RADIOACTIVE WASTE FROM NUCLEAR POWER PLANTS AND BACK END NUCLEAR FUEL CYCLE OPERATIONS: THE FRENCH APPROACH TO SAFETY

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Abstract

The Centre de l'Aube Disposal Facility (Centre de Stockage de l'Aube) is designed to receive a wide variety of waste produced by nuclear power plants, reprocessing, decommissioning, as well as the industry, hospitals and armed forces. Such a variety of wastes incur highly different risks which must be grasped in the safety analysis of the Centre. This article attempts to show how a number of safety analysis tools are used to meet the highly varied needs of the waste producers and guarantee safe disposal. They involve functional analysis, risk analysis and safety calculations. The paper show that the most important acceptance criteria for the first containment barrier, namely the waste package, are containment, durability, activity limitation and biological shielding. And a method is proposed to determine some of these criteria from safety scenarios (scenarios of accidents in operation, intrusion in the post-institutional control phase). Over the years, however, the waste producers have asked the Agence Nationale pour la gestion des Déchets Radioactifs (ANDRA) to accept new types of waste not initially anticipated in the design criteria, and the safety analysis must imagine new scenarios and develop new acceptance criteria. The paper gives the example of sealed sources, closure heads of NPP vessels, racks for fuel elements, contaminated manipulators, irradiating waste, etc, which incur specific risks. In fact, some of this waste represent a source of unusual irradiation, a risk of further contamination in an accidental situation, or simply increase the likelihood of occurrence of certain scenarios, such as retrieval in the post-institutional control phase. The safety analysis must adapt and imagine specific scenarios to judge the acceptability of such waste, and must identify the acceptance criteria commensurate with the risks. The paper offers examples of research, some of it still under way at ANDRA.

1. INTRODUCTION: A WIDE VARIETY OF WASTES

In France, low and medium level, short-lived radioactive waste is disposed of in a near-surface facility, at Centre de l'Aube. This Centre, 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 waste, and for which the administrative formalities for a permit for closure and transition to the institutional control phase are under way.

ANDRA performs the industrial management of Centre de l'Aube. Generally speaking, French law has assigned ANDRA the mission to design, build and manage the disposal centers designed to accommodate all the radioactive waste produced in France.

A wide variety of radioactive waste is destined for disposal, produced by operations connected with power generation, fuel fabrication, spent fuel reprocessing, research reactors and pilot facilities, as well as medicine, research and industries which use radioactivity.

The main types of conventional waste to be managed are hence: technological waste (safety equipment, materials for laboratory work, metal and plastic parts, water filters and ventilation circuits filters), process waste (concentrates, sludge, bitumen, ion exchange resins, filters), decommissioning waste (obsolete equipment and machines, spent glove boxes, concrete and structural materials, glass wool, graphite sleeves from gas-cooled reactors).

Centre de l'Aube was designed to accommodate this wide variety of waste, both from the standpoint of long-term safety and protection of the workers during operations.

2. TOOLS FOR THE SAFETY ANALYSIS OF CENTRE DE L'AUBE

The different analyses conducted in designing the Centre in 1986 and during the operation of the facility rely on a number of tools allowing qualitative and quantitative analysis of the safety of the Centre. These include functional analysis, risk analysis and quantitative safety assessment.

2.1. Functional analysis

Every near-surface disposal facility in France must comply with the safety objectives set by the regulatory body, the Nuclear Installation Safety Directorate (DSIN). These objectives are aimed to guarantee the immediate and future protection of the public and the environment, and to permit the limitation of the institutional control period of the facility to 300 years.

In the design phase, functional analysis enabled ANDRA to show how Centre de l'Aube could receive the waste while meeting the safety objectives assigned to it. In this method, these needs are first expressed in an overall manner by safety functions of the facility, for each use situation. This corresponds to the external functional analysis. The system analyzed is first defined accurately, its life phases distinguished, followed by identification of the external elements of the system, and culminating in the determination of the functions which formalize the interactions of the system with the external elements and which enable it to perform its mission. The functions identified for Centre de l'Aube are isolation of the radioactivity during the first two life phases, limitation and delay of transfer of the radionuclides to the biosphere, and limitation of personal exposure.

In the second step, it is necessary to carry out an internal functional analysis by distinguishing the different components of the facility, in order to find technical solutions for meeting the needs expressed in the external functional analysis.

The French concept is based on the interposition of a three-component multi-barrier system: the waste package, the disposal structure, and the host site. For the first containment barrier, the safety functions are reflected by the most important acceptance criteria:

- Radioactivity containment performance, guaranteed by a matrix or a concrete envelope,
- Durability in time in order to protect the materials containing external aggression, guaranteed by a concrete envelope,
- Limitation of the activity present in the packages, and
- Effectiveness of its biological shielding provided by a shield or by the envelope.

2.2. Risk analysis

On completion of this analysis leading to a technical choice of a disposal concept, ANDRA carried out a qualitative analysis of the risks. The approach serves to guarantee the robustness of the facility faced with external aggressions and internal failures. The aim is twofold:

- To demonstrate that the different potential aggressions of the multi-barrier system will not give rise to unacceptable consequences, thanks to the preventive and protective measures,
- To derive representative scenarios of potential changes in the system, which could be used later for the quantitative safety assessment.

The first part includes a preliminary risk analysis, carried out in several steps:

- Identification of all events liable to aggress the system design; they correspond either to natural phenomena or to events of external or internal origin. The relevance of the method is contingent on the completeness of the events identified,

- Search for the causes of each event. The knowledge of these possible causes helps determine the probabilities of occurrence of the associated scenarios, and also to select the appropriate preventive measures,
- Identification of the consequences of the different events, qualitatively or quantitatively. This step serves to select the ideal protective measures and to group the events by level of consequences. Each of the consequences can be assigned a degree of gravity,
- Selection of the appropriate preventive measures,
- Selection of appropriate protective measures, aimed to limit the consequences identified earlier.

In the second step, probabilities of occurrence can be assigned to the different events and to their consequences, considering the preventive measures selected earlier. An inventory can then be compiled of the potential scenarios, taking account of the different gravities and probabilities of occurrence. After grouping, a selection of pertinent representative scenarios can be produced.

It may be noted that while this method was used to derive the safety analysis scenarios of Centre de l'Aube, ANDRA has also very often used other methods including failure trees, event trees, cause-effect diagrams, as well as expert judgement.

2.3. Quantitative safety assessment by scenarios

The development of the scenarios, the key point of the facility safety demonstration, set the stage for the quantitative safety assessment. We shall restrict ourselves here to describing the main scenarios used by ANDRA to account for the risks associated with the disposal of the standard waste described in the introduction at Centre de l'Aube. However, a number of details concerning the disposal concept first need to be clarified.

At Centre de l'Aube, the waste is immobilized by a matrix in a concrete or steel container. The packages thus produced are placed in structures comprising, from the bottom up, a raft, shells and, after filling, a concrete closure slab. Steel containers are themselves immobilized by a grout filling the different disposal structures, while concrete containers are surrounded by gravel. The tightness of the structures is monitored via line networks in underground galleries. The disposal structures are built on a low-permeability sand, which itself overlies a watertight clay layer. When the operating phase is terminated, the facility is covered by a final cap. The site hosting the facility has a relatively simple hydrogeological system with a single, accurately identified outlet.

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

The tools described above, used to make the safety assessments of the facility, can also be used to determine the acceptance criteria for the waste packages. We shall examine this in a second part.

3. THE DERIVATION OF CERTAIN WASTE ACCEPTANCE CRITERIA FOR CENTRE DE L'AUBE

The limitation of radionuclide transfer to man and the limitation of personnel exposure (safety functions of Centre de l'Aube identified in 2.1) in all situations considered plausible (listed in 2.2) requires limiting the total activity stored at the Centre as well as the activity of each package. We shall briefly show how ANDRA has derived the activity-related acceptance criteria, based on the safety analysis.

3.1. Important radionuclides for safety

It should first be observed that 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, 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.

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

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

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 an 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 utilization scenarios after the institutional control phase (road building site or residence) 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).

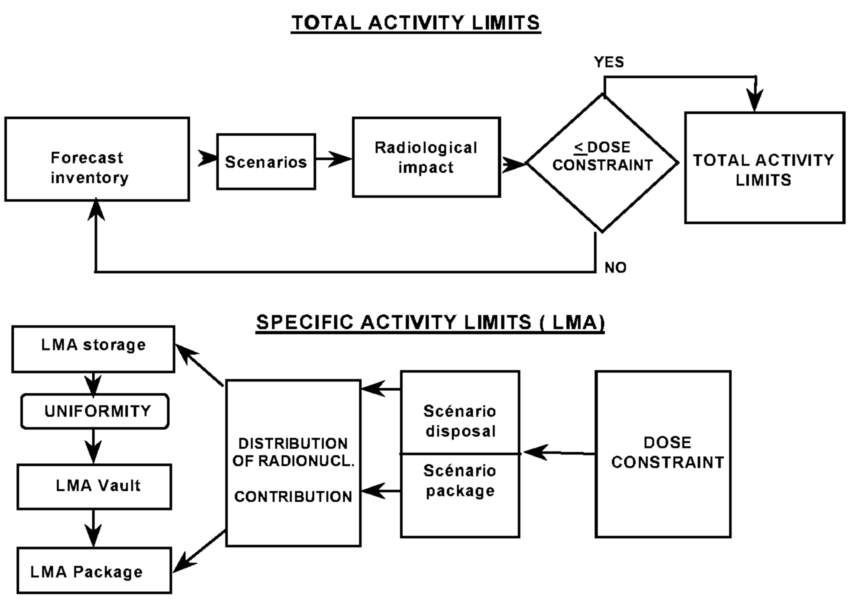
Accident scenarios in the operating phase concerning a package or a group of packages were also used to derive the activity limits per package.

It may be noted as mentioned in 2.3, that 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 structures have also led to a limit on the activity of each package.

3.3. Calculation of total activity limits and 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 1 below.

Figure 1 Methodology for calculating total activity limits and specific activity limits



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 1):

- Compilation of a forecast inventory with the waste producers, often using safety margins for radionuclides difficult to measure,
- Application of the scenarios selected in 2.3.,
- 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.

For the 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. The different steps are as follows (Figure 1):

- Setting of a dose limitation per scenario,
- Application of pertinent scenarios selected in 2.3.,
- Assumptions on the radionuclide distribution in the packages and on the contribution allocated to each in the consumption of the dose constraints. The knowledge of the content of certain packages, if any, can be used here,
- Direct derivation of the activity limits per package for the scenarios concerning a package,
- Derivation of a limit for all the packages, and then for each package for scenarios concerning the entire disposal. A limitation on the package is applied in this case to respect the assumption of uniform activity distribution mentioned in 2.3., as explained in 3.2.

The activity limits per package currently applied for the packages disposed of at Centre de l'Aube are, for example:

- Short- and medium-lived emitters:
  $^{60}Co = 5.10^4$ GBq/t, $^{137}Cs = 3.3.10^2$ GBq/t, $^{90}Sr = 9.1.10^1$ GBq/t
- Long-lived emitters, fission products:
  $^{129}I = 4.6.10^{-2}$ GBq/t, $^{99}Tc = 1$ GBq/t, $^{151}Sm = 1.6.10^3$ GBq/t
- Long-lived emitters, activation products:
  $^{63}Ni = 1.2.10^{-4}$ GBq/t, $^{94}Nb = 1.2.10^{-1}$ GBq/t, $^{14}C = 2.10^2$ GBq/t

The activity acceptance criteria listed above result directly from the safety scenarios, which are themselves derived from the risk analysis discussed in 2.2. These criteria represent a quantification of the performance required by the functional analysis described in 2.1. However, it is important to note that certain types of waste which do not fit into the basic assumption of the risk analysis cannot be accepted without an additional safety analysis. This is the subject of the next section.

4. ACCEPTANCE OF NEW TYPES OF WASTE AT CENTRE DE L'AUBE

Since the first safety analyses of Centre de l'Aube, the French radioactive waste producers asked ANDRA to accept very special types of waste not initially anticipated at Centre de l'Aube. Some of these, which will now be discussed, are distinguished from standard waste by their volume and the particular risks which they incur. We shall show how new safety analyses served to demonstrate the acceptability of these types of waste at Centre de l'Aube, starting with the largest-volume waste.

4.1. Closure vessels heads from EDF power plants

In the framework of French NPP ten-years maintenance program, technological waste consisting of large pieces of equipment is generated by *Electricité De France* (EDF). In the mid-nineties, ANDRA was asked by EDF to investigate the feasibility of direct disposal of closure vessels heads.

These lids, measuring 5.30 meters in diameter, 2.80 meters high and weighing 80 tons (for the largest from 1300 MWe reactors) are internally lined with stainless steel.

Specific safety studies have been conducted, based on the use of the tools presented in Section 2.

- A risk study served to inventory the situations liable to lead to worker exposure. These mainly include placing the lids in a hold situation, removal of the transport envelope, placement in the structure, and the grouting operations which generate the highest dose rates. The limitation of operating time, use of the containment unit, zone marking, and the rehearsal of the operations with an inactive mock-up are the main preventive and protective measures taken.
- The concept of the disposal structure and the handling tools have been adapted to the dimensions of the lids so that the latter can be placed individually in a special bin, and then embedded in concrete.
- A model adapted to the dimensions and characteristics of the lids was developed to quantify the release of radioactivity. Since it was not as easy to characterize containment performance as with small packages, the assumptions were increased to ignore certain concrete volumes in the calculations.
- The presence of large masses of undegraded and low-contaminated steel after the institutional control phase led to the consideration of new scenarios pertaining to the exposure of these lids in the long term.

4.2. Metallic racks for fuel assemblies from NPP fuel storage ponds

In 1995, EDF required ANDRA to assess the feasibility of disposing the metallic racks for fuel assemblies at the Centre de l'Aube. The racks consists of ten modules each comprising 63 compartments, was designed to receive new or irradiated fuel assemblies.

ANDRA investigated the possibilities of disposing of this waste directly in its transport casks. Each cavity is in the form of a square-section tube, and the module dimensions are: length 5 meters, width 2.6 meters, height 2 meters. The ten modules will be conditioned directly at the time of disposal, in their transport envelope corresponding to a 46 m^3 metal vessel.

A specific study demonstrated that despite the special shape of the waste, modeling of radioactivity releases in normal and accident situations via water and air for standard packages is appropriate for this specific case.

However, since the transcontainers are too large to be passed above the structural shell, as in the normal procedure, it was decided to condition them directly at their disposal location. ANDRA demonstrated that the associated risks were acceptable in view of the low irradiation levels of this waste and the arrangements adopted, and also in light of the measures taken to prevent any contamination during the grouting operations.

4.3. Manipulator robot used in hot cells

A similar study to that of the rack was aimed to demonstrate the acceptability of a manipulator robot. The manipulator used to be introduced into the cell to conduct operations associated with the decommissioning of a building. It consisted of a hood, trolley and two lids.

The conditioning mode developed included removal in a metal chest and immobilization of the waste in its envelope directly in the structure, at the time of disposal. The highest risks relate to the internal exposure of the workers during the grouting operations, because the waste contains part of its contamination which is labile. One of the measures selected for these operations is the use of filters.

4.4. Gamma irradiating packages

Some packages generated by COGEMA, at the La Hague fuel reprocessing plant contain radioactive waste with significant gamma radiation levels. As the package contact dose rate exceed ANDRA's Centre de l'Aube acceptance criteria, ANDRA was asked by COGEMA to investigate the consequences of accepting waste packages with higher dose rate.

In the past, this type of waste was conditioned in concrete shells with a lead shield. To help reduce the chemical toxics at the Centre, ANDRA examined the possibility of modifying its operating conditions in order to accommodate such packages at Centre de l'Aube.

These modifications primarily consisted of using an offset cab for the disposal operator, and controlling the distribution of irradiating packages in the structure. In the past, however, at Centre de la Manche, special structures were built with shells designed to act as biological shielding, and this practice would be technically feasible today at Centre de l'Aube.

4.5. Sealed radioactive sources

Certain studies also concern very small waste such as radioactive sources. These sources are produced by medical applications, industry, laboratories, as well as nuclear power plants.

This waste incurs risks which are not considered by the safety scenarios discussed in 2.3. Since the items are small, they can be handled, and some clearly offer an attractive option. The risk analysis of their disposal led to the consideration of new scenarios for a quantitative assessment of the impact of the disposal of such waste.

ANDRA is hence investigating new safety scenarios including the recovery of intact sources after the institutional control phase of the Centre. These scenarios are based on accidents that have already occurred with children, and will serve to determine the acceptance criteria, such as the limitation of the activity of each source and the limitations of hot spots in a package.

5. CONCLUSION: THE INDUSTRIAL CHARACTER OF CENTRE DE L'AUBE

The few examples discussed above show that Centre de l'Aube can accommodate a very wide variety of radioactive waste. Although ANDRA is investigating other near-surface and sub-surface disposal concepts for very low level waste, radium-bearing waste, graphite waste and tritiated waste, Centre de l'Aube already offers an alternative today to storage at the production centers, thanks to the robustness of its safety demonstration.

In addition to standard waste conditioned in concrete matrices and exotic waste described in the previous section, graphite sleeves, fusion ingots and bituminized waste has already been passed by ANDRA for disposal.

Following the safety exercise conducted on the occasion of the final safety report presenting the feedback from the first seven years of operation, ANDRA will be authorized by the safety authority to increase the quantities of activity to be received at Centre de l'Aube.