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DERIVATION OF WASTE ACCEPTANCE CRITERIA FOR LOW AND INTERMEDIATE LEVEL WASTE IN SURFACE DISPOSAL FACILITY. ACTIVITY LIMITS AT THE CENTRE DE L'AUBE (France)

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ABSTRACT

In France, low- and intermediate-level radioactive wastes are disposed in a near-surface facility, at Centre de l'Aube disposal facility. This facility, which was commissioned in 1992, has a disposal capacity of one million cubic meters, and will be operated up to about 2050. It took over the job from Centre de la Manche, which was commissioned in 1969 and shut down in 1994, after having received about 520,000 cubic meters of wastes. The Centre de l'Aube disposal facility is designed to receive a many types of waste produced by nuclear power plants, reprocessing, decommissioning, as well as by the industry, hospitals and armed forces. The limitation of radioactive transfer to man and the limitation of personnel exposure in all situations considered plausible require limiting the total activity of the waste disposed in the facility as well as the activity of each package. The paper presents how ANDRA has derived the activity-related acceptance criteria, based on the safety analysis. In the French methodology, activity is considered as end-point for deriving the concentration limits per package, whereas it is the starting point for deriving the total activity limits. For the concentration limits (called here LMA) the approach consists of five steps:

- the determination of radionuclides important for safety with regards to operational and long-term safety,
- the use of relevant safety scenarios as a tool to derive quantitative limits,
- the setting of dose constraint per situation associated with scenarios,
- the setting of contribution factor per radionuclide, and
- the calculation of concentration activity limits.

An exhaustive survey has been performed and has shown that the totality of waste packages which should be delivered by waste generators are acceptable in terms of activity limits in the Centre de l'Aube. Examples of concentration activity limits derived from this methodology are presented. Furthermore those limits have been accepted by the French regulatory body and constitute a key point of ANDRA waste acceptance criteria.

1. INTRODUCTION

Most of low- and intermediate-level wastes are generated by the different branches of nuclear industry in France. Fuel cycle waste include wastes from conversion, enrichment of uranium and fabrication of fuel assemblies. Power plants represent the major part of the deliveries though it considerably decreased by a factor 3.5 in 15 years. Waste from reprocessing plants are chiefly technological wastes. The French Atomic Energy Commission delivers research wastes. There is also a small waste stream from small producers: hospital, health industry and non nuclear-industry. They come from about 1,000 sites and are collected by a special division at ANDRA, which characterises and conditions them for disposal.

The deliveries to Centre de l'Aube in 1998 were 12 763 m^3: Power plants 5632 m^3, Reprocessing 3347 m^3, Research 1476 m^3, Fuel cycle 850 m^3, Small producers 293 m^3, ANDRA 165 m^3.

2. THE NEED TO LIMIT QUANTITIES OF ACTIVITY TO BE DISPOSED OF

2.1 Consequences of the safety objectives of a near-surface repository

The disposing of this wide variety of waste must comply with two basic safety objectives defined by the French regulatory body: the immediate and future protection of the public and the environment, and the limitation to 300 years of the surveillance period required.

ANDRA adopted the following design criteria to meet these objectives:

- isolation of the radioactivity during the first two phases of the life of the facility,
- limitation and/or delay of radionuclide transfer to the biosphere,
- limitation of personal exposure.

ANDRA accordingly took the following technical safety decisions:

- insertion of a multi-barrier system which enhances waste isolation as well as radionuclide containment,
- limitation of the quantities of activity initially disposed of throughout the facility and in each waste package,
- uniform distribution of the activity disposed at the facility.

To guarantee the safety level required by the facility, it was therefore necessary to determine the radiological capacity of the facility (total activity limits which can be stored), and the concentration activity limits for each waste package. Furthermore, to guarantee the uniform distribution of the activity over the facility, it was necessary to define the concentration activity limits for the waste package of each disposal structure.

2.2. Important radionuclides for safety

Since it is neither possible nor relevant to take into account very large number of radionuclides for a safety assessment, it becomes necessary to perform a selection. The concept of importance for safety of a radionuclide is appreciated as a function of:

- the quantity of activity to be stored for this radionuclide,
- its radioactive half-life,
- its radiotoxicity expressed by dose factors (ingestion, inhalation, external exposure), and
- various parameters characterizing its transport (in concretes, soil, the biosphere, and air) in the scenarios employed.

Hence this concept is closely linked to the knowledge of the wastes to be disposed of and the disposal concept adopted.

In its technical acceptance criteria, ANDRA normally distinguishes between two types of radionuclide according to their radioactive half-life. For radionuclides with short half-lives compared with the duration of the institutional control phase, it is only necessary at Centre de l'Aube to limit the activity in each package in order to protect the workers in case of an operating accident. For the other radionuclides, it is also necessary to limit the total activity to be disposed of at Centre de l'Aube, in order to protect the public in case of accidental deterioration of the facility or intrusion in the long term.

Through the safety analyses conducted by ANDRA in its different safety reports, the list of important radionuclides for safety has lengthened with the growing knowledge, by the waste generators, of long-lived beta-gamma emitter radionuclides which are difficult to measure. Hence the radionuclides of which the total quantity and the quantity per package accepted for disposal is limited, was:

- in 1986 and then 1987, in the preliminary safety report, intended to obtain the permit for creation of the Centre: ^{3}H, ^{60}Co, ^{90}Sr, ^{137}Cs, ^{239}Pu, and ^{241}Am; then ^{14}C, ^{63}Ni, ^{94}Nb, ^{241}Pu, ^{237}Np, and ^{238}U,
- in 1991, in a provisional safety report, designed to obtain the permit for active operation of the facility, eight long-lived beta-gamma emitter radionuclides were added: ^{59}Ni, ^{129}I, ^{99}Tc, ^{93}Zr, ^{93}Mo, ^{107}Pd, ^{151}Sm, ^{135}Cs, and ^{238}U,
- in 1996, in the final safety report, designed to obtain the final operating permit, four long-lived beta-gamma emitters and three alpha emitters were added: ^{36}Cl, ^{41}Ca, ^{79}Se, ^{108m}Ag, ^{238}Pu, ^{240}Pu, and ^{224}U.

Furthermore, the radionuclides of which it is only necessary to limit the activity per package at Centre de l'Aube are: ^{22}Na, ^{54}Mn, ^{55}Fe, ^{65}Zn, ^{106}Ru, ^{110m}Ag, ^{119m}Sn, ^{125}Sb, ^{134}Cs, ^{144}Ce, ^{147}Pm, ^{152}Eu, ^{204}Tl, ^{210}Pb, and ^{227}Ac. These activities are expressed as specific activities.

3. THE SAFETY SCENARIOS AS A TOOL FOR CALCULATING ACTIVITY LIMITS

3.1. The safety scenarios

In the safety analysis of a disposal facility, the design, modelling and quantification of the scenarios help evaluate the repercussions of all events which could disturb the facility. Thus the quantification of the scenarios serves to establish a direct link between the quantity of activity stored and its radiological impact.

Besides the normal change scenario, which accounts for the migration of the radionuclides outside the waste packages, through the disposal structures, into the aquifer and up to the biosphere, for seepage conditions through the cap and normal degradation conditions of the containment barriers, ANDRA has analysed a number of deterioration scenarios. These scenarios are distinguished according to whether they correspond to radioactivity transfers by water or by air. In all cases, the activity is presumed to be uniformly distributed in the 400 structures planned.

Deterioration scenarios with the water pathway are the collapse of the cap of a structure, loss of containment performance of the second barrier (the structures) during the institutional control phase, and the use of a well respectively at the exterior and directly above the facility during the institutional control and post-institutional control phases.

Operating accidents have also been analysed, including package dropping during handling at the Centre and during positioning operations, plus a package fire. Intrusion scenarios were also analysed for the post-institutional control phase, such as a road building site and the construction of a permanent residence on the facility location with playground for children. These scenarios involve the air pathway.

3.2. Links between the safety scenarios and the representative volumes of the facility

In order to establish a link between the safety analysis of the disposal facility and acceptance criteria for the waste packages, it is important to highlight the correspondence between the different safety scenarios, and the elementary volumes of the facility to which they apply. The figure 1 illustrates this correspondence.

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Water pathway scenarios apply in most cases to all the disposal structures as a whole. This means that a quantity of activity hypothetically present at an outlet in a alterated operating scenario of a well, or in the normal change scenario in the river, will be due to the contribution of the release of all the structures of the Centre. In these conditions, these scenarios were used to derive the total acceptable activity limits in the facility, particularly for the long-lived beta-gamma emitter radionuclides.

The site intrusion and utilisation scenarios after the institutional control phase (road building site, residence, or playground) also concern all the disposal structures. These scenarios, which describe radioactivity transfers by air, were used to derive the total acceptable activity limits for radiating radionuclides and for alpha emitters (inhalation).

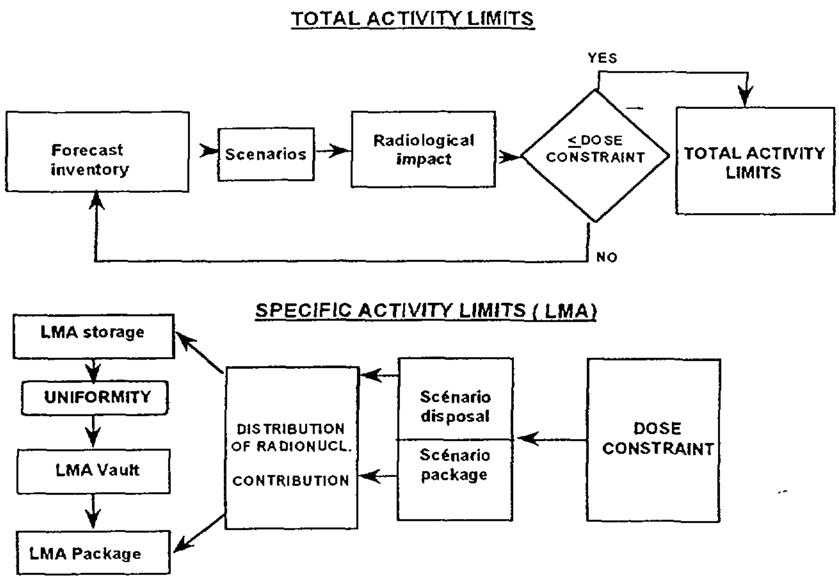
Accident scenarios in the operating phase concerning a package or a group of packages were also used to derive the concentration activity limits per package for short-lived emitters.

Note that, as mentioned in 3.1., since the activity is assumed to be uniformly distributed in all the scenarios, ANDRA has chosen an additional constraint to guarantee this hypothesis. This activity management constraint is reflected by the fact that the air pathway scenarios concerning all the facility (site) have also led to a limit on the activity concentration of each structure and each waste package, as shown in figure 1 by the word "uniformity".

4. CALCULATION OF TOTAL ACTIVITY LIMITS AND CONCENTRATION ACTIVITY LIMITS PER PACKAGE

The methodology for calculating the total activity limits for the facility and the specific activity limits per package stemmed from substantially different approaches, as shown in Figure 2 below.

Figure 2 Methodology for calculating total activity limits and specific activity limits



4.1. Methodology for calculating the total activity limits for the repository

For the total activity limits, the forecast activities are the input points of the calculation, and their acceptability is checked. The different steps are as follows (Figure 2):

- compilation of a forecast inventory with the waste producers, often using safety margins for radionuclides difficult to measure,
- application of the scenarios selected in 3.1.,
- assessment of the radiological impact associated with each scenario,
- comparison of this radiological impact with the dose constraints selected for each scenario,
- if the total radiological impact is lower than the dose constraints for all the scenarios, the forecast inventory can become the total activity limits,
- if not, the margins used in compiling the forecast inventory must be reduced, or it is impossible to receive all of the activities specified.

4.2. Methodology for deriving the specific activity limits for each package

For the concentration activity limits per package, the activities are the output points of the calculation, and the input points are the dose limitations. The problem to be solved here is one for, which there is no single solution, so that several sets of solutions are acceptable to give an acceptable radiological impact. After the selection of radionuclides important for safety and the selection of relevant scenarios, the different steps are as follows (Figure 2):

- application of the scenarios selected in 3.1.,
- setting of a dose constraints per scenario (taking into account the critical group, the probability of the scenario, and the exposure duration),
- assumptions on the radionuclide distribution in the packages and choice of the contribution allocated to each radionuclide in the consumption of the dose constraints,

- direct derivation of the concentration activity limits per package for the scenarios concerning a package (paragraphe 4.3.),
- derivation of a limit for all the packages of the site, and then for each structure and each package for scenarios concerning the entire disposal (paragraph 4.4., 4.5., and 4.6.). A limitation on the package is applied in this case to respect the assumption of uniform activity distribution mentioned in 3.1., as explained in 3.2.

Some details concerning the two last steps of this methodology are presented below.

4.3. Direct calculation of the waste package concentration activity limits (LMA) for short-lived emitters

For short-lived emitters, two accident scenarios in the operating phase were considered with the same dose constraint (workers in accidental situation) for each radionuclide, yielding two values of acceptability activity limits, and then, by assuming the mass of the packages, two values of limit concentration activities Ad and Af respectively for dropping and for fire. Then the LMA for the radionuclide concerned is the lower of the two values: LMA = min (Ad,Af).

4.4. Calculation of site (facility) concentration activity limits for medium- and long-lived β/γ radionuclides and α emitters

For medium- and long-lived emitters, three intrusion scenarios in the post-surveillance phase were considered, with relevant dose constraints (public, taken into account the probability of occurrence of the scenario and the exposure duration). For each radionuclide and for each scenario, the mass activity was calculated giving the dose factors multiplied by the contribution factor of the radionuclide.

Three concentration activities, Aro (road construction), Ar (residential area) and Ap (playground) were obtained for each radionuclide, and the facility LMA is the minimum of these three values:

facility LMA = min (Aro,Ar,Ap)

The LMA for α emitters were calculated directly on the date of the scenario, i.e. t = 350 years (300 years of decay). The LMA for the other radionuclides are calculated for t = 0.

4.5. Calculation of structure LMA for medium- and long-lived β/γ radionuclides and α emitters

To satisfy some degree of uniformity in the activity distribution in the facility, a constraint was imposed on the structures. It was assumed that the site LMA can be satisfied if the structure LMA do not exceed three times the site LMA. This assumption is a management requirement imposed on the operator to guarantee the range of validity of the scenarios applied. It is illustrated in figure 3.

4.6. Calculation of package LMA for medium- and long-lived β/γ radionuclides and α emitters

It was also assumed that the site LMA can be satisfied if the package LMA do not exceed ten times the facility LMA.

However, as recalled in figure 3, the package LMA thus calculated for the medium- and long-lived emitters must be lower than the concentration activities permitted for these radionuclides by the application of the accident scenarios in the operating phase. This was always the case in practice. The formula applied for the packages is accordingly:

$$\text{package LMA} = \min(\min(Ad, Af),\ 10 \times (\min(Aro, Ar, Ap)))$$

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5. CONCLUSION

The methodology presented have been used by ANDRA in 1991, and updated in 1998 for the derivation of waste acceptance criteria in the Centre de l'Aube. The concentration activity limits per package derived from this approach, and currently applied since September 1999 for the packages disposed of at Centre de l'Aube are, for example for some radionuclides:

- short- and medium-lived emitters:
  $^{60}Co = 1{,}3.10^5$ GBq/t, $^{137}Cs = 3.3.10^2$ GBq/t, $^{90}Sr = 6.10^3$ GBq/t
- long-lived emitters, fission products:
  $^{129}I = 1{,}4$ GBq/t, $^{99}Tc = 4{,}4.10^1$ GBq/t, $^{151}Sm = 4{,}5.10^2$ GBq/t
- long-lived emitters, activation products:
  $^{63}Ni = 3{,}2.10^3$ GBq/t, $^{94}Nb = 1.2.10^{-1}$ GBq/t, $^{14}C = 9{,}2.10^1$ GBq/t
- total alpha emitters: 3,7 GBq/t

Some evolutions can be noted in the limits between 1991 and 1999, they are due to the larger number of radionuclides to be taken into account and the updating of the safety assessment performed for the Centre de l'Aube.

An exhaustive survey has been performed (taken into account all the waste identified by waste generators), and has shown that the totality of waste packages which should be delivered are acceptable in terms of activity limits in the Centre de l'Aube. But it must be mentioned that for two new radionuclides: ^{108m}Ag and ^{126}Sn, the margin between expected activity levels and derived limits are the smallest.

The limits derived have been accepted by the French regulatory body in September 1999 in a Technical Prescription and constitute a key point of ANDRA waste acceptance criteria.

Furthermore ANDRA is participating since 1998 to IAEA consultant meeting in order to provide technical material on the work undertaken in 1997 for the development and illustration of such an approach for generic near surface repository.

The work undertaken by ANDRA now is to derive acceptance criteria for sealed sources which could constitute an hot-spot inside a waste package and for which concentration activity limits per package are not relevant. A first authorisation has been obtained from the French regulatory body for short-lived sealed sources.