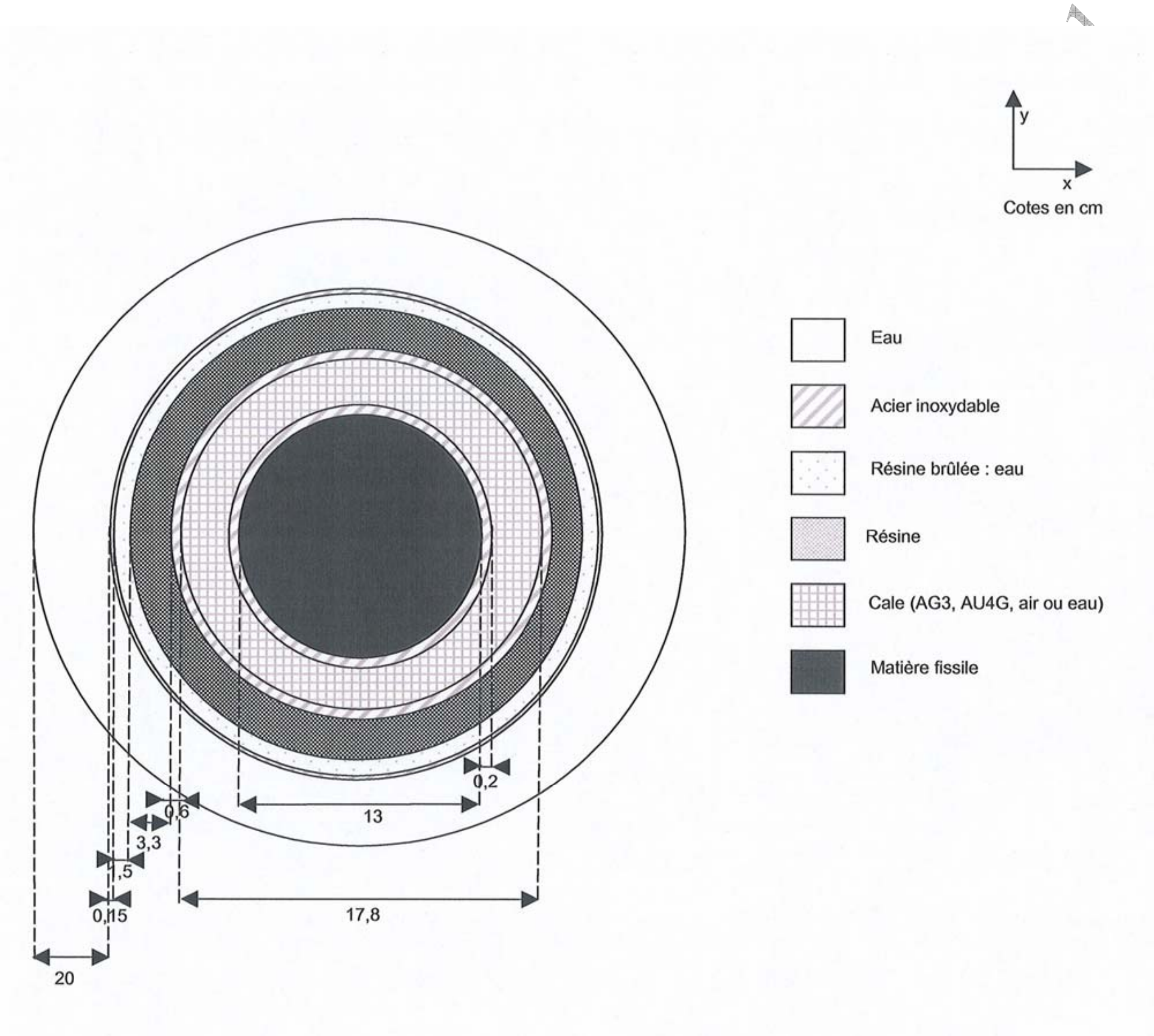
FIGURES

FIGURE 1: NON-AIR TRANSPORT - ISOLATED PACKAGES DURING ACT - AXIAL SECTION



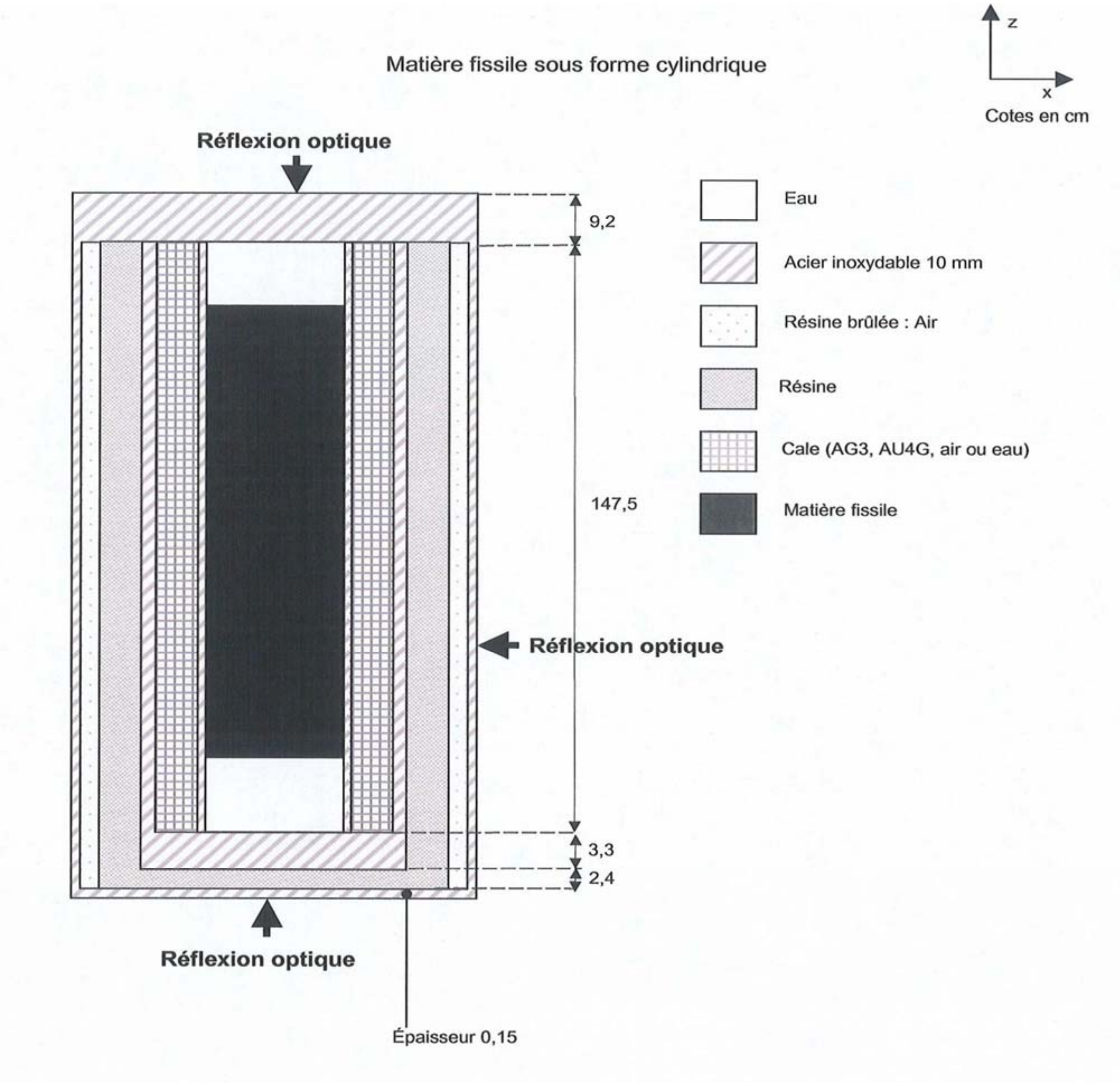
Matière fissile sous forme cylindrique	Fissile material in cylindrical form
Cotes en cm	Dimensions in cm
Eau	Water
Acier inoxydable	Stainless steel
Résine brûlée: eau	Burnt resin: water
Résine	Resin
Cale (AG3, AU4G, air ou eau)	Chock (AG3, AU4G, air or water)
Matière fissile	Fissile material
Epaisseur 0,15	0.15 thick

FIGURE 2: NON-AIR TRANSPORT - ISOLATED PACKAGES DURING ACT - RADIAL SECTION



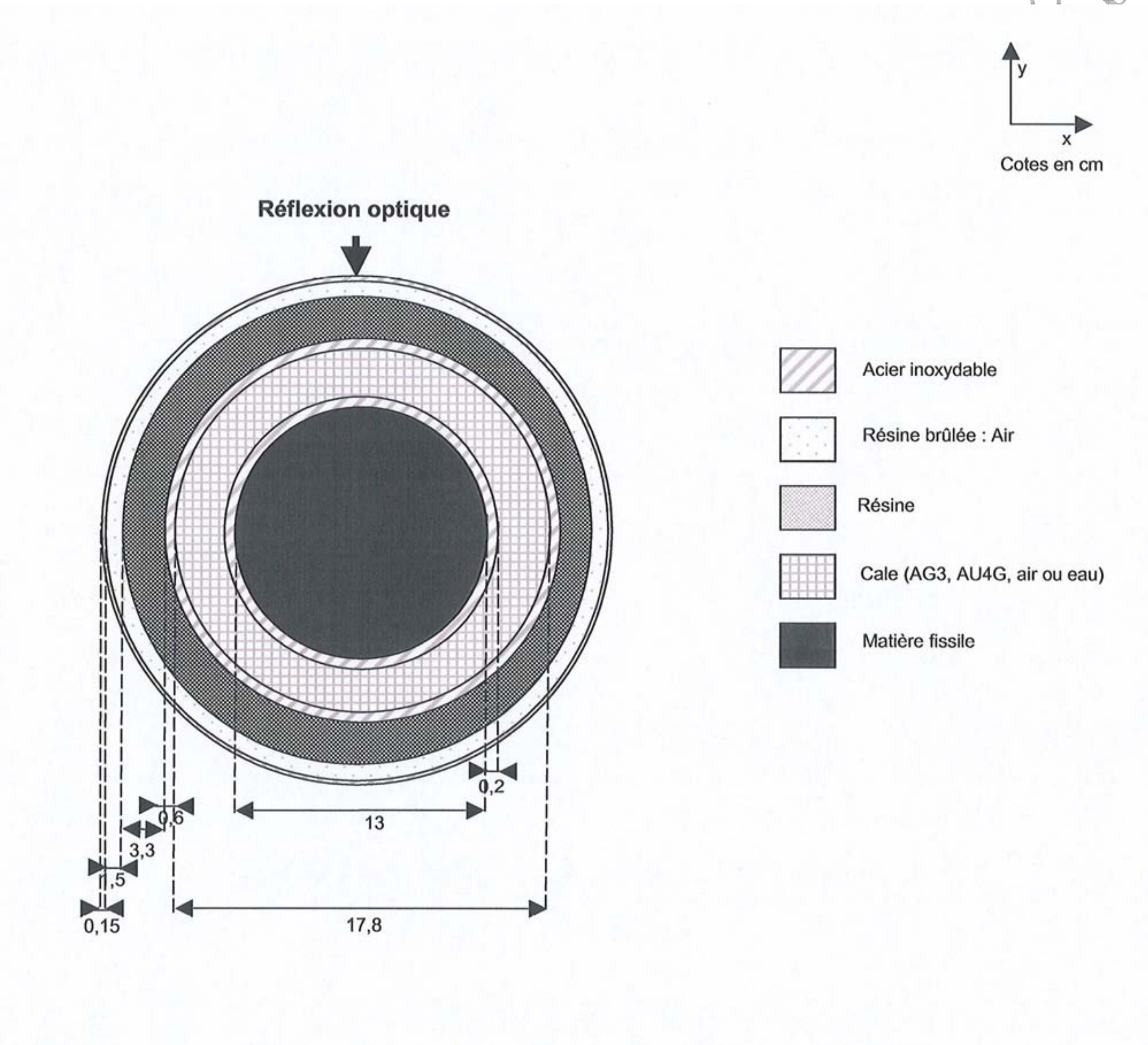
Cotes en cm	Dimensions in cm
Eau	Water
Acier inoxydable	Stainless steel
Résine brûlée: eau	Burnt resin: water
Résine	Resin
Cale (AG3, AU4G, air ou eau)	Chock (AG3, AU4G, air or water)
Matière fissile	Fissile material

FIGURE 3: NON-AIR TRANSPORT - PACKAGE NETWORK DURING ACT - AXIAL SECTION



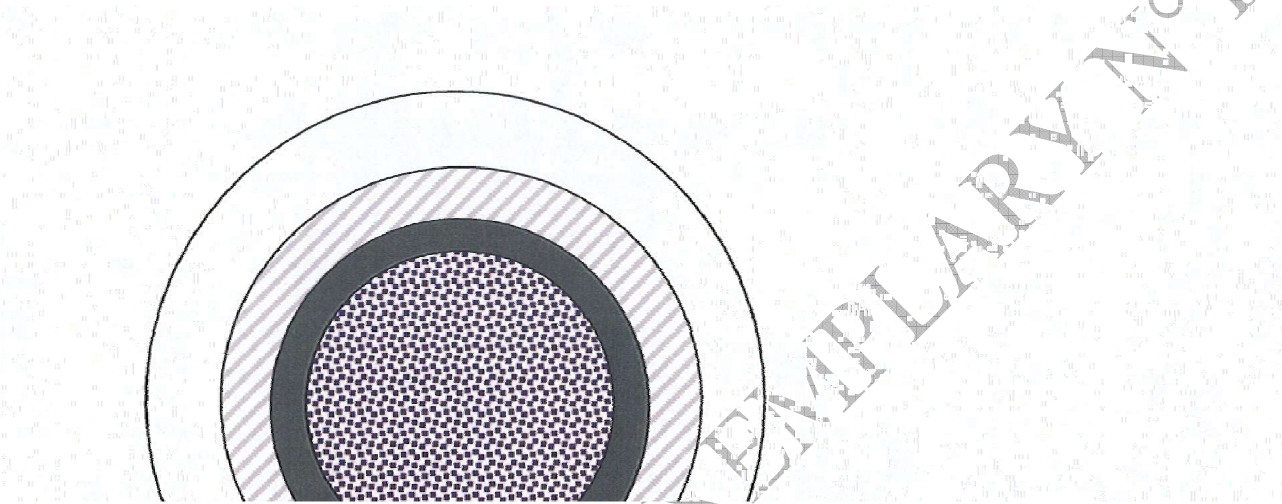
Matière fissile sous forme cylindrique	Fissile material in cylindrical form
Cotes en cm	Dimensions in cm
Réflexion optique	Optical reflectivity
Eau	Water
Acier inoxydable 10 mm	10 mm stainless steel
Résine brûlée: Air	Burnt resin: air
Résine	Resin
Cale (AG3, AU4G, air ou eau)	Chock (AG3, AU4G, air or water)
Matière fissile	Fissile material
Epaisseur 0,15	0.15 thick

FIGURE 4: NON-AIR TRANSPORT - PACKAGE NETWORK DURING ACT - RADIAL SECTION



Cotes en cm	Dimensions in cm
Réflexion optique	Optical reflectivity
Acier inoxydable	Stainless steel
Résine brûlée: Air	Burnt resin: air
Résine	Resin
Cale (AG3, AU4G, air ou eau)	Chock (AG3, AU4G, air or water)
Matière fissile	Fissile material

FIGURE 5: AIR TRANSPORT - RADIAL SECTION



ORIGINAL

PROGRAMME: RENEWAL OF THE CEA PACKAGING STOCK

TITLE: TN BGC1 PACKAGING SAFETY FILE

SECTION 9 APPENDIX 6

CRITICALITY-SAFETY OF THE TN-BGC1 PACKAGE MODEL LOADED WITH CONTENT NO.26: TRIGA FUEL

COPY

Summary:

- This appendix justifies the criticality-safety of the TN BGC1 package model loaded with content no.26, TRIGA fuel. It also examines the possibility of air transport.

Initials	[signature]	[signature]	[signature]
Date	18/07/03	04/08/2003	07/08/2003
Name	C. MATHON	T. CUVILLIER	D. LALLEMAND
Unit	ATR Engineering Dept	DEN/DTAP/SPI/GET	DEN/DTAP/SPI/GET
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	Written by	**Checked by**	**Approved by**

E	M	B		T	N	B	G	C		P	B	C		D	J	S		C	A	0	0	0	3	8	6	A
1	2	3		4	5	6	7	8		9	10	11		12	13	14		15	16	17	18	19	20	22	22	23

LIST OF CHANGES

ISSUE	DATE	Type of change	Pages modified
A	07/08/03	Document issued	

This appendix comprises note EMB TNBGC PBC NTT CA000260A (without appendix 4), a total of 65 pages.

ORIGINAL

PROGRAMME: RENEWAL OF THE CEA PACKAGING STOCK

TITLE:

TN/BGC 1 CERTIFICATE OF APPROVAL EXTENSION TRIGA fuel

Summary:

- This note constitutes the justification that the TN BGC1 package loaded with TRIGA fuel complies with the IAEA 85 regulatory controls concerning B(U)F packages. Compliance of the package with the IAEA 96 stipulations relating to air transport is also demonstrated.

Initials	[signature]	[signature]	[signature]
Date	10/02/03	11/02/03	11/02/2003
Name	T. CUVILLIER	S. CLAVERIE-FORGUES	D. LALLEMAND
Unit	DEN/DTAP/SPI/GET	DEN/DTAP/SPI/GET	DEN/DTAP/SPI/GET
Function	Lab research officer	Lab research officer	Head of the GET
	Written by	**Checked by**	**Approved by**

E	M	B		T	N	B	G	C		P	B	C		N	T	T		C	A	0	0	0	2	6	0	A
1	2	3		4	5	6	7	8		9	10	11		12	13	14		15	16	17	18	19	20	22	22	23

LIST OF CHANGES			
ISSUE	DATE	Type of change	Pages modified
A	February 2003	Document issued	

CIRCULATION

CIRCULATION GRID																			
Addressees		Number of examples for each circulation index																	
Unit	Name or function	A	B	C	D	E	F	G	H	J	K	L	M	N	P	Q	R	S	T
DSNQ/MSN	M. BRUHL	5																	
DTAP/SPI/GET	M. LALLEMAND	1																	
DTAP/SPI/GET	M. CLAVERIE-FORGUES	1																	
DTAP/SPI/GET	M. CUVILLIER	1																	
	TOTAL	8																	

CONTENTS

I INTRODUCTION

The last F/313/B(U)F-85 (Gp) approval extension certificate in the guise of content no.11, authorises the transport of uranium in any solid form (excluding metal powders) in TN/BGC 1.

Alloys of uranium with the following metals are notably authorised: aluminium, molybdenum, silicon.

The purpose of this request is to authorise the transport of TRIGA fuel which differs from the current description of content no.11 for two reasons: these elements contain zirconium and are, for the most part, hydrogenated (see description in II).

II DESCRIPTION OF THE TRIGA FUEL

TRIGA fuel rods are made of U-ZrHx (x varying between 0 and 2). There are two types, standard and thin, both of which are cylindrical with the following geometric characteristics:

Standard: 3.63 cm in diameter, and 12.7 cm long,

Thin: 1.29 cm in diameter, and 18.6 cm long.

A hole is made in the centre of the standard fuel rods prior to hydridation. The diameter of said hole is 6.35 mm.

The Utotal weight content varies between 8 and 47% in accordance with the fuel element. The uranium 235 enrichment is 20%. The table shown in appendix 1 gives the composition of the TRIGA fuel elements.

The diagram of both the standard and thin TRIGA fuel elements is shown in appendix 2.

III JUSTIFICATION OF THE PACKAGE SAFETY

We envisage exactly the same constraints in terms of internal arrangement for the transport of this content as for content no.11 (packing in TN90, AA203, AA204, or AA41).

As far as the thermal risk is concerned, this new fuel is made from uranium and, therefore, comes within the current scope of the certificate of approval.

The presence of Zr and H has no impact in terms of safety in relation to content no.11 on the certificate of approval as far as releases and the study of health physics are concerned. In fact, the study of health physics was carried out for the conservative content as regards transport in TN/BGC 1, namely content no.16 composed of 17 kg of plutonium. As far as releases are concerned, the set criterion of 5 lusec makes it possible to identify very significant margins in relation to the regulatory criteria during NTC and ACT.

Concerning mechanical, thermal, release and health physics aspects, TRIGA fuel has no detrimental influence in terms of safety in relation to content no.11 on the certificate of approval. This package is, therefore, covered by the safety analysis report [1] studies and by the note [3], relating to the transport of content no.11.

However, the fact that the rods comprise zirconium and hydrogen has consequences in terms of criticality. In fact:

- zirconium is a diffusing element,

- the presence of hydrogen means that the medium has intrinsic moderation.

The FRAMATOME ANP criticality study, ref. FF JN DC 0098, dated 05/11/2002, shown in appendix 3, relates to the transport of TRIGA fuel elements in TN/BGC 1. It justifies the safety of this package under normal and accidental transport conditions with the following weight limitations:

U (Weight %)	ZrH (weight %)	U-Zr (g/cm³)	Maximum weight of uranium (kg)
8	92	6.90	9
12	88	7.10	14
21	79	7.40	27
31	69	8.10	43
47	53	9.30	76

It should be noted that the hypotheses taken into account as regards the geometry of the package during NTC or ACT comply with those used in the criticality studies shown in the report [1], notably for content no.11.

Moreover, it also justifies the safety of the air transport of such fuel elements with a weight limitation which is as follows:

U (Weight %)	ZrH (weight %)	U-Zr (g/cm³)	Maximum weight of uranium (kg)
8	92	6.90	1.1
12	88	7.10	1.7
21	79	7.40	3.3
31	69	8.10	5.3
47	53	9.30	6.6

IV CONCLUSION

The safety of the package loaded with TRIGA fuel elements is, therefore, demonstrated, subject to the weight limitations mentioned in the previous section.

The description stipulated as part of the certificate of approval project is given in appendix 4.

V REFERENCES

[1] Safety analysis report EMB TNBGC PBC DS- CA000001 A dated 24 November 2000
[2] Certificate of approval F/313/B(U)F-85 (Gp)
[3] Note EMB TNBGC PBC NTT CA000023A dated 7 December 2001

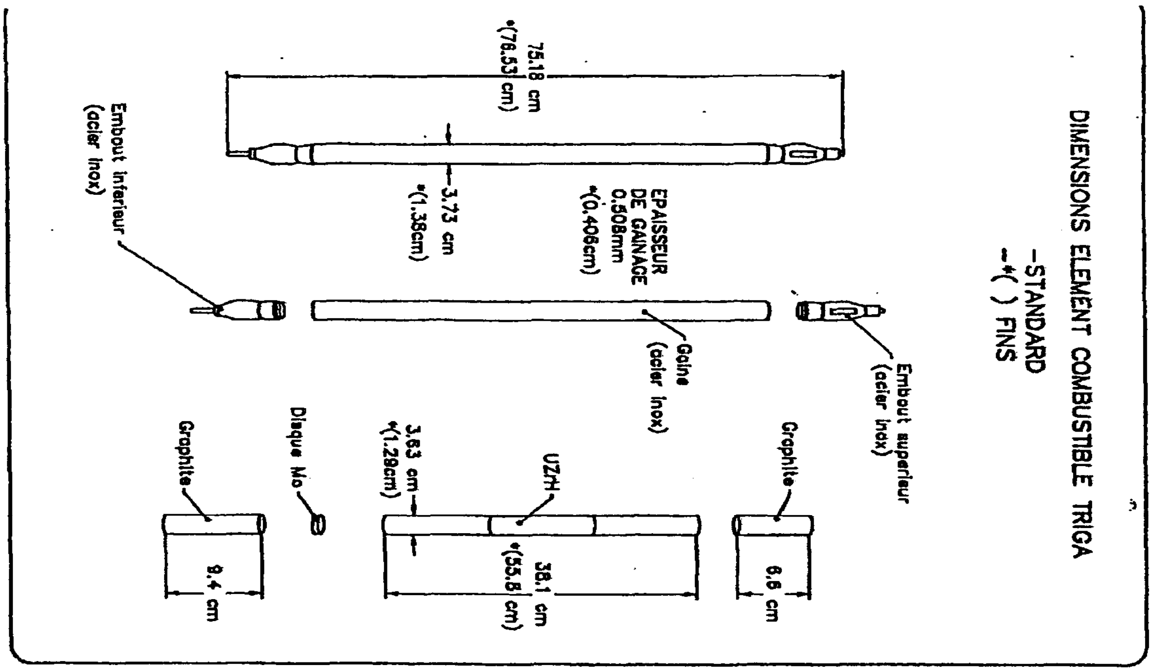
VI APPENDICES

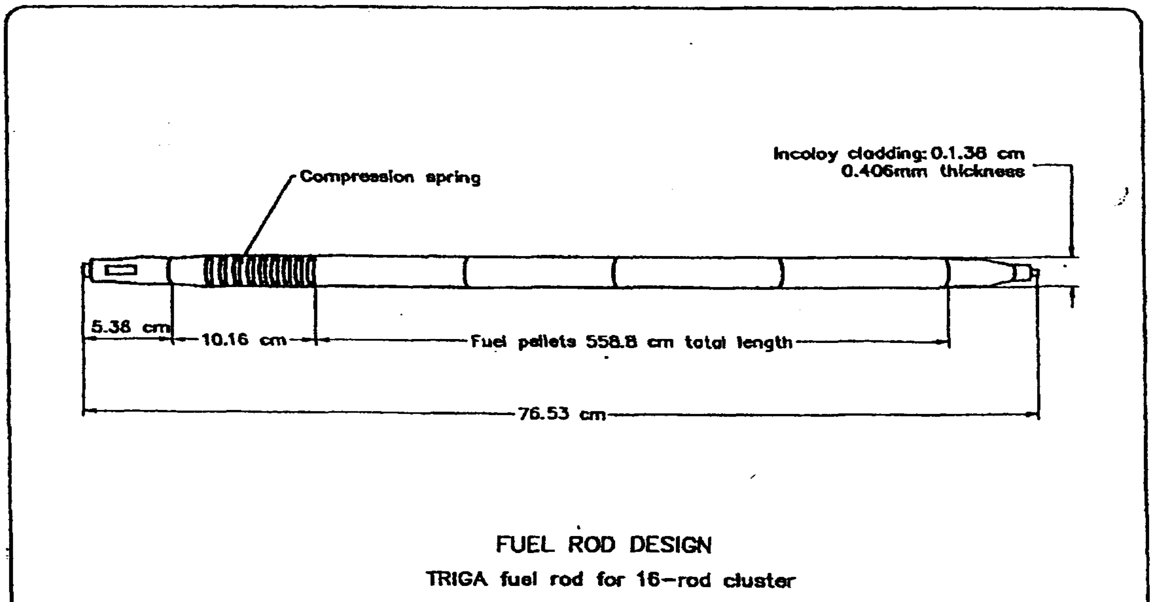
Appendix 1: composition of the TRIGA fuel elements
Appendix 2: diagram of the TRIGA fuel elements
Appendix 3: criticality study FF JN DC 0098 dated 05/11/2002
Appendix 4: certificate of approval project: content TRIGA fuel

Appendix 1: composition of the TRIGA fuel elements

TYPE	U (weight %)	ZrH (weight %)	U-Zr (g/cm³)	U-ZrH2 (g/cm³)
Composition of the standard TRIGA fuel elements - uranium 235 enrichment is 20%				
103	8	92	6.90	6.04
105	12	88	7.10	6.22
107	12	88	7.10	6.22
117	21	79	7.40	6.64
119	31	69	8.10	7.24
Composition of the thin TRIGA fuel elements - uranium 235 enrichment is 20%				
424	47	53	9.3	8.4

Appendix 2: diagram of the TRIGA fuel elements





DIMENSIONS ELEMENT COMBUSTIBLE TRIGA	TRIGA FUEL ELEMENT DIMENSIONS
- STANDARD	- STANDARD
-*( ) FINS	-*( ) THIN
Embout supérieur (acier inox)	Top end-piece (stainless steel)
Graphite	Graphite
Gaine (acier inox)	Cladding (stainless steel)
EPAISSEUR DE GAINAGE	CLADDING THICKNESS
Disque Mo	Mo disc
Graphite	Graphite
Embout inférieur (acier inox)	Bottom end-piece (stainless steel)

Appendix 3: criticality study FF JN DC 0098 dated 05/11/2002

Design & Sales Department	69456 LYON CEDEX 06	Class F	No. of pages: 54 No. of appendices: 0

Document title

CRITICALITY-SAFETY STUDY OF THE TN BGC1 PACKAGING FOR TRANSPORTING TRIGA FUEL ELEMENTS

KEY WORDS: TRANSPORT - SAFETY - CRITICALITY - TRIGA - TNBGC

Circulation	FFJI	FFJN	FFJN				FFBF	
Circulation purpose	GED	Customer circulation	For info.				For info.	
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C	05/11 2002	N. COMTE[signature]	M. DOUCET[signature]	Minor corrections	BPE	M. DOUCET[signature]
REV	DATE	WRITTEN BY	CHECKED BY	MODIFICATIONS - OBSERVATIONS	STATE OF VALIDITY	APPROVED BY

CLASSIFICATION

U.D.:

IN-HOUSE IDENTIFICATION NUMBER														
	F	F	J	N			D	C		0	0	9	8	

CONTENTS

REFERENCES

[1] IAEA safety standards - Radioactive material transport regulations - 1985 edition (Revised in 1990)

[2] IAEA Safety Standards Collection. Radioactive material transport regulations.
1996 edition - ST-1.

[3) Advisory material for the regulations of the safe transport of radioactive material (1996 edition) IAEA safety standards series No. TS-G-1

[4] Note SEC/T/96.020 - F.B.F.C. ROMANS - ROMANS FACTORY - BNI 63 TRIGA PROJECT - Standards relating to homogeneous U media (20% U235)-ZrHx-H2O or in the form of a rod network, x varying from 0 to 2.

[5] Note TN9990-Z-5A-9 - STUDY OF THE CRITICALITY-SAFETY OF THE TN BGC1 PACKAGING LOADED WITH:
- URANIUM OXIDE PELLETS
- MIXED OXIDE PELLETS
- URANIUM OXIDE RODS
- MIXED OXIDE RODS
- URANIUM AND/OR PLUTONIUM-BEARING MATERIAL IN ANY SOLID FORM

[6] Comments on the requirements for packages containing fissile materials
L. Heulers, S. Claverie-Forgues - Patram 2001 Chicago.

[7] LA-12625- M issued March 1997:
MCNP - A general Monte Carlo N-particle transport code - Version 4B

[8] F/313/B(U)F-85 On - Certificate of approval for a package model

LIST O F TABLES

LIST OF FIGURES

1. INTRODUCTION

The purpose of this study is to demonstrate the sub-criticality of the transport of TRIGA fuel elements in the TN BGC1 packaging loaded with a TN 90 packing container.

Studies have dealt with the determination of the number N of packages and the number of fuel elements or the quantity of rejects that can be transported in each package in accordance with the IAEA 1985 /1/ and IAEA 1996 - ST1 /2/ regulations.

Justification of the safety of air transport is also demonstrated, based on the latest IAEA /3/ working group recommendations.

2. CODES AND VERSIONS

Calculations are made using:

- CIGALES (version v1.0 for PC)
- APOLLO 2 version 2.4.3
- MORET 4 version 4.A.4

These three software packages are part of the CRISTAL VO.1 form.

Additional calculations and the justification for air transport were carried out using the MCNP4B code

3. CRITICALITY-SAFETY CRITERIA

IAEA /1/ and IAEA /2/ standards do not clearly define the sub-criticality margins to be observed when transporting fissile material.

In France, the IRSN has defined the following rules:

- isolated packages under normal and accidental conditions: Keff ≤ 0.95 (for all uncertainties)
- network of packages under normal and accidental conditions: Keff ≤ 0.98 (for all uncertainties)

Not all countries adopt said criteria. The criterion set for this study is Keff ≤ 0.95 for all calculations in order to resolve questions and justifications from overseas Safety Authorities applying different rules.

4. STUDY SOURCE DATA

4.1. TRIGA fuel element characteristics

The main characteristics of TRIGA fuel elements are reiterated in table 1; they are taken from note /4/. The dimensions of the fuel elements are given in figure 1.

4.2. TN BGC1 packaging characteristics

The characteristics of the packaging are identical to those set out in /5/. Figure 2 shows this packaging loaded with a TN 90 container.

The chemical composition of the structure materials from note /5/ is reiterated below:

Shells: stainless steel:	
- Density:	7.9 g/cm^3
- Atomic concentrations (1024 atoms / cm^3):	Fe: 6.1341.10-2
	Cr: 1.6467.10-2
	Ni: 8.1070.10-3

Boron-loaded resin	
- Density:	1.186 g/cm^3
- Atomic concentrations (1024 atoms / cm^3):	H: 4.0616.10-2
	C: 2.3803.10-2
	O: 2.3580.10-2
	B: 9.4597.10-4

5. CALCULATION HYPOTHESES FOR TRANSPORT

5.1. Geometric model of the package

The most conservative hypotheses are taken into account in all the calculations whatever the regulatory conditions to be observed (normal and accidental conditions); they are taken from note /5/.

We take the most conservative hypothesis that the radial thickness of the burnt resin is 15 mm. At the ends of the packaging (near the covers), said resin has not undergone any damage and is still 24 mm thick. The 2 mm thick stainless

steel containment enclosure has an internal diameter of 120 mm. The distortion undergone by the container is integrated by expanding the internal diameter to 130 mm throughout the height of the container.

The geometric characteristics of the packaging model taken into account are, therefore:

- a cylindrical cavity with an internal Ø = 181 mm, and an effective height h = 1,475 mm,
- a steel shell with a radial thickness of 6 mm,
- a ring of boron-loaded resin with a radial thickness of 33 mm
- a 15 mm thick ring of burnt resin, replaced by air or water (of variable density) in accordance with the worst case scenario configuration,
- a steel shell 1.5 mm thick with an internal Ø = 292 mm,
- the thickness of the resin at the ends of the packaging is 24 mm,
- the internal thickness of steel at the ends is 33 mm,
- the external thickness of steel at the ends of the packaging is 1.5 mm,
- a containment enclosure with an internal Ø = 130 mm, made of 2 mm thick steel and with a height h = 1,475 mm.

5.2. Contents

There are two geometric types of TRIGA fuel elements that can be transported: standard or thin.
In an enclosure with an internal Ø = 120 mm, we can mathematically transport:
- 10 standard fuel elements,
- 73 thin fuel elements.

5.2.1. Heterogeneous fissile medium

In APOLLO 2, the diameter of the fissile medium is taken as being equal to the diameter of a fuel element:
- 37.3 mm (standard fuel element)
- 14 mm (thin fuel element).

In MORET 4, the height of the fissile medium is taken as being equal to the total height of the fuel element:

- 752 mm (standard fuel element),
- 770 mm (thin fuel element)

The diameter of fissile material is taken as being equal to the internal diameter of the container which is 130 mm.

5.2.2. Homogeneous fissile medium

So as to be able to transport fuel element rejects and given that the integrity of the fuel elements cannot be guaranteed, section 6.4 deals with any height of fissile material, compared to a homogeneous mixture UZrH2-H2O, with variable density and moderation. This varies the weight and moderation in order to find the maximum weight that can be transported, independently of the maximum number of fuel elements contained in the package.

6 RESULTS

6.1 APOLLO2 calculations

Initially, the fissile material under consideration is a heterogeneous medium, UZrH2 in water or air, with various compositions in keeping with the TRIGA fuel element types and a Vm/Vf variable in accordance with the number of fuel elements transported.

The thermalisation matrix used is H in ZrH and Zr in ZrH.

Table 2 gives the comparison of the B^2m and the infinite K for the different standard fuel elements and for the heterogeneous fissile medium UZrH2-H2O.

Figure 3 shows this comparison.

Table 3 gives the comparison of the B^2m and the infinite K for the different standard fuel elements and for the heterogeneous fissile medium UZrH2-Air.

Figure 4 shows this comparison.

Table 4 gives the values of the B^2m and the infinite K for the thin fuel element and for the heterogeneous fissile medium UZrH2-H2O.

Figure 5 sets out the values of the B^2m for the heterogeneous fissile media UZrH2-H2O and UZrH2-Air for this type of fuel element.

Given these results, the reference fissile medium shall be compared either to a heterogeneous mixture UZrH2-H2O or UZrH2-Air corresponding to standard fuel elements 119 and 117 (both these types have a B^2m that is very close and far greater than other types) and thin fuel element 424.

Subsequently, in order to deal with the rejects, the fissile medium under consideration is a homogeneous medium UZrH2-H2O, with different compositions in keeping with the TRIGA fuel element types and with a variable total uranium concentration. Table 5 gives the values of the B^2m and infinite K for standard fuel element 119 and thin fuel element 424.

6.2 MORET4 calculations

6.2.1. Study of the damaged isolated package

The packaging is surrounded by a 20 cm ring of water. The fissile material is centred axially in the internal container. It radially fills the entire diameter of the internal container which is 130 mm.

The packaging modelling is as described in section 5.1.

Figure 6 shows a radial section of said modelling.

Figure 7 shows an axial section of said modelling.

Fuel elements 119

Initially, the free spaces in the packaging, including the burnt resin, are filled with water.

The results for a heterogeneous medium UZrH2-H2O are given in table 6 and shown in figure 8.

The maximum Keff is obtained for a maximum number of fuel elements.

This configuration leads to a **Keff = 0.830** (Keff+3σ) for 10 standard fuel elements.

The container is filled with air. The other free spaces, including the burnt resin, are filled with water. This configuration leads to a **Keff = 0.822** (Keff+3σ) for 10 fuel elements.

The cylindrical cavity with an internal Ø = 181 mm is filled with air. The other free spaces, including the burnt resin, are filled with water. This configuration leads to a **Keff = 0.702** (Keff+3σ) for 10 fuel elements.

The cylindrical cavity with an internal Ø = 181 mm is filled with air. The burnt resin is replaced by air and the other free spaces are filled with water. This configuration leads to a **Keff = 0.702** (Keff+3σ) for 10 fuel elements.

The configuration with water in the container and in the cavity and water in place of the burnt resin is the most reactive configuration under isolated conditions, for a medium UZrH2-H2O.

The results for a heterogeneous medium UZrH2-Air cannot be provided using the CRISTAL form; there is currently a thermalisation matrix problem with this type of medium.

Fuel elements 117

The free spaces in the packaging, including the burnt resin, are filled with water.

The results for a heterogeneous medium UZrH2-H2O are given in table 7 and shown in figure 8.

The maximum Keff is obtained for a maximum number of fuel elements.

This configuration leads to a **Keff = 0.818** (Keff+3σ) for 10 standard fuel elements.

The container is filled with air. The other free spaces, including the burnt resin, are filled with water. This configuration leads to a **Keff = 0.807** (Keff+3σ) for 10 fuel elements.

Fuel elements 424

The free spaces in the packaging, including the burnt resin, are filled with water.

The results for a heterogeneous medium UZrH2-H2O are given in table 8 and shown in figure 9.

The maximum Keff is obtained for a maximum number of fuel elements.

This configuration leads to a **Keff = 0.828** (Keff+3σ) for 73 thin fuel elements.

The container is filled with air. The other free spaces, including the burnt resin, are filled with water. This configuration leads to a **Keff = 0.823** (Keff+3σ) for 73 fuel elements.

The reactivity of the isolated package is far below the criterion.

6.2.2. Study of the damaged package network

The package is modelled in the same way as in section 6.2.1. but total reflection conditions are applied to the radial and axial limits of the package.

These accidental transport conditions enclose the normal conditions.

Only standard fuel element 119 was studied as it leads to the maximum reactivity under isolated conditions. The fissile

medium is a heterogeneous mixture UZrH2-H2O.

Looking for the most conservative configuration:

Initially the free spaces in the packaging, including the burnt resin, are filled with water.

This configuration leads to a **Keff = 0.871** (Keff+3σ) for 10 standard fuel elements.

The container is filled with water, the cavity with air and the burnt resin is replaced by water. This configuration leads to a **Keff = 0.777** (Keff+3σ) for 10 fuel elements.

The container is filled with water, the cavity with air and the burnt resin is replaced with air. This configuration leads to a **Keff = 0.862** (Keff+3σ) for 10 fuel elements.

The container is filled with water, the cavity with water and the burnt resin is replaced with air. This configuration leads to a **Keff = 0.921** (Keff+3σ) for 10 fuel elements.

The container is filled with air, the cavity with water and the burnt resin is replaced with water. This configuration leads to a **Keff = 0.867** (Keff+3σ) for 10 fuel elements.

The container is filled with air, the cavity with air and the burnt resin is replaced with water. This configuration leads to a **Keff = 0.776** (Keff+3σ) for 10 fuel elements.

The container is filled with air, the cavity with air and the burnt resin is replaced with air. This configuration leads to a **Keff = 0.850** (Keff+3σ) for 10 fuel elements.

The container is filled with air, the cavity with water and the burnt resin is replaced with air. This configuration leads to a **Keff = 0.912** (Keff+3σ) for 10 fuel elements.

The most conservative configuration is **water in the container and the cavity and air in the place of the burnt resin**.

Calculations to check on the impact of the axial position of the fissile medium were carried out. The content was placed as close as possible to the top of the packaging. This configuration leads to a **Keff = 0.920** (Keff+3σ) for 10 fuel elements. The axial position of the content in the packaging has no impact.

The results for a heterogeneous medium UZrH2-H2O in accordance with the number of fuel elements contained in the package and in the most conservative configuration are given in table 9 and shown in figure 10.

The penalizing situation is obtained when the package is filled with a maximum of fuel elements.

6.3 MCNP4B calculations

Given that it is not possible to check that the medium UZrH2-H2O is more detrimental than the medium UZrH2-Air using the CRISTAL form, an additional study was carried out using the MCNP4B code /7/. Said code uses a library of ad hoc cross-sections which rigorously deal with the different spectral situations linked to moderation variations.

10 standard fuel elements 119 are modelled (most conservative configuration found using the MORET4 code). The fuels elements are explicitly modelled.

Figure 11 shows a radial section of said modelling in a square grid arrangement.

Figure 12 shows an axial section of said modelling in a square grid arrangement.

The fuel elements UZrH2 are in air. The container above and below the fissile material and the cavity are filled with water. The burnt resin is replaced with air. This configuration leads to a **Keff = 0.802** (Keff+3σ).

The fuel elements UZrH2 are in air. The container above and below the fissile material and the cavity are filled with air. The burnt resin is replaced with air. This configuration leads to a **Keff = 0.718** (Keff+3σ).

A parametric study is carried out, varying the density of water in each free space of the container. The density is varied in just one space at each stage. The most conservative configuration vis-à-vis Keff is adopted for the following stage.

Initially, the density of water around the fuel elements UZrH2 is varied. The container above and below the fissile material and the cavity are filled with water or air. The burnt resin is replaced with air.

The results are shown in table 10. The package network has the most conservative when the fissile material is surrounded by water and the cavity is filled with water. This configuration leads to a **Keff = 0.893** (Keff+3σ).

Subsequently, the density of water in the container above and below the fissile material and in the cavity is varied. The burnt resin is replaced with air. The fissile elements are in the water.

The results are shown in table 11. The package network has the most conservative when the container above and below the fissile material and the cavity are filled with water. This configuration leads to a **Keff = 0.894** (Keff+3σ).

Further to this, the density of the water in the part corresponding to the burnt resin is varied. The free spaces inside are filled with water.

The results are shown in table 12. The package network has the most conservative when the burnt resin is compared to air. This configuration leads to a **Keff = 0.893** (Keff+3σ).

The most conservative configuration is that of water in the container and the cavity and air in place of the burnt resin. The fuel elements UZrH2 are in the water.

Using this configuration, if the reflection conditions are applied directly on the outer steel shell as in the MORET model, the reactivity increases and the Keff becomes **Keff = 0.909** (Keff+3σ) to be compared with **Keff = 0.921** (Keff+3σ) obtained using CRISTAL.

The configurations studied using the CRISTAL form are indeed the most conservative and we notice that this system is always more conservative than the MCNP4B code.

6.4. Variation of the height of the fissile medium in the container (CRISTAL calculations)

So as to be able to transport the fuel element rejects and given that the integrity of the fuel elements cannot be guaranteed, we use the hypothesis that the fuel elements or chips of fuel elements are compared to a homogeneous mixture UZRH2-H2O, of variable height, with a variable concentration in uranium and, therefore, a variable moderation.

6.4.1. Results for 119

Initially, the free spaces in the packaging, including the burnt resin, are filled with water.

An initial calculation is made with a fissile height of 752 mm, a fissile diameter of 130 mm and a U concentration close to 1.85 g/cm³ corresponding to a weight of 10 standard fuel elements. Case similar to the heterogeneous case for validating our hypothesis of the fissile medium used. This configuration in an isolated configuration leads to a **Keff = 0.827** (Keff+3σ) to be compared with the result using the heterogeneous fissile medium **Keff = 0.830** (Keff+3σ). These results are statistically equal and validate our hypothesis.

Subsequently, for several densities of uranium and, therefore, several H/U, the height of the fissile material and, therefore, the weight is varied, independently of the maximum number of fuel elements that can be contained in the package. This approach not only covers the “unguaranteed” integrity of the fuel elements but also makes it possible not to limit the weight of the rejects to a weight equal to 10 standard fuel elements or 73 thin fuel elements.

The results for the damaged isolated package are given in table 13 and shown in figure 13.

The maximum Keff is obtained when the package is full of a homogeneous mixture UZRH2-H2O with an almost maximum concentration (C(U)=2.2 g/cm³- H/U = 0.2). This is equal to a uranium weight of 42.9 kg.

This configuration leads to a **Keff = 0.846** (Keff+3σ).

The results for the maximum concentration C(U) = 2.23 g/cm³ corresponding to a H/U = 0 cannot be provided using the CRISTAL form; there is currently a thermalisation matrix problem. It is likely that this configuration is slightly more reactive. But given the low values obtained for Keff, this is not sufficient to bring these conclusions into question.

The cavity is filled with water and the burnt resin is replaced with air. This configuration leads to a **Keff = 0.844** (Keff+3σ) for an isolated package full of a homogeneous mixture UZRH2-H2O (C(U) = 2.20 g/cm³.

The cavity is filled with air and the burnt resin is replaced with air, this configuration leads to a **Keff = 0.719** (Keff+3σ) for an isolated package filled with a homogeneous mixture UZRH2-H2O (C(U) = 2.20 g/cm³ - H/U = 0.2).

The network of damaged packages filled with a homogeneous mixture UZRH2-H2O (C(U) = 2.20 g/cm³ - H/U = 0.2), the free spaces in the packaging, including the burnt resin, being filled with water, leads to a **Keff = 0.892** (Keff+3σ).

The cavity is filled with water and the burnt resin is replaced with air. This configuration leads to a **Keff = 0.941** (Keff+3σ) for a network of packages filled with a homogeneous mixture UZRH2-H2O (C(U) = 2.20 g/cm³ - H/U = 0.2).

6.4.2. <u>Results for 424</u>

The results for the damaged isolated package are given in table 14 and shown in figure 14. The free spaces in the packaging, including the burnt resin, are filled with water.

The maximum Keff is obtained when the package is filled with a homogeneous mixture UZRH2-H2O with an almost maximum concentration (C(U) = 3.85 g/cm³ - H/U = 0.17). This is equal to a uranium weight of 75.3 kg.

This configuration leads to a **Keff = 0.834** (Keff+3σ).

The network of damaged packages filled with a homogeneous mixture UZRH2-H2O (3.85 g/cm³ - H/U = 0.17), the free spaces in the packaging being filled with water and the burnt resin replaced with air, leads to a **Keff = 0.935** (Keff+3σ).

6.5. <u>Air transport justification</u>

Article 680 of the TSR1 states that there is no intrusion of water in the package but it makes no mention of any mixture of the fissile material with the hydrogen compounds of the package.

Taking the moderation that can be made by the package's hydrogen materials into account can lead to more reactive configurations. The different configurations having to be justified for air transport were discussed during a meeting of experts at the IAEA and shall be included in the next revision of the TS-G-1 /3/. The practical interpretation of these methods was presented during the last PATRAM in September 2001 in Chicago /6/. The results given in the next section take this approach into account.

6.5.1 Calculation hypotheses

The calculations are made using the MCNP4B code /7/ and using a library of ad hoc sections which rigorously deals with the different spectral situations linked to the moderation variations to be envisaged in this study.

The calculations are made using spherical geometry.

The configurations studied are:

- the dry fissile material reflected by 20 cm of water (configuration a),
- the dry fissile material reflected by the package steel and 20 cm of water (configuration b),
- the fissile material moderated by the hydrogen elements contained in the package (configuration c).

Moderation comes, on the one hand, from the hydrogen compounds contained in the container containing the fuel and, on the other, from the wood contained in the cover. In accordance with note /8/ the total weight of water contained in the wood is less than 1,670 g.

The boxes used in order to protect the fuel are compared to water (H2O) for a worst case scenario with a weight of 200 g per box for the standard fuel elements and a weight of 84.2 g per box for the thin fuel elements, the weight of water equal to the wood is taken to be 1,700 g. The moderator contained in the neutron-absorbing poison does not participate in the moderation /6/.

Given the external Ø of the boxes of 40 mm (standard fuel elements) and 18 mm (thin fuel elements), we can only mathematically put 44 thin fuel elements in the TN 90. This represents 2 litres of water for the 10 standard elements and 3.7 litres of water for the 44 thin elements (corresponding to the box), plus 1.7 litres corresponding to the wood.

Using the moderating material contained in the package, we are looking for the maximum reactivity for the maximum number of standard and thin fuel elements that can be transported. This can lead to the fuel being dealt with in a heterogeneous manner:

- a number x of moderated elements placed in the centre of the sphere,
- the remaining elements in the form of a dry sphere playing the role of reflector around the volume of wet fuel.

The 5 concentric spherical regions are:

- “wet” fuel,
- dry fuel
- steel, with preservation of the volume if this configuration is the most conservative one.
- resin, with preservation of the volume,
- 20 cm of water.

The diagram of this modelling is given in figure 15.

6.5.2. Calculation results

The calculations were made using the following characteristics for the fuel:

- X standard fuel elements 119, 39 cm in height (height of the fissile medium),
- Y thin fuel elements 424, 56 cm in height (height of the fissile medium),
- X and Y correspond to the maximum number of standard and thin fuel elements that can be transported, such that all the configurations observe the criticality-safety criteria of Keff ≤ 0.95.

Configuration a (dry sphere + 20 cm of water)

CONFIGURATION	Keff	σ (pcm)	Keff + 3σ
10 fuel elements 119	0.901	174	0.906
7 fuel elements 119	0.823	175	0.828
6 fuel elements 119	0.793	194	0.799
44 fuel elements 424	0.858	168	0.863
40 fuel elements 424	0.837	181	0.843
23 fuel elements 424	0.727	172	0.732

Configuration b (dry sphere + steel + 20 cm of water)

Not knowing the weight of steel contained in the container, the calculations were made using a weight equal to the

weight of the empty packaging which is 300 kg of steel. This represents a steel ring varying between 10 and 13 cm in accordance with the fissile volume.

The results are provided below:

CONFIGURATION	Keff	σ (pcm)	Keff + 3σ
10 fuel elements 119	0.907	183	0.912
7 fuel elements 119	0.825	165	0.830
6 fuel elements 119	0.784	164	0.789
44 fuel elements 424	0.855	155	0.860
40 fuel elements 424	0.826	160	0.831
23 fuel elements 424	0.702	158	0.706

When comparing the results of configurations a and b, we note that the steel can have a role of absorption or reflection in accordance with the sizes of the different spheres.

<u>Configuration c (moderate fuel)</u>

Results for the standard fuel elements 119:

TOTAL NUMBER OF FUEL ELEMENTS CONTAINED IN THE CONTAINER	NUMBER OF MODERATED FUEL ELEMENTS	Quantity of water (litres)	Keff	σ (pcm)	Keff + 3σ
10	10	3.7	1.03	181	**1.036**
7	5	3.1	0.962	191	**0.968**
	0		0.821	165	0.826
	1		0.877	181	0.882
	2		0.914	186	0.920
6	3	2.9	0.922	194	0.928
	4		0.925	195	0.931
	5		0.930	189	**0.936**
	6		0.927	156	0.932

The calculations show that in order to be able to observe the safety criterion, the package cannot contain more than 6 standard fuel elements.

So as to find out the impact of the steel on this configuration, an additional calculation is made by removing it, but the volume of resin is preserved. This most conservative configuration concerning the worst case scenario (5 moderated fuel elements out of 6) leads to a **Keff = 0.802** (Keff+3σ). We clearly highlight here the role of reflector that the steel represents in this configuration.

A second additional calculation is made by removing the resin from the same case whilst preserving the volume of steel. This configuration concerning the worst case scenario (5 moderated fuel elements out of 6) leads to a **Keff = 0.928** (Keff+3σ). Statistically, this result is equal to the result with resin. This confirms the result found previously, the steel playing the role of reflector, the resin which is to be found afterwards, therefore, has only very little impact.

If the quantity of water is reduced, the reactivity of the container is reduced as the results given below show. All the hydrogenated materials from the package have to be taken into account.

TOTAL NUMBER OF FUEL ELEMENTS CONTAINED IN THE CONTAINER	NUMBER OF MODERATED FUEL ELEMENTS	Quantity of water (litres)	Keff	σ (pcm)	Keff + 3σ
6	1	2	0.873	150	0.877
	2		0.889	186	0.895
	3		0.896	163	0.901
	4		0.898	160	0.902
	5		0.898	189	0.903
	6		0.898	173	0.904

Figure 16 shows these results.

Results for thin fuel elements 424:

TOTAL NUMBER OF FUEL ELEMENTS CONTAINED IN THE CONTAINER	NUMBER OF MODERATED FUEL ELEMENTS	Quantity of water (litres)	Keff	σ (pcm)	Keff + 3σ
44	44	5.4	1.056	182	**1.062**

TOTAL NUMBER OF FUEL ELEMENTS CONTAINED IN THE CONTAINER	NUMBER OF MODERATED FUEL ELEMENTS	Quantity of water (litres)	Keff	σ (pcm)	Keff + 3σ
40	40	5.1	1.025	177	**1.031**
30	30	4.3	0.976	182	**0.981**
23	0	3.65	0.757	169	0.762
	1		0.774	166	0.779
	5		0.912	155	0.916
	10		0.932	172	0.937
	15		0.939	163	**0.944**
	20		0.935	182	0.940
	23		0.931	197	0.937

These calculations show that in order to be able to observe the safety criterion, the package cannot contain more than 23 thin fuel elements.

Figure 17 shows these results.

7. SAFETY ANALYSIS

The purpose of this section is to reiterate the conservative hypotheses taken into account in this study.

Non-air transport

Non-air transport has been demonstrated for a number of fuel elements that is greater than that contained in the container. The diameter of the container is taken as being equal to 130 mm throughout its height whereas its actual diameter is 120 mm. The weight of fissile material which has been taken into account is greater than it actually is as the study deals with a height of 1,475 mm whereas the TN 90 has an effective height of 1,397 mm.

Air transport

Fuel modelling in the form of a sphere is enclosure of the reality. The weight of steel taken into account is greater than it actually is as it is equal to the weight of the empty packaging whereas said packaging contains, amongst other things, 70 kg of resin and 1.7 kg of wood.

8. CONCLUSION

The purpose of this study is to demonstrate the criticality-safety of the transport of TRIGA fuel elements in the TN BGC1 packaging loaded with the TN 90 container. The sub-criticality of this transport package is guaranteed by the use of a neutron-absorbing resin.

Non-air transport

The sub-criticality is demonstrated for a maximum number of standard or thin fuel elements or pieces of fuel element able to be contained in the package (whatever the height of the fissile medium).

The maximum weights of uranium that can be transported are:

U (weight %)	**ZrH (weight %)**	**U-ZR (g/cm^3)**	**Maximum weight of uranium (kg)**
8	92	6.90	9
12	88	7.10	14
12	88	7.10	14
21	79	7.40	27
31	69	8.10	43
47	53	9.3	76

The calculation conditions taking the hypothetical consequences of the results of the regulatory drop tests and the thermal test into account to give a worst case scenario are used to guarantee that the transport of TRIGA fuel elements is safe provided that they are the same type as that defined in table 1.

Calculations have demonstrated that the criticality-safety criterion of $K_{eff} \leq 0.95$ was observed both under normal and accidental transport conditions for an infinite number N of packages for TRIGA fuel elements of the same type as that defined in table 1.

Air transport

The demonstration of criticality-safety using the special air transport configurations is carried out for the contents below:

- 6 standard fuel elements of the type defined in table 1
- 23 thin fuel elements 424 as defined in table 1.

The maximum weights of uranium that can be transported by air are:

U (weight %)	ZrH (weight %)	U-ZR (g/cm^3)	Maximum weight of uranium (kg)
8	92	6.90	1.1
12	88	7.10	1.7
12	88	7.10	1.7
21	79	7.40	3.3
31	69	8.10	5.3
47	53	9.3	6.6

Calculations have demonstrated that the criticality-safety of Keff $\leq$ 0.95 was observed.

TABLE 1: Composition of the TRIGA fuel elements (data taken for the calculations)

TYPE	U (weight %)	ZrH (weight %)	U-Zr (g/cm^3)	U-ZrH2 (g/cm^3)
Composition of the standard TRIGA fuel elements - uranium 235 enrichment is 20%				
103	8	92	6.90	6.04
105	12	88	7.10	6.22
107	12	88	7.10	6.22
117	21	79	7.40	6.64
119	31	69	8.10	7.24
Composition of the thin TRIGA fuel elements - uranium 235 enrichment is 20%				
424	47	53	9.3	8.4

TABLE 2: Comparisons of the B²m and infinite k in a heterogeneous medium UZrH2-H2O - Standard TRIGA fuel elements studied

NUMBER OF ELEMENTS	B²m (cm-2)			
	103	**105**	**117**	**119**
10	2.84E-02	3.27E-02	**3.51E-02**	3.45E-02
9	2.65E-02	3.10E-02	3.40E-02	3.40E-02
8	2.42E-02	2.90E-02	3.26E-02	3.32E-02
7	2.14E-02	2.65E-02	3.08E-02	3.20E-02
6	1.80E-02	2.34E-02	2.83E-02	3.01E-02
5	1.37E-02	1.93E-02	2.49E-02	2.72E-02
4	8.21E-03	1.39E-02	1.99E-02	2.28E-02
3	9.84E-04	6.45E-03	1.25E-02	1.57E-02
2	< 0	< 0	1.50E-03	4.60E-03
1	< 0	< 0	< 0	< 0

NUMBER OF ELEMENTS	Infinite K			
	103	**105**	**117**	**119**
10	1.52666	1.59907	1.63842	1.62530
9	1.50913	1.59088	1.64323	1.63925
8	1.48313	1.57525	1.64259	1.64954
7	1.44486	1.54823	1.63252	1.65226
6	1.38934	1.50428	1.60709	1.64132
5	1.30955	1.43527	1.55708	1.60697
4	1.19320	1.32680	1.46585	1.53152
3	1.02420	1.15818	1.30728	1.38554
2			1.03846	1.11829
1				

TABLE 3: Comparisons of the B^2m and infinite k in a heterogeneous medium UZrH2-Air - Standard TRIGA fuel elements studied

NUMBER OF ELEMENTS	**B^2m (cm-2)**			
	103	**105**	**117**	**119**
10	2.31E-02	2.60E-02	**2.72E-02**	2.62E-02
9	1.94E-02	2.20E-02	2.30E-02	2.22E-02
8	1.59E-02	1.80E-02	1.90E-02	1.82E-02
7	1.26E-02	1.44E-02	1.52E-02	1.46E-02
6	9.50E-03	1.08E-02	1.15E-02	1.11E-02
5	6.73E-03	7.71E-03	8.15E-03	7.86E-03
4	4.73E-03	5.04E-03	5.36E-03	5.14E-03
3	2.46E-03	2.84E-03	3.03E-03	2.92E-03
2	1.08E-03	1.24E-03	1.320E-03	1.28E-03
1	2.56E-04	2.94E-04	3.11E-04	3.00E-04

NUMBER OF ELEMENTS	**Infinite K**			
	103	**105**	**117**	**119**
10	1.54435	1.60119	1.62038	1.59435
9	1.54388	1.60067	1.61989	1.59401
8	1.54332	1.60019	1.61935	1.59371
7	1.54282	1.59942	1.61864	1.59303
6	1.54228	1.59893	1.61809	1.59255
5	1.54193	1.59858	1.61806	1.59274
4	1.54141	1.59801	1.61750	1.59234
3	1.54079	1.59769	1.61732	1.59219
2	1.53980	1.59711	1.61728	1.59231
1	1.53655	1.59512	1.61640	1.59171

TABLE 4: Values of the B²m and infinite k in a heterogeneous medium UZrH2-Air or H2O -Thin TRIGA fuel elements 424

NUMBER OF ELEMENTS	Vm/Vf	UZrH2-H2O		UZrH2-Air	
		B²m (cm-2)	Infinite k	B²m (cm-2)	Infinite k
73	0.181	3.18E-02	1.56696	2.42E-02	1.53636
70	0.232	3.18E-02	1.57525	2.29E-02	1.53622
65	0.327	3.18E-02	1.58995	2.02E-02	1.52660
60	0.437	3.18E-02	1.60553	1.78E-02	1.53660
55	0.568	3.18E-02	1.62194	1.50E-02	1.53578
50	0.724	3.17E-02	1.63858	1.28E-02	1.53653
45	0.916	3.16E-02	1.65493	1.04E-02	1.53563
40	1.156	3.14E-02	1.66995	8.35E-03	1.53638
35	1.464	3.09E-02	1.68177	6.41E-03	1.53632
30	1.874	3.01E-02	1.68768	4.76E-03	1.53609
25	2.449	2.89E-02	1.68270	3.30E-03	1.53628
20	3.311	2.68E-02	1.65699	2.07E-03	1.53590
15	4.748	2.31E-02	1.59047	1.13E-03	1.53591
10	7.622	1.62E-02	1.43321	4.91E-04	1.53557
8	9.778	1.17E-02	1.31949	3.09E-04	1.53549
5	16.245	1.46E-03	1.04102	1.18E-04	1.53437

TABLE 5: Values of the B²m and infinite k in a homogeneous medium UZrH2-H2O - TRIGA fuel elements 119 and 424

119			
C(U) (g/cm³)	**H/U**	**B²m (cm-2)**	**Infinite k**
2.23	0	3.56E-02	1.59635
2.195	0.2	3.55E-02	1.59848
2	1.4	3.49E-02	1.61030
1.848	2.5	3.44E-02	1.61959
1.5	5.8	3.31E-02	1.64044
1	14.6	3.06E-02	1.66403
0.942	16.2	3.03E-02	1.66545

424			
C(U) (g/cm³)	**H/U**	**B²m (cm-2)**	**Infinite k**
3.94	0	3.17E-02	1.53713
3.85	0.2	3.17E-02	1.54080
3.5	0.9	3.15E-02	1.55568
3	2.1	3.12E-02	1.57889
2.5	3.9	3.08E-02	1.60425
2	6.5	3.03E-02	1.63143
1.5	10.9	2.96E-02	1.65865
1	19.7	2.84E-02	1.67902
0.5	46.0	2.50E-02	1.65276
0.4	59.2	2.35E-02	1.62488
0.3	81.2	2.10E-02	1.57274
0.2	125.1	1.67E-02	1.46785
0.1	256.9	7.08E-03	1.20543

TABLE 6: Isolated package MORET4 results - TRIGA fuel element 119 - UZrH2-H2O

NUMBER OF ELEMENTS	**Keff**	**σ (pcm)**	**Keff + 3σ**
10	0.821	295	**0.830**
9	0.813	294	0.822
8	0.805	292	0.814
7	0.783	278	0.792
6	0.745	294	0.754
5	0.709	285	0.718
4	0.642	290	0.651
3	0.539	267	0.547
2	0.401	196	0.407

TABLE 7: Isolated package MORET4 results - TRIGA fuel element 117 - UZrH2-H2O

NUMBER OF ELEMENTS	Keff	σ (pcm)	Keff + 3σ
10	0.809	290	**0.818**
9	0.790	280	0.798
8	0.776	280	0.784
7	0.752	291	0.760
6	0.711	277	0.719
5	0.671	268	0.679
4	0.594	275	0.602
3	0.498	185	0.503
2	0.364	201	0.370

TABLE 8: Isolated package MORET4 results - TRIGA fuel element 424 - UZrH2-H2O

NUMBER OF ELEMENTS	Container filled with air			Container filled with water		
	Keff	**σ (pcm)**	**Keff + 3σ**	**Keff**	**σ (pcm)**	**Keff + 3σ**
73	0.815	262	**0.823**	0.819	291	**0.828**
70	0.807	271	0.815	0.814	257	0.822
65	0.813	264	0.821	0.819	245	0.826
60	0.814	281	0.822	0.818	261	0.826
55	0.808	269	0.816	0.813	264	0.821
50	0.804	287	0.813	0.811	262	0.819
45	0.802	224	0.808	0.806	292	0.815
40	0.797	291	0.806	0.800	280	0.809
35	0.787	243	0.794	0.790	275	0.798
30	0.771	281	0.780	0.768	248	0.776
25	0.742	266	0.750	0.761	292	0.770
20	0.716	247	0.723	0.715	240	0.722
15	0.647	215	0.653	0.653	186	0.659
10	0.543	191	0.549	0.547	212	0.553
5	0.353	159	0.358	0.485	188	0.491

TABLE 9: Package network MORET4 results - TRIGA fuel element 119 - UZrH2-H2O

NUMBER OF ELEMENTS	Keff	σ (pcm)	Keff + 3σ
10	0.912	290	**0.921**
9	0.902	294	0.911
8	0.890	291	0.899
7	0.873	263	0.881
6	0.833	286	0.841
5	0.787	292	0.796
4	0.714	283	0.723
3	0.608	260	0.615
2	0.444	188	0.450

TABLE 10: Package network MCNP4B results - TRIGA fuel element 119

UZrH2-H2O (variable density)

DENSITY (g/cm^3)	Keff	σ (pcm)	Keff + 3σ water in cavity	Keff	σ (pcm)	Keff + 3σ air in cavity
0	0.797	192	0.802	0.712	197	0.718
0.1	0.806	215	0.813	0.722	227	0.729
0.2	0.816	214	0.823	0.735	199	0.741
0.3	0.827	224	0.833	0.749	192	0.754
0.4	0.837	200	0.843	0.763	224	0.769
0.5	0.843	206	0.849	0.774	188	0.780
0.6	0.848	209	0.854	0.784	216	0.791
0.7	0.862	182	0.867	0.796	197	0.802
0.8	0.869	195	0.875	0.807	193	0.813
0.9	0.882	248	0.889	0.819	207	0.826
1	0.886	206	**0.893***	0.831	227	**0.837****

*** MORET4 CODE RESULTS FOR THE SAME CONFIGURATION Keff + 3σ = 0.921**

**** MORET4 CODE RESULTS FOR THE SAME CONFIGURATION Keff + 3σ = 0.867**

TABLE 11: Package network MCNP4B results - TRIGA fuel element 119 UZrH2-H2O (variable density of water in the container and the cavity)

DENSITY (g/cm^3)	**Keff**	**σ (pcm)**	**Keff + 3σ**
0	0.831	227	0.837
0.1	0.836	210	0.842
0.2	0.843	176	0.848
0.3	0.854	217	0.860
0.4	0.853	212	0.859
0.5	0.861	264	0.869
0.6	0.869	199	0.875
0.7	0.868	210	0.874
0.8	0.876	191	0.882
0.9	0.887	230	**0.894**
1	0.886	206	0.893

TABLE 12: Package network MCNP4B results - TRIGA fuel element 119 UZrH2-H2O (variable density of water - burnt resin)

DENSITY (g/cm^3)	Keff	σ (pcm)	Keff + 3σ
0	0.886	206	**0.893**
0.1	0.884	196	0.890
0.2	0.880	216	0.887
0.3	0.873	208	0.879
0.4	0.869	221	0.876
0.5	0.861	211	0.868
0.6	0.863	209	0.870
0.7	0.859	195	0.865
0.8	0.856	226	0.863
0.9	0.849	214	0.856
1	0.847	216	0.853

TABLE 13: Isolated packages MORET4 results - TRIGA fuel element 119 homogeneous UZrH2-H2O (variable concentration, variable height)

C(U) (g/cm^3)	Fissile height (cm)	Keff	σ (pcm)	Keff + 3σ
1	20	0.671	198	0.677
	40	0.753	196	0.759
	60	0.768	197	0.774
	80	0.778	195	0.783
	100	0.789	197	0.795
	120	0.788	195	0.794
	147.5	0.790	198	0.796
1.5	20	0.708	195	0.714
	40	0.781	199	0.787
	60	0.797	198	0.803
	80	0.808	196	0.814
	100	0.816	196	0.822
	120	0.819	192	0.825
	147.5	0.816	194	0.822
2	20	0.720	198	0.726
	40	0.795	197	0.801
	60	0.815	196	0.821
	80	0.828	196	0.834
	100	0.832	197	0.838
	120	0.836	198	0.842
	147.5	0.835	195	0.841
2.20	20	0.731	197	0.737
	40	0.803	198	0.809
	60	0.823	198	0.829
	80	0.834	196	0.840
	100	0.833	198	0.839
	120	0.837	195	0.843
	147.5	**0.840**	**192**	**0.846**

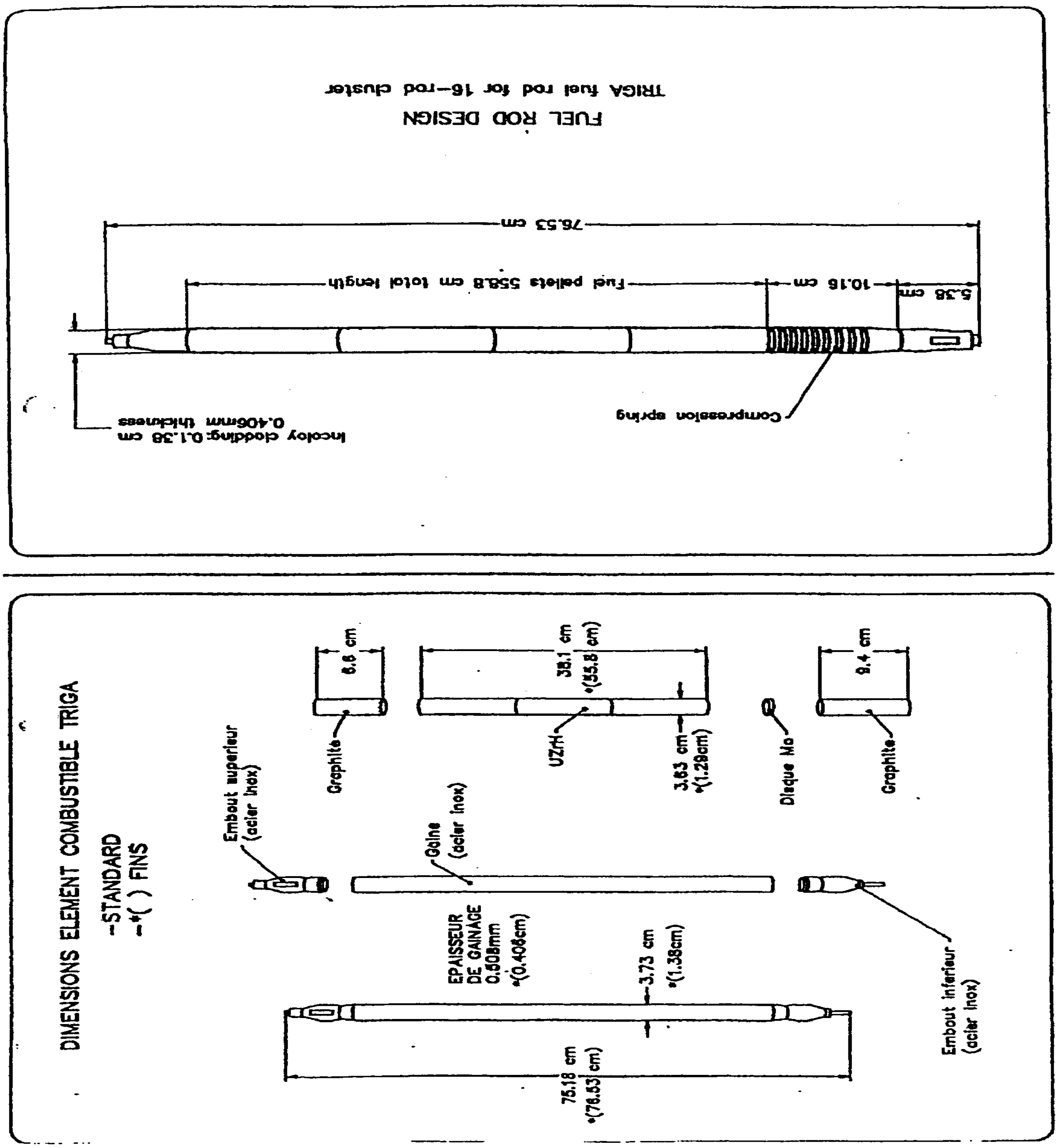
TABLE 14: Isolated packages MORET4 results - TRIGA fuel element 424 homogeneous UZrH2-H2O (variable concentration, variable height)

C(U) (g/cm^3)	**Fissile height (cm)**	**Keff**	**σ (pcm)**	**Keff + 3σ**
0.1	10	0.216	200	0.267
	20	0.348	196	0.354
	40	0.395	198	0.401
	60	0.413	194	0.418
	80	0.420	195	0.426
	100	0.417	198	0.423
	120	0.424	198	0.430
	147.5	0.421	192	0.427
0.5	10	0.477	197	0.483
	20	0.606	199	0.612
	40	0.672	196	0.678
	60	0.692	196	0.698
	80	0.709	193	0.714
	100	0.711	195	0.716
	120	0.713	193	0.718
	147.5	0.716	198	0.722
1	10	0.542	198	0.548
	20	0.664	196	0.670
	40	0.736	198	0.742
	60	0.758	196	0.764
	80	0.769	198	0.775
	100	0.775	196	0.780
	120	0.780	198	0.785
	147.5	0.780	199	0.786

C(U) (g/cm³)	Fissile height (cm)	Keff	σ (pcm)	Keff + 3σ
1.5	10	0.562	194	0.568
	20	0.686	198	0.691
	40	0.759	197	0.765
	60	0.780	198	0.786
	80	0.792	199	0.798
	100	0.794	199	0.800
	120	0.798	194	0.804
	147.5	0.801	197	0.807
2	10	0.579	199	0.585
	20	0.703	196	0.708
	40	0.769	198	0.775
	60	0.792	199	0.798
	80	0.800	197	0.806
	100	0.803	198	0.809
	120	0.808	197	0.814
	147.5	0.814	198	0.820
2.5	10	0.586	197	0.592
	20	0.706	199	0.712
	40	0.784	196	0.789
	60	0.799	198	0.805
	80	0.809	197	0.815
	100	0.812	196	0.818
	120	0.813	195	0.819
	147.5	0.818	198	0.824
3	10	0.597	198	0.602
	20	0.711	193	0.717
	40	0.780	195	0.786
	60	0.801	197	0.807
	80	0.810	198	0.816
	100	0.818	195	0.824
	120	0.819	196	0.825
	147.5	0.824	195	0.830

C(U) (g/cm³)	Fissile height (cm)	Keff	σ (pcm)	Keff + 3σ
3.85	10	0.603	198	0.609
	20	0.721	196	0.727
	40	0.789	196	0.795
	60	0.807	196	0.813
	80	0.814	194	0.820
	100	0.820	197	0.826
	120	0.822	198	0.828
	147.5	0.828	197	0.834

FIGURE 1: Dimensions of the TRIGA fuel elements

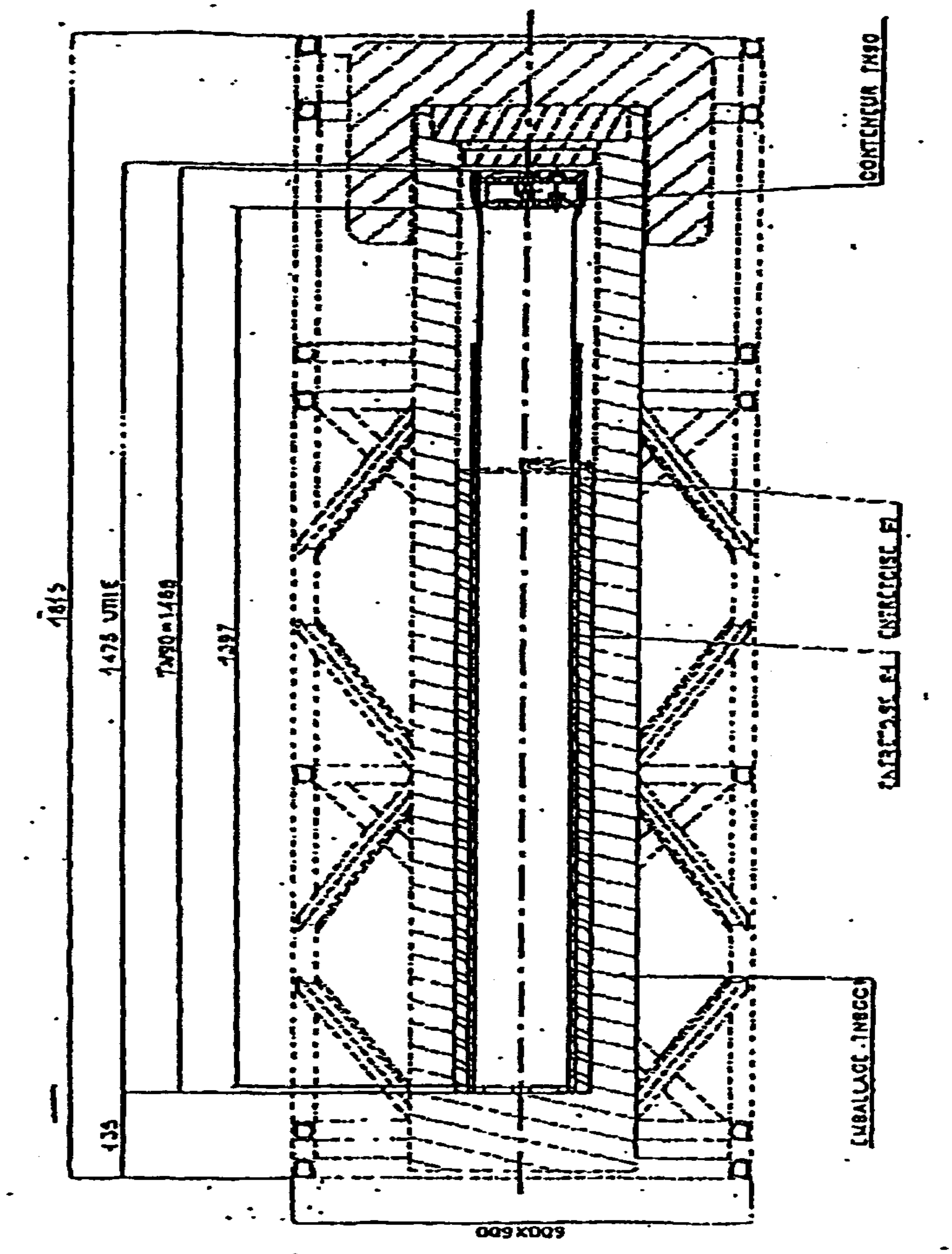


TRIGA FUEL ELEMENT DIMENSIONS

- STANDARD
- *( ) THIN

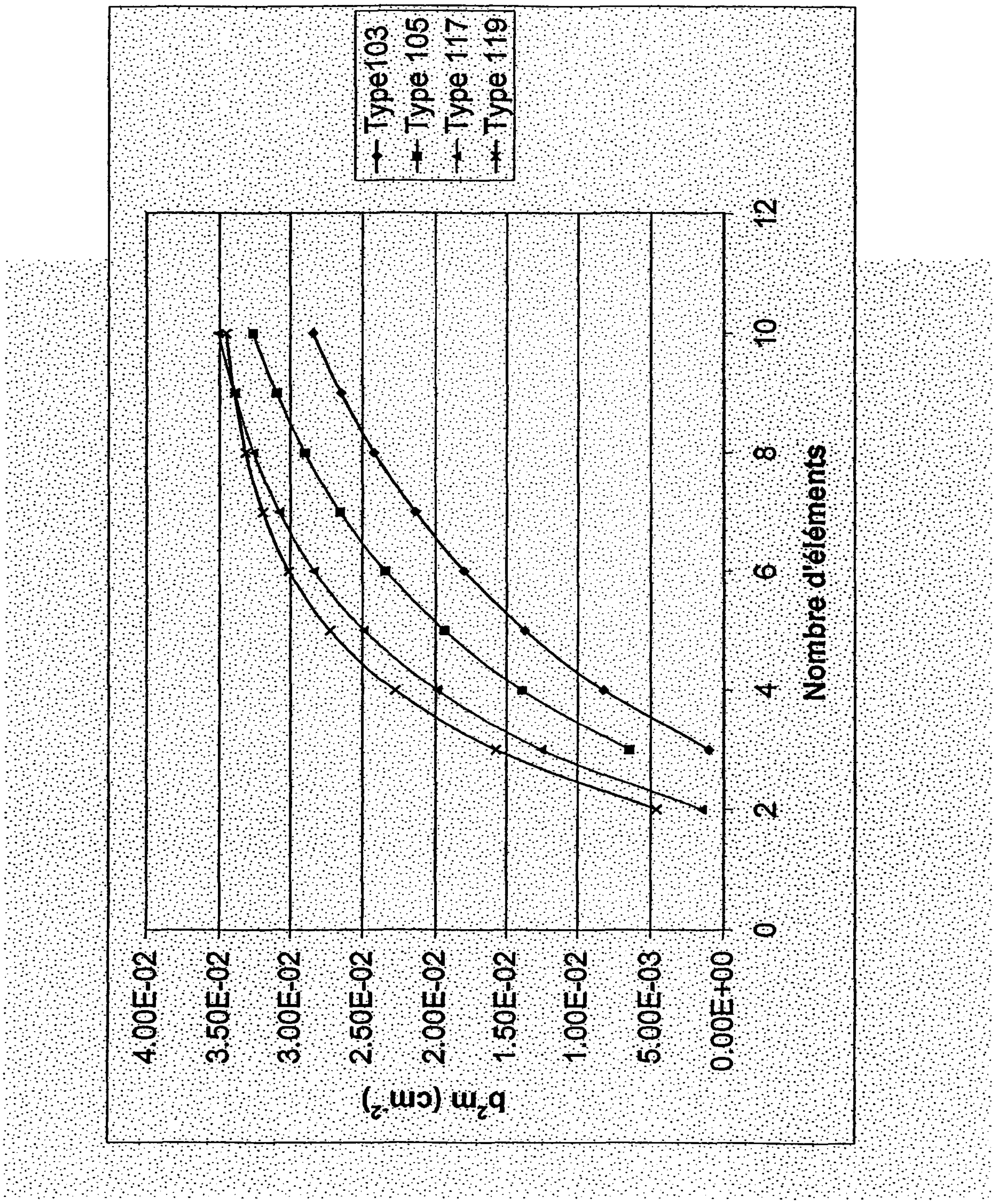
Embout supérieur (acier inox)	Top end-piece (stainless steel)
Graphite	Graphite
Gaine (acier inox)	Cladding (stainless steel)
EPAISSEUR DE GAINAGE	CLADDING THICKNESS
Disque Mo	Mo disc
Embout inférieur (acier inox)	Bottom end-piece (stainless steel)

FIGURE 2: Diagram of the TN BGC1 packaging loaded with a TN 90 packing container



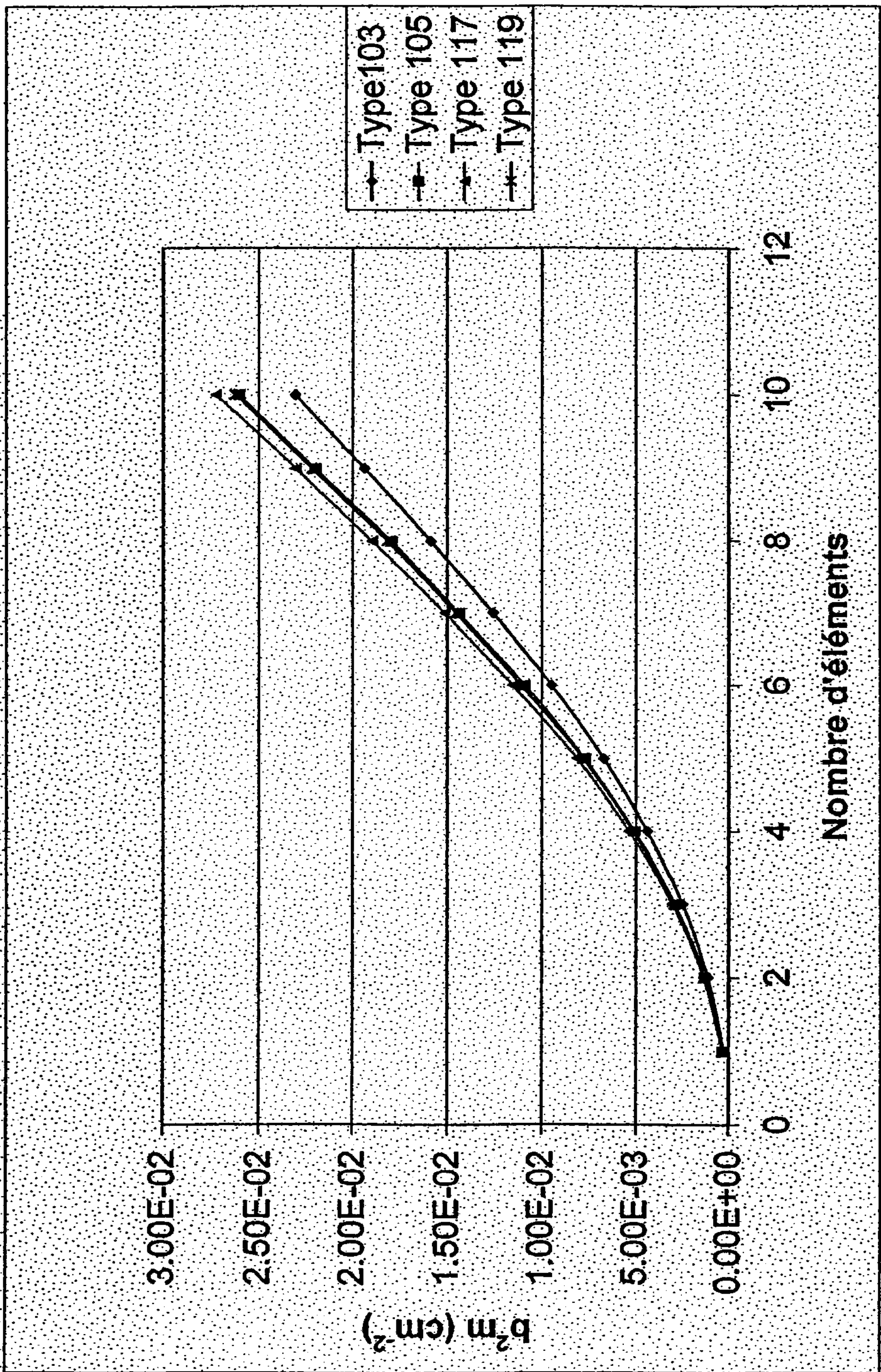
UTILE	EFFECTIVE
EMBALLAGE TN-BGC 1	TN-BGC 1 PACKAGING
ENTRETOISE E1	E1 STRUT
ENTRETOISE E2	E2 STRUT
CONTENEUR TN 90	TN 90 CONTAINER

FIGURE 3: Comparison of fissile media UZrH2-H2O - Standard fuel elements



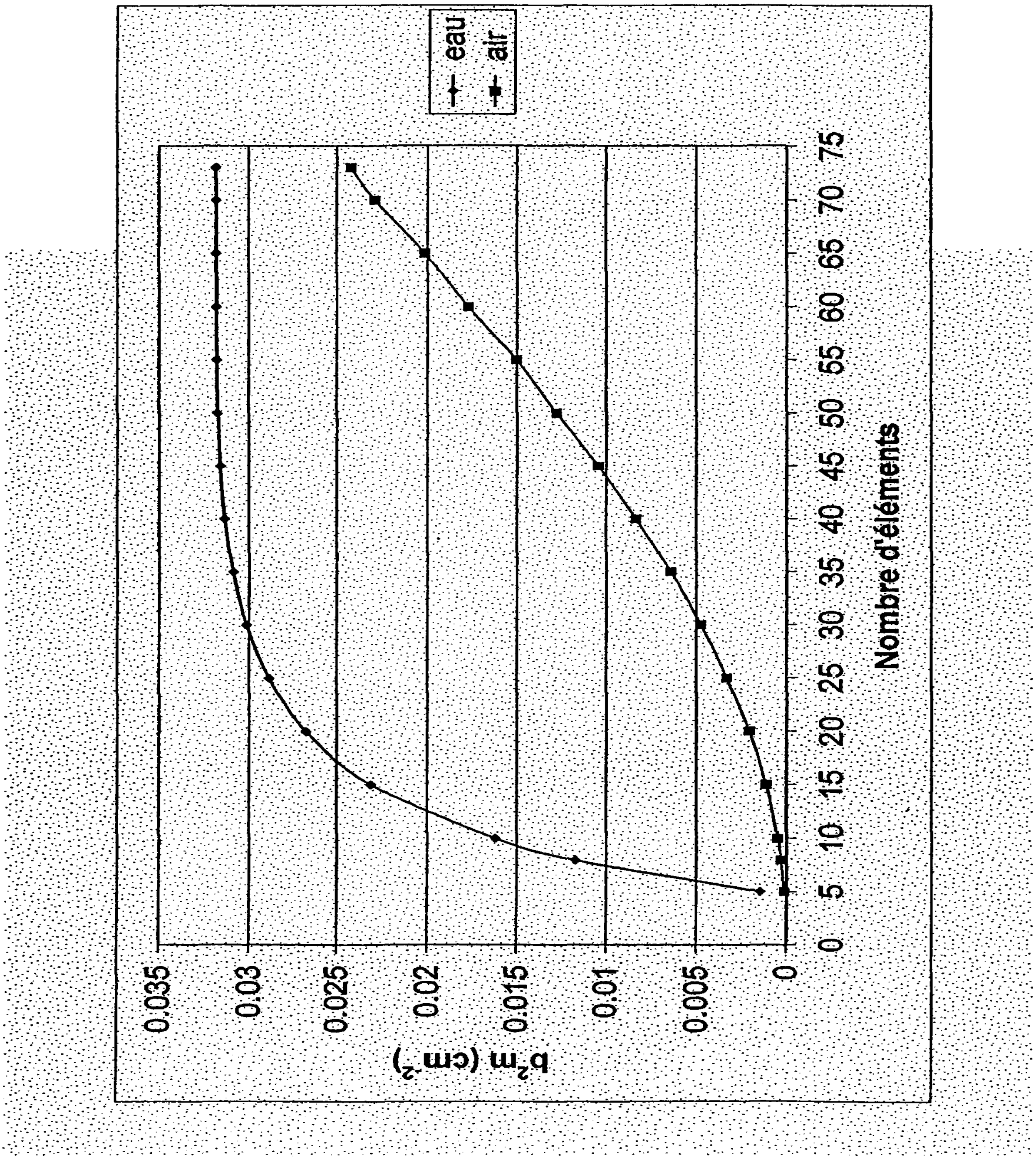
b²m (cm-2)	b²m (cm-2)
Nombre d'éléments	Number of elements

FIGURE 4: Comparison of fissile media UZrH2-Air - Standard fuel elements



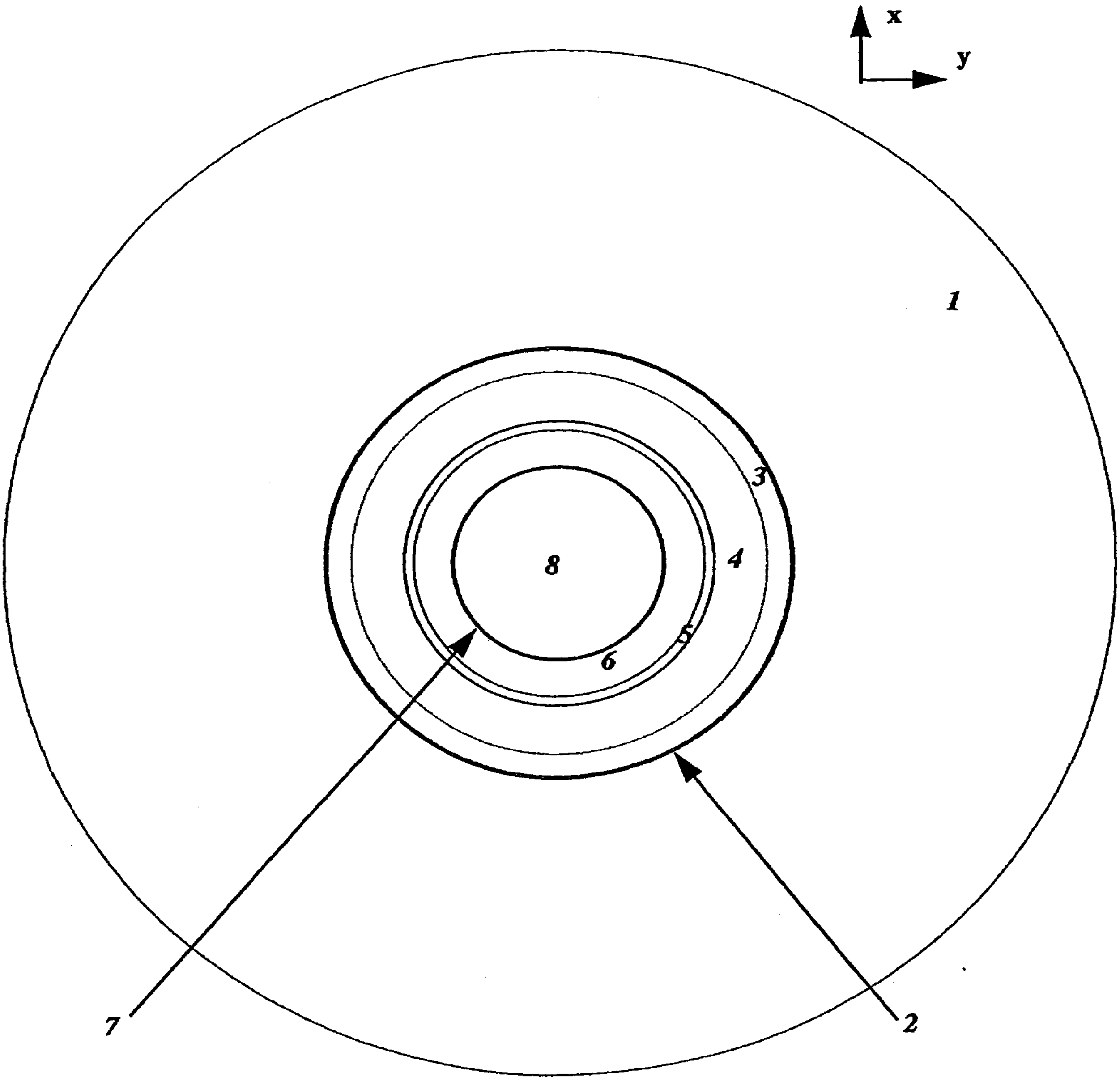
b²m (cm-2)	b²m (cm-2)
Nombre d'éléments	Number of elements

FIGURE 5: Comparison of fissile media UZrH2-H2O and Air - Thin fuel elements



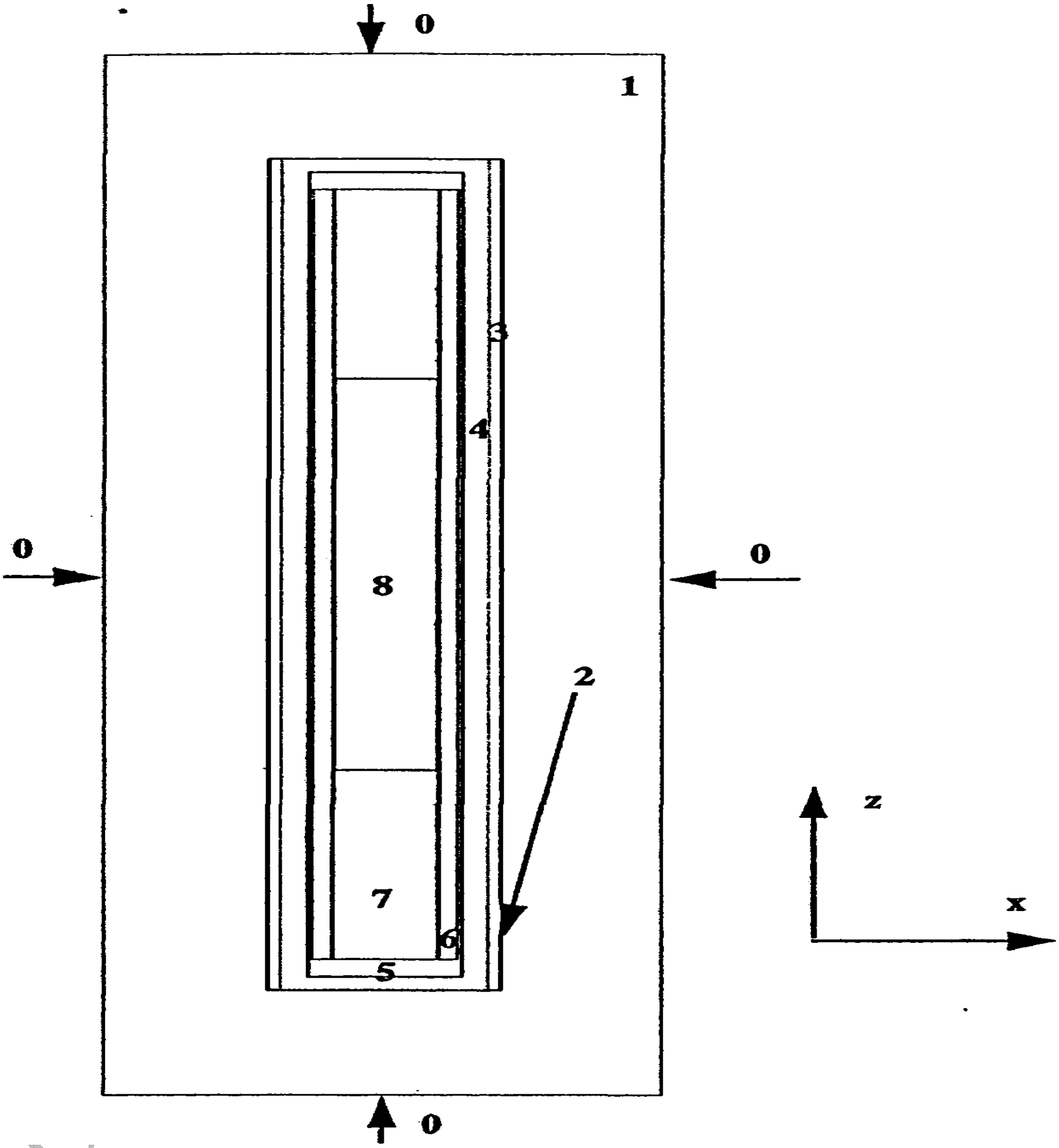
b²m (cm-2)	b²m (cm-2)
Nombre d'éléments	Number of elements
eau	water
air	air

FIGURE 6: MORET4 modelling - Radial section of the isolated package



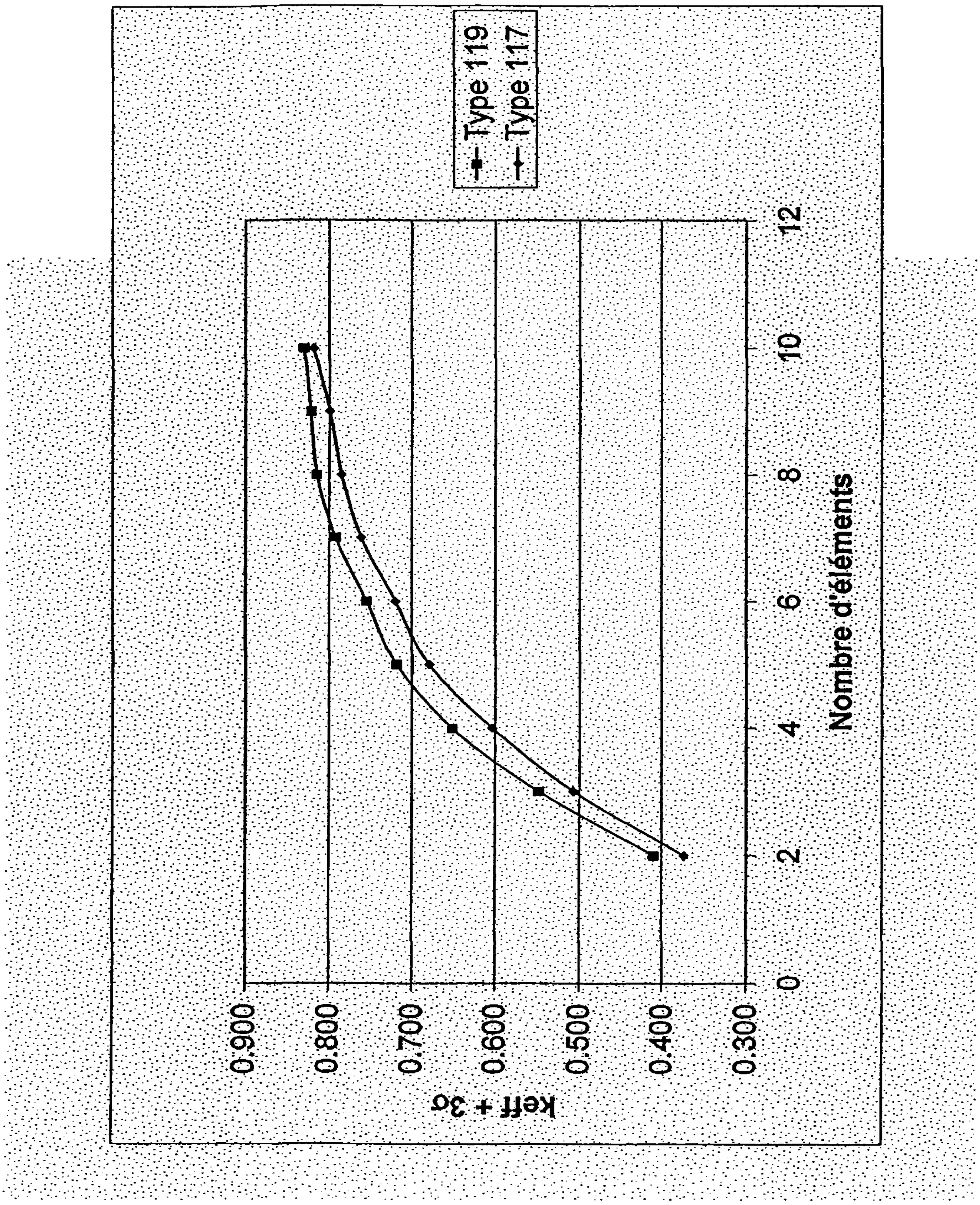
1:	couronne d'eau (20 cm)	ring of water (20 cm)
2:	virole externe d'acier e = 1.5 mm (Ø externe = 292 mm)	outer steel shell 1.5 mm thick (external Ø = 292 mm)
3:	résine brûlée (e = 15 mm)	burnt resin (15 mm thick)
4:	résine (e = 33 mm)	resin (33 mm thick)
5:	virole en acier (e = 6 mm)	steel shell (6 mm thick)
6:	cavité (Ø interne = 181 mm, hauteur = 1475 mm)	cavity (internal Ø = 181 mm, height = 1,475 mm)
7:	conteneur interne en acier (Ø interne = 130 mm, e = 2 mm)	internal steel container (internal Ø = 130 mm, 2 mm thick)
8:	milieu fissile (hauteur = 752 ou 770 mm, Ø 130 mm)	fissile medium (height = 752 or 770 mm, Ø 130 mm)

FIGURE 7: MORET4 modelling - Axial section of the isolated package



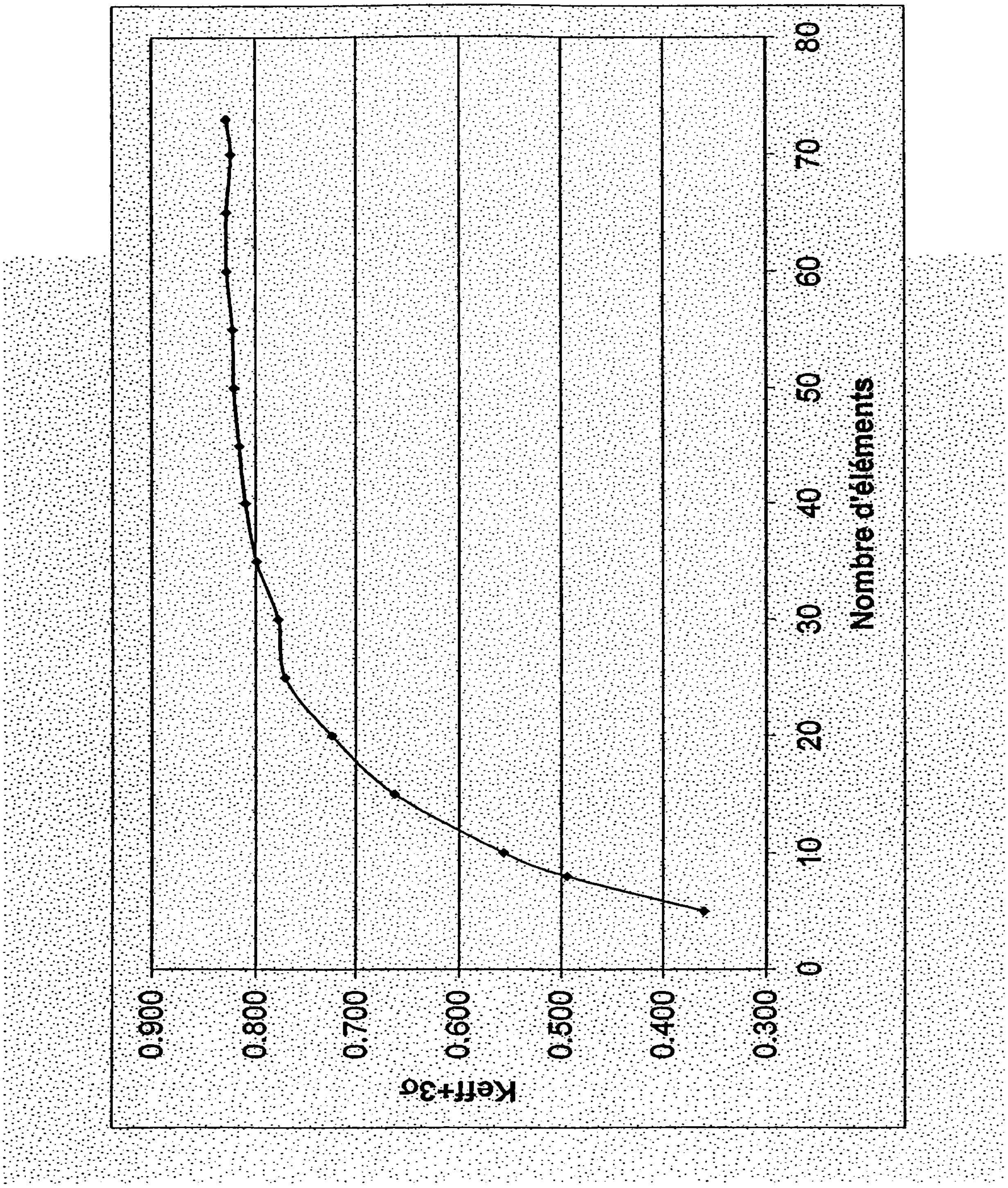
1:	couronne d'eau (20 cm)	ring of water (20 cm)
2:	virole externe d'acier e = 1.5 mm (Ø externe = 292 mm)	outer steel shell 1.5 mm thick (external Ø = 292 mm)
3:	résine brûlée (e = 15 mm)	burnt resin (15 mm thick)
4:	résine (e = 33 mm)	resin (33 mm thick)
5:	virole en acier (e = 6 mm)	steel shell (6 mm thick)
6:	cavité (Ø interne = 181 mm, hauteur = 1475 mm)	cavity (internal Ø = 181 mm, height = 1,475 mm)
7:	conteneur interne en acier (Ø interne = 130 mm, e = 2 mm)	internal steel container (internal Ø = 130 mm, 2 mm thick)
8:	milieu fissile (hauteur = 752 ou 770 mm, Ø 130 mm)	fissile medium (height = 752 or 770 mm, Ø 130 mm)

FIGURE 8: Isolated package results - TRIGA fuel elements 119 - 117 - UZrH2-H2O



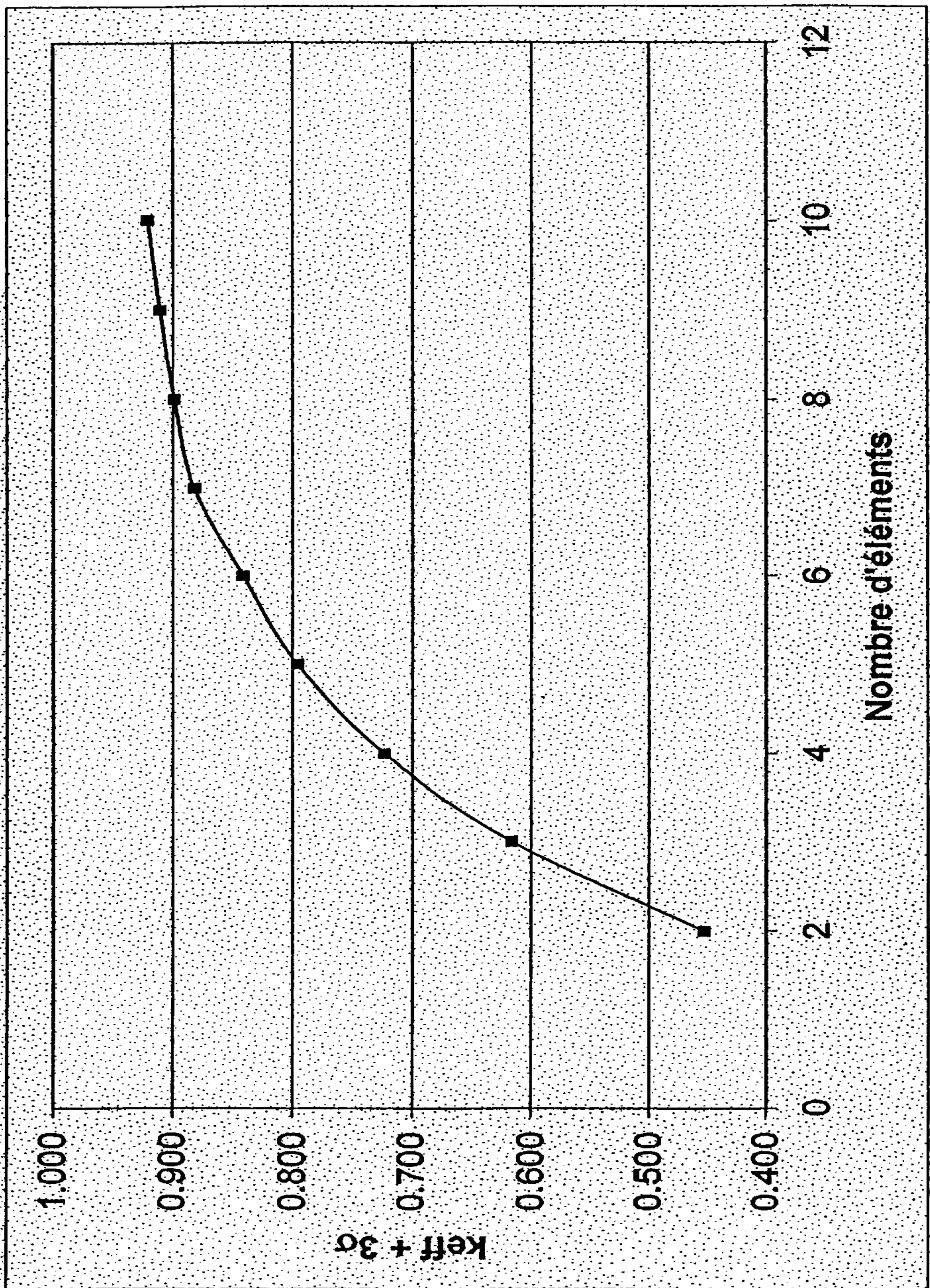
Keff + 3σ	Keff + 3σ
Nombre d'éléments	Number of elements

FIGURE 9: Isolated package results - TRIGA fuel elements 424 - UZrH2-H2O



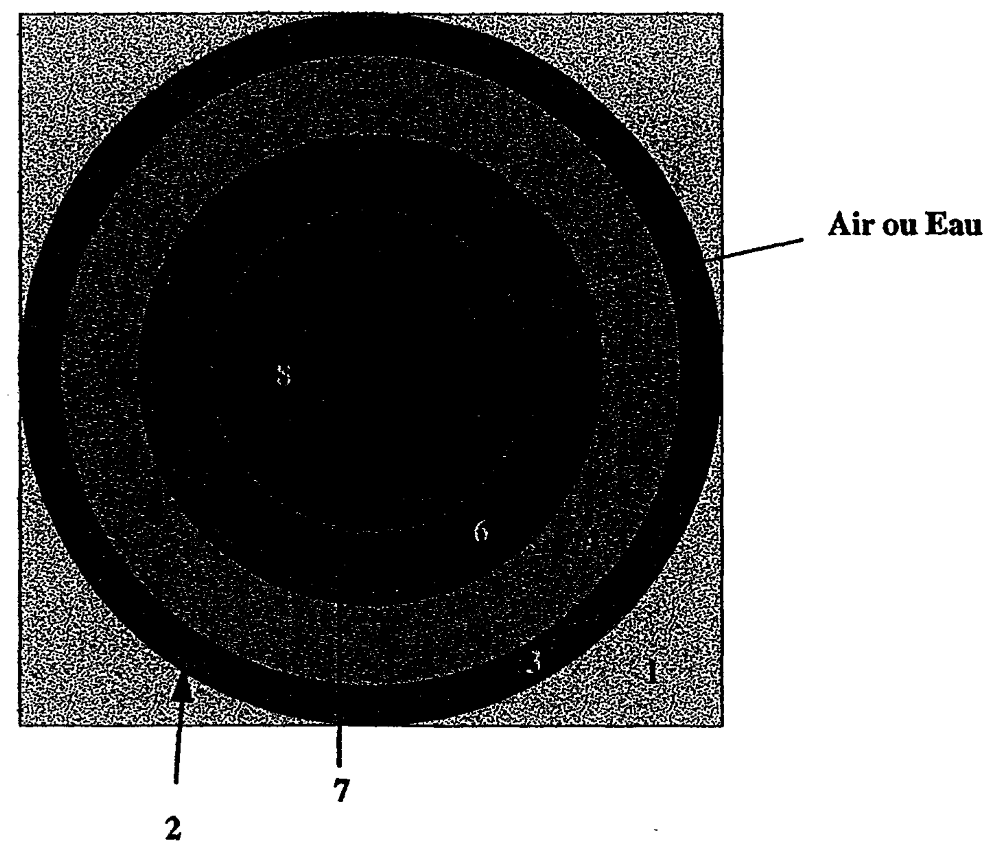
Keff + 3σ	Keff + 3σ
Nombre d'éléments	Number of elements

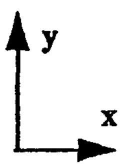
FIGURE 10: Package network results - TRIGA fuel element 119 - UZrH2-H2O



Keff + 3σ	Keff + 3σ
Nombre d'éléments	Number of elements

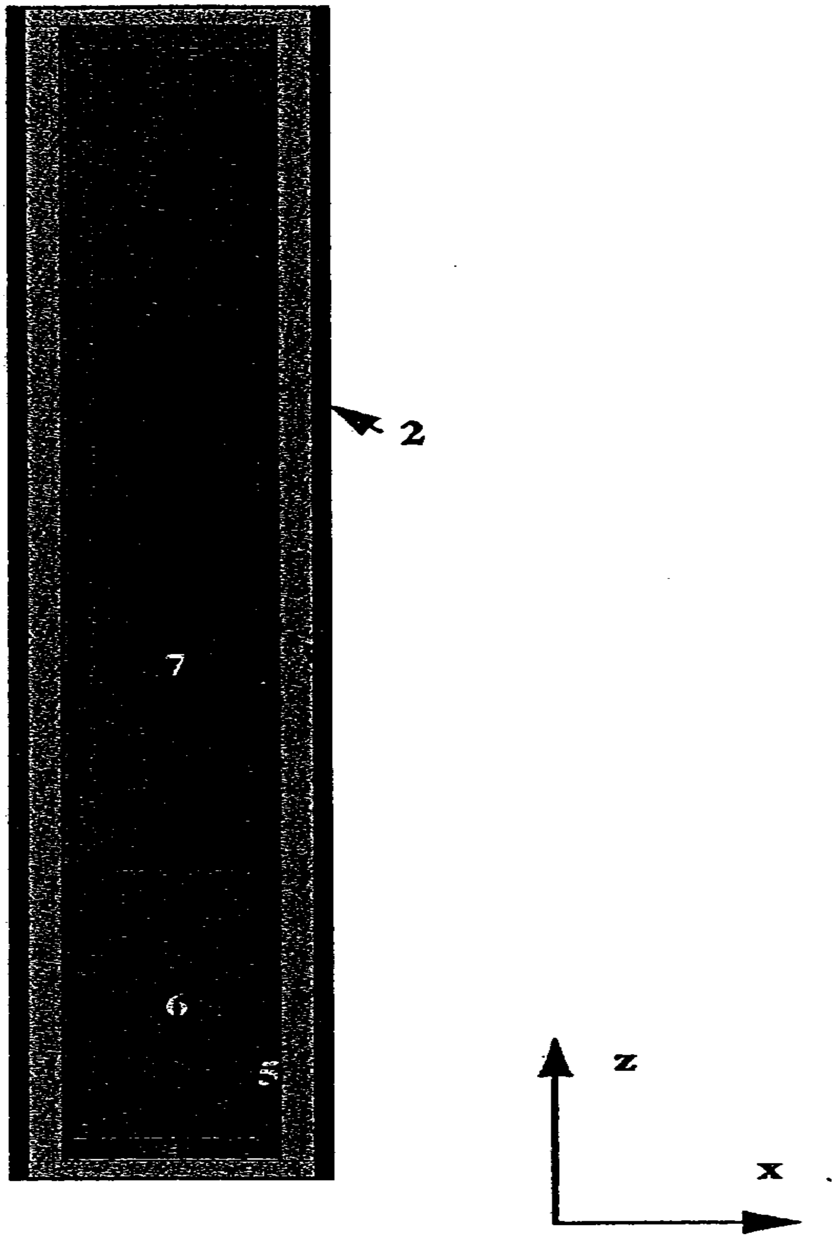
FIGURE 11: MCNP4B modelling - Radial section of the package network





	Air ou Eau	Air or Water
1:	air	air
2:	virole externe d'acier e = 1.5 mm (Ø externe = 292 mm)	outer steel shell 1.5 mm thick (external Ø = 292 mm)
3:	résine brûlée (e = 15 mm)	burnt resin (15 mm thick)
4:	résine	resin
5:	virole en acier (e = 6 mm)	steel shell (6 mm thick)
6:	cavité (Ø interne = 181 mm, hauteur = 1475 mm)	cavity (internal Ø = 181 mm, height = 1,475 mm)
7:	conteneur interne en acier (Ø interne = 130 mm, e = 2 mm)	internal steel container (internal Ø = 130 mm, 2 mm thick)
8:	milieu fissile (hauteur = 752)	fissile medium (height = 752)

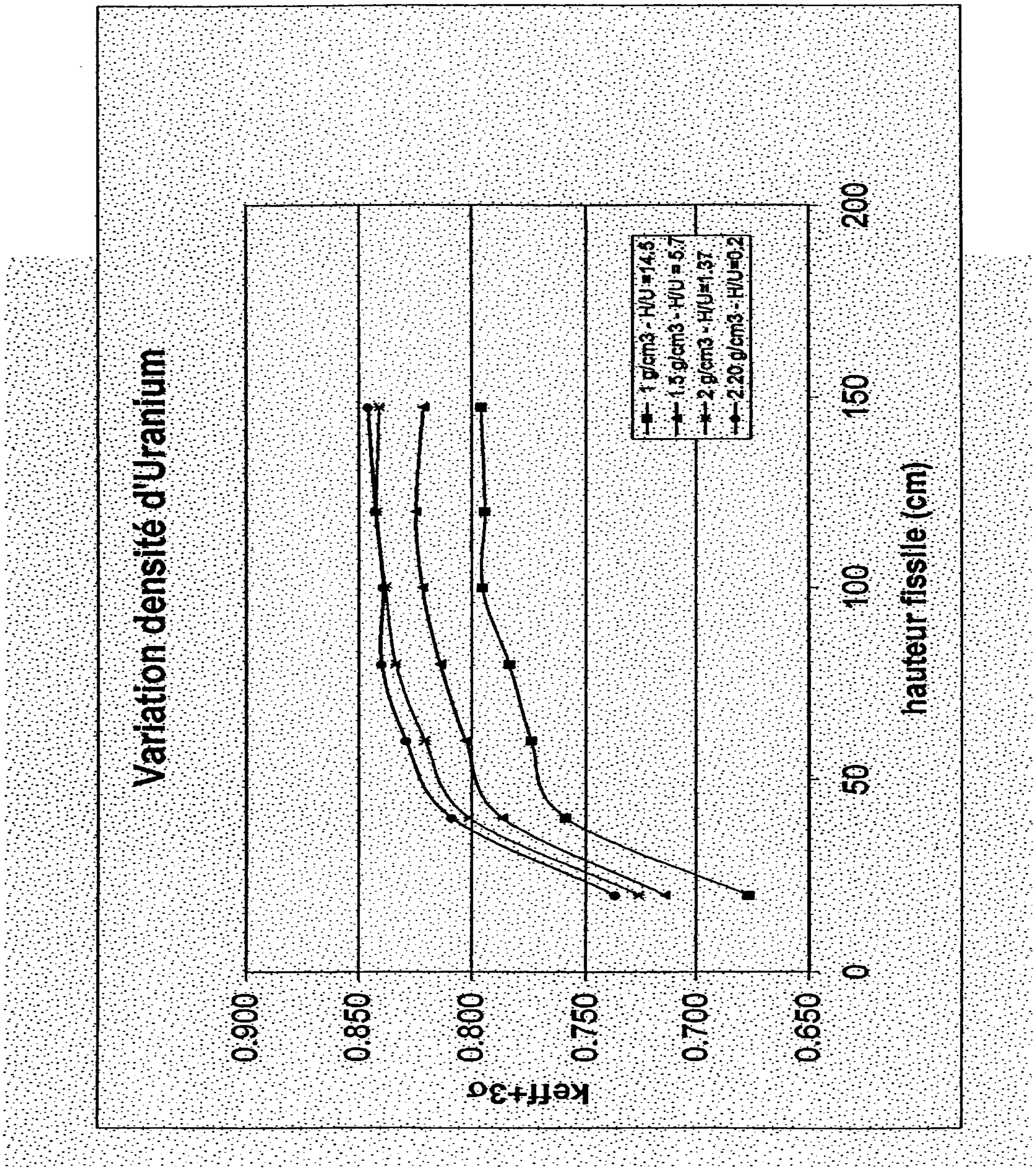
FIGURE 12: MCNP4B modelling - Axial section of the package network



1:	virole externe d'acier e = 1.5 mm (Ø externe = 292 mm)	outer steel shell 1.5 mm thick (external Ø = 292 mm)
2:	résine brûlée (e = 15 mm)	burnt resin (15 mm thick)
3:	résine (e = 33 mm)	resin (33 mm thick)
4:	virole en acier (e = 6 mm)	steel shell (6 mm thick)
5:	cavité (Ø interne = 181 mm, hauteur = 1475 mm)	cavity (internal Ø = 181 mm, height = 1,475 mm)
6:	conteneur interne en acier (Ø interne = 130 mm, e = 2 mm)	internal steel container (internal Ø = 130 mm, 2 mm thick)
7:	milieu fissile (hauteur = 752)	fissile medium (height = 752)

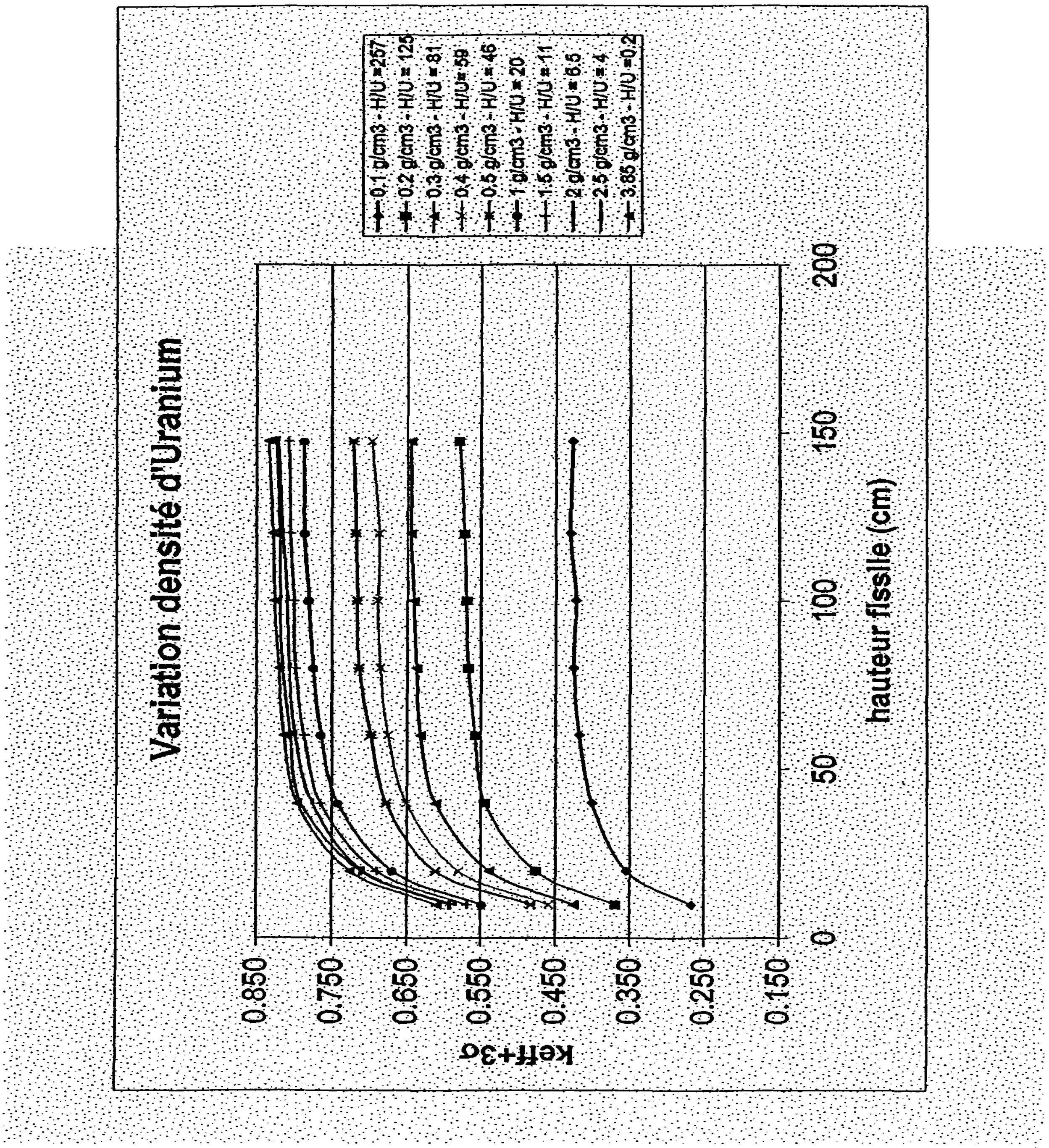
FIGURE 13: Isolated package MORET4 results - TRIGA fuel element 119

homogeneous UZrH2-H2O (variable concentration, variable height)



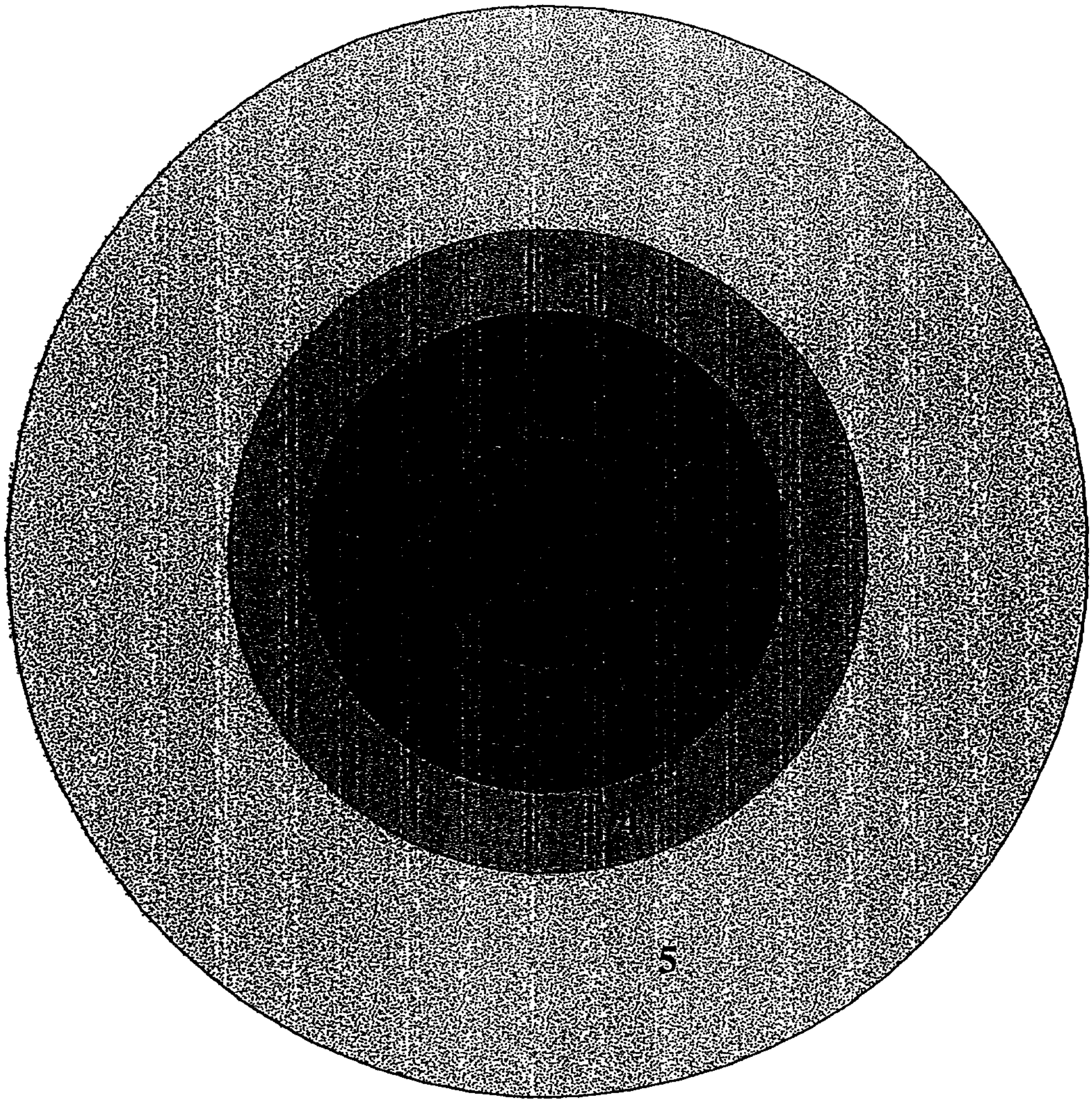
Variation densité d'Uranium	Uranium density variation
Keff + 3σ	Keff + 3σ
hauteur fissile (cm)	fissile height (cm)

FIGURE 14: Isolated package MORET4 results - TRIGA fuel element 424 homogeneous UZrH2-H2O (variable concentration, variable height)



Variation densité d'Uranium	Uranium density variation
Keff + 3σ	Keff + 3σ
hauteur fissile (cm)	fissile height (cm)

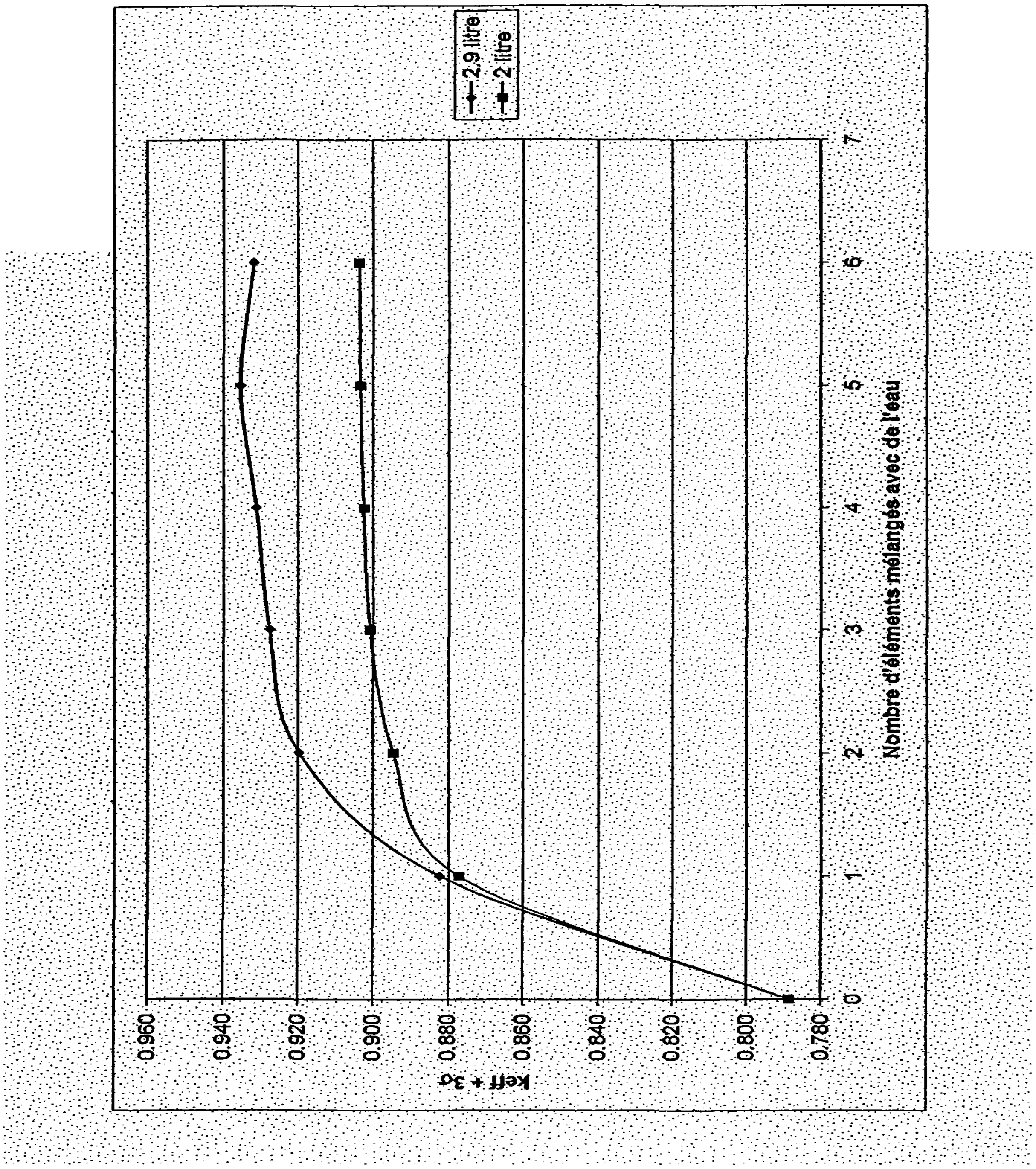
FIGURE 15: MCNP4B modelling - Configuration for air transport



1:	milieu fissile « mouillé »	“wet” fissile medium
2:	milieu fissile sec	dry fissile medium
3:	virole externe d’acier	outer steel shell
4:	résine	resin
5:	couronne d’eau 20 cm	20 cm ring of water

FIGURE 16: MCNP4B results - Air transport

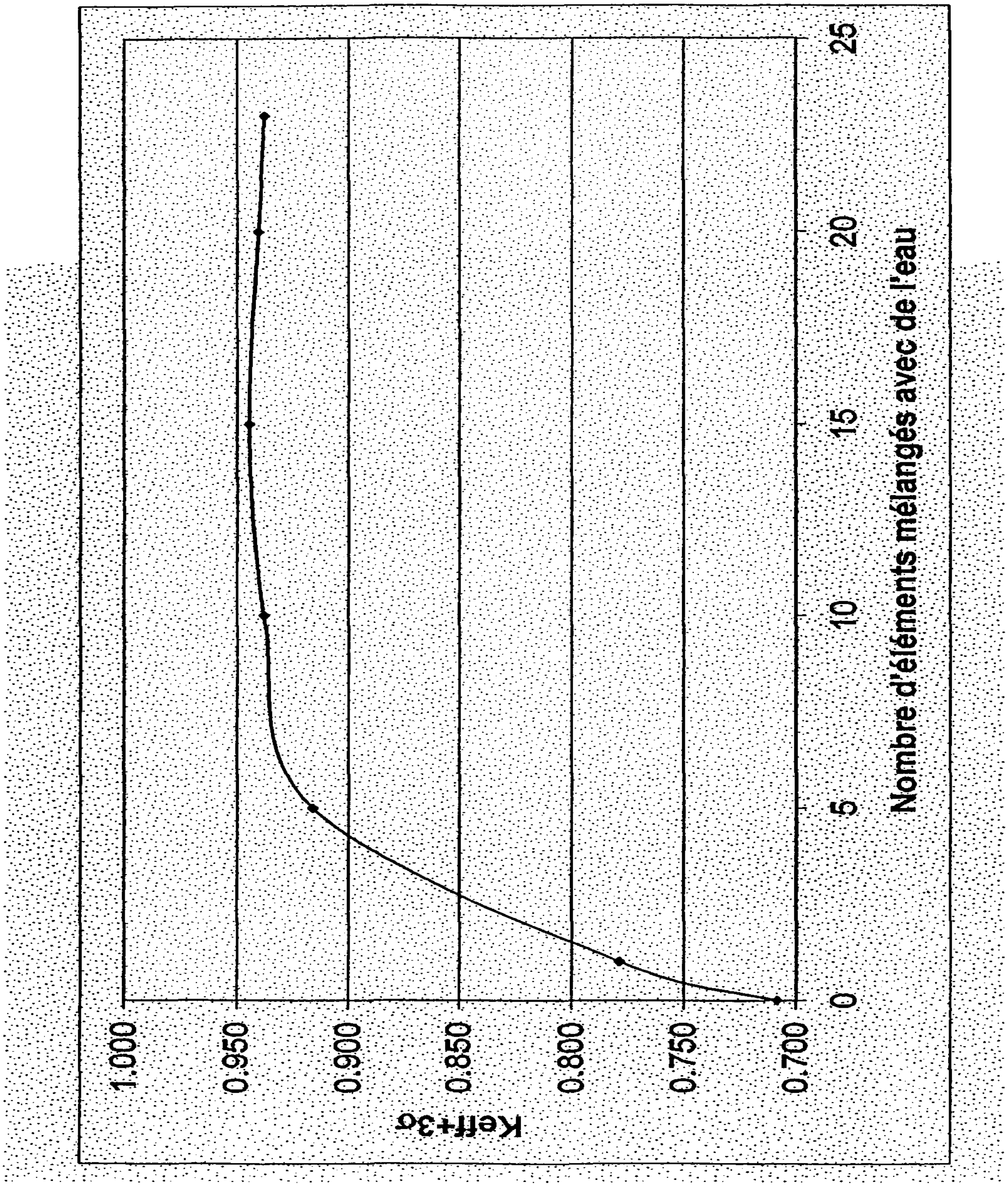
Results for 6 TRIGA fuel elements 119 - 2.9 and 2 litres of water



Keff + 3σ	Keff + 3σ
nombre d'éléments mélangés avec de l'eau	number of elements mixed with water

FIGURE 17: MCNP4B - Air transport

Results for 23 TRIGA fuel elements 424 - 3.65 litres of water



Keff + 3σ	Keff + 3σ
nombre d'éléments mélangés avec de l'eau	number of elements mixed with water

NUCLEAR ENERGY DIVISION
Department of Installation and Packaging Projects
CEA Transport package service

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Purpose of the document:

This chapter specifies the instructions for use and the acceptance and maintenance test programmes

CEA/DEN/CAD/DPIE/SET
DO 89 27/02/08
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Field of application and summary:

APPENDICES (included in this document and therefore in global page numbers)			ATTACHMENTS (separate page numbers, identification and formal procedures)		
No.	TITLES	N° of pages	No.	TITLES	N° of pages

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Issue	Date	Comments/Purpose of the change of issue
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Visa		Cf. page 2	
	WRITTEN BY	**CHECKED BY**	**APPROVED BY**

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Issue	Checked by	Visa
A	J.M. MURE (CEA/DEN/DPIE/SA2S)	
	S. CLAVERIE-FORGUES (CEA/DEN/DPIE/SET)	

TABLE OF CONTENTS

1 INSTRUCTIONS FOR USE

1.1 PREAMBLE

This paragraph specifies the instructions for use provided for the TN-BGC 1 packaging.

These instructions for use are likely to be modified in order to take account of the specific needs of each user, any special precautions required by the competent authority, the regulations and codes applicable in countries in which the packaging is used and in the light of experience.

Special procedures are drawn up by the users on the basis of these instructions and include a check-list to ensure that the various operations are carried out correctly.

The packaging is designed to transport extremely varied uranium- and/or plutonium-bearing materials contained in an internal arrangement loaded in the body of the packaging.

The loading (or unloading) and the transport takes place dry, with inerting of the cavity for certain contents.

1.2 INSTRUCTIONS FOR LOADING THE PACKAGING

The configuration is such that the packaging is already located in the loading area.

1.2.1 Preparation

A radiological inspection (measuring dose rates and surface contamination) takes place before any operation.

It is also important:

- inspect the seal faces of the body and the plug as well as the O-rings and replace them where appropriate,
- check for deterioration in the quick-connect coupling and the viewing port and for leaktightness of the containment seals.

The operations listed below are then carried out:

- detach the grips, turn the cover so as to release the fixing ring anchoring feet, lift the cover with its handles (using a crane if necessary),
- loosen the quick-connect cap tightening nut and remove the nut and the cap,
- install the plug prestressing tool, fix it to the internal threads in the body flange provided for that purpose, connect the pump and pump until a force of 30 kN is reached,
- loosen the compression ring until the bayonet ring can be disconnected,
- remove the plug prestressing tool,
- remove the compression ring, the bayonet ring and the plug.

1.2.2 Loading the internal arrangement

The operations listed below are carried out:

- install the necessary transport fillers (based on the arrangement and the content - see Chapter 1) in the cavity it that has not already been done during a previous transport operation,
- install the internal arrangement in the packaging.

1.2.3 Closing the packaging

The operations listed below are carried out:

- clean the flange and make sure that the flange seal face is clean (no scratching, no dust which could cause a leak),
- clean the plug seals with alcohol,
- insert the plug, the bayonet ring and the compression ring,
- install the plug prestressing tool, fix it to the internal threads in the body flange provided for that purpose, connect the pump until a force of 30 kN is reached,
- inspect visually (via the holes in the flange provided for this purpose) that the plug seals are flattened properly,
- connect the bayonet ring + compression ring assembly and tighten the compression ring,
- remove the plug prestressing tool.

When the cavity has to be made inert (see the stipulations in Chapter 1 for this):

- connect the quick-connect coupling to the vacuum system in the packaging and pressurise the packaging to 100 Pa absolute,
- connect the quick-connect coupling to the inert gas system and pressurise the packaging to the desired pressure ($0.2.10^5$ Pa absolute),
- clean the bearing face on the quick-connect cap and make sure that the seal face is clean (no scratching, no dust which could cause a leak),
- clean the cap seals with alcohol,
- insert the quick-connect cap and the tightening nut; tighten the nut to a torque of 50 N.m,
- loosen the test plug,
- check using a suitable method that the leak rate measured is less than the value indicated in Chapter 3.4 for the content being transported,
- clean the test plug and its seal and screw in the plug (torque 10 N.m).

When the cavity does not have to be made inert (see the stipulations in Chapter 1 for this):

- clean the bearing face on the quick-connect cap and make sure that the seal face is clean (no scratching, no dust which could cause a leak),
- clean the cap seals with alcohol,
- insert the quick-connect cap and the tightening nut; tighten the nut to a torque of 50 N.m,
- loosen the test plug,
- check using a suitable method that the leak rate measured is less than the value indicated in Chapter 3.4 for the content being transported,
- clean the test plug and its seal and screw in the plug (torque 10 N.m). Every handling point (upper oblique bar in the cage) is subjected to a force equal to 1.5 times its nominal load.

1.2.4 Fixing the cover

The operations listed below are carried out:

- position the cover on the top of the packaging (move with the handles),
- turn it so that the anchoring feet slot into the corresponding fixing lugs,
- attach the grips,
- fix the seals to the grips.

1.2.5 Despatching the packaging

The operations listed below are carried out:

- measure the radiation, temperatures and contamination and fill in the transport file, adding any results from sampling analyses performed,
- measure temperatures to check that the temperature on the accessible packaging surfaces in the shade, when thermal equilibrium is reached, does not exceed 50°C for transport under non-exclusive use and 85°C for transport under exclusive use. Accessible surface means here the outside walls of the cage and the upper part of the packaging cover or the walls of the protective housing where appropriate,
- fill in the regulatory labels and affix them to the packaging,
- transfer the packaging to the vehicle; the packaging is handled using a forklift (two fork slots are provided) or via the bars in the upper part using straps or slings. When the packaging being handled is horizontal, tip it then lift it with straps slung around the cage.

  Comment: it is recommended to transport the packaging vertically,
- stow the packaging on the vehicle; this stowing depends on the vehicle and whether or not a protective transport housing is used,
- measure the radiation around the vehicle and fill in the transport file,
- fill in the regulatory labels and affix them to the vehicle,

1.3 INSTRUCTIONS FOR UNLOADING THE PACKAGING

1.3.1 Arrival

The operations listed below are carried out:

- familiarise yourself with the transport file and any comments therein,
- inspection the vehicle: radiation, contamination, temperatures,
- unstow the packaging from the vehicle or its protective transport housing,
- unload the packaging; the packaging is handled using a forklift (two fork slots are provided) or via the bars in the upper part using straps or slings. When the packaging being handled is horizontal, lift it with straps slung around the cage.
- inspect the packaging: radiation, contamination, temperatures.

1.3.2 Preparation

The operations listed below are carried out:

- detach the grips, turn the cover so as to release the fixing ring anchoring feet, lift the cover with its handles (using a crane if necessary),
- loosen the compression ring on the quick-connect cap and remove it,

- take a gaseous sample if necessary and return the containment to atmospheric pressure if necessary.
- install the plug prestressing tool, fix it to the internal threads in the body flange provided for that purpose, connect the pump and pump until a force of 30 kN is reached,
- loosen the compression ring until the bayonet ring can be disconnected,
- remove the plug prestressing tool,
- remove the compression ring, the bayonet ring and the plug.

1.3.3 Unloading the internal arrangement

The operations listed below are carried out:

- remove the internal arrangement
- extract the fillers if necessary.

2 ACCEPTANCE AND MAINTENANCE TEST PROGRAMMES

This paragraph describes the acceptance, commissioning and maintenance programmes scheduled to take place upon receipt and during the life of the packaging.

2.1 ACCEPTANCE TEST PROGRAMME

2.1.1 Examining purchasing and manufacturing documents

This examination concerns all inspection and testing documents and reports relating to procurement and manufacture in order to ensure that the packaging has been manufactured in accordance with the established specifications and Quality Assurance Programme.

This examination will cover the following points in particular:

- compliance of materials with the specifications set forth in Chapters 1 and 2,
- dimensional conformity of the various components: the dimensions must comply with those marked on the design drawings appended to this safety file (attachment 2-1) with the exception of justified deviations or repairs marked on the assembly drawings,
- conformity of performance of processes and operator qualification; these processes cover in particular:
  - ➢ welds,
  - ➢ installation of the neutron shielding,
  - ➢ non-destructive testing (acceptance of equipment and check on performance of certain welds).
- conformity of acceptance results with the criteria.
- inspector qualification.

2.1.2 Visual examination

All accessible surfaces are visually inspected to ensure that they are free from grease, oil and any other

contaminating deposit; the surfaces will be cleaned if necessary.

The internal and external surface conditions are inspected in order to ensure that they comply with manufacturing specifications. Any non-compliance must be repaired or justified.

2.1.3 Handling point tests

Every handling point (upper oblique bar in the cage) is subjected to a force equal to 1.5 times its nominal load.

A liquid penetrant inspection of the welds tested is carried out before each test. The test is considered satisfactory if no permanent deformation is detected and if the liquid penetrant inspection is satisfactory.

2.1.4 Leaktightness tests

The packaging is checked for leaktightness by measuring the leak rate from the containment.

The test is considered satisfactory if the overall leak from the containment does not exceed 10^{-8} $Pa.m^3.s^{-1}$ SHeLR (standard helium conditions according to ISO standard 12 807).

2.1.5 Satisfactory Operation Test

The purpose of these tests is to check that the various packaging components can be assembled and operated easily without jamming or abnormal friction and that the packaging is suitable for its intended use. These tests cover the following points:

- fitting the seals into their throats,
- installing and removing closing parts,
- installing and removing the cover.

2.1.6 Markings

The regulation packaging identification plates will be attached after the examinations, inspections and tests have been performed and deemed satisfactory.

2.2 MAINTENANCE PROGRAMME

The following paragraphs describe the projected checks, tests and critical component replacements scheduled during packaging maintenance.

This schedule is subject to change in line with experience acquired during use of the packaging.

It provides for control sheets drawn up for each packaging as to its compliance or otherwise with the operations performed. A non-conformity sheet is opened in the event of a non-conformity.

2.2.1 Small maintenance tasks

These tasks take place every fifteen transport operations or every three years. They cover:

- changing O-rings,
- inspecting threads,
- inspecting closing parts,

- inspecting packaging dimensions and appearance,
- implementing leaktightness tests on the closing system: criterion 10^{-6} $Pa.m^3.s^{-1}$ SLR when the leaktightness criterion checked before each despatch is strictly less than $6.6.10^{-4}$ $Pa.^{m3}{\cdot}s^{-1}$; or criterion $6.6.10^{-5}$ $Pa.m^3.s^{-1}$ SLR in the other cases.

Note: the identification numbers given below relate to the drawings supplied in attachment 2-1 to this file.

2.2.1.1 Changing O-rings

The operations described below are carried out:

- 1/ open the packaging and especially remove rings,
- 2/ replace the O-rings in the plug closure,
- 3/ replace the O-ring in the test plug,
- 4/ replace the O-rings in the Staubli plug and connection,
- 5/ close the packaging, especially:
  - ➢ refit the test plug to the plug closure (item 2), complying with the tightening torque values (10 N.m),
  - ➢ refit the Staubli connector to the plug closure (item 2), complying with the tightening torque values (30 N.m),
  - ➢ loosen the tightening nut (item 8), complying with the tightening torque values (50 N.m),

2.2.1.2 Inspecting threads, closing parts, dimensions and appearance

The threads can be inspected by screwing and loosening the opposing parts (no jamming or major play) or by using the related calibrated buffers.

The seal faces are inspected visually, using a magnifying lens where needed.

The principle is the following:

<u>1/ Visual inspection</u>

The visual inspection will be made with suitable lighting of:

- all closing system parts, more especially the seal faces,
- the internal and external surfaces of packagings.

The following defects will be sought:

- adhesion of foreign bodies,
- abrasions, notches, bumps and shocks,
- scratches on the seal faces,

They will be dealt with accordingly if they are not found in a part providing a leaktightness function. Otherwise the part concerned should be reworked, replaced or rejected depending on the scale of the defect and the type of component.

2/ Functional inspections

All the threads noted on the control sheets must be checked using a suitable procedure.

All closing system parts are tested for their assembly and disassembly.

The packaging plug STAUBLI stem must be checked that it is working properly.

2.2.1.3 TN-BGC 1 packaging leaktightness test

For type B(U) packaging, leaktightness is controlled by a rise in pressure; compliance with a leak rate of 6.66×10^{-5} Pa .$m^3.s^{-1}$ is mandatory.

For the type B(M) packaging, leaktightness is controlled by helium detection; compliance with a leak rate of 10^{-6} Pa .$m^3.s^{-1}$ (1.5 Pa .$m^3.s^{-1}$ per orifice) is mandatory.

The tests are performed on the seals delimiting the containment (packaging plug seal and Staubli plug seal).

The rise in pressure test for leaktightness can only be applied by personnel trained in the use of TN-BGC 1 packaging and acknowledged capable by the instructor (named certificate of aptitude). Cofrend approval is required for the helium detection test (level 1 for carrying it out, level 2 for validating it).

2.2.2 Large-scale maintenance operations

These operations take place every forty transport operations or every six years. They cover:

- the neutron inspection to check the effectiveness of the neutron shielding,
- an overload test on handling points (upper oblique bars in the cage),
- a helium leaktightness test on the containment (closing system and body); criterion 10^{-7} Pa.$m^3.s^{-1}$ SHeLR (standard helium conditions according to ISO standard 12 807).

2.2.2.1 Neutron inspection

2.2.2.1.1 Principle of the method

This involves using a non-destructive test to ensure that the packaging's neutron protection is fulfilling its role correctly.

This test is based on the attenuation of a neutron flux from a radioactive source according to a so-called transmission geometry.

The instrumentation includes a transmitter unit containing the source inserted into a mass reflector-diffuser and a receiver unit containing the detector. These two components are placed either side of the body of the packaging to be tested.

A test part is used as a minimum absorption reference. It includes a boron/hydrogen rate so that the absorption is slightly higher than the safety dossier criterion.

This test part is used to check the quality of the neutron-absorbing power of the borated resin. The packaging is considered of sufficiently high quality when the absorption is greater than that of the test part (measurement value by the lower transmission).

2.2.2.1.2 Acceptance criteria

Measurements are made on the test part at the beginning and end of packaging inspection. This produces an average value Σo and an acceptance limit equal to:

$$\Sigma o - 3 \sqrt{(\Sigma o/n)}$$ where "n" is the number of measurements to calculate Σo.

The packaging is only declared compliant if each one of 24 measurements, increased by its statistical uncertainty ($M+3\sqrt{M}$), remains less than the acceptance limit.

If not, the packaging will be inspected again at the end of the campaign.

Three options are possible during the new test:

- 1/ the new measurements are all less than the acceptance limit. The packaging is declared compliant,
- 2/ one or more of the new measurements have values higher than the acceptance limit, but remain within 2% of this limit; the packaging is declared compliant,
- 3/ at least one of the new measurements has a value higher than 2% of the acceptance limit; the packaging is declared non-compliant.

2.2.2.2 Handling point overload test

This test is performed by an approved inspection body.

The following should be carried out:

- sling the packaging from the crane hook. Raise the packaging then position it on the ballast support frame weighing the same as the packaging (about 300 kg),
- install the two coupling bars on both parts,
- lift the assembly by 5 to 10 cm.

  The suspension time to detect any defect is fifteen minutes.
- put the packaging and its load back down,
- remove the coupling bars from the packaging and the ballast support frame,
- put the packaging back in its storage area,
- check the state of the bars (residual deformation or not).

2.2.2.3 Containment leaktightness test

The operations described below must only be applied by Cofrend 1 or 2 personnel (results validated by Cofrend 2).

The packaging is checked for leaktightness by measuring the leak rate from the containment.

The test is considered satisfactory if the overall leak from the containment does not exceed 10^{-7} Pa.m^3.s^{-1} SHeLR (standard helium conditions according to ISO standard 12 807).

IMPORTANT: for type B(U) packaging, the containment leaktightness test will be performed with a packaging plug fitted with Viton seals.

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This chapter presents the quality assurance principles applicable to the package model.

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TABLE

1 QUALITY MANAGEMENT SYSTEM AT CEA/DEN/CAD/DTAP/SET

The main mission of the CEA's Transport Package Service (SET) is to make available to CEA units and programmes suitable packaging for the safe transport of radioactive materials.

For this purpose, SET designs new packaging, modifies or manufactures other and creates dossiers to obtain authorisation (approval or homologation) from the Safety Authorities.

Another SET activity is managing (maintenance and availability) the packaging it holds.

The quality management system in place at SET is taken from the document CEA 160 EMBAL PFM NOT 04001362, the last revision in force.

This document specifies which processes, procedures and associated resources must be applied to fulfil the missions entrusted to SET.

2 QUALITY ASSURANCE PRINCIPLES

The transport regulations in force make it mandatory to apply quality assurance programmes for:

- design,
- manufacture,
- tests, drawing up of documents,
- use,
- maintenance and inspection,
- transport operations.

These activities are all performed by different parties (designer, owner, prime contractor, manufacturers, users, consignors, shippers, maintenance companies and so on) who must all draw up appropriate quality assurance programmes and produce and conserve the documents (records) justifying their activity.

The quality assurance programmes must comply with the requirements of one or other of the following documents:

- Code 50-C-QA revision 1 "Nuclear Power Plant Safety Code: Quality Assurance", IAEA,
- Safety Series No. 113 "Quality Assurance for the Safe Transport of Radioactive Material," International Atomic Energy Agency;
- ISO 9001, 1994 "Quality System: Model for quality assurance in design, development, production, installation and related services".

They must also take the Order dated 10 August 1984 into account: "Quality of the design, construction and operation of basic nuclear facilities".

The classification of the various activities linked to the package model is recalled below.

ACTIVITY	**Applicable QA class**
Packaging and components	
– Design and modification	Q1
– Manufacture	Q2

– Maintenance	Q2
Package (including empty packaging) – Loading, unloading, shipment – Transport commissioning – Transport	 Q2 Q2 NC

Class Q1 corresponds to the application of Code 50-C-QA.

Class Q2 corresponds to the application of Code 50-C-QA with the exception of design.

Class Q3 corresponds to the application of Code 50-C-QA restricted to final inspections and tests.

Not classified (NC) is applied to standard, catalogue or very simple equipment for which only a certificate of compliance is required.

3 QUALITY ASSURANCE SYSTEM FOR THE DESIGN AND SAFETY STUDIES

The design and safety studies developed in relation to the package model have been organised in agreement with standard ISO 9001.

The requirements of standard ISO 9001 - V 2000 include:

- managing documents issued within the framework of this project,
- when necessary, checking documents by individuals competent in the specified domain(s).

4 QUALITY ASSURANCE SYSTEM FOR MANUFACTURE AND QUALIFICATION

The manufacture of the packaging under standard ISO 9001 requires a manufacturer data file, in which feature in particular the material certificates, the welds and related inspections, the tests, the dimensional inspections, the weighings, the non-conformities and the certificates of conformity for each item of equipment produced.

Remember that the manufacturer must be ready to provide the competent authority with the means of carrying out its inspections during manufacture and to prove that the manufacturing methods and materials used comply with the specifications of the approved model.

5 QUALITY ASSURANCE SYSTEM FOR USE, MAINTENANCE AND INSPECTION

5.1 ORGANISATION

The responsibilities of the various units involved in using the packaging are defined in writing with their respective interfaces (NIG no. 467 of 11 May 2001 on the CEA's transport of radioactive materials, Circular DCS 38 of 15 June 2000 on the regulations for the transport of radioactive materials outside CEA sites).

Measures are taken to ensure that:

- the documents relating to the use of the packaging are provided to the operator,
- in terms of traceability, the first issue and subsequent revisions of the packaging use documents are valid,

Remember that the shipper or user must be ready to provide the competent authority with the means of carrying out its inspections during use and to prove that all the packaging is inspected periodically and, if appropriate, repaired and maintained in good condition so that it continues to satisfy all the relevant stipulations and specifications, even after repeated use.

5.2 CONTROLLING THE DRAWING UP OF DOCUMENTS

Measures are taken to ensure the validity of the first issue and subsequent revisions of packaging use documents.

5.3 PROCEDURE FOR USE

A manual and user programme ensure correct use of the packaging compatible with the safety rules and the instructions for use defined in Chapter 4.

Every packaging user will have at his disposal a packaging user manual, drawn up in compliance with the regulations in force and with Chapter 4 of the latest version of the safety dossier.

To ensure that all transport-related operations comply with safety rules, the user manual describes in detail the rules to be observed during the principal operations such as:

- loading, unloading,
- regulatory inspections,

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- stowing,
- handling,
- interim storage.

5.4 MAINTENANCE AND INSPECTION

The maintenance and inspection modalities for the packaging model are defined in Chapter 4 of this dossier and by a technical maintenance specification.

This specification ensures compliance with measures stipulated in the safety dossier and trouble-free maintenance operations.

These maintenance operations are characterised by:

- inspections during use,
- basic regulatory maintenance,
- principal regulatory maintenance,
- and additional inspections when necessary.

Maintenance will be carried out by a company certified ISO 9001: 2000.

6 CLASSIFICATION OF PACKAGING PARTS

To ensure that the quality of the packaging parts matches the safety requirements, these parts are classified according to the expected construction level (N1, N2 and N3) for each part corresponding to their respective significance for safety:

- Level 1: components with a direct influence on the leaktightness or radiological protection of packages, or for fissile material packages, components which directly influence the geometry and, therefore, the criticality check, etc.
- Level 2: structures, components or systems where failure could have an indirect influence on safety, but only if associated with an event or a secondary failure,
- Level 3: structures, components or systems with no influence on the effectiveness of the packaging should they fail to function correctly and, therefore, in all likelihood no influence on safety.

These design, manufacturing, inspection and test requirements incorporate the IAEA recommendations (data package no. 37, appendix IV: quality assurance modulation) in terms of quality assurance modulation.

Table 5-1 lists and classifies the main packaging components.

7 MAJOR COMPONENTS FOR SAFETY

Safety functions	Major components for safety	Essential parameters
Controlling the containment of radioactive materials	Internal leaktightness O-rings Containment	Seal characteristic performance (double O-ring at the plug and quick-connect cap; seal material) Integrated internal cavity (shell and stainless steel bottom assembled by seam welds; steel plug)
External exposure restriction	<u>Lateral shielding</u>: Thickness of external steel shell Thickness of internal steel shell Thickness of resin <u>Axial shielding (bottom)</u>: Stainless steel in the bottom Stainless steel in two closure plates Carbon steel in the diffuser plate Resin <u>Axial shielding (top)</u>: Stainless steel in the plug Resin	 1.5 mm 6 mm 48 mm 8 mm 2 x 1.5 mm 25 mm 24 mm 59 mm 24 mm

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Safety functions	Major components for safety	Essential parameters
Controlling criticality-safety	<u>Cage</u> Cage width: <u>Radial part of the packaging body</u>: Useful diameter of the cavity Thickness of the internal stainless steel shell Borated resin Thickness of the external stainless steel shell <u>Plug</u>: Stainless steel <u>Bottom (from the inside outwards)</u>: Stainless steel shell and carbon steel diffuser plate Resin Stainless steel shell Useful diameter and thickness of internal arrangements Fissile material	 60 cm x 60 cm (in NTC) 178 mm (181 mm max.) 6 mm in NTC, thickness of 48 mm + composition; in TAC, thickness of 33 mm + composition 1.5 mm thickness of 92 mm thickness of 33 mm. thickness of 24 mm + composition thickness of 1.5 mm. weight and composition
Residual heat removal	Thickness of resin Internal shell thickness External shell thickness	48 mm 6 mm 1.5 mm

TABLE 5-1: CLASSIFICATION OF MAIN PACKAGING COMPONENTS

Part reference	Component name	Material	Safety role	Classification level
1	BODY			
	Flange, shells, bottoms, closing plates	Z6 CN 18.09 or Z2 CN 18.10	Very important	N1
1d	Diffuser plate	39 CD 4	Very important	N1
1g	Resin filler	Resin	Very important	N1
1h1	Wood filler	Poplar	Average importance	N2
1h2	Wood filler	Balsa	Average importance	N2
1j	M8 axis	Z6 CN 18.09 or Z2 CN 18.10	Average importance	N2
1k	M8 nut	Z6 CN 18.09 or Z2 CN 18.10	Average importance	N2
1l	Grip	Stainless steel 304	Not important	N3
1m	Fixing lug	Stainless steel 304	Not important	N3
1n	Identification plate	Stainless steel 304L	Not important	N3
1o	Regulatory plate	Stainless steel 304L	Not important	N3
1p	Rubber skid	Neoprene 40 sh	Not important	N3
1q	Pellet holder	Z6 CN 18.09 or Z2 CN 18.10	Not important	N3
1r	0.9 plastic plug	Polyethelene	Not important	N3
1s	Poral pellet	Sintered stainless steel	Not important	N3
1t	Cover end stop	Z6 CN 18.09 or Z2 CN 18.10	Not important	N3
1u	Screw HM6 - 50	Z6 CN 18.09 or Z2 CN 18.10	Not important	N3
1v	Loosening-resistant nut H6	Z6 CN 18.09 or Z2 CN 18.10	Not important	N3
1w	Sheet metal screw	Z6 CN 18.09 or Z2 CN 18.10	Not important	N3
1x	Washer	Z6 CN 18.09 or Z2 CN 18.10	Not important	N3

Part reference	Component name	Material	Safety role	Classification level

2	PLUG			
2a	Plug	Z6 CN 18.09 or Z2 CN 18.10	Very important	N1
2b	Polyethylene disc	HD 100	Average importance	N2
2c	Closure plate	Z6 CN 18.09 or Z2 CN 18.10	Average importance	N2
3	CAGE			
	Structure	6082	Very important	N1
3e	Square end pieces	Polyethylene or aluminium	Not important	N3
3f	Flats	6060	Average importance	N2
3g	Support legs	6060	Average importance	N2
3 hrs	Angle bracket	6060	Average importance	N2
3i	Plastic rectangular end pieces	Polyethelene	Not important	N3
3j	Perforated plates	Aluminium	Not important	N3
4	SHOCK-ABSORBING COVER			
	Bottoms and shells	Z6 CN 18.09 or Z2 CN 18.10	Average importance	N2
4e1	Wood filler	Poplar	Average importance	N2
4e2	Wood filler	Balsa	Average importance	N2
4f	Folding handle	Stainless steel 304	Not important	N3
4g	Boss for grip and end stop	Z6 CN 18.09 or Z2 CN 18.10	Not important	N3
4 hrs	Threaded rod	Z6 CN 18.09 or Z2 CN 18.10	Not important	N3
4i	Stop nut	Z6 CN 18.09 or Z2 CN 18.10	Not important	N3
4j	Resin filler	Resin	Very important	N1
5	BAYONET RING	Z6 CN 18.09 or Z2 CN 18.10	Very important	N1
5a	Stop pin	Z6 CN 18.09 or Z2 CN 18.10	Average importance	N2
6	COMPRESSION RING	Cu Al9 Ni5 Fe5 Y20	Average importance	N2
7	STAUBLI PLUG	Z6 CN 18.09 or Z2 CN 18.10	Average importance	N2
8	TIGHTENING NUT	CuSn12	Average importance	N2
9	TEST PLUG	Bronze ASTM BI 51	Average importance	N2
10	STAUBLI CONNECTOR	Stainless steel 304	Average importance	N2
11	O-ring $\varnothing_{int}$ 196.25	Viton (low temperature) or silicone	Very important	N1
12	O-ring $\varnothing_{int}$ 228	Viton	Average importance	N2
13	O-ring $\varnothing_{int}$ 32.915	Viton (low temperature) or silicone	Very important	N1

14	O-ring $\varnothing_{int}$ 53.57	Viton	Average importance	N2