AREVA-TN International transportation procedure for used fuel casks: Transportabilité tool
Presentation, methodology, qualification and benchmarking

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ABSTRACT

AREVA-TN International works in collaboration with nuclear industry operators to manage the transportation of fuel assemblies in casks. In order to optimize the management and evacuation of spent fuel from nuclear power plants, AREVA-TN International developed the "*Transportabilité*" calculation tool.

The aim of such a tool is to quickly determine the dose equivalent rates and thermal powers for a chosen fuel batch in the suitable cask among those available, and to check whether the transportation is possible according to the regulatory criteria. For each type of cask used, a calculation model is associated in the tool. The addition of a new model in "*Transportabilité*" is systematically carried out with a three-step methodology. First, the cask to be implanted is modelled and dose rate calculations are made using a three-dimensional Monte Carlo code. Then, three dose rate measurement experiments are done around the loaded cask. Finally, the model created in step 1 is tuned according to the experimental measurements while taking into account margins in order to cover uncertainties associated with the statistical calculations and the measurement instruments.

The increase of PWR MOX (Mixed OXide) fuel use in nuclear power plants during the last decade has led TN International to design a new cask called the TN®112. This cask will allow the loading of 12 MOX spent fuel assemblies per transport, which offers an economic interest for both operator and our firm. Consequently, a new model has been added in "*Transportabilité*" to continue optimizing the management of spent fuel, and particularly MOX fuel.

The goal of this article is to present the three-step methodology used for the TN®112 cask, notably the calculation procedure with evolution codes ORIGEN and DARWIN and the 3D Monte Carlo code TRIPOLI-4.3. This article also deals with the global benchmark done around the cask and the comparisons between measured and calculated dose rates.

INTRODUCTION

The transportation of nuclear materials presents a particular risk in the nuclear industry because it is the only moment when nuclear materials are outside nuclear power plants and fuel cycle facilities. During transportation, these materials are very close to the environment, which requires the management of a variety of risks (incidents, accidents, terrorism, losses, leakages...). While most of the risks are controlled by the very transportation cask (protection against radiation, confinement, sub-criticality…), another important way to prevent risks is to minimize the number of transports. Indeed, reducing the transport frequency directly decreases the probability of an incident or accident.

TN International seeks to decrease the number of shipments of transport casks while maintaining the global flow of transfers by continuously improving the variety of cask loading scenarios. Further improvement of the shipment of nuclear plants is possible by increasing the capacity of packages and by optimizing the margins with respect to the regulatory criteria. As a result, shielding analysis for the new casks in TN International is becoming more important as dose rates approach the limits set by the regulatory authorities. In order to achieve credible shielding analysis methods while guaranteeing safety, advanced computational analysis tools, techniques and data are implemented. The validation of these tools is systematically carried out by comparison between numerical calculations and measurements.

Among the software tools developed by TN International, "*Transportabilité*" allows the management of fuel assemblies in casks to be optimized by determining the dose equivalent rates and thermal powers in order to check the feasibility of a transport scenario according to the regulatory criteria.

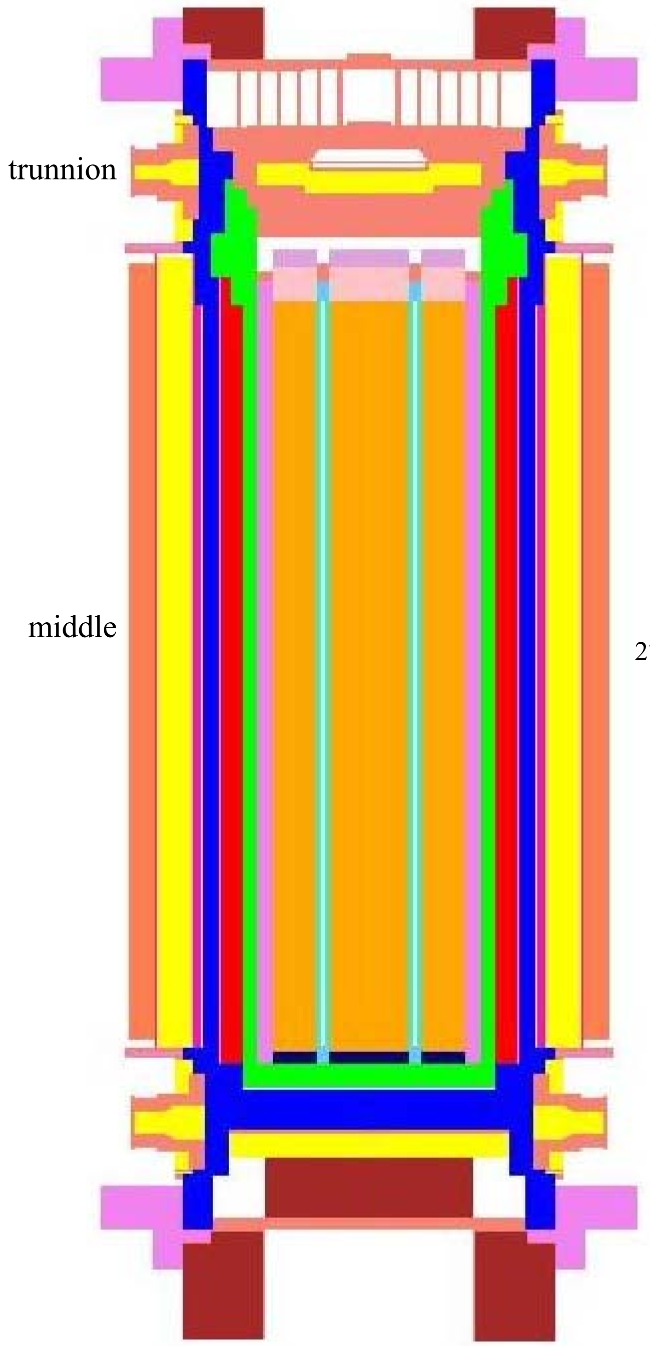
CALCULATION METHODS AND VALIDATION PROCESS

With the improvement of computer performance during the last decades, it is now possible to use 3D Monte Carlo codes coupled with evolution codes for all radioprotection calculations (pre-design, transportation and exploitation configurations, benchmarking…) without significant increase in the computing time.

In our shielding calculation methods, the evolution codes used to evaluate the radioactive sources are ORIGEN2 <1>, ORIGEN-S <2> and DARWIN <3>. We also use APOLLO2 <4> for the activation products of the fuel assemblies ends. Once the sources are calculated according to the fuel assemblies data (fuel type, rod array, irradiation time, burn up, cooling time…), they are imported in the Monte-Carlo code TRIPOLI4 <5>. This code allows a full 3D geometry description that minimizes modelling approximations, and a point wise cross section which eliminates the multigroup approximation. Thanks to this realistic physical description in models, it is possible to calculate dose equivalent rate values very close to those which can be measured around a package. Moreover, it is possible to precisely determine dose equivalent rates in singular areas – such as trunnions or orifices – which have less shielding.

The shielding calculations for TN International are validated by comparisons between the calculated and measured dose rates of the TN International casks. This benchmarking process has been done for a large number of experiments on many casks.

As part of the development of the new TN®112 cask for MOX fuels, a complete 3D model has been made in normal conditions of transport in order to estimate dose equivalent rates around the cask (in radial position at the mid-plane, in axial position at the bottom and at the top of the package, and at the level of the trunnions) in contact, and at a distance of 1 meter and 2 meters from the cask. The sources have been evaluated with ORIGEN-ARP / ORIGEN-S on the basis of real used MOX fuel assemblies data from EDF nuclear power plants. The TRIPOLI4.3 model of the cask is shown in figures 1 and 2 below:



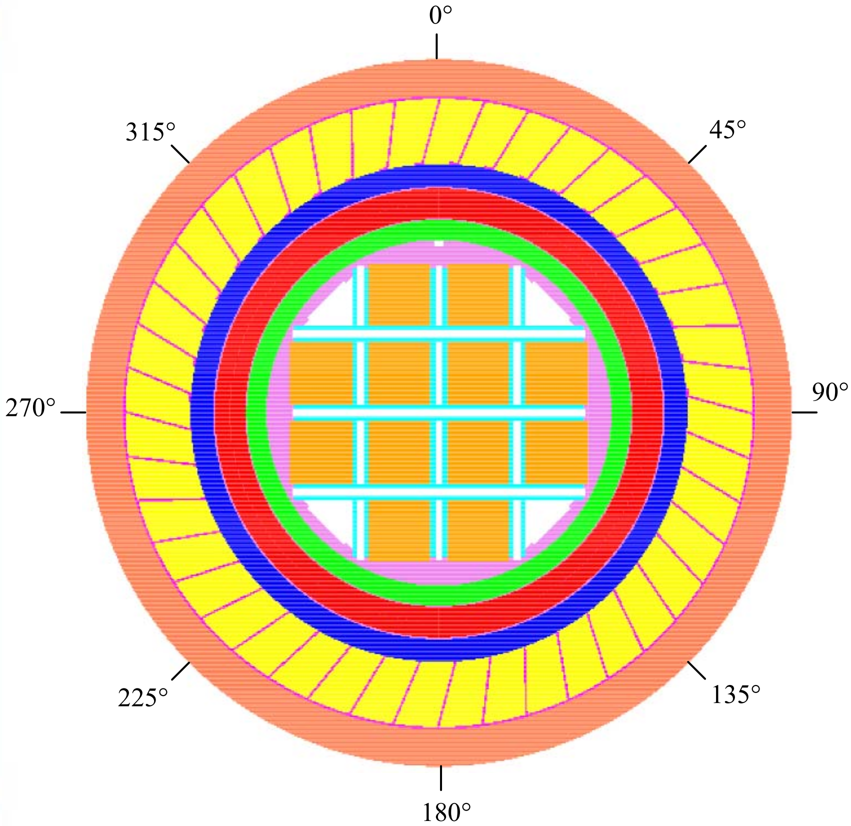


Figure 2 : TN®112 TRIPOLI4.3 model
Radial cross section

Figure 1 : TN®112 TRIPOLI4.3 model
Axial cross section

Currently, these models have been validated by two dose rate measurement experiments around TN®112 casks loaded with 12 MOX fuel assemblies. The measurements were made all around the package at the same positions as the calculation points. Around the middle of the package, a 360° angular sweeping was made to determine the dose rate variations with the radial position.

In order to minimize the measurement dispersion and uncertainty, we use precise tools calibrated with sources adapted to the energy field encountered around casks (for instance, a ^{137}Cs source instead of a ^{60}Co one for gamma-ray measurements), and some measurements are made twice with two different tools. The measurement tools used are:

– Berthold and Cramal for neutron dose rates,
– Babyline for gamma-ray dose rates.

The percentage differences between measured and computed dose rates around the cask for one of the measurement experiments are given in the following tables:

Table 1: Summary of differences between measured and computed dose rates at the middle of the TN®112 cask in radial position for one of the measurement experiments (TRI3.84 loading):

	Contact			**2 metres**		
Position	Measurements (10^{-2} mSv/h)	Calculations (10^{-2} mSv/h)	Differences* (%)	Measurements (10^{-2} mSv/h)	Calculations (10^{-2} mSv/h)	Differences* (%)
0°	8.97	9.52	+6.2%	2.44	2.60	+6.6%
45°	8.69	9.74	+12.0%	2.22	2.53	+14.1%
90°	9.23	9.62	+4.3%	2.83	2.54	-10.2%
135°	9.43	9.58	+1.6%	2.70	2.56	-5.2%
180°	9.13	9.51	+4.1%	-	2.57	-
225°	8.80	9.82	+11.7%	2.73	2.53	-7.5%
270°	9.23	9.50	+2.9%	2.88	2.54	-11.7%
315°	8.01	9.72	+21.4%	2.38	2.55	+7.1%
Average	**8.94**	**9.63**	**+7.7%**	**2.60**	**2.55**	**-1.7%**
Max	**9.43**	**9.82**	**+4.2%**	**2.88**	**2.60**	**-9.7%**
Min	**8.01**	**9.50**	**+18.6%**	**2.22**	**2.53**	**+14.1%**

* (calculated/measured -1) × 100%

Table 2: Summary of differences between measured and computed dose rates at the level of the bottom trunnions of the TN®112 cask for one of the measurement experiments (TRI3.84 loading):

	90° bottom trunnion			**270° bottom trunnion**		
Position	Measurements (10^{-2} mSv/h)	Calculations (10^{-2} mSv/h)	Differences* (%)	Measurements (10^{-2} mSv/h)	Calculations (10^{-2} mSv/h)	Differences* (%)
Base	24.73	45.53	+84.1%	25.48	45.86	+80.0%
Flat area	6.25	7.42	+18.7%	7.37	7.54	+2.2%
2 metres	1.56	1.98	+27.2%	1.92	1.92	0.0%

* (calculated/measured -1) × 100%

The comparison between calculated and measured dose rates around the TN®112 cask shows that the calculations made with the TN International shielding calculation methods are very close to the measurements (less than 10% disparity on average, except for the trunnions).
With the conservative assumptions taken into account, the calculations generally overestimate the dose rates, particularly in contact with the cask where small surface estimators are used in TRIPOLI whereas measurement instruments integrate bigger surfaces and at some distance of the cask. This effect can be significant in singular areas where shielding is weaker, such as at the base of the trunnions.
The underestimated dose rates calculated at 2 metres from the cask can be explained by the large measurement uncertainty with respect to the relatively low dose rates for this loading.

THE “TRANSPORTABILITE” TOOL

Loading plans are an effective mean to manage and optimize loadings in casks. In radioprotection, the most intense radioactive sources are placed at the centre of the package contents to take advantage of the shielding by the less intense surrounding sources. With regard to the irradiated fuel transport packaging, the cooling time before evacuation from the power plant is between 1 to 2 years for UOX fuels and between 2 to 3 years for MOX fuels. An optimized loading can decrease the evacuation time by several months, which can have a significant impact for nuclear industry operators.

As a result of the interest in optimizing the fuel loading in the cask, TN International has developed the “*Transportabilité*” tool to manage the evacuation of irradiated fuels from nuclear power plants.

The Transportabilité tool is based on:

- The utility fuel assembly database.
- The ORIGEN2 code for the evaluation of the neutron and gamma-ray sources.
- Models of cask and internal arrangements (basket) establishing unitary responses of the dose equivalent rates contribution at various calculation points around the packaging.
- A validation of every model by comparing calculations with measurements.
- A margin on the calculated dose rates to cover measurement uncertainties.

The input data for the tool are mass, initial composition (mass-fractions) and irradiation history of the fuel assemblies. With these characteristics, the ORIGEN2 evolution code gives neutron and gamma-ray sources for each assembly. The output data are then imported in the calculation model according to the position of the assembly in the basket.

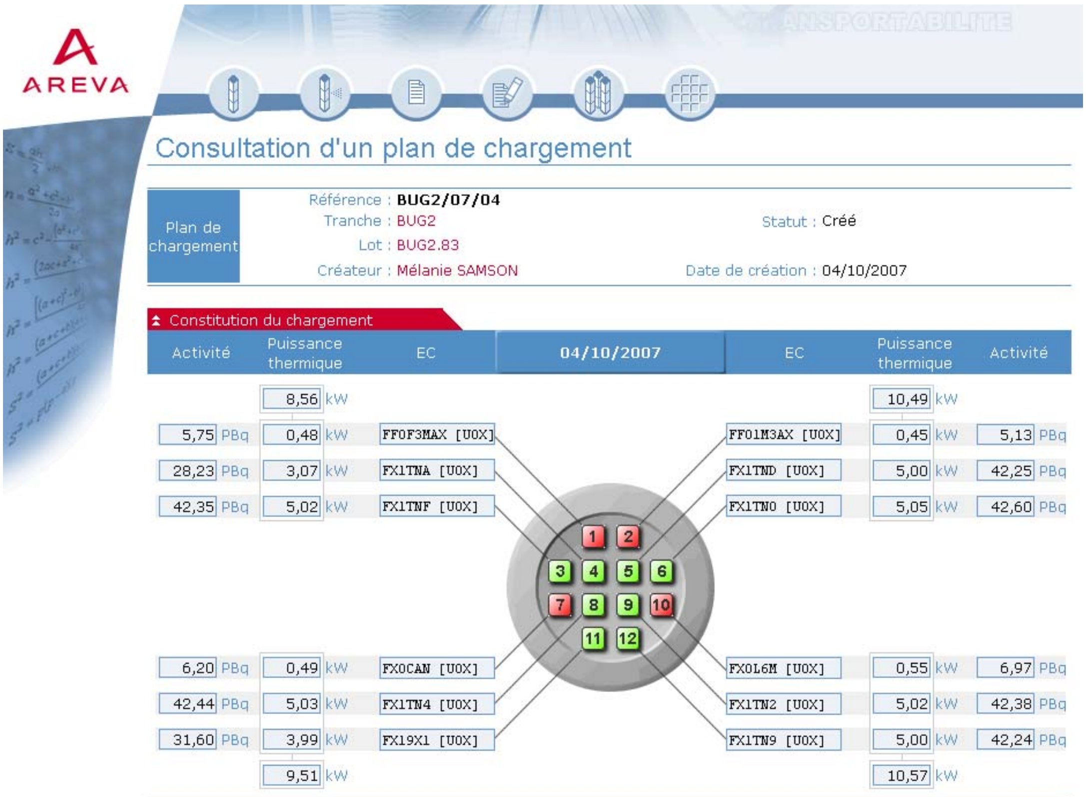


Figure 3: Fuel assembly loading map in the “*Transportabilité”* tool

In "*Transportabilité*", the model associated with an internal arrangement is an array of unitary contributions of every assembly of the loading at the various calculation points around the packaging. These unitary contributions are calculated in advance by the TRIPOLI4 code with a 3D model and with two neutron energy spectra (spontaneous fission and α-n) for neutron and gamma-ray from capture dose rates. Concerning gamma-rays, 7 energy groups from 0.3 keV to 3.5 MeV are considered.

The contribution to the dose equivalent rates of an assembly at a calculation point in "*Transportabilité*" is then obtained by the product of the sources of this assembly, calculated by ORIGEN2 with its characteristics, and its associated unitary response previously calculated by TRIPOLI 4 and available in the array. The total dose equivalent rate (DER) of the whole loading is given by summing the responses of each assembly:

$$DER_{tot} = \sum_i s_i \, . \, UnitDER_i \qquad \text{with } 1 \leq i \leq \text{number of assemblies,}$$

where s_i is the neutron and gamma-ray source of the i^{th} assembly, and $UnitDER_i$ the unitary response of the i^{th} assembly.

The validation of a model is made by a comparison of the calculated values with the measured ones carried out within the framework of three independent measurement experiments. These three measurement experiments characterize the reproducibility of the measurement/calculation coefficients and guarantee the reliability of the model fitting. To take into account possible measurement uncertainties and ensure covering calculation values, an additional 20 % margin is associated with the gamma and neutron dose equivalent rates.

As the new TN®112 cask has been developed for used in the same exploitation conditions as the current TN International casks used for EDF (in particular for MOX fuels), it will be imported in "*Transportabilité*" for the management of the future MOX loadings. Like the other cask models, the addition of the TN®112 model in "*Transportabilité*" is carried out with a three-step methodology:

– Modelling: the three-dimensional TN®112 model constructed for the benchmarking is used to calculate the unitary contribution of each of the 12 MOX assemblies to dose equivalent rates around the cask.
– Measurements: two dose rate measurement experiments have already been done around the TN®112 cask (cf. benchmarking), and the third one is planned.
– Model fitting: the calculation results given by the TN®112 model and the associated methodology have been tuned according to the two measurement experiments already carried out. The definitive model-fitting will not be done until the last measurement experiment is finished.

The comparisons between dose equivalent rates around the TN®112 cask measured and calculated with "*Transportabilité*" are given in the tables on next pages.

In tables 3 and 4, the "*Transportabilité*" values are presented for the before model fitting without the 20% margin.

Table 5 gives the comparison between measurements and "*Transportabilité*" values after model fitting and taking into account the measurement uncertainty.

The methodology used to fit the calculation model according to the measurements while guaranteeing conservative assumptions consists in considering the average ratio Measurements/Calculations of the three measurement experiments for each calculation point (middle of the cask, trunnions…) in contact and at 2 metres from the cask, and then applying this ratio as a fitting factor at this point for all the calculations with other loadings.

Table 3: Comparison between measured and computed dose rates in contact and at 2 metres from the TN®112 cask for the first measurement experiment (TRI3.79 loading)

	Contact			2 metres		
Position	Measurements (10^{-2} mSv/h)	Calculations (10^{-2} mSv/h)	Ratio M/C	Measurements (10^{-2} mSv/h)	Calculations (10^{-2} mSv/h)	Ratio M/C
0°	6.98	7.84	0.891	1.92	2.03	0.948
45°	7.08	8.53	0.830	2.44	2.20	1.108
90°	7.75	8.65	0.896	2.46	2.27	1.083
135°	7.80	8.31	0.938	2.10	2.13	0.988
180°	8.24	8.18	1.007	-	2.13	-
225°	8.11	8.13	0.998	1.92	2.05	0.936
270°	7.26	7.65	0.949	2.10	1.93	1.088
315°	7.36	8.03	0.916	2.36	2.02	1.169

Table 4: Comparison between measured and computed dose rates in contact and at 2 metres from the TN®112 cask for the second measurement experiment (TRI3.84 loading)

	Contact			2 metres		
Position	Measurements (10^{-2} mSv/h)	Calculations (10^{-2} mSv/h)	Ratio M/C	Measurements (10^{-2} mSv/h)	Calculations (10^{-2} mSv/h)	Ratio M/C
0°	8.97	9.82	0.914	2.44	2.73	0.893
45°	8.69	9.82	0.885	2.22	2.68	0.826
90°	9.23	9.86	0.936	2.83	2.73	1.037
135°	9.43	9.87	0.955	2.70	2.69	1.004
180°	9.13	9.85	0.928	-	2.73	-
225°	8.80	9.89	0.889	2.73	2.70	1.011
270°	9.23	9.86	0.936	2.88	2.74	1.051
315°	8.01	9.84	0.813	2.38	2.69	0.885

As for the benchmarking calculation results, the computed values with the unitary response method are very close to the measurements, which involves ratios generally close to 1. However, some values give more important differences.
To take into account measurement uncertainties and ensure conservative calculation values, an additional 20 % margin is applied to the gamma-ray and neutron dose equivalent after the fitting.

As an example, the model fitting methodology is illustrated in table 5 for a calculation 2 metres from the middle of the cask for the TRI3.84 loading. Currently, the fitting factors are determined with the two measurement experiments (TRI3.79 and TRI3.84) carried out.
NB: the results given are not definitive, they are just presented here to illustrate the method.

Table 5: Comparison between measured and computed dose rates with "*Transportabilité*" at 2 metres from the TN®112 cask:

Position	Fitting Factor (ff)	Neutron Dose Rates (10^{-2} mSv/h)	Gamma Dose Rates (10^{-2} mSv/h)	Neutron × ff + 20%	Gamma × ff + 20%	Final tuned Dose Rates (10^{-2} mSv/h)	Measure-ments (10^{-2} mSv/h)	Diffe-rences (%)
0°	0.920	0.78	1.95	0.87	2.15	**3.01**	2.44	+23.7%
45°	0.967	0.79	1.90	0.91	2.20	**3.11**	2.22	+40.5%
90°	1.060	0.79	1.94	1.01	2.47	**3.47**	2.83	+22.7%
135°	0.996	0.79	1.90	0.94	2.27	**3.22**	2.70	+19.1%
180°		0.79	1.94	-	-	**-**	-	-
225°	0.974	0.79	1.91	0.92	2.23	**3.16**	2.73	+15.5%
270°	1.069	0.79	1.95	1.01	2.50	**3.52**	2.88	+22.1%
315°	1.027	0.78	1.91	0.97	2.35	**3.32**	2.38	+39.3%

The final results show that the calculated dose equivalent rates are all larger than the measured values, which guarantees the safety margin for the loading with respect to the regulatory criteria. However, the calculated values are also close enough to the measurements that it is still possible to optimize the fuel loading scenario as the conservatism in the calculation remains realistic.

CONCLUSION

The widespread use of the three-dimensional Monte Carlo codes, the greater qualification of nuclear data and calculation methods, as well as the development of tools to optimize loadings with the close cooperation with utility companies, allows TN International to improve the radiological performances of the transport casks.

These calculation tools allow the packages to be loaded to 80 % of the radioprotection criterion imposed by the regulations instead of approximately 50 % previously. This allows to compensate the increase of the intensity of the sources to transport and to maintain or increase the capacity of packagings while improving the global safety of the transport of radioactive materials.

REFERENCES

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<5> TRIPOLI 4: "TRIPOLI-4 Monte-Carlo transport code - User's Manual" : Technical report CEA : SERMA/LEPP/RT/01-2901/C.