# Characterization of the Back-End Wastes in View of their Transportation

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## NATURE OF THE PROBLEM

The designer of a packaging has to face two main problems: to fulfill the goal set forth by the user and to meet the safety requirement defined by the International Atomic Energy Agency (IAEA) and Competent Authorities.

The demonstration of the second point is brought by the Safety Analysis Report of the package. This document must include a description of the packaging and of the content. When the transport takes place, the consignor has to demonstrate that the characteristics of the material to be loaded in the cask are not worse, as concerns safety, than the content described in the Safety Analysis Report.

For this purpose, the description of the content will include a list of pertinent parameters allowing its characterization and clearly defining an "allowable content". These parameters must be as detailed as necessary and as simple as possible. But above all it must be possible to confirm, before shipment, that the package fulfills all safety criteria.

Up to now, the main type B or type F packagings have involved fuel or radioactive sources which are particularly well defined and each packaging is dedicated to a given content. The parameters which characterize them are well known. For instance, for spent fuel assemblies, the list provides geometry of the assemblies (diameter of pellets, diameter and thickness of cladding, pitch of the rods), enrichment in uranium 235, burn-up, specific power, cooling time,... This type of material is well defined and is followed throughout its life by a Quality Assurance system, allowing a perfect knowledge of its characteristics and of its history.

Because nuclear energy is now long established, more and more shipments and packagings are concerned by waste generated by the back end of the fuel cycle. The definition and characterization of the wastes are not so easy. One reason among others, for instance, is the waste processing, which generates a mixture of products of various origins.

The paper describes some cases TRANSNUCLEAIRE had to face concerning high level, transuranic and irradiating wastes and the way they have been solved in order to provide the most reliable and flexible envelope characterization of the contents.

## HIGH LEVEL VITRIFIED WASTE

The reprocessing of spent fuel generates different kinds of wastes. Among them, fission products and non-fissile actinides represent 98 percent of the radioactivity; these wastes are separated, concentrated, mixed with molten glass and poured into stainless steel containers.

To design the TN 28 cask, which will be used to transport these wastes, typical cases were considered for the radioactive sources. These cases were defined from typical fuel assemblies (for instance PWR 17 x 17 assembly with an initial enrichment in U335 of 3.5 %), typical irradiation and cooling time (for instance burn up of 33000 MW.d/tU and cooling time of 8 years: 3 years from reactor unloading to reprocessing, 1 year from reprocessing to vitrification and 4 years from vitrification to transport) and typical quantities of fission products and actinides per canister.

This method clearly defines the source term which may be used for the design: the activity of each radionuclide is determined and therefore the spectrum to be used for shielding evaluation, the heat load per canister for thermal evaluation or the different nuclides for containment evaluation.

With this method, we obtain a typical source which may be quite representative of the radioactive material to be transported but its representativity only reflects a global point of view.

But if an allowable content must be defined, this method appears too restrictive: it is not possible to define in this case the allowable content by wastes coming from PWR 17 x 17 assemblies with a maximum burn-up of 33 000 MW.d/tU and a maximum cooling time of 8 years. This is impossible for various reasons, for instance, because there is no traceability between the fuel which is reprocessed and the radioactive material poured in the glass canister.

In fact, from the reprocessor, the only data available for the canisters is the list of activity of the main radionuclides and the heat power released.

Therefore the allowable content must be defined by reference with this data.

The first idea was to compare, isotope by isotope, the list of values of activities of each canister with the list used for the design. This also appears to be too restrictive. Waste generated by fuel with lower burn-up and cooling time may have the same "strength" concerning dose rate or heat power but may have a quite different spectrum.

Finally, the solution which was chosen was broken down according to the various influences on safety.

For heat power, the values of residual power released by each canister at the time of transport will be available. Therefore, it will only be necessary to compare the values used for the design and the safety analysis with the actual power.

Concerning dose rates, a maximum quantity Qi (expressed as an activity for fission products and a mass for actinides) to be transported in each canister was defined for each isotope. This quantity is the maximum for which the allowed dose rates around the cask are reached: at surface and at one meter in normal conditions of transport and at one meter after accident conditions. The actual quantities to be transported (qi) are acceptable if the following inequation is checked:

$$\Sigma \frac{qi}{Qi} \leq 1$$

Safety and/or corrective factors may be added as necessary, for instance, to take into account isotopes not previously considered.

At last, the containment analysis was based in France on the only consideration of gas release. As the vitrified waste is basically a solid material, the quantity of gas was calculated on the basis of the equilibrium vapour pressure between the solid and the gas phases. This allows us to demonstrate that there was no limitation to apply concerning this safety aspect. A different analysis was performed in Japan, which should lead to a rather similar conclusion.

As a conclusion concerning the transport of vitrified high level waste, the allowable content is defined using only two conditions: a limitation on the thermal power per canister and an inequation connecting the quantity of the different isotopes. This method allows us to satisfy all the safety aspects and provides in the meantime a very large flexibility.

HIGH LEVEL ACTIVATED CORE COMPONENTS

The so-called TN 17 CC casks are operated to transport high level activated core components.

This cask model is derived from the well-known TN 17/2 packaging used for the transport of PWR or BWR spent fuel assemblies. It is a simplified version: neutron shielding and copper fins used for thermal dissipation have been designed out.

Up to now, this TN 17 CC cask was licensed to transport very well-defined BWR core components such as control rods, fuel boxes, temporary neutron absorbers, detector assembly support tubes, ... For these components, the geometrical and mass characteristics are perfectly known. Equally well defined is the radioactive spectrum, assuming as necessary conservative assumptions. Therefore, for each type of component, the total activity gives a perfectly well defined characterisation of the radioactive content.

Nowadays, the users wish to transport miscellaneous core components such as screws, grids, spacers, nuts,... in a new type of canister. For this kind of material, nearly nothing is known in advance.

The approach followed by TRANSNUCLEAIRE to define the allowable content was nearly the same as previously explained for high level vitrified wastes. But an additional difficulty came to light. In the case of wastes mixed with glass poured in a canister, the binding agent is well defined as much in weight as in chemical composition or in location in the cavity of the cask. This is not the case for these activated core components.

This led us to make a preliminary survey of the influence of the various possible densities of material. As a result, it appeared that the specific activity (i.e., activity per unit of weight of the content) was a good parameter: the allowable values of specific activity are rather steady according to the variation of density.

As a conclusion concerning the transport of high level activated core components, if we focus on the main problem, which is the dose rate limitation, the allowable content may be defined with one inequation connecting the quantity of the different isotopes per unit of weight of content. This method allows us to satisfy the dose rate limitation aspects and provides as well a very large flexibility.

## MEDIUM LEVEL WASTE IN FRANCE

For a long time, medium level waste generated by reactors (such as ion-exchange resins and filters) has been immobilized within a solid matrix and packed in concrete cylinders. These concrete cylinders are then sent to storage disposal to be stored for several centuries.

Since the implementation of IAEA 1985, some of these packagings - with the more irradiating contents - are no longer classified as IP but as type B packagings, because the dose rate at 3 m of the unshielded content exceeds 1 rem/h.

The solution suggested by TRANSNUCLEAIRE is the use of an overpack, allowing the existing models of concrete cylinders to be kept in use, therefore saving the efforts of qualifying new ones for long-term disposal. It is not required that packagings be totally leaktight but activity release limits must be met in normal and accident conditions. Therefore this overpack is not fitted with a sealing arrangement: the demonstration of type B requirements concerning the release of activity will be done through the characteristics of the binding agent itself.

The characterization of the waste has various steps.

The first is the nature of the radionuclides involved. Their origin is known, from the process, and the radioisotopes are activation products such as Co58, Co60 or Mn54. Actinides or fission products are not present.

The second is the value of the activity. The actual gamma dose rates around the concrete cylinder are measured and "transformed" in activity with the help of graphs and/or correlations, obtained by calculation validated by benchmarks. With a similar method, the value of the dose rate at 3 m of the unshielded material is obtained.

At last, according to the result of the evaluation of the dose rate at 3 m, the package may be classified either as needing the overpack (type B) or acceptable as it is (IP: the criteria concerning LSA II material is checked, even considering only the most pessimistic value of A2 among all the values for the actual radioisotopes).

It must be pointed out that our analysis will show that due to the characteristics of the binding agent, the concrete cylinder and the overpack, the acceptable activity will be several times higher than what should be transported. Therefore, even if the total activity has to be checked, it is only purely formal.

In the case of medium-level waste immobilized in a solid matrix, such as concrete or resin, and packed in a concrete cylinder fitted, as necessary, with a steel overpack, it is demonstrated that the characterisation of the content may be very simple. As the nature of the radioactive elements is established, a control of the gamma dose rate around the concrete cylinder provides the value of the activity to be transported and of the dose rate at 3 m of the unshielded material. This gives all the information needed to classify the material and to conclude whether an Industrial Package or a type B container is necessary. The other verifications are purely formal.

**ALPHA WASTE**

TRANSNUCLEAIRE is now developing a large-dimension type B container, called TN-GEMINI, for the transportation of transuranic waste. It may accommodate forty 200-l drums, or sixty 100-l drums, but also larger drums or cubic large dimensions concrete containers. The maximum quantity of plutonium transported is 400 g. In this part we will focus on the content of drums, which consists of paper, vinyl, cotton, wood, concrete, metallic pieces,....

To avoid the accumulation of different risks other than radioactive, the following types of material are prohibited: pyrophoric, explosive, highly burnable or corrosive materials, items containing pressurized or burnable gases and reactive chemicals. A good chemical compatibility among the wastes is also required.

It must be pointed out that the content of each drum is known from data sheets following the life of the drum, at least as concerns the type of material and the quantity of plutonium.

It appeared very quickly that, for the characterization of the content as concerns safety, only two problems were specific in the design of this package.

The first one is the thermal analysis: the material to be transported has a very wide range in nature. Nevertheless, with some rough hypothesis concerning the isotopic composition of plutonium, it was demonstrated that a total heat power of 10 watts covers the content of 400 g of plutonium for which the container is designed. This low value of thermal power leads to very low temperature in the content even for pessimistic assumptions.

The second problem concerns the generation and build-up of gas due to radiolysis.

A survey has been made in the literature for different materials. The worst case was selected and considered in the study. This led to a pressure increase of 1.2 bar, for the duration of one year to be considered for transport licensing. This pressure value was added to that resulting from other processes (increase of temperature, thermal decompostion, ...), to show that the total pressure was lower than the design pressure (2.0 bar gauge in normal conditions of transport and 3.0 bar gauge in accident conditions).

The consideration of the most pessimistic value of factor G led, for a transport duration limited to 45 days, to an hydrogen content of 3.1 %, less than 4 % which is the limit of burnable or explosive mixture in the cavity.

The analysis assumed that the drum contents have a void rate of 50 % and that there is no hydrogen build-up prior to transport.

It implies that the drums have to be vented: either the leak rate of the drums was initially high enough to avoid this build-up or actions have to be taken to eliminate the gas contained. This can be achieved for instance by opening the lid or making a hole in the drums.

It may be concluded that, after elimination of wastes which present non-radioactive risks, the transport of transuranic waste is demonstrated safe with respect to only two conditions: limitation of the plutonium quantity per shipment and venting of the drums. All the other parameters have been shown to be non-pertinent. This advantage, compared to existing designs, derives from the use of very large envelope definition, which, as a counterpart, allows the user to transport a wide range of material with the minimum amount of control.

## CONCLUSION

Through these various examples, we have shown TRANSNUCLEAIRE's involvement and philosophy in the field of the back end waste characterization. We have in particular illustrated how the designer can contribute to the most efficient use of its packagings through a practical approach of the contents definition. Customers and packaging operators can, therefore, benefit from a large flexibility and minimum control operations prior to shipment, while keeping the high level of safety required by national and international authorities.