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PRELIMINARY EVALUATION OF THE IMPACT AND INTER-GENERATION RISK TRANSFERS RELATED TO THE RELEASE AND DISPOSAL OF RADIOACTIVE WASTE FROM THE NUCLEAR FUEL CYCLE

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SUMMARY

Reflection about the consequences of decisions involving the long term raises various theoretical and complex issues related to the validity of the quantitative assessment of what could be future risks, but also to the ethical position we are adopting towards future generations. In this perspective, decision-making in the field of radioactive waste management with a view to maintaining present and future radiation exposures as low as reasonably achievable implies being able to discriminate among alternative options, i.e., being in a position to evaluate the differential impacts of potential consequences. Because of the complex and multi-dimensional nature of the distant future consequences of waste management options, their comparison involves expressing this impact using various aggregated indicators, taking into account the time during which radionuclides remain in the environment and their local, regional, or world-wide dispersion.

This report is an attempt to contribute to the development of such a framework. This is preliminary work, mainly focused on the risk transfer dimension inherent to waste disposal management. Any decision to protect people now against the potential impacts of radioactive releases into the environment leads inevitably to the exposure of present workers and potentially of future generations. In this perspective, one of the key questions related to waste management is to decide on the best compromise between present dilution-dispersion into the environment or concentration in surface or underground disposal sites. The objective of this study, sponsored by IPSN, is to illustrate, using the French nuclear fuel cycle context, the relative impact of some simple waste management options, focusing especially on inter-generational risk transfers.

All the values presented in this report are related to six particular radionuclides, for a limited number of waste management options. Even if the selected radionuclides are among the most important in terms of radiological impact, a complete assessment of waste impact should include all the radionuclides contained in the waste, to be compared to the different categories of impact for the entire fuel cycle. This is particularly true for underground waste disposal, for both normal evolution scenarios and intrusion.

The alternative waste management options analysed are of a one-off nature, i.e., they consist in the total disposal or total release of a radionuclide. These assumptions lead to some extremely high exposures such as for the integral release of Cs-137. But between total disposal and total release, an infinity of choices are available. These intermediate options are, on the whole, more realistic from a technical point of view. The analysis of this kind of intermediate management options could give an estimation of the most appropriate options in an ALARA perspective.

All these evaluations are preliminary. The problem is to estimate whether this kind of time and space presentation provides the right information for helping decision-making processes and facilitating the communication with non-specialists on the risks associated with the waste management options, as well as the inter-generation risk transfer.

1. INTRODUCTION

Reflection about the consequences of decisions involving the long term raises various theoretical and complex issues related to the validity of the quantitative assessment of what could be future risks, but also to the ethical position we are adopting towards future generations. However, from a practical point of view, a responsible attitude implies using as best we can all available information on the possible consequences of our present actions, even if this information does reflect the limitation of our knowledge, given the deficiencies of our modern instruments in the assessment of consequences far in the future.

In this perspective, decision-making in the field of radioactive waste management with a view to maintaining present and future radiation exposures as low as reasonably achievable implies being able to discriminate among alternative options, i.e., being in a position to evaluate the differential impacts of potential consequences. Because of the complex and multi-dimensional nature of the distant future consequences of waste management options, their comparison involves expressing this impact using various aggregated indicators, taking into account the time during which radionuclides remain in the environment and their local, regional, or world-wide dispersion. With respect to the radiation impact, there is an ongoing debate on the use of the concept of collective dose, which allows expressing the impact on populations in space and time, whatever the level of individual exposures.

Two types of arguments have been raised against the use of the collective dose to assess the impact of waste management options:

- the aggregation on huge populations of extremely low individual doses such as those resulting from waste management, which leads to significant collective doses, together with the use of the linear, no threshold dose-effect relationship to assess the corresponding potential health impact,

- the existence of increasing uncertainties, especially with time, which weaken the relevance of estimating impacts in the distant future.

These criticisms are highly valid, although any responsible decision-making process on waste management options cannot avoid taking into account, for each option, the

magnitude of radionuclide releases into the environment, the time during which these radionuclides remain a source of exposure and how widely they are dispersed geographically, i.e., how large is the size of the exposed population.

In view of these difficulties, the evaluation of impacts can be performed either qualitatively by describing, in the most detailed way, the various aspects presented above, or partly quantitatively. Quantification must rely on the best current scientific knowledge of the various mechanisms involved in the radionuclides dispersion and exposure of human beings, as well as the best guess as to the behaviour and size of future generations. So far, the use of the individual dose concept remains the most appropriate performance indicator to assess whether any waste management option is likely to put people in danger (deterministic effects) or keep them at an acceptable level of exposure. The collective dose concept allows evaluating whether these options have a significant impact on public health (stochastic effects). The problem is less the accuracy of these concepts than in the difficulty of using them in an adequate and comprehensive framework allowing responsible decisions to be taken, i.e., with all the key dimensions and consequences being estimated and valued. Any approach relying on the cut-off of collective exposure either in time, space, or individual level of exposure is misleading in terms of understanding and communication with the public.

This report is an attempt to contribute to the development of such a framework. This is preliminary work, mainly focused on the risk transfer dimension inherent to waste disposal management. Any decision to protect people now against the potential impacts of radioactive releases into the environment leads inevitably to the exposure of present workers and potentially of future generations. In this perspective, one of the key questions related to waste management is to decide on the best compromise between present dilution-dispersion into the environment or concentration in surface or underground disposal sites. The objective is to illustrate, using the French nuclear fuel cycle context, the relative impact of some simple waste management options, focusing especially on inter-generation risk transfers.

The following chapters summarise a study sponsored by the French Institute for Nuclear Protection and Safety (IPSN).

2. THE NUCLEAR FUEL CYCLE WASTE MANAGEMENT SYSTEM

The nuclear fuel cycle consists of a succession of stages, which may differ depending on the type of reactor (PWR, GCR, CANDU, etc.) and the various options that exist for fuel management (with or without reprocessing of spent fuel, recycling of fissile matter, etc).

Each of the stages produces radioactive waste in liquid, gaseous or solid forms. The methods of managing this radioactive waste can vary considerably from one facility to another, with the choice of a waste management option depending mainly on the characteristics of the waste being considered. The waste produced is composed of a group of radionuclides, each of which having its own particular characteristics, in terms of both chemical and physical properties (radioactive half-life, mode of disintegration, environmental behaviour, chemical form, radiological impact). As a result, the management of radioactive waste necessarily implies specific management for each radionuclide, or family of radionuclides.

Figure 1 illustrates schematically the different management methods for waste produced by the nuclear fuel cycle. Most of the waste is submitted to an initial processing phase. This processing can differ considerably depending on the radionuclide concerned: it may be partial elimination as a result of radioactive decay or various physical and chemical processes.

Following this processing, a portion of the radioactive waste can be directly released into the environment in the form of liquid or gaseous effluents (dilution/dispersion principle). In this case, the radionuclides are immediately dispersed into the biosphere, the extent of this dispersion differing according to the geographical characteristics of the site. (This dilution/dispersion process mainly concerns radionuclides having a limited radiological impact or those discharged in very small quantities).

Waste which is not released into the environment is immobilized with a view to disposal (concentration/containment principle). This disposal can be envisaged either on the surface for low- or medium-level waste, or underground for higher level waste. The radioactivity contained in the confined waste is held for a period which can stretch from some hundreds to some hundreds of thousands of years, during which time part of this activity is eliminated as a result of radioactive decay. Then, following this retention phase, there is a gradual release into the biosphere of the remaining radionuclides

(delayed dilution/dispersion), usually via underground water.

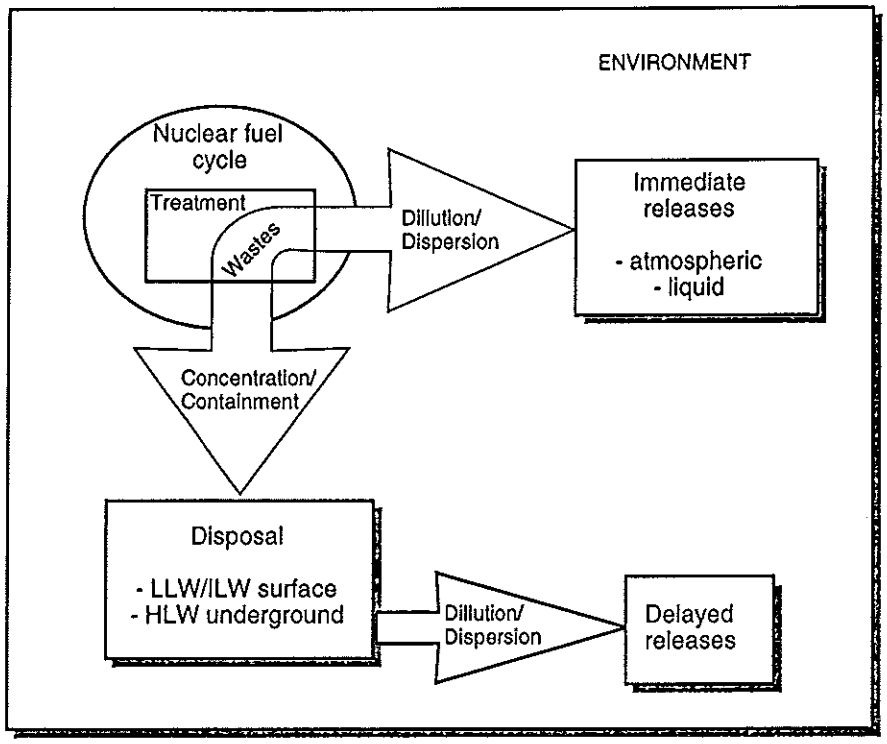

Figure 1. Waste production system and management options (release or disposal)

An estimation of the impact associated with nuclear fuel cycle waste requires a breakdown by radionuclide. It is therefore necessary, for each radionuclide produced which is not recycled and therefore constitutes waste, to assess the impact resulting from its release or its disposal (when this is technically feasible), vis-à-vis both the public and workers.

3. SPACE AND TIME DIMENSIONS

A major difficulty associated with the analysis of long-term consequences is defining the time and space dimensions at stake. The space dimension is considered with respect to the size of the population potentially affected either at the local level (reference group from a few to a few hundred people) in the vincinity of the facilities, or at the regional and even world-wide levels. As regards the time dimension, the driving parameter is the radioactive decay of the radionuclide. Thus, the question is to determine, according to its half-life, the persistence of the radionuclide in the environment due to releases, as well as the effect of underground disposal. At this stage, a large uncertainty is introduced into the evaluation framework.

Broadly speaking, it is possible to appreciate these effects in a simple space and time matrix as follows:

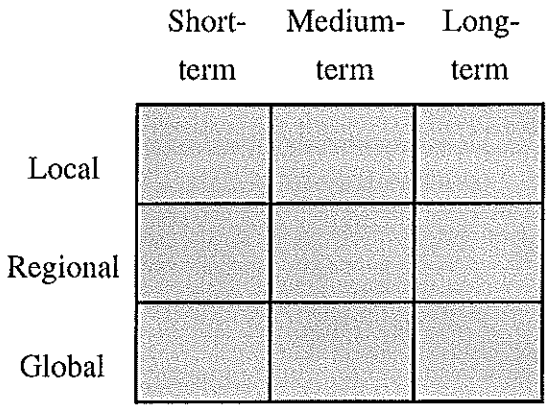

Figure 2. Time and space matrix

Figure 3 illustrates the time scales which can be reasonably envisaged for direct releases into the environment and those associated with near-surface and underground disposal, according to our present knowledge and expert consensus. It does not include considerations on associated exposures.

The first set of releases is due to nuclear energy production (here, an equivalent of 60 years of production is assumed). After this period, direct releases from nuclear facilities are stopped, but will induce exposure of the public according to the dispersion of the radionuclides into the environment.

The first delayed releases are due to surface disposal. It should be noted that releases start quite early with H-3. For the other radionuclides, releases occur only after several hundreds of years and become quite low after a few tens of thousands of years.

With respect to the underground disposal, it should be noted that there are no significant releases before a thousand years, but, on the other hand, releases of transuranium elements still exist after 100 000 years. Because of the time scale considered for these releases, one of the problems concerns the evolution of the environment such as, for example, hypothetical glaciation. At this level, for the purpose of evaluating the potential impact of an extensive disturbance of a waste disposal site, a simulation of the occurrence of events, which have a reasonable probability of occurrence, is generally accepted, taking into account past experience.

The concept of generation is introduced in this figure, which describes the evolution of the number of generations concerned according to the time scale considered (for example: 300 generations for releases lasting 10 000 years).

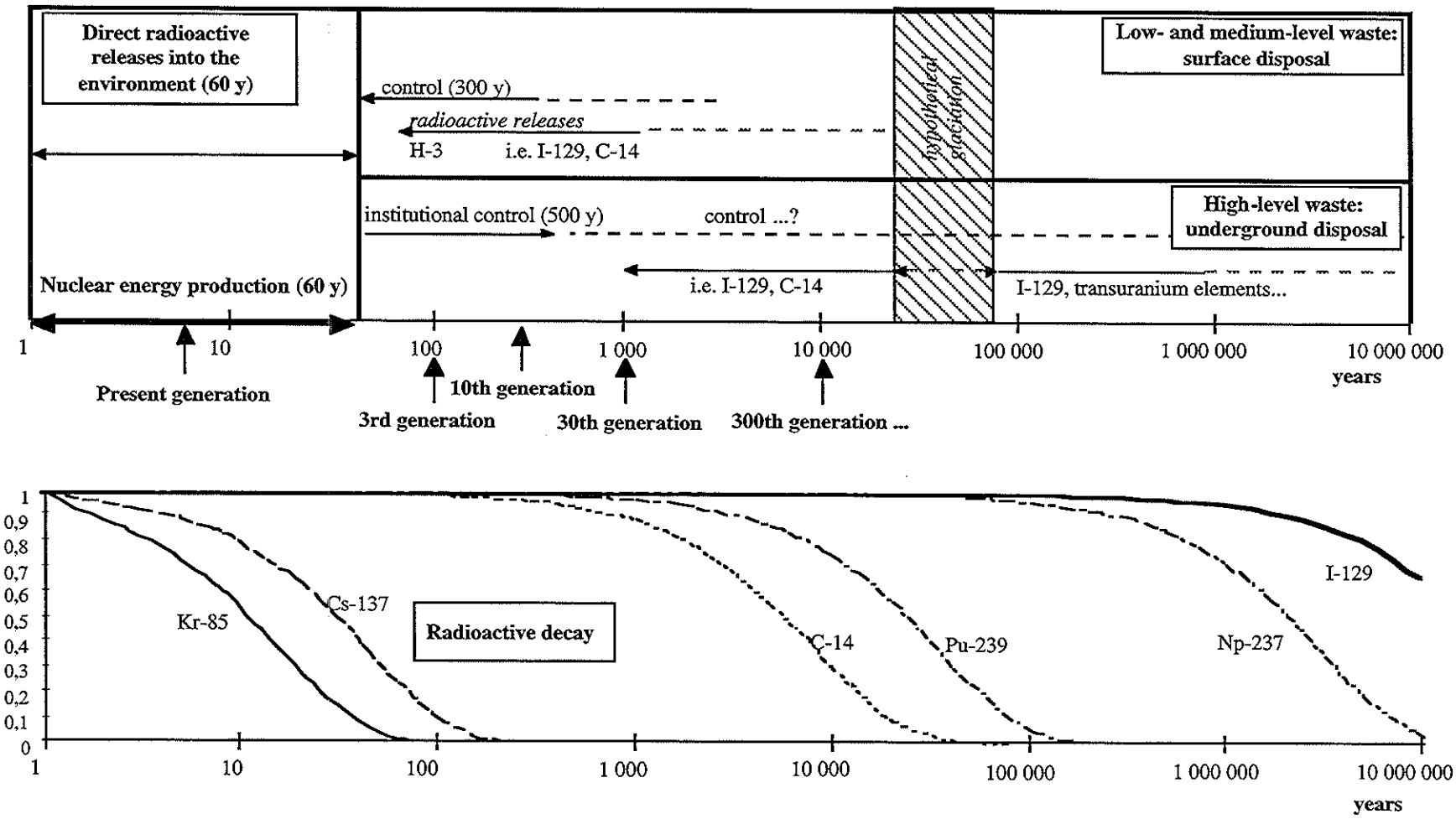

Figure 3. Time aspects of nuclear fuel cycle waste management

4. RISK TRANSFERS

Whatever the option selected, waste management leads to a radiological impact. Depending on the radionuclide concerned and the management option, the most exposed population group can be either the workers or the public, in the short, medium or long term.

With respect to the different options related to the cycle, the transfer of risks between these different categories of population has to be examined, as well as between these different time frames. These risk transfers may occur at three levels:

- Between the public and workers: for example, choosing to trap and store a radionuclide which is usually released into the environment would lead to a reduction in public exposure, but would increase the exposure of workers by creating new operations for processing this radionuclide, hence transferring the risk from the public to the workers.

- Between generations: as previously pointed out, storing a radionuclide postpones its release into the environment, except for the fraction eliminated through radioactive decay during the retention period. In this case, choosing disposal rather than immediate release implies transferring risks from current to future generations.

- Between local and global populations: here again choosing to store and therefore concentrate a radionuclide, rather than disperse it early into the environment, can lead to a transfer of part of the risk from global to local populations (particularly as regards the risk of intrusion in a disposal site).

In terms of radiological risk management, any option envisaged either to dilute/disperse or to concentrate and confine must be evaluated taking into account transfers between exposed groups in space and time. Final decisions rely on a compromise between the short-term protection of the population and the risk transferred to future generations, a compromise which is essentially of an ethical nature.

5. EVALUATION OF WASTE MANAGEMENT OPTIONS

The assessment of the impact of radioactive releases, directly from the operating fuel cycle or after disposal, is illustrated for the six particular radionuclides corresponding to some of the most important elements in terms of radiological impact (C-14, I-129, Kr-85, Cs-137, Pu-239 and Np-237), produced by the French nuclear fuel cycle. The stages involving these radionuclides are the production of electricity and the reprocessing of spent fuel. The release sites selected for the evaluation are the Flamanville nuclear power plant and the reprocessing plant at La Hague. The disposal sites considered are the French surface disposal centre in Aube (CSA) for low- and medium-level radioactive waste and an hypothetical underground disposal site in granite for high-level waste as described in the latest EC "Everest" study [1].

Table 1 shows, on the one hand, the various options for the managing activities involved for the six radionuclides considered in the context of the standard method currently used for waste, and, on the other hand, the various alternative options envisaged, namely:

- Underground disposal of all C-14 (half-life 5700 y), Kr-85 (half-life 10.8 y) and I-129 (half-life 16 10^6 y),

- Total release, either in the sea or in the atmosphere, of Cs-127 (half-life 30 y), Pu-239 (half-life 2.4 10^4 y) (excluding the recycled fraction), and Np-237 (half-life 2.1 10^6 y).

For the direct release of radioactivity into the biosphere, associated exposures to the reference groups were calculated using classic environmental dispersion models, such as the Gaussian plume model for atmospheric releases and the compartment marine model for liquid releases into the ocean.

With respect to surface disposal, the results were taken from the EC ExternE study [2], which is based on the radiological inventory and the characteristics of the Aube disposal centre (CSA). Where underground disposal is concerned, the findings of the "Everest" study, including the different intrusion scenarios adopted, are taken into account insofar as they are applicable.

Table 1. Distribution of the activities involved for each type of management

	C-14	Kr-85	I-129
Current management			
Gaseous releases	37.4 %	100 %	1.6 %
Liquid releases	12.5 %	0 %	86.7 %
Surface disposal	49.9 %	0 %	7.4 %
Underground disposal	0.2 %	0 %	4.2 %
Alternative management			
Underground disposal	100 %	100 %	100 %

	Cs-137	Pu-239	Np-237
Current management			
Gaseous releases	< 0.0001 %	< 0.0001 %	0 %
Liquid releases	< 0.0001 %	0.01 %	0.05 %
Surface disposal	< 0.0001 %	5.9 %	0.2 %
Underground disposal	99.99 %	94 %	99.7 %
Alternative management			
1- Gaseous releases	100 %	100 %	100 %
2- Liquid releases	100 %	100 %	100 %

In order to be able to illustrate the actual exposures linked to each waste management option, a hypothetical reference fuel cycle was defined as follows:

- An "Everest type" underground disposal site in granite intended to receive all the waste from 100 000 tons of reprocessed fuel,

- Two CSA-type near-surface disposal sites for low-and medium-level waste, each designed to receive the equivalent of 30 years' production of fuel cycle operations,

- A reprocessing plant enabling all of the spent fuel to be reprocessed, or the equivalent of around 60 years of operations at La Hague (capacity 1600 t/year),

- A set of nuclear power plants producing the 100 000 tons of irradiated fuel during 60 years of operation.

6. RESULTS

6.1. Maximum individual doses (excluding intrusion)

The maximum individual doses from the hypothetical cycle, for current management as well as for alternative management, are presented in Table 2. This table also shows the site responsible for the maximum exposure, together with the "most contributing" pathway.

Table 2. Individual doses to the reference group (mSv/year) associated with waste management options for a hypothetical cycle

Radionuclide	Current management (mainly disposal)	Underground disposal (excluding intrusion)
C-14	$6\ 10^{-3}$ (sea releases, reprocessing)	$6.6\ 10^{-4}$
Kr-85	$1.1\ 10^{-2}$ (atm. releases, reprocessing)	0
I-129	$4.8\ 10^{-2}$ (surface disposal)	$4.4\ 10^{-3}$

Radionuclide	Current management (mainly disposal)	Atmospheric releases	Marine releases
Cs-137	$1.4\ 10^{-3}$ (sea releases, reprocessing)	$> 10^{5}$	900
Pu-239	$3.6\ 10^{-4}$ (sea releases, reprocessing)	100	2.9
Np-237	$4.3\ 10^{-5}$ (atm. releases, reprocessing)	13	$9\ 10^{-2}$

For all of the radionuclides examined, it appears that the management method predominantly involving underground disposal (which corresponds to the current situation for Cs-137, Pu-239 and Np-237) leads to individual doses to the reference groups below those associated with the management methods based on immediate release into the environment. This is due:

- on the one hand, to the retention of radionuclides in underground disposal structures during a period sufficiently long for the initial activity to disappear entirely through radioactive decay, as in the case of Kr-85 or Cs-137, or partially disappear, as for C-14, of which only 3% of the initial inventory reappears in the biosphere for an underground disposal in granite and,

- on the other, to gradual releases over time from the disposal site, with a lower release rate than that for immediate release even if, several million years later some very long-lived radionuclides, such as I-129, are totally reinjected into the environment.

The previous table shows that the maximum annual individual doses related to current waste management remain well below the limit of 1 mSv/year set for the members of the public by the ICRP in its Publication 60. The same applies for the management option which consists in trapping and storing C-14, Kr-85 and I-129.

On the other hand, as regards the options consisting in totally discharging Cs-137, Pu-239 and Np-237, all of these management methods would lead to high levels of individual dose, from some mSv to several Sv per annum.

Beyond the direct comparison of maximum individual doses, an important parameter is the moment when these doses are received. Here again, the retention of the underground disposal site leads to a considerable postponement of these doses, ranging from 9 000 years for C-14 up to several million years for Np-237, whereas for direct release peak doses are not postponed. This illustrates the risk of a transfer between present and future generations that can arise due to the choice of storing the waste.

The evolution in time of the maximum exposure is illustrated in the following figures, together with the populations concerned (local reference group or global population), and this for two different radionuclides:

- one with a long half-life: C-14 (5 700 years), and
- the other with a short half-life: Kr-85 (10.8 years).

It is worthwhile noting that these figures do not take into account all the doses received by the regional or national populations, but only the exposed groups with the maximum individual doses for each relevant time period.

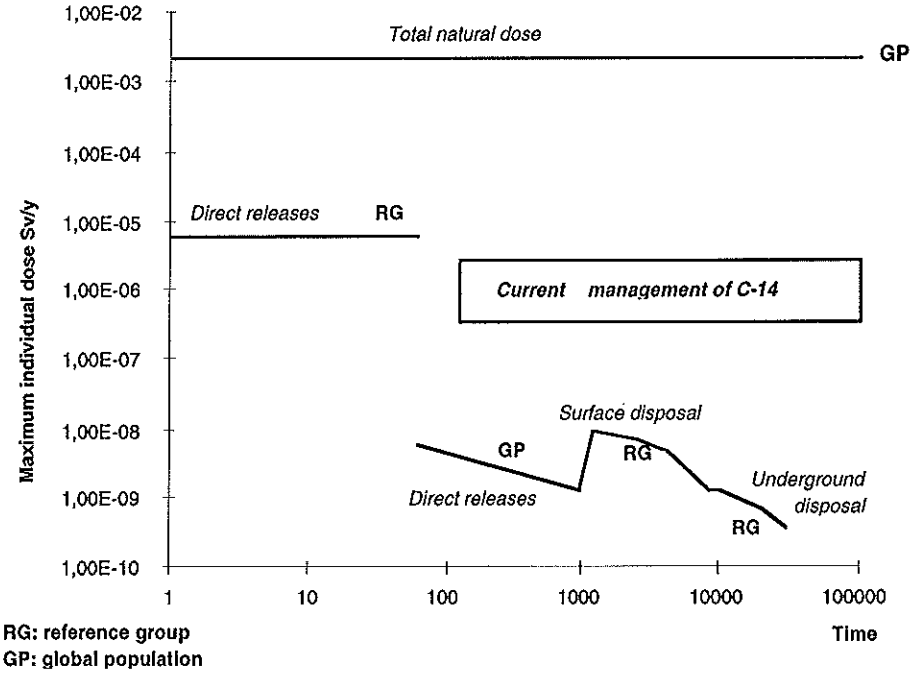

Figure 4. **Time and space distribution of the maximum individual doses associated with current C-14 management**

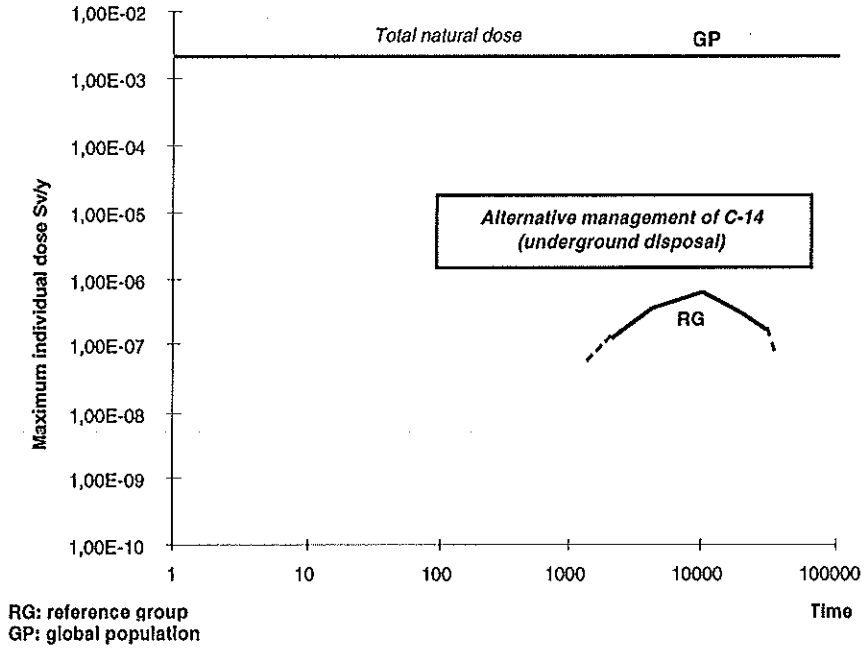

Figure 5. **Time and space distribution of the maximum individual doses associated with alternative C-14 management: underground disposal**

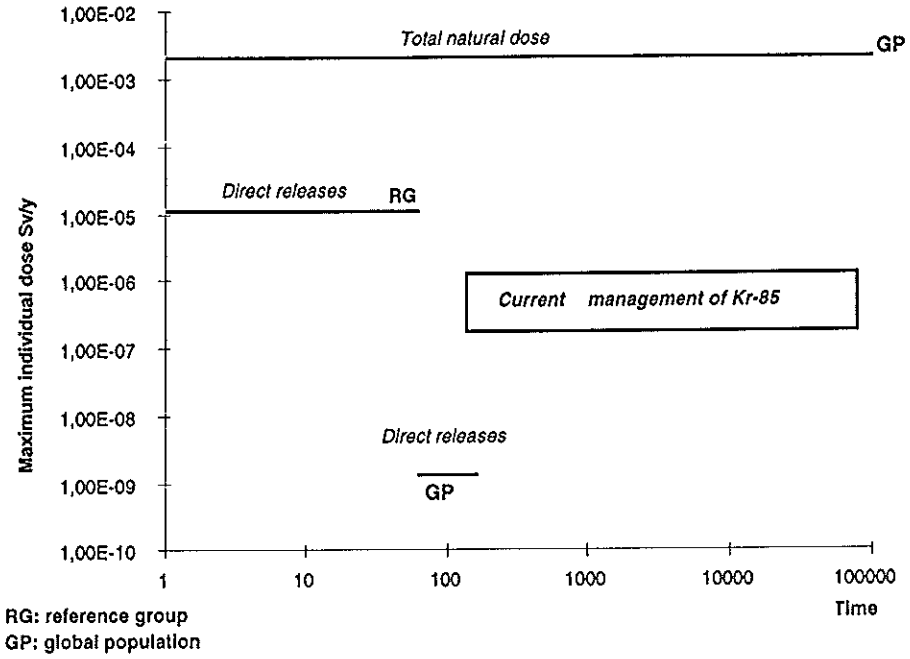

Figure 6. Time and space distribution of the maximum individual doses associated with current Kr-85 management

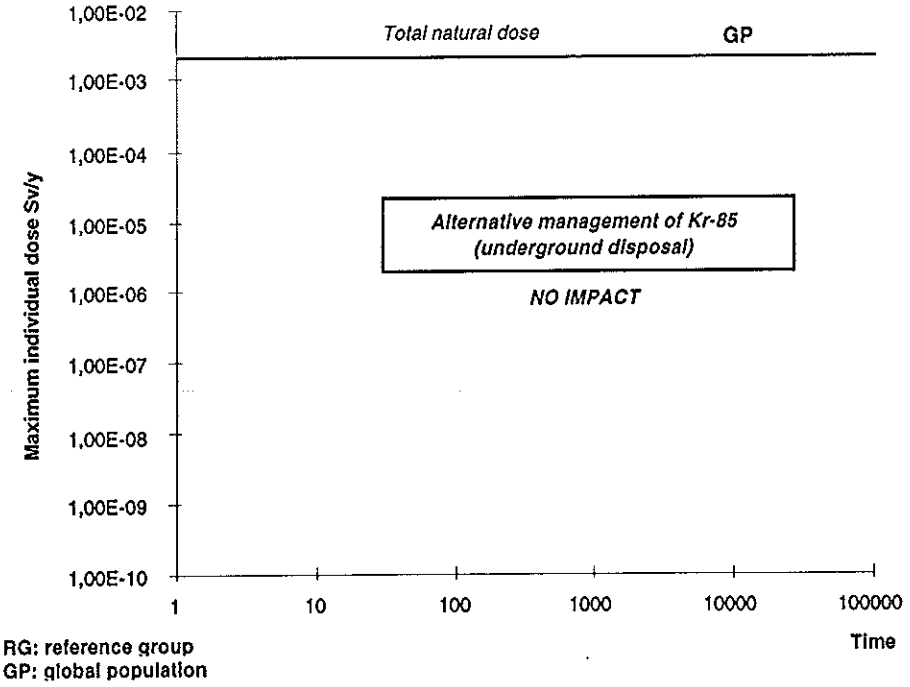

Figure 7. Time and space distribution of the maximum individual doses associated with alternative Kr-85 management: underground disposal

6.2. Individual doses related to intrusion in underground disposal

For the entire nuclear fuel cycle, the doses related to the additional disposal option of the radionuclides reduces the maximal doses to the reference group by deferring them. However, this option gives rise to the risk of intrusion within the disposal site, with the possibility for the individuals concerned of receiving higher individual doses.

The maximum individual doses linked to intrusion scenarios for an underground disposal site are shown in the following table for current waste management, as well as for the additional options examined. The intrusion scenarios considered are the drilling of a well near the disposal site (one at 610 m from the high-level vitrified waste and the other at 1230 m from the medium-level waste). The doses are expressed for the entire disposal site as defined in the Everest study. Probabilities of intrusion are not considered here. It should also be pointed out that these doses are deferred for a considerable length of time, from several thousands to several millions of years.

Table 3. Maximum individual doses linked to intrusion scenarios for underground disposal (mSv/year) excluding the probability of intrusion

	Current management	Alternative management
C-14	$1.2\ 10^{-5}$ - $5.3\ 10^{-4}$	$5.6\ 10^{-3}$ - 0.25
Kr-85	0	0
I-129	$3.8\ 10^{-3}$ - 1.8	0.89 - 42
Cs-137	0	0
Pu-239	$1.6\ 10^{-4}$ - $2.9\ 10^{-4}$	0
Np-237	$3.1\ 10^{-5}$ - $2.5\ 10^{-3}$	0

These maximum individual doses are, on the whole, several orders of magnitude above exposures linked to the normal evolution of the disposal site alone. As regards I-129 in

particular, they can reach levels of several mSv/year for the site as a whole, both for current waste management and the additional options envisaged.

6.3. Global collective exposure

Some of the selected radionuclides such as C-14, Kr-85 and I-129 may be dispersed world-wide, given their lengthy half-life or environmental mobility. Specific models have been used to assess the global collective exposures associated with these elements [3], taking into account the percentage of activity initially contained in waste released into the environment after the retention periods.

The results presented for Pu-239 and Np-237 are only indicative, as these radionuclides are not dispersed at global scale. Moreover, it is pointed out that, given their short radioactive half-lives, Kr-85 and Cs-137 disappear by decay before being able to re-emerge in the biosphere. The retention periods in granite vary between 1 000 and 2 000 years for C-14 and I-129, but are much longer for Pu-239 and Np-237 (around 100 000 years).

Table 4. Percentage of activity initially contained in waste being released through the outlets for underground disposal

	Medium level waste	High level waste
C-14	3 %	-
I-129	100 %	-
Pu-239	0.007%	0.00005 %
Np-237	1 %	0.04 %

Table 5 presents the global collective exposure, together with the population concerned and the approximate duration of the exposure. These values correspond to the hypothetical cycle defined above. The main results for the maximum individual doses, presented previously, are confirmed as far as the global collective doses are concerned, i.e.:

- deferred exposures for underground disposal,

- the disappearance through the radioactive decay of some radionuclides during disposal (partial disappearance for C-14 and total disappearance for Kr-85),

- the reappearance into the environment of all I-129 after several hundred thousand years,

- the transfer of risks from present to future generations.

Table 5. Average world collective doses associated with different waste management options (man.Sv)

Radionuclide		Population concerned	Duration of exposure	Global collective dose (man.Sv)			
				current management		underground disposal	
				total	per year	total	per year
C-14	0-100 years			9 400	94	0	0
	> 100 years	10^{10}	≈ 60 000 y	200 000	3	7 100	0.1
Kr-85	0-100 years			1 200	12	0	0
	> 100 years	10^{10}	≈ 100 y	0	0	0	0
I-129	0-100 years			90	0.9	0	0
	> 100 years	10^{10}	> 10^{8} y	150 000	0.001	150 000	0.001

It is important to point out that, even if the global collective doses are high, they result from the aggregation of very low average individual exposures over long periods of time. To illustrate the order of magnitude of the expected health impact, the ICRP 60 dose response function for fatal cancers (0.05 fatal cancer per man.Sv) can be applied to the average annual world collective dose for C-14 of 94 man.Sv. The expected associated health effects are about 5 deaths per year in a global population of 10 billion people. This value can be compared to the present world mortality by cancer of 7 000 000 deaths per year.

7. PERSPECTIVES

All the values presented in this report are related to six particular radionuclides, for a limited number of waste management options. Even if the selected radionuclides are among the most important in terms of radiological impact, a complete assessment of waste impact should include all the radionuclides contained in the waste, to be compared to the different categories of impact for the entire fuel cycle. This is particularly true for underground waste disposal, for both normal evolution scenarios and intrusion.

Concerning the intrusion scenarios, at this time no probability has been considered. This raises the problem of the weighting of certain exposures (normal evolution scenarios) as well as hypothetical exposures with a very low probability of occurrence. Both surface and underground disposals are concerned. It should be noted that no intrusion scenario for near-surface disposal sites has been considered in this analysis.

The alternative waste management options analysed are of a one-off nature, i.e., they consist in the total disposal or total release of a radionuclide. These assumptions lead to some extremely high exposures such as for the integral release of Cs-137. But between total disposal and total release, an infinity of choices are available. These intermediate options are, on the whole, more realistic from a technical point of view. The analysis of this kind of intermediate management options could give an estimation of the most appropriate options in an ALARA perspective.

Moreover, the risk transfers from the public to workers, which have not been examined in this report, could be an important indicator, together with the cost associated with each waste management option selected.

All these evaluations are preliminary. Their objective is to define a framework of reflection for decision-making on radiation protection actions which may have consequences in time and space. Improving the presentation of the results could lead to a better appreciation of the uncertainties associated with long-term and global exposures.

The problem is to estimate whether this kind of time and space presentation provides the right information for helping decision-making processes and facilitating the communication with non-specialists on the risks associated with the waste management

options, as well as the inter-generation risk transfer. Changing the time unit considered by using generations as an unit instead of years could be another possible presentation to be tested.

REFERENCES

[1] INSTITUT DE PROTECTION ET DE SURETE NUCLEAIRE, **Programme Européen EVEREST: Etude comparative des résultats obtenus par l'IPSN concernant les formations sédimentaires et granitiques**, Rapport IPSN/DES n° 278, 1996.

[2] DREICER M., TORT V., MANEN P., **Nuclear Fuel Cycle: Estimation of Physical Impacts and Monetary Valuation for Priority Pathways**, CEPN-R-234, February 1995.

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, **The Radiological Impact of Radionuclides Dispersed on a Regional and Global Scale: Methods for Assessment and their Application**, IAEA Technical Reports Series No. 250, Vienna, Austria, 1985.

APPENDIX: DETAILS OF THE CALCULATIONS

1. PRESENTATION OF MODELS USED

1.1. Dispersion in oceans

The CEPN has developed the "Poséidon" computer code [1] enabling the assessment of medium- and long-term radiological impacts arising from radioactive release in a marine environment and their widescale dispersion. The Poséidon code uses the modelling of European seas proposed in the framework of the European Commission MARINA project.

The approach is based on the subdivision of European seas into 44 compartments, each having, it is assumed, a similar distribution of activity and unchanging physical parameters (depth, rate of sedimentation, etc.). Exchanges between compartments are reflected by transfers of water mass. The kinetics disperses the radioactive pollutant, which is simultaneously subjected to the phenomena of sedimentation and decrease. Given the radiological importance of certain elements produced by disintegration, impact calculations have been extended to radionuclides produced by filiation. Their consideration can briefly be summarized in each compartment by a constant contribution of transmuted radionuclides depending on the quantity of ascendants present and the half-life of the latter.

On the basis of the definition of the release (location and source term), simulation enables the changes in concentration in each compartment to be determined. The next step consists in quantitatively assessing the collective intake of radionuclides on the basis of quantities harvested (fish, shellfish, molluscs and algae) in the different compartments and the transfer coefficients of the radionuclides to sea products. The final stage is the calculation of the collective dose and changes in it over time on the basis of collective intake previously calculated. For each observation time adopted, the code supplies the collective doses associated with each radionuclide selected from the filiation chain. It is therefore possible to ensure the contribution of descendants to the total collective dose. The results in terms of collective dose can also be obtained for the

different countries of the European Union on the basis of the origin and quantity of marine products consumed by each country. The impact in terms of individual doses is estimated from the concentration of radioactivity calculated in a local compartment (near the coasts) for fish, molluscs and shellfish and from the annual consumption of the most exposed population group (fishermen) of this food.

1.2. Atmospheric dispersion

The CEPN uses a standard Gaussian type model ("Calconc") developed on the basis of the model suggested by the CEA and NRPB. This model, on the basis of local meteorological conditions (wind speed, rainfall), assesses the dispersion of a unitary and continued release of radionuclides emitted at a given height. The calculation takes into account the decrease in radioactivity of the element together with plume depletion by dry and wet deposition. The model calculates atmospheric concentrations as well as radioactive deposits on the soil according to the distance from the point of disposal for different angular sectors (which correspond to the angular sectors defined for the wind rose) [2].

The transfer of the radioactivity deposited on the ground (dry and wet deposition), to the radioactivity contained in agricultural produce can be calculated using the transfer coefficients of the NRPB Farmland Model [3]. Each radionuclide has its own transfer coefficient, which varies over time, for each category of produce (grains, green vegetables, root vegetables, beef, mutton and milk). These coefficients can be applied throughout Europe, even though there may be considerable variations depending on climate or soil type. The transfer of C-14 to food does not follow a similar model, but uses a particular model based on the balance of the specific activity between the air and the food at a given point. The calculation of the ingested activity is based on the concentration of radioactivity in food (Bq/kg) and the annual food consumption of an adult.

Calculation of the inhaled activity is based on the concentration of radioactivity in the air (Bq/m^3) and the average respiratory rate of an adult (8 030 m^3/year).

The transfer of the activity both ingested and inhaled, expressed in Bq, to the dose expressed in Sv, can be calculated using the conversion coefficients published in ICPR 68 for adults [4].

External exposure due to the deposition on the soil of radionuclides emitted in a gaseous

form is assessed on the basis of the rate of deposit on the soil calculated by the atmospheric dispersion model and the coefficients particular to each element enabling the received dose to be directly calculated (Sv/year) at a point where the average annual deposit rate is known ($Bq/m^2.s$) [2]. Direct external exposure due to the plume is calculated using a finite cloud model for the γ and an infinite cloud model for the β. These models enable coefficients to be calculated for the transfer of the concentration of radioactivity in the air (Bq/m^3) to the dose received by external radiation [2].

Calculation of the individual dose to the critical group is based on an individual living under the plume, who only consumes food produced in this zone.

1.3. Global dispersion

Some of the selected radionuclides may be dispersed world-wide, given either their lengthy half-life or environmental mobility. Specific models are used for these elements (C-14, I-129 et Kr-85). These models are shown further on in this report for each of the radionuclides.

1.4. Underground disposal

The findings presented in this report are taken from the "Everest" study on underground disposal in granite formations. This study assesses the radiological impact of release in a biosphere of reference for a hypothetical granite site, in terms of individual doses related to the following affected pathways: ingestion of river water and fish, of agricultural produce irrigated with river water, external exposure and inhalation. The stored waste stems from the reprocessing of 100 000 tons of fuel and is composed of high-level vitrified waste (HA) and medium-level waste (hulls and caps - MA). The scenarios considered are, on the one hand, the normal development of the disposal site and, on the other, an intrusion scenario with the drilling of a well (service life 100 years) near the disposal site (a well at 610 m from the HA vitrified waste and a well at 1 230 m from the MA waste). No collective dose assessment has been made in the framework of this project. It is difficult to assess collective doses in the same way as for continuous and constant releases in the sea or atmosphere (local, regional and global doses). Nevertheless, the global dose can be estimated by taking into account the percentage of the activity which was initially contained and subsequently released through the outlets.

It is important to point out that the Everest study also relates to the disposal of waste in

clayey formations. The assessments made in this part of the study lead to findings in terms of radiological impact which is lower by several orders of magnitude than the impact related to the granite site. These doses are, moreover, deferred by one order of magnitude. The values presented in the next part of this report therefore correspond to the site leading to the highest exposures, that is to say, the granite site.

1.5. Surface disposal

As regards the surface disposal of waste, the site under consideration is the Aube disposal centre. The results, taken from the ExternE study, are based on the radiological capacity of this centre. This inventory corresponds to the waste originating from around 30 years of nuclear electricity production, which represents about 10 000 TWh.

2. ASSESSING THE IMPACT OF SELECTED RADIONUCLIDES

This section presents an assessment of the radiological impacts related to the current management of the six radionuclides already mentioned. The results as a whole are related to the production of energy (normalization by electric TWh).

2.1. C-14

C-14 is a radionuclide with a half-life of 5 700 years, naturally present into the environment and leading to an average annual dose of around 0.012 mSv [5]. In nuclear reactors, it is produced both in the fuel and in the moderator by reaction to O-17 and nitrogen respectively.

The C-14 produced in the reactors is released into the environment by nuclear power plants and the La Hague reprocessing plant without any retention whatsoever. A fraction is, however, found in solid waste resulting from reprocessing operations on irradiated fuels. The quantity of C-14 in question is around 0.08 TBq/TWh, or in the region of 50 TBq/year.

A part of this C-14 is released in gaseous form (around 0.013 TBq/TWh released by power plants and 0.017 TBq/TWh through reprocessing). Liquid releases are only evident at the reprocessing stage and represent around 0.011 TBq/TWh.

The calculation of the radiological impact of liquid or gaseous emissions of C-14 facilities from the fuel cycle (nuclear power plants and reprocessing) varies in accordance with the distance to the release point under consideration. Local and regional impacts can be assessed using standard tools, adapted to most radionuclides such as Gaussian plume dispersion models for atmospheric releases or dispersion models with compartments for liquid releases.

Nevertheless, C-14 has a special feature: given its long half-life and because it is a carbon isotope, it will follow the carbon cycle at planetary scale and disperse world-wide. Thus, the dispersion of this C-14 at global scale can lead to a radiological impact on the entire global population. A compartment model used by the IAEA (1985) enables calculations to be made using a constant world population of 10 billion individuals [6]. This model has been used in particular to assess doses due to C-14 in the framework of the ExternE project. The world individual and collective doses for 1 MBq of C-14

released in the atmosphere are shown in the following table. Given the fact that the cycles of carbon present in the atmosphere and oceans are closely linked, it will be considered that, in the long term, the values given for the doses associated with gaseous releases are applicable to liquid releases.

Table 1. Dose rates and total cumulated world rate associated with a unitary gaseous release of C-14

	Dose rate (Sv/y by MBq)		Total cumulated dose (Sv by MBq)	
time (years)	individual	collective	collective	distribution (%)
10	2.3E-17	2.3E-7	4.1E-6	3 %
100	4.0E-18	4.0-E-8	1.1E-5	8 %
1 000	1.5E-18	1.5E-8	3.1E-5	22 %
10 000	5.4E-19	5.4E-9	9.1E-5	65 %
infinite (100 000)	0	0	1.4E-4	100 %

With regard to the underground disposal of C-14, the values shown in the Everest project are based on the quantity of C-14 contained in medium-level waste (3.7 TBq for 100 000 t of reprocessed fuel, or around 1.7 10^{-4} TBq/TWh). For the normal evolution scenario, the maximal individual dose calculated is 3.5 10^{-10} Sv/year for 1 TBq of stored C-14. This maximal dose is reached 9 400 years after the beginning of release.

The maximal doses related to the two intrusion scenarios are 1.2 10^{-8} à 5.3 10^{-7} Sv/year for the stored activity and correspond to wells drilled at 21 000 and 8 500 years respectively.

It is important to note that only 3% of the C-14 initially contained in the waste is found in the biosphere, i.e., 5 10^{-6} TBq/TWh, the remainder having disappeared through radioactive decay.

At the level of surface disposal, the C-14 contained in the radiological inventory of the Centre de l'Aube (CSA) is 400 TBq, or around 0.04 TBq/TWh. The maximum dose estimated is 10^{-8} Sv/year [7].

The following table shows the radiological impact in terms of individual doses to the reference group and collective dose related to current management of C-14 (excluding intrusion). Although the dispersion of the C-14 may lead to relatively high collective

doses on a global scale, given its lengthy half-life and environmental mobility, the associated individual doses are very low, in the region of the μSv for 1 TBq released.

Table 2. Dosimetric impact of current C-14 management for a release of around 0.03 TBq/TWh in gaseous form and 0.01 TBq/TWh in liquid form, 1.7 10^4 TBq/TWh intended for underground disposal and 0.04 TBq/TWh for surface disposal

	Atmospher. releases	Marine releases	Surface disposal	Deep disposal	Total
Max. individual dose to the reference group (Sv/year by TWh)	6 10^{-9} - 1 10^{-8}	1.7 10^{-8}	10^{-12}	10^{-13}	1.7 10^{-8} (max.)
Collective dose (man. Sv/TWh)					
local 0-100 years	1.1 10^{-4}	*	0	0	1.1 10^{-4}
local 0-100 000 years	1.1 10^{-4}	*	*ND*	*ND*	*ND*
regional 0-100 years	6.9 10^{-4}	3.9 10^{-3}	0	0	4.6 10^{-3}
regional 0-100 000 years	6.9 10^{-4}	5.3 10^{-3}	*ND*	*ND*	*ND*
world 0-100 years	0.38	0.13	0	0	0.51
world 0-100 000 years	4.3	1.4	< 5.6	7 10^{-4}	11.3

* included in the regional dose
ND: not determined

2.2. Kr-85

Kr-85 is a noble gas with a radioactive half-life of 10.8 years. This fission product created in the core of the reactor is released in large quantities into the environment by nuclear power plants and the La Hague reprocessing plant, as no retention procedure is currently used. As a noble gas, it is only eliminated from the atmosphere through radioactive decay (disintegration β^-) and, as it is neither deposited on the soil nor absorbed by the organism, the only mode of exposure for man is external irradiation. It is released, of course only in gaseous form.

The quantities of Kr-85 produced are in the region of 633 TBq/TWh, mainly released during reprocessing. A small fraction is released by nuclear power plants (0.07 TBq/TWh).

The radiological impact of Kr-85 emissions from cycle facilities at local and regional levels can be calculated using the Gaussian plume atmospheric dispersion model. Like C-14, Kr-85 is dispersed at planetary level. A very simple compartment model enables the impact of this dispersion to be assessed, using a constant world population of

10 billion as a basis [6].

The world individual and collective doses for 1 MBq of Kr-85 released in the atmosphere are shown in the following table.

Table 3. Dose rates and total cumulated world rate associated with a unitary release of Kr-85

	Dose rate (Sv/y by MBq)		Total cumulated dose (Sv by MBq)	
time (years)	individual	collective	collective	distribution (%)
10	2.8E-22	2.8E-12	4.2E-11	48 %
20	1.4E-22	1.4E-12	6.0E-11	69 %
30	7.5E-23	7.5E-13	7.5E-11	86 %
100	8E-25	8E-15	8.7E-11	100 %
100 000	0	0	8.7E-11	100 %

The following table shows the radiological impact in terms of individual dose to the reference group and collective dose related to current Kr-85 management.

Table 4. Dosimetric impact of gaseous release of 633 TBq/TWh of Kr-85

	Atmospheric release
Max. individual dose to the reference group (Sv/year)	$3.2\ 10^{-8}$
Collective dose (man.Sv)	
local 0-100 years	$5\ 10^{-5}$
local 0-100 000 years	$5\ 10^{-5}$
regional 0-100 years	$6.4\ 10^{-3}$
regional 0-100 000 years	$6.4\ 10^{-3}$
world 0-100 years	$5.6\ 10^{-2}$
world 0-100 000 years	$5.6\ 10^{-2}$

2.3. I-129

I-129 is a very long-lived radionuclide: 17 million years. It disintegrates by β^- emission and reaches man through every exposure pathway: inhalation, ingestion, and external exposure. Formed in the core of a nuclear reactor by uranium fission, most I-129 is released into the environment during reprocessing operations, as it is not currently submitted to any trapping procedure. A small fraction is, however, found in solid waste

intended for disposal.

The quantity of I-129 called into question is around 4 10^{-3} TBq/TWh. Most of this I-129, approximately 3.5 10^{-3} TBq/TWh, is released in liquid form in the Channel. A small fraction is emitted in the form of atmospheric releases, in the region of 6.6 10^{-5} TBq/TWh, whereas around 3 10^{-4} TBq/TWh passes into solid waste intended for surface disposal and 1.7 10^{-4} TBq/TWh into waste intended for underground disposal.

The local and regional impact of liquid and gaseous I-129 emissions can be assessed using the dispersion models already described. On the other hand, specific models must be applied when calculating the impacts of its dispersion at planetary level as, like C-14 and Kr-85, its very long half-life means that it is distributed in the longer term throughout all the compartments of the environment. The world individual and collective doses for 1 MBq of I-129 released in the atmosphere and the ocean are shown in the following tables [6].

Table 5. Dose rates and total cumulated world rate associated with a unitary release of I-129 in the atmosphere

	Dose rate (Sv/y by MBq)		Total cumulated dose (Sv by MBq)	
time (years)	individual	collective	collective	distribution (%)
10	1E-17	1E-7	4.4E-5	2 %
100	9.9E-18	9.9E-8	5.3E-5	2.4 %
1 000	8.1E-18	8.1E-8	1.3E-4	6 %
10 000	1.1E-18	1.1E-8	4.4E-4	20 %
100 000	6.1E-20	6.1E-10	5.5E-4	25 %
infinite	0	0	2.2E-3	100 %

Table 6. Dose rates and total cumulated world rate associated with a unitary release of I-129 in the ocean

	Dose rate (Sv/y by MBq)		Total cumulated dose (Sv by MBq)	
time (years)	individual	collective	collective	distribution (%)
10	4.3E-19	4.3E-9	5.5E-8	0.003 %
100	3.2E-20	3.2E-10	1.6E-7	0.009 %
1 000	3.8E-20	3.8E-10	4.6E-7	0.03 %
10 000	6.9E-20	6.9E-10	5.8E-6	0.3 %
10 0000	6.1E-20	6.1E-10	6.6E-5	4 %
infinite	0	0	1.7E-3	100 %

Where the deep disposal of I-129 is concerned, the values shown in the Everest project are based on the quantity of I-129 contained in medium-level waste (3.7 TBq for 100 000 t of reprocessed fuel). Given the very long half-life of I-129, all the initial activity contained in the waste is released through the outlets (integration over 10^7 years). For the normal evolution scenario, the maximum individual dose calculated is 5.1 10^{-8} Sv/year for 1 TBq of I-129 stored. This maximum dose is reached 20 000 years after the beginning of disposal.

The doses related to both intrusion scenarios vary from 3.8 10^{-5} to 1.8 10^{-3} Sv/year for the stored activity and correspond to wells drilled at 82 000 and 100 000 years respectively.

As regards of surface disposal, the maximum dose due to I-129 is 1.8 10^{-5} Sv/year, for a stored quantity of 3 TBq (3 10^{-4} TBq/TWh). This dose is reached 1100 years after the beginning of disposal. The total initial inventory is released into the environment.

The following table presents the radiological impact in terms of individual dose to the reference group and collective dose related to current I-129 management (excluding intrusion).

Table 7. **Dosimetric impact of current I-129 management, with a gaseous release of around 6.6 10^{-5} TBq/TWh and liquid release of 3.5 10^{-3} TBq/TWh, 1.7 10^{-4} TBq/TWh intended for underground disposal and 3 10^{-4} TBq/TWh intended for surface disposal.**

	Atmospher. release	Marine release	Surface disposal	Deep disposal	Total
Individual max. dose to the reference group (Sv/year)	1.3 10^{-8}	6 10^{-10}	1.8 10^{-9}	9 10^{-12}	1.3 10^{-8} (max.)
Collective dose (man.Sv)					
local 0-100 years	1.9 10^{-4}	*	0	0	1.9 10^{-4}
local 0-100 000 years	2.2 10^{-4}	*	*ND*	*ND*	*ND*
regional 0-100 years	3.5 10^{-3}	4.6 10^{-4}	0	0	4 10^{-3}
regional 0-100 000 years	4.1 10^{-3}	4.6 10^{-4}	*ND*	*ND*	*ND*
world 0-100 years	3.5 10^{-3}	5.6 10^{-4}	0	0	4.1 10^{-3}
world 0-100 000 years	3.6 10^{-2}	0.23	*ND*	*ND*	*ND*
world 0-infinite	0.15	5.95	0.51	0.3	6.91

* included in the regional dose
ND: not determined

2.4. Cs-137

Cs-137, which is one of the main fission products created in a nuclear reactor during irradiation of the fuel has a half-life of 30 years. The quantities produced are in the region of 5 to 6 million TBq a year, or around 15 000 TBq/TWh.

Most of the Cs-137 contained in the irradiated fuel reprocessed at La Hague is currently kept and stored. Nevertheless, some releases in liquid and gaseous form into the environment can be noted: 0.025 TBq/TWh released in liquid form (mainly from reprocessing) and 5 10^{-7} TBq/TWh in gaseous form (mainly from power plants, this being an envelope value).

The dispersion of liquid and gaseous release can be assessed using the models described above. With respect to disposal, given its 30-year half-life, it can be assumed that the Cs-137 will have disappeared through decay before being able to enter the biosphere.

The following table shows the radiological impact in terms of individual dose to the reference group and collective dose related to current Cs-137 management.

Table 8. Dosimetric impact of current Cs-137 management, with a gaseous release of around 5 10^{-7} TBq/TWh and a liquid release of 0.025 TBq/TWh, the remainder being stored

	Atmospher. release	Marine release	Disposal	Total
Max. individual dose to the reference group (Sv/year)	1.5 10^{-9}	4 10^{-9}	0	4 10^{-9} (max.)
Collective dose (man.Sv)				
local 0-100 years	7 10^{-7}	*	0	7 10^{-7}
local 0-100 000 years	7 10^{-7}	*	0	7 10^{-7}
regional 0-100 years	4 10^{-6}	9 10^{-4}	0	9 10^{-4}
regional 0-100 000 years	4 10^{-6}	9 10^{-4}	0	9 10^{-4}
world 0-100 years	-	-	-	-
world 0-100 000 years	-	-	-	-

* included in the regional dose

2.5. Pu-239

Pu-239 is an α emitter radionuclide with a half-life of 24000 years, produced in nuclear reactors during irradiation of the fuel. During reprocessing operations this Pu-239 is, like the uranium remaining in the irradiated fuel elements, separated and recovered for recycling. Although most of the Pu-239 contained in the irradiated fuel may be recovered, there nevertheless remains some traces of this radionuclide in the waste. The remaining quantity is currently in the region of 0.12 % of the initial quantity, which represents around 64 GBq/TWh. Nevertheless, the quantities of Pu-239 for disposal given in this report for the disposal correspond to less recent inventories, in which the quantity of retained Pu is higher and corresponds to a total activity of 0.4 TBq/TWh (or around 0.7% of the initial activity of the irradiated fuel).

Most of this Pu-239 is found in solid waste (0.38 TBq/TWh in underground disposal and 0.024 TBq/TWh in the surface disposal). However, small quantities of Pu-239 are released in the sea and the atmosphere during reprocessing operations (48 Bq/TWh in gaseous form and 0.07 GBq/TWh in liquid form).

The dispersion of liquid and gaseous releases can be assessed using the models described above. Where underground disposal is concerned, the values presented in the Everest project are based on the quantity of Pu-239 contained in medium-level waste and vitrified waste (8330 TBq for 100 000 t of reprocessed fuel, of which 94% is vitrified and 6% is of medium level). For the normal evolution scenario, the maximum individual dose calculated is $6\ 10^{-14}$ Sv/year for 1 TBq of Pu-239 stored in high-level waste (reached at 150 000 years) and 10^{-12} Sv/year for 1 TBq of Pu-239 stored in medium-level waste (reached at 200 000 years).

The doses related to both intrusion scenarios vary from $1.6\ 10^{-7}$ to $2.9\ 10^{-7}$ Sv/year for the stored activity and correspond to wells drilled at 150 000 and 200 000 years respectively.

With respect to surface disposal, the maximum dose to the reference group assessed for the CSA is 10^{-9} Sv/year for a quantity of Pu-239 of 240 TBq (0.024 TBq/TWh). This dose is reached around 100 000 years after the beginning of disposal.

The following table shows the radiological impact in terms of individual dose to the reference group and of collective dose related to current Pu-239 management (excluding intrusion).

Table 9. **Dosimetric impact of current Pu-239 management, with a gaseous release of around 48 Bq/TWh and liquid release of 0.07 GBq/TWh, 0.024 TBq/TWh intended for surface disposal and 0.38 TBq/TWh intended for underground disposal**

	Atmospher. release	Marine release	Surface disposal	Deep disposal	Total
Max. individual dose to the reference group (Sv/year)	$< 10^{-13}$	10^{-9}	10^{-13}	$< 10^{-13}$	10^{-9} (max.)
Collective dose (man.Sv)					
local 0-100 years	$3\ 10^{-10}$	*	0	0	$3\ 10^{-10}$
local 0-100 000 years	$3\ 10^{-10}$	*	*ND*	*ND*	*ND*
regional 0-100 years	$2\ 10^{-9}$	$7\ 10^{-4}$	0	0	$7\ 10^{-4}$
regional 0-100 000 years	$2\ 10^{-9}$	$7\ 10^{-4}$	*ND*	*ND*	*ND*
world0-100 years	-	-	-	-	-
world 0-100 000 years	-	-	-	-	-

* included in the regional dose
ND: not determined

2.6. Np-237

Np-237 is an α emitter actinide with a half-life of 2.1 million years and is produced by the disintegration of Am-241.

The total quantity of Np-237 involved each year is around 0.053 TBq/TWh. Most of the Np-237 contained in irradiated fuel reprocessed at La Hague is currently retained and stored (0.053 TBq/TWh in underground disposal and 10^{-4} TBq/TWh in surface disposal). Nevertheless, some liquid releases into the environment about 0.025 GBq/TWh. can be noted.

The dispersion of this liquid release can be assessed using the model described above. Where underground disposal is concerned, the values shown in the Everest project are based on the quantity of Np-237 contained in medium-level waste and in vitrified waste (for 100 000 t of reprocessed fuel: 1158 TBq in vitrified waste and 0.22 TBq in medium-level waste). For the normal evolution scenario, the maximum individual dose calculated is $9\ 10^{-10}$ Sv/year for 1 TBq of Np-237 stored in medium-level waste (reached after 10 million years) and $2\ 10^{-12}$ Sv/year for 1 TBq of Np-237 stored in high-level waste (reached at 6 million years).

The doses related to both intrusion scenarios vary from 3.1 10^{-8} to 2.5 10^{-6} Sv/year for the stored activity and correspond to wells drilled at 800 000 and $>10^{7}$ years respectively.

With respect to surface disposal, the maximum dose to the reference group assessed for the CSA is 3 10^{-8} Sv/year for 1 TBq of Np-237 stored (10^{-4} TBq/TWh).

The following table shows the radiological impact in terms of individual dose to the reference group and collective dose related to current Np-237 management (excluding intrusion).

Table 10. **Dosimetric impact of current Np-237 management, with a liquid release of around 0.025 GBq/TWh, 0.053 TBq/TWh intended for underground disposal and 10^{-4} TBq/TWh intended for surface disposal**

	Marine release	Surface disposal	Deep disposal	Total
Max. individual dose to the reference group (Sv/year)	1.2 10^{-10}	3 10^{-12}	10^{-13}	1.2 10^{-10} (max.)
Collective dose (man.Sv)				
local 0-100 years	*	0	0	0
local 0-100 000 years	*	*ND*	*ND*	*ND*
regional 0-100 years	4.3 10^{-5}	0	0	4.3 10^{-5}
regional 0-100 000 years	5.5 10^{-5}	*ND*	*ND*	*ND*
world 0-100 years	-	-	-	-
world 0-100 000 years	-	-	-	-

* included in the regional dose
ND: not determined

3. EXAMINATION OF THE VARIOUS ADDITIONAL OPTIONS

The analysis of the variations in radiological impact associated with the different options for managing radioactive waste is illustrated in this chapter for the six radionuclides examined above. The additional options assessed are as follows:

- Underground disposal (total confinement) of C-14, I-129 and Kr-85, which are currently almost entirely released by the La Hague reprocessing plant and, with respect to C-14 and Kr-85 by nuclear power plants.

- Continuous atmospheric and marine releases into the environment (total dilution), of the radionuclides Cs-137, Pu-239 (excluding the portion stored for recycling) and Np-237, currently practically entirely stored with a view to their final disposal.

3.1. Disposal of C-14

The following table shows the radiological impact in terms of individual dose to the reference group and collective dose related to C-14 management options: current management and trapping/underground disposal (excluding intrusion). Where underground disposal is concerned, the individual doses are taken from the Everest study (findings based on the quantities of C-14 contained the hulls and caps).

Table 11. Impact of management options of 0.08 TBq/TWh of C-14

Management option	Current management	Underground disposal
Max. individual dose to the reference group (Sv/year)	$1.7\ 10^{-8}$ (max.)	$3\ 10^{-11}$
Collective dose (man.Sv)		
local 0-100 years	$1.1\ 10^{-4}$	0
local 0-100 000 years	*ND*	*ND*
regional 0-100 years	$4.6\ 10^{-3}$	0
regional 0-100 000 years	*ND*	*ND*
world 0-100 years	0.51	0
world 0-100 000 years	11.3	0.34

* included in the regional dose *ND*: not determined

3.2. Disposal of Kr-85

The following table shows the radiological impact in terms of individual dose to the reference group and collective dose related to the Kr-85 management options: current management and trapping/underground disposal.

As regards the underground disposal of trapped Kr-85, considering, on the one hand, its relatively short half-life of 10.8 years and, on the other, the lapse of time between the moment when the radionuclide is stored and when the confinement structures begin to allow leakage of its radioactivity, the Kr-85 will have disappeared through radioactive decay before being able to enter the biosphere, resulting in zero radiological impact.

Table 12. Dosimetric impact of management options for 633 TBq/TWh of Kr-85

Management option	Current management	Underground disposal
Max. individual dose to the reference group (Sv/year)	$3.2\ 10^{-8}$	0
Collective dose (man.Sv)		
local 0-100 years	$5\ 10^{-5}$	0
local 0-100 000 years	$5\ 10^{-5}$	0
regional 0-100 years	$6.4\ 10^{-3}$	0
regional 0-100 000 years	$6.4\ 10^{-3}$	0
world 0-100 years	$5.6\ 10^{-2}$	0
world 0-100 000 years	$5.6\ 10^{-2}$	0

3.3. Disposal of I-129

The following table shows the radiological impact in terms of individual dose to the reference group and collective dose related to I-129 management options: current management and trapping/underground disposal (excluding intrusion). The results are expressed for $4\ 10^{-3}$ TBq/TWh.

Where underground disposal is concerned, the values have been taken from the Everest study (quantity of I-129 contained in hulls and caps).

Table 13. Dosimetric impact of management options for 4 10^{-3} TBq/TWh of I-129

Management option	Current management	Underground disposal
Max. individual dose to the reference group (Sv/year)	1.3 10^{-8} (max.)	2 10^{-10}
Collective dose (man.Sv)		
local 0-100 years	1.9 10^{-4}	0
local 0-100 000 years	*ND*	*ND*
regional 0-100 years	4 10^{-3}	0
regional 0-100 000 years	*ND*	*ND*
world 0-100 years	4.1 10^{-3}	0
world 0-infinite	6.9	6.9

* included in the regional dose
ND: not determined

3.4. Cs-137 Release

The following table shows the radiological impact in terms of individual dose to the reference group and collective dose related to Cs-137 management options: current management, total release in the sea and atmosphere (from the La Hague plant). The results are expressed for 15 000 TBq/TWh.

Table 14. Dosimetric impact of management options for 15000 TBq/TWh of Cs-137

Management option	Current management	Atmospheric releases	Marine releases
Max. individual dose to the reference group (Sv/year)	4 10^{-9} (max.)	> 10	2.4 10^{-3}
Collective dose (man.Sv)			
local 0-100 years	7 10^{-7}	21 000	*
local 0-100 000 years	7 10^{-7}	22 000	*
regional 0-100 years	9 10^{-4}	116 000	540
regional 0-100 000 years	9 10^{-4}	118 000	540
world 0-100 years	-	-	-
world 0-100 000 years	-	-	-

* included in the regional dose

It is important to note that the impressive dose levels obtained for the total release of Cs-137 in gaseous form are due to the fact that the activity taken into account for this option is more than 10^{11} times higher than the standard release of Cs-137 from cycle facilities.

3.5. Release of Pu-239

The following table shows the radiological impact in terms of individual dose to the reference group and collective dose related to Pu-239 management options: current management (excluding intrusion), total release in the sea and atmosphere (from La Hague site). The results are expressed for 0.4 TBq/TWh (0.12 % of the total Pu-239 content).

Table 15. Dosimetric impact of management options for 0.4 TBq/TWh of Pu-239

Management option	Current management	Atmospheric disposal	Marine disposal
Max. individual dose to the reference group (Sv/year)	10^{-9} (max.)	$3\ 10^{-4}$	$8\ 10^{-6}$
Collective dose (man.Sv)			
local 0-100 years	$3\ 10^{-10}$	2.7	*
local 0-100 000 years	*ND*	2.7	*
regional 0-100 years	$7\ 10^{-4}$	16.5	3.9
regional 0-100 000 years	*ND*	16.5	4
world 0-100 years	-	-	-
world 0-100 000 years	-	-	-

* included in the regional dose
ND: not determined

3.6. Np-237 Release

The following table shows the radiological impact in terms of individual dose to the reference group and collective dose related to Np-237 management options: current management (excluding intrusion), total release in the sea and atmosphere (from the La Hague site). The results are expressed for 53 GBq/TWh.

Table 16. Dosimetric impact of management options for 53 GBq/TWh of Np-237

Management option	Current management	Atmospheric releases	Marine releases
Max/ individual dose to the reference group (Sv/year)	$1.2\ 10^{-10}$ (max.)	$3.7\ 10^{-5}$	$2.5\ 10^{-7}$
Collective dose (man.Sv)			
local 0-100 years	0	0.46	*
local 0-100 000 years	*ND*	0.46	*
regional 0-100 years	$4.3\ 10^{-5}$	2.9	$9\ 10^{-2}$
regional 0-100 000 years	*ND*	2.9	0.12
world 0-100 years	-	-	-
world 0-100 000 years	-	-	-

* included in the regional dose
ND: not determined

APPENDIX REFERENCES

[1] D. RAFFESTIN, S. LEPICARD, **Poseidon: un modèle de dispersion de matières radioactives en milieu maritime**, Rapport CEPN n°236, Fontenay-aux-Roses, 1995.

[2] EUROPEAN COMMISSION, Radiation Protection 72, **Methodology for assessing the radiological consequences of routine releases of radionuclides to the environment**, Report EUR 15760 EN, 1995.

[3] C.A. ROBINSON et al., **Critical Group Doses Around Nuclear Sites in England and Wales**, NRPB-R271, UK, 1994.

[4] INTERNATIONAL COMMISSION OF RADIOLOGICAL PROTECTION, **Dose Coefficients for Intakes of Radionuclides by Workers**, ICRP Publication 68, 1994.

[5] UNITED NATIONS SCIENTIFIC COMMITTEE ON THE EFFECTS OF ATOMIC RADIATION, **Sources, Effects and Risks of Ionizing Radiation, UNSCEAR 1988 report,** United Nations, New York, USA, 1988.

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, **The Radiological Impact of Radionuclides Dispersed on a Regional and Global Scale: Methods for Assessment and their Application**, IAEA Technical Reports Series No. 250, Vienna, Austria, 1985.

[7] DREICER M., TORT V., MANEN P., **Nuclear Fuel Cycle: Estimation of Physical Impacts and Monetary Valuation for Priority Pathways**, CEPN-R-234, February 1995.