French National Plan

for the Management of Radioactive Materials and Waste

2013–2015

Liberté • Égalité • Fraternité
RÉPUBLIQUE FRANÇAISE

Ministère de l'Écologie, du Développement durable et de l'Énergie

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Foreword

Radioactive materials and waste must be managed sustainably, to protect individual health, security and the environment. It is also essential to limit the burden to be supported by future generations.

The National Plan for Radioactive Materials and Waste Management (PNGMDR) is a key tool in ensuring the long-term implementation of the principles laid down in the Act of 28th June 2006 concerning the sustainable management of radioactive materials and waste. It aims primarily to produce a regular inventory of radioactive substances management policy, to evaluate new requirements and to determine the objectives to be met in the future, particularly with regard to studies and research. The strength of the PNGMDR is its comprehensiveness: it concerns both ultimate waste and reusable radioactive materials, both existing management routes and those planned, under development or to be defined, both highly radioactive and less radioactive waste, even including that which is not strictly considered to be radioactive and regardless of origin. Its validity was confirmed at a European level by the adoption on 19th July 2011 of the Council's 2011/70/Euratom directive establishing a community framework for the responsible and safe management of spent fuel and radioactive waste.

For this new edition of the PNGMDR, the third since 2007, we have sought to take account of operating experience feedback as well as comments made regarding the previous version of the PNGMDR, in particular those from the Parliamentary Office for the evaluation of scientific and technological choices. The General Directorate for Energy and Climate (DGEC) and ASN have, with the successive editions, sought to continuously improve both the presentation and the content of the PNGMDR. All and any suggestions from the readers enabling the Plan to be improved and made clearer are therefore naturally welcome.

Notwithstanding the management framework implemented in France, radioactive waste still all too often triggers a disproportionate level of fear and negative reactions. In order to build confidence, transparency and high-quality information are crucial. Similarly, dialogue and consultation, especially with the representatives of civil society, must also be central considerations when drafting public policies.

In aiming to share views and proposals, the DGEC and ASN decided once again to draft the PNGMDR on the basis of presentations and discussions from a pluralistic working group, comprising environmental protection associations, regulatory authorities and regulatory assessment bodies, alongside the producers and the managers of radioactive waste. 38 meetings of this working group have therefore been held since 2003. We would like to extend our warmest thanks to all the members of this working group for their participation and to congratulate them for their contribution, the quality of which must be underlined and without whom the level of progress achieved in just a few years would not have been possible.

In order to guarantee transparency and in accordance with Article L.542-1-2 of the Environment Code, the PNGMDR will be made public and will be available for consultation on the ASN and DGEC websites. An educational and informative summary will also be published so that it is accessible to the greatest possible number of readers.

The PNGMDR proposes possible solutions for improving the management of all radioactive materials and waste. These proposals are the result of extensive work carried out since the first version of the PNGMDR covering the period 2007-2009, in particular the performance and subsequent assessment of the studies required by the Government. Though much progress has been made, it is essential that the work be continued. It should be noted that under the polluter-pays principle, all this work will continue to be directly or indirectly financed by the producers of the radioactive materials and waste.

Even though radioactive materials and waste are now safely managed under the control of the nuclear safety regulators, we cannot over-emphasise how essential we feel that it is to implement the recommendations of this PNGMDR. The aim is to maintain progress in the sustainable management of radioactive materials and waste, by defining final, long-term solutions for all these substances. It is our responsibility not to pass this burden on to future generations.

Pierre-Franck Chevet Chairman of ASN (French nuclear safety regulator)	Pierre-Marie Abadie Director for Energy (DGEC)

Introduction

When faced with the considerable diversity of radioactive materials and wastes, it may be hard to grasp the pertinence and consistency of the management framework put into place. The National Plan For Radioactive Materials And Waste Management (PNGMDR) aims to clarify and improve this management framework. To do this, it draws up an inventory of management policy, assesses requirements and determines objectives for the future.

The effectiveness of the PNGMDR has been confirmed by Parliament. The PNGMDR 2010-2012 evaluation report from the Parliamentary Office for the Evaluation of Scientific and Technical Choices[1] (OPECST) thus acknowledges the "globally positive results achieved by implementation of the nuclear waste management system and the functioning of its working group". It underlines the "constructive dialogue between the administration, industry and the associations" created within the context of the work linked to the Plan and its drafting. In addition to the "significant progress" highlighted by the Office in its assessment report, it identified a number of areas for improvement in recommendations concerning the content of the next PNGMDR and the organisation of the PNGMDR working group. Most of these recommendations are in line with the areas for work identified by the Directorate General for Energy and Climate (DGEC) and ASN as a result of analysis of operating experience feedback from the year prior to publication of the report. The steps already taken in this direction were therefore quite naturally continued. Whenever necessary, they were added to in the light of the opinion of the OPECST.

The PNGMDR gives the general public an overview of radioactive materials and waste management, covering both topical and less newsworthy subjects. The main interest of the PNGMDR is thus its clearly stated goal of comprehensiveness.

Article L.542-1-2 of the Environment Code defines the PNGMDR's objectives more precisely : it "identifies existing management modes for radioactive materials and wastes, inventories the foreseeable need for storage or disposal facilities, clarifies the capacity that will be needed in these facilities and the storage durations and, for radioactive waste for which no management mode has yet been defined, determines the objectives to be attained". This article also states that "the national plan organises research and studies into the management of radioactive materials and waste, by setting deadlines for the implementation of new management modes, the creation of facilities or the modification of existing facilities […]", and that "it comprises an appendix summarising achievements and research in foreign countries".

The structure of the 2013-2015 PNGMDR was thus revised, so that the information concerning a given technology is grouped together and so that the single document is compatible with different reading strategies. The document is thus divided into three main parts. The first recalls the principles and objectives of radioactive materials and waste management, including a presentation of the legal and institutional framework. The results and development prospects of the existing management solutions are then presented. This is followed by the needs and prospects for the management solutions to be put into place. A number of appendices are included: a presentation of the studies carried out on conservation of an archive; a summary of achievements and research carried out in foreign countries; a detailed analysis of the adequacy of storage capacity for the anticipated volumes of radioactive waste; a detailed presentation of studies and research to be carried out in the coming years concerning the management of radioactive materials and waste; a presentation of concepts and plans for the period following repository closure and the list of inter-governmental agreements signed by France with other countries, concerning spent fuel or radioactive waste management.

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[1] « Déchets nucléaires, se méfier du paradoxe de la tranquillité – évaluation du PNGMDR 2010-2012 ». (Nuclear waste, be wary of the paradox of tranquillity – 2010-2012 PNGMDR assessment)

1 Management of radioactive materials and waste: principles and objectives

1.1 Presentation of radioactive materials and waste

Of the radioactive substances, some are materials, when their subsequent use is planned or envisaged. This chiefly concerns fuels in use or spent fuels, natural, enriched, depleted or reprocessed uranium, plutonium and thorium. The radioactive materials and waste produced since the beginning of the 20th century comes primarily from five economic sectors: nuclear power generation, research, defence, industry unrelated to nuclear power generation, and medical.

Concerning radioactive waste, the usual French classification is based on two parameters: the activity level of the radioactive elements and their half-life. This classification comprises the following main categories:

- high level waste (HLW);
- intermediate level, long-lived waste (ILW-LL);
- low level, long-lived waste (LLW-LL);
- low and intermediate level, short-lived waste (LLW/ILW-SL);
- very low level waste (VLLW);
- very short half-life waste (<100 days), managed by radioactive decay and disposed of in conventional routes.

A National Inventory of these radioactive materials and waste is produced every three years by Andra. The 2012 edition identifies the existing materials and waste as at the end of 2010 and presents the forecasts for the end of 2020 and the end of 2030.

1.1.1 Definitions

The Environment Code (Art L.542-1-1) states:

- that "*a radioactive substance is a substance containing natural or artificial radionuclides, the activity or concentration of which justifies radiation protection*";
- that "*a radioactive material is defined as being a radioactive substance for which subsequent use is planned or envisaged, if necessary after processing*". In the electricity generating process, for example, the spent fuel still contains materials which can be used. These materials are reprocessed in France in order to extract the plutonium and the uranium;
- that "*radioactive wastes are radioactive substances for which no subsequent use is planned or envisaged*";
- that "*ultimate radioactive waste is radioactive waste which can no longer be reprocessed in the technical and economic conditions of the time, in particular by extracting their reusable part or by reducing their polluting or hazardous nature*".

In France, there is no single activity level or concentration per radionuclide such as to determine whether or not radiation protection is justified.

Having regard to Council directive 96/29/Euratom dated 13th May 1996 setting out basic standards for the health protection of the general public and workers against the dangers arising from ionising radiation; most countries have adopted an approach based on a clearance or exemption level, which entails removing a material from the scope of the regulations. Some countries use clearance or exemption levels expressed in terms of specific activity (Bq/g), that are either universal (regardless of the material, its origin and its destination), or dependent on the material, its origin and its destination.

In France, for nuclear activities covered by the basic nuclear installations (BNI) and defence BNI (INBS) regime, as well as for nuclear activities authorised or notified in accordance with Article L.1333-4 of the Public Health Code, mentioned in Article R.1333-12 of the same code, any waste that is contaminated, activated, or liable to be so must, as an interim measure, be the subject of specific and reinforced management, which notably includes the disposal of ultimate waste in a facility dedicated[2] to radioactive waste. The French regulations make no provision for clearance or exemption of very low level waste.

For the other nuclear activities, the justification or otherwise for radiation protection monitoring is assessed in accordance with the provisions of the Public Health Code, taking account of the three fundamental radiation protection principles: justification, optimisation and radiation dose limitation, and of the fact that the sum of the effective doses due to nuclear activities received by any member of the public, must not exceed 1 mSv per year[3]. Therefore, when the waste management radiological impact acceptability study demonstrates that radiation protection monitoring is not warranted, the waste can, in certain conditions, be accepted in conventional disposal facilities. This is particularly the case for waste containing TENORM, for which the management conditions are described in chapter 2.9 of this report.

1.1.2 Origin of radioactive materials and waste

Radioactive substances may be of natural origin or the result of human activity, although the boundary between the two is not always easy to define. For example, in the case of technologically enhanced naturally occurring radioactive materials (TENORM), certain natural materials can be used by man in such a way that the radioactivity is concentrated, even though their radioactive properties are not necessarily used.

The are many sources of naturally occurring radiation: ore and materials containing radionuclides naturally present in our environment (isotopes of uranium and of thorium, tritium, potassium 40, carbon 14, or daughter elements such as radium and radon), cosmic radiation and so on. These natural radionuclides are present in all compartments of the environment. Moreover, the concentration of radionuclides varies widely depending on the material and its origin: exposure to radionuclides of natural origin can vary by more than an order of magnitude in the various regions of the world (from an average of 2.4 mSv/year in France, to more than 250 mSv/year in some parts of India or Brazil).

Furthermore, since the beginning of the 20th century, human activities handling radioactive substances have produced radioactive materials and waste, originating in five main economic sectors:

- the **nuclear power generating sector**, primarily nuclear power plants generating electricity, plus the plants dedicated to the fabrication and reprocessing of nuclear fuel

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[2] Except for the waste managed by radioactive decay, pursuant to ASN resolution 2008-DC-0095 approved by the order of 23rd July 2008.

[3] Article R.1333-8 of the Public Health Code states that "the sum of the effective doses received by any individual not in the categories mentioned in Article R. 1333-9, as a result of nuclear activities, must not exceed 1 mSv/year."

(extraction and processing or uranium ore, chemical conversion of uranium concentrate, enrichment and fabrication of fuel, reprocessing of spent fuel and recycling);

- the **research sector,** comprising research in the civil nuclear field (mainly CEA research activities), medical research, particle physics, agronomy, chemistry, etc. laboratories;
- the **defence sector**: mainly activities related to the nuclear deterrent force, including the nuclear propulsion of certain ships or submarines, as well as the corresponding research activities;
- **industry other than for nuclear power generation,** notably the extraction of rare earths, the manufacture of sealed sources, but also various applications such as weld inspections, the sterilisation of medical equipment, the sterilisation and conservation of food products, etc.;
- **the medical sector,** comprising therapeutic, diagnostic and research activities.

Those sectors which historically contributed the most to the production of radioactive waste in France are the nuclear power generation, research and defence sectors (see part 1.1.4 concerning the national inventory of radioactive materials and wastes).

1.1.3 Usual classification of radioactive materials and waste

In accordance with the definitions given in section 1.1.1, a distinction is made between radioactive materials, for which subsequent use is planned or envisaged, and radioactive waste, for which no subsequent use is planned or envisaged.

Radioactive materials are not covered by any particular classification. This mainly concerns uranium (natural, enriched or depleted), fuels (in use or spent), uranium and plutonium separated by reprocessing of spent fuels, and reusable materials from industries other than nuclear power generation (mainly materials containing thorium).

With regard to radioactive waste, the usual French classification rests on two main parameters, when defining the appropriate management method: the activity level of the radioactive elements contained and their half-life. A particular distinction is made between waste primarily containing radionuclides with a half-life of less than 31 years (waste referred to as "short lived – SL") and waste primarily containing radionuclides with a half-life of more than 31 years (waste said to be "long-lived – LL").

This classification comprises the following main categories:

- **high-level waste (HLW),** mainly consisting of vitrified waste packages from spent fuels after reprocessing. These waste packages contain most of the radioactivity from all of the waste, whether fission products or minor actinides. The activity level of this waste is about several billion Bq per gram;
- **intermediate level, long-lived waste (ILW-LL),** also mainly from spent fuels after reprocessing and activities involved in the operation and maintenance of fuel reprocessing plants. This comprises structural waste, hulls and end-pieces making up the nuclear fuel cladding, conditioned in cement-encapsulated or compacted waste packages, along with technological waste (used tools, equipment, etc.) or waste resulting from the treatment of effluents, such as bituminised sludges; The activity level of this waste is about one million to one billion Bq per gram;
- **low level, long-lived waste (LLW-LL),** mainly graphite waste and radium-bearing waste. Graphite waste comes primarily from decommissioning of the gas-cooled reactors. The graphite from these reactors contains long-lived radionuclides such as carbon 14

(half-life 5,700 years). Its radioactivity level is several hundred thousand Bq per gram. Radium-bearing waste is mainly produced by the nuclear industry unrelated to power generation (such as the processing of ores containing rare earths) and has a level of between several tens and several thousands of Bq per gram. This LLW-LL category also comprises other types of waste, such as certain legacy bitumen packages, uranium conversion treatment residues from the Comurhex plant in Malvési, and so on;

- **low level and intermediate level, short-lived waste (LL/ILW-SL),** mainly from the operation, maintenance and decommissioning of nuclear power plants, fuel cycle facilities, research centres and, to a far lesser extent, from medical research activities. The level of this waste is between a few hundred and a million Bq per gram;
- **very low level waste (VLLW),** mainly from the operation, maintenance and decommissioning of nuclear power plants, fuel cycle facilities and research centres. The activity level of this waste is generally less than 100 Bq per gram;
- **very short lived waste**, mainly from the medical and research sectors. It is stored on the site on which it was used to allow radioactive decay, before disposal through a conventional route, corresponding to its physical, chemical and biological characteristics.

In schematic terms, this classification enables each waste category to be associated with one or more management solutions, which will be described in more detail below. They are summarised in the following table.

<table>
<tr><th></th><th>Very short lived waste containing radionuclides with a half-life of < 100 days</th><th>Short lived waste in which the radioactivity comes mainly from radionuclides with a half-life ≤ 31 years</th><th>Long-lived waste containing a significant quantity of radionuclides with a half-life > 31 years</th></tr>
<tr><td>Very low level (VLL)</td><td rowspan="3">Waste management by radioactive decay</td><td colspan="2">Recycling or dedicated surface disposal
(Industrial centre for collection, storage and disposal (Cires) disposal facility in the Aube département</td></tr>
<tr><td>Low level (LL)</td><td rowspan="2">Surface disposal
(Aube waste disposal facility)</td><td>Shallow depth disposal
(being studied pursuant to the Act of 28 June 2006)</td></tr>
<tr><td>Intermediate level (IL)</td><td rowspan="2">Deep geological disposal (being planned pursuant to the 28th June 2006 Act)</td></tr>
<tr><td>High level (HL)</td><td>Not applicable[4]</td><td></td></tr>
</table>

Radioactive materials classification principles

Two important aspects must be underlined with regard to the radioactive waste classification:

- there is no single classification criterion for determining the class of a waste. The radioactivity of the various radionuclides present in the waste must first be studied in order to classify it in a particular category. However, despite the absence of a single

[4] The category of high level, short-lived waste does not exist.

criterion, the wastes in each category are generally within a specific radioactivity range as shown above;
- waste can fall into a particular category, but may not be accepted by the corresponding management route because of other properties (for example its chemical composition).

1.1.4 Summary of the national inventory of radioactive materials and waste

In compliance with the legislative and regulatory provisions detailed in part 1.3.1, a National Inventory of radioactive materials and waste is produced, updated and published every three years by the French National Radioactive Waste Management Agency (Andra).

The 2012 edition of the National Inventory details the radioactive waste holdings stored or of disposed of as at the end of 2012, their location and their breakdown per category, economic sector and owner. This edition presents the waste production forecasts for the end of 2020, the end of 2030 as well as after the lifetime of the existing or authorised facilities, according to two intentionally contrasting energy scenarios. This Inventory also identifies radioactive material stocks and forecasts. This edition of the Inventory also contains information on *in situ* legacy waste stocks, polluted sites, radioactive waste immersed between 1967 and 1982, the management of used radioactive sources and TENORM waste.

A summary of the data contained in this National Inventory is presented below. **On the basis of annual declarations by the producers, Andra will present an update of the quantities of waste disposed of or stored at the meeting of the PNGMDR working group.**

<u>Radioactive waste</u>

The following table summarises the stocks of radioactive waste at the end of 2010 and the forecasts for the end of 2020 and 2030 for each category. The forecasts considered here are based on an estimate of the waste produced as at the dates in question by the facilities in operation or for which creation has been authorised, and scheduled to be accepted by the Andra disposal facilities. They do not take account of the waste already produced by the COMURHEX conversion plant in Malvesi, for which a long-term management solution is currently being examined, nor the "legacy" waste, such as:
- uranium ore processing residues, disposed of on certain former mining sites. The National Inventory identifies 20 sites on which these residues are permanently stored;
- waste disposed of "in situ" which in the past had been disposed of near to the nuclear facilities or plants. This usually takes the form of embankments, backfill or ponds;
- waste immersed by France in the North-East Atlantic in 1967 and 1969 and in the territorial waters of French Polynesia.

The quantities of radioactive waste are given in equivalent packaged m^3 (volume of waste once contained in a primary package). In this summary table, the figures are rounded off to the nearest hundred m^3 for the HLW waste and waste for which there is no disposal route, to the nearest thousand m^3 for ILW-LL and LLW-LL waste and to the nearest ten thousand m^3 for the other waste.

In m3 equivalent packaged	Existing waste at end 2010	Forecasts at end 2020	Forecasts at end 2030
HLW	2 700	4 000	5 300
ILW-LL	40 000	45 000	49 000
LLW-LL	87 000	89 000	133 000
LL/ILW-SL	830 000	1 000 000	1 200 000
VLLW	360 000	762 000	1 300 000
Waste with no disposal route	3 600		
Total	~1 320 000	~1 900 000	~2 700 000

Radioactive waste: stocks at end 2010 and forecasts for end 2020 and end 2030 for each category

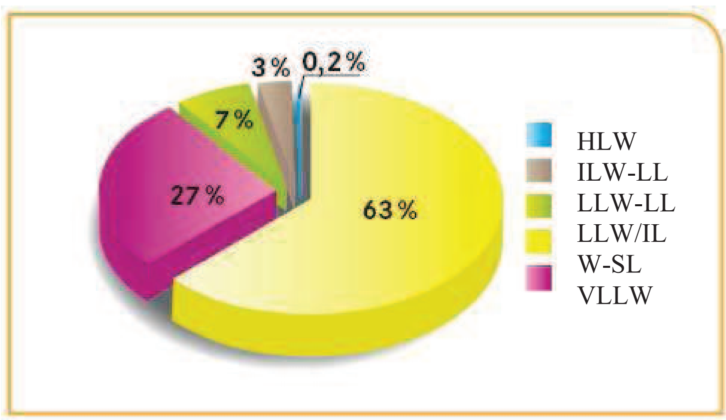
At the end of 2010, 3,600 m^3 of radioactive waste were still without a management route (existing or planned), either because it is insufficiently characterised, or because its chemical or physical form prevents it from being directly associated with an existing or planned management route. Studies are under way to identify solutions for these wastes. Moreover, the stocks as at the end of 2010 and the forecasts for end of 2020 and 2030 for the waste from the COMURHEX plant in Malvési, for which the long-term management mode remains to be defined, are as follows:

(In m^3 gross waste)	Existing waste at end 2010	Forecasts at end 2020	Forecasts at end 2030
UCTR*	**600 000**	**635 000**	**688 000**

Stocks as at end of 2010 and forecasts for end of 2020 and 2030 for waste from the COMURHEX plant in Malvési

**UCTR: Uranium conversion reprocessing residues*

The breakdown by category of the volume of radioactive waste as at the end of 2010 (excluding "legacy" waste and waste already produced by the COMURHEX plant in Malvési) is given below:



Volumes of radioactive waste by category as at end of 2010

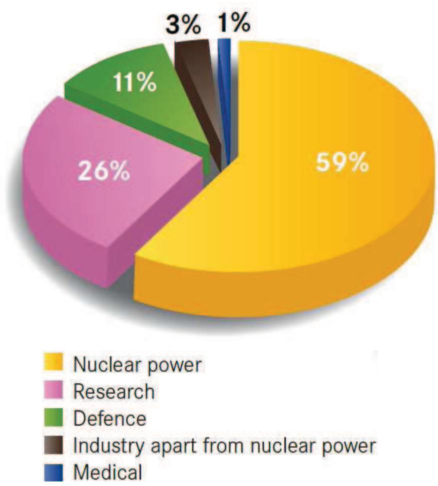
Although it only represents 0.2% of the total volume, as shown in the above figure, high level waste nonetheless accounts for 95% of the radioactivity:

Category	α (TBq)	Short-lived β/γ (TBq)	Long-lived β/γ (TBq)	Total radioactivity (TBq)
HLW	3,000,000	102,000,000	300,000	~105,000,000
ILW-LL	30,000	3,800,000	1,000,000	~4,800,000
LLW-LL	300	8,000	4,000	~12,000
LILW-SL (including T-LILW-SL)	800	19,000	7,000	~27,000
VLLW	2	2	1	5
UCTR	100*	-	-	100*

** Declared radioactivity level.*

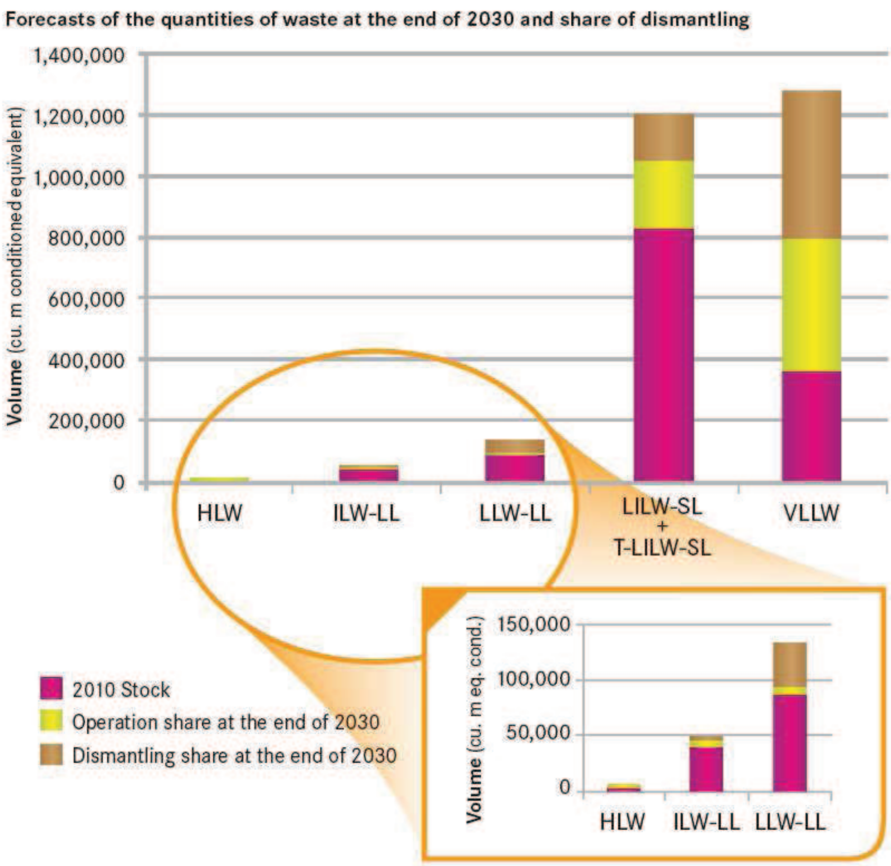
Radioactive waste: type and level of radioactivity for each category
*(UCTR: Uranium conversion reprocessing residues from COMURHEX in Malvési, * notified activity)*

Radioactive waste comes from five economic sectors: nuclear power generation, research, defence, industry other than nuclear power generation, and medical. The respective contributions of each of these sectors to the stocks of waste as at the end of 2010 (excluding "legacy" waste and waste already produced by the COMURHEX plant in Malvési) is given in the following figure. With regard to the research sector, which accounts for 26% of the volume of waste, 95% of this volume comes from the CEA research sector and 5% from other research organisations (CNRS, universities, etc.).



Breakdown per economic sector of the volume of waste as at the end of 2010

The following graph presents the forecast quantities of waste to be produced by the end of 2030 according to the category, differentiating the waste resulting from decommissioning. Most of the radioactive waste from decommissioning operations is VLL waste and, to a lesser extent, LL/ILW-SL waste. The decommissioning of the first generation of gas-cooled reactors produces LLW-LL waste.



Forecast quantities of waste produced as at end of 2030, according to category

Radioactive materials

The following table presents the quantities of radioactive materials notified as at end of 2010, plus the forecasts for the end of 2020 and end of 2030.

Material	2010	2020	2030
Natural uranium (tHM)*	15,913	25,013	28,013
Enriched uranium (tHM)	2,954	2,344	2,764
Depleted uranium (tHM)	271,481	345,275	454,275
Recycled uranium (tHM)	24,100	40,020	40,020
Fuel in use (tHM)	4,932	5,120	4,320
Spent fuel (tHM)	13,929	15,251	18,362
Plutonium	80	55	53
Thorium (t)	9,407	9,334	9,224
SS (t)	23,454	0	0

*tHM tonne of heavy metal: tonne of uranium and plutonium contained in fuel before irradiation.

Radioactive materials: quantities notified as at end of 2010 and forecasts for end of 2020 and end of 2030.
SS: Suspended solids, by-products of processing of rare earths containing thorium
**tHM: ton equivalent heavy metal*[5]

[5] tHM: ton of uranium or plutonium contained in the fuel before burn-up.

Inventory forecasts

The National Inventory also gives forecasts of the waste and materials that would be produced by all the facilities until the end of their life. These quantities are presented according to two intentionally contrasting nuclear power generating policy scenarios. This in no way anticipates which French energy policy will ultimately be adopted. The activity of the economic sectors other than nuclear power generation is assumed to be identical in both scenarios.

In both cases, the inventory only concerns the waste produced by facilities which have a creation authorisation decree by the end of 2010, even though the "continuation" scenario implies the commissioning of new facilities.

Scenario 1: continuation of nuclear power generation

This scenario envisages the continued production of electricity from nuclear power, plus the current strategy of spent fuel reprocessing. It considers an operating life of 50 years for all the reactors. All the fuel consumed by the reactors authorised as at the end of 2010 is thus assumed to be reprocessed for separation of materials (uranium, plutonium) from the ultimate waste. No spent fuel is then directly disposed of and all the plutonium extracted from the spent fuels is assumed to be recycled, in either the existing or the future reactors. Given the number of reactors today authorised to use MOX fuel (22 reactors as at end 2012), the NPPs in operation will be able to reuse the separated plutonium until 2029. Beyond that date, the rate of reprocessing of spent fuel and thus the production of plutonium will depend directly on the rate of deployment of the new reactors consuming it. 30,000 tons of spent PWR fuels will then remain to be reprocessed. Assuming a staggering of these operations over 40 years, this would represent an annual average reprocessing flow of 700 to 1000 tons of UOX and MOX fuel, and thus an annual flow of about 10 to 13 tons of plutonium.

Scenario 2: non-renewal of the NPP fleet

This scenario implies non-renewal of the existing fleet, leading to the cessation of spent fuel reprocessing before shutdown of the reactors, to avoid producing separated plutonium. The operating life of the reactors is assumed to be 40 years.

In this scenario, plutonium recycling is limited to the fabrication of the MOX fuel needed to operate the reactors today authorised to use this type of fuel. In the light of the shutdown dates for these reactors, their operation will require no further separation of plutonium by spent fuel reprocessing after 2019.

In this scenario, about 28,000 tons of spent fuel, UOx, MOX and fuel from fast-neutron type reactors, will become waste and will need to be disposed of (in the same conditions as HLW waste).

The following table gives an estimate of the waste produced by both envisaged scenarios.

Category	Ongoing electricity production using nuclear power
HLW	10,000
ILW-LL	70,000
LLW-LL	165,000
LILW-SL	1,600,000
VLLW	2,000,000

Category		Non-renewal of electricity production using nuclear power
HLW	Spent UOX Fuel	~ 50,000 assemblies*
	Spent ENR Fuel	~ 1,000 assemblies*
	Spent MOX Fuel	~ 6,000 assemblies*
	Vitrified waste	3,500
ILW-LL		59,000
LLW-LL		165,000
LILW-SL		1,500,000
VLLW		1,900,000

Estimation of waste produced in the envisaged continuation and non-renewal scenarios.
** the unpackaged volume of an assembly is about 0.2 m.^{3}*

Spent fuels are not today considered to be waste and are not therefore packaged for disposal in a repository. The average volume of a fuel assembly being about 0.2 m^3, these unpackaged assemblies represent a volume of 12,000 m^3. Andra checked the feasibility of disposing of spent fuels in 2005. The disposal container concepts used for this demonstration led to a disposal package volume of about 89,000 m^3 (about 8 times more than the unpackaged volume).

Management of reprocessed uranium

In both scenarios, the reprocessing of spent fuel also produces reprocessed uranium which is a valuable material. Assuming that the current level of recycling is maintained (in the 4 Cruas reactors until they are finally shut down), in the continuation scenario, a stock of 40,000 tons of reprocessed uranium would remain at the end of the life of the existing NPPs and 10,000 tons assuming the non-renewal scenario. In this latter case, if it cannot be reused in reactors outside France, the reprocessed uranium would become waste once the nuclear programme is shut down. However, it is technically possible to resorb this entire stock in both scenarios: it presupposes the fabrication of fuel from this reprocessed uranium which could, subject to the corresponding administrative authorisations being obtained, be consumed in all or some of the existing or future reactors, in a manner comparable to the recycling at present carried out in Cruas.

1.2 The principles to be considered when defining the management routes

The usual waste classification, taking account of the activity level of the radioactive substances and their half-life, is a simplified guide to the orientation of radioactive waste and the identification of its disposal solutions. It does not however take into account certain complicating factors which lead to a management solution being chosen which is different from the waste's particular category. Other criteria, such as stability, or the presence of toxic chemical substances and even the attractiveness of the waste (for used sealed sources in particular), must be considered.

Moreover, the definition of a management method must take account of the general principles stipulated in book V, title IV, chapter 1 of the Environment Code, more specifically the need to reduce the volume and harmfulness of the ultimate radioactive waste. Furthermore, optimisation of the waste processing operations and optimisation of the dose received by the staff and a reduction in the impact on the populations must be sought.

Radioactive waste management falls within the general framework defined in book V, part IV, chapter I of the Environment Code. In particular, radioactive waste must be managed, as far as is reasonably feasible, with a view to:

- preventing and reducing at source the production and harmfulness of the waste;
- ranking the reprocessing modes, with preference being given to preparation with a view to reuse; recycling; recovery; disposal;
- ensuring waste management without endangering human health and without harming the environment;
- limiting waste transport in large volumes and over long distances;
- ensuring public information of the effects of the waste production and management operations on the environment and on public health.

To define long-term management solutions for radioactive waste, it is important to take account of the principle of proportionality with regard to the risk and the impact and of optimisation by trading off the costs (financial, human, etc.) against the expected benefits of creating a particular management solution. This principle is hard to apply simply, particularly because it means that costs and benefits must be considered over different time-scales, sometimes a long time into the future.

1.2.1 Management modes tailored to the nature and diversity of the waste

The radioactive waste classification presented in § 1.1.3 identifies the management solution adopted or envisaged, according to the nature of the waste (activity and half-life).

This classification leads in practice to differentiating between very low level waste (VLL), low and intermediate level, short-lived waste (LL/ILW-SL), low level, long-lived waste (LLW-LL) and, finally, high level or intermediate level, long-lived waste (HL/ILW-LL). VLL waste has very low radiotoxicity and rarely presents an immediate health risk on contact. Waste with higher activity can, on contact, through inhalation or ingestion of small quantities, induce the maximum allowable annual dose for the public in a short space of time, which means that waste containment and isolation measures must be taken, along with handling precautions. Low and intermediate level, short-lived waste is therefore managed in surface repositories able to protect the waste from hazards, in particular from the circulation of water, for the duration of what is referred to as the oversight phase, conventionally set at 300 years. The periodically updated facility safety reports, including during the oversight phase, must be able to check that the activity contained in the waste reaches a residual level such that human and environmental exposure are not unacceptable, even in the event of a significant loss of the containment properties of the facility. Concerning low level, long-lived waste, the disposal concepts must ensure effective containment for a period of several tens of thousands of years, to allow decay of the majority of the radioactive substances contained in this category of waste. After this duration, the activity contained in the waste must have reached a residual level such that human and environmental exposure is not unacceptable, even in the event of a significant loss of the containment properties of the facility. For high level and intermediate level long-lived ultimate radioactive waste from the current NPPs (especially vitrified fission products and minor actinides), international experts have reached a consensus wherein deep geological disposal is the reference, safe, long-term management solution. This is also contained in the European 2011/70/Euratom directive of 19th July 2011 concerning the safe and responsible management of spent fuels and radioactive waste, which considers that *"it is commonly accepted that from the technical viewpoint, deep geological disposal is currently the safest and most sustainable solution as the final step in the management of high level waste and spent fuel considered to be waste"*. In France, the Environment Code identifies deep geological disposal as the solution for the long-term management of ultimate radioactive waste which cannot be disposed of on the surface or at shallow depth, for nuclear safety or radiation protection reasons. The Act of 28th June 2006 requires commissioning of a reversible deep geological disposal facility in 2025. The research carried out within the framework of the 30th December 1991 Act and the codified 28th June 2006 Act has enabled significant results to be obtained on the Bure site concerning the feasibility and safety of such a repository.

The classification does not take account of certain degrees of complexity. For example, nuclear industry operating waste, even if it generally contains a vast majority of short-lived radionuclides, also often contains traces of long-lived radionuclides which must be taken into account in the safety analysis of a repository. Similarly, the differentiation between VLL and higher level waste, based on the immediate radiological impact in the event of non-specific utilisation, is a simplification with regard to long-term management routes for which many other parameters have to be taken into account, such as the toxicity and chemical reactivity. Waste can also fall into a particular category, but may not be accepted by the corresponding management route because of other properties (stability, presence of certain chemical elements such as niobium used in an alloy with zirconium for the fuel rod envelopes). Consequently the waste category is not necessarily compatible with its management route.

Furthermore, identification of the appropriate management routes implies taking account of other criteria: the possible need for storage (provisional management solution adopted for waste awaiting reprocessing and ultimate radioactive waste that cannot be accepted by existing facilities or which would be unacceptable in a solution being studied); management solutions adopted in the past for certain waste categories for which the applicable regulations have significantly changed but for which recovery of the waste for management in the existing or planned repositories is not as yet envisaged. In addition, for certain waste categories, parameters particular to them must be taken into account. This is the case with the management of used sealed sources which, because of their specific nature (attractiveness, size, etc.), mean that parameters other than simply activity and lifetime must be taken into account (details of the management methods proposed are presented in chapter 3.1).

Specific management must also be defined for historical objects containing radioactivity and in the possession of private individuals or small local authorities (educational kits, objects containing radium, lightning conductors, etc.), sometimes without their knowledge. As part of its duties as set out in Article L.542-12 of the Environment Code, Andra informs the public about the management of radioactive waste and notably information enabling them to identify objects containing radioactivity. It also takes charge of this waste. A public subsidy enables it to take charge of radioactive objects free or with significant assistance in certain conditions. The Andra services manage the requests according to two criteria: the status of the person in possession and the nature of these objects. The service is provided free of charge to private individuals, to the public security services (fire brigade, gendarmerie, etc.), to rural communities and to schools. Above a certain amount, corresponding to a stock of several objects, assistance is decided on a case by case basis by the French National Funding Commission for Radioactive Matters, the CNAR[6].

As one of the main aims of the PNGMDR is to identify and implement management solutions for all the waste produced, the current classification is easy to apply to the majority of radioactive waste produced, in order to identify the solutions available for the various types of waste, even if it does not give a completely exhaustive view of all the management routes.

1.2.2 The parameters to be considered when defining the management modes

When defining a waste management mode, several criteria must be considered, the main ones being as follows.

Reducing the volume and harmfulness of the waste intended for disposal:

The storage capacity of radioactive waste disposal facilities is a rare resource, which must be preserved by sending only ultimate waste for disposal. The current waste management policy requires that the waste producers subject to the BNI regime or a nuclear activity covered by Article L.1333-4 of the Public Health Code, produce a “waste assessment” which takes account of this objective. This assessment must contain waste zoning differentiating between nuclear waste zones[7] in which contaminated waste is produced, activated or liable to be so, from conventional waste zones. The aim of this zoning is to produce less waste. This however implies that reducing the amount of materials and equipment entering the nuclear waste zone must be studied upstream thoroughly enough to allow a reduction in the quantity of waste produced.

For all activities generating radioactive waste, the management of radioactive waste must therefore begin with certain measures taken at source, on the actual site producing the waste. Sorting is the first step in reducing the volume of waste produced. This enables the waste to be separated according to its characteristics, notably the half-life of the radionuclides it contains. Sorting is followed by processing before packaging of the waste. The processing techniques vary according to the nature of the waste, which can be compacted, melted (solid radioactive waste) or incinerated. Incineration is described in more detail in chapter 2.5 of this Plan.

[6] The CNAR is chaired by the Andra Director General and comprises representatives from the supervisory ministries (DGEC, DGPR, DGS), ASN, the IRSN, the Association of Mayors of France, environmental defence associations and qualified personalities.

[7] As of 1st July 2013, these zones will be called “possible nuclear waste production zones” in the BNIs, in accordance with the provisions of the order of 7th February 2012 setting the general rules applicable to BNIs.

It would also be useful to consider the materials introduced into the waste. For example, this could lead to a reduction in the amount of organic waste (such as vinyl bags for packaging) which is accepted by the disposal facilities in very limited quantities.

Another area of progress for reducing the volume of waste to be disposed of is to reuse some of the waste produced. Initiatives have been taken or are being envisaged by certain licensees, for VLL metals and rubble. They are mentioned in chapter 2.6 of this report.

Optimisation of waste processing management

Nuclear facilities must be operated with the aim of concentrating and containing the radioactivity in solid waste, by limiting its volume and toxicity in technically and economically acceptable conditions.

The licensee must therefore take all steps to minimise facility discharges, as of the design phase. The processes implemented are therefore the subject of optimisation studies, covering all the phases in the lifetime of the facilities (design, operation and decommissioning) with the aim of mitigating their impacts on the environment.

There is a limit below which radionuclides or other chemical substances in the effluents can no longer be recovered in technically and economically acceptable conditions. This break-even point is determined on a case by case basis, through a multi-criterion optimisation process which, among other things, takes account of lifecycle analyses of products and services, safety and radiation protection and which, for constant production levels, is constantly being lowered as technology progresses. A requirement linked to the optimisation process is imposed by the current regulations.

The authorised discharges from nuclear facilities comply with limits such that the resulting exposure remains below natural exposure levels for the populations and the environment. When technically possible, discharges continue to be reduced by means of the best available techniques and their associated environmental performance, which is a regulatory requirement contained in the Environment Code.

These measures fall within the framework of the goal defined by the 1998 Sintra ministerial declaration (Ospar Convention) which states that, by 2020, environmental levels must be reached that are close to the ambient levels in the case of substances that are naturally present, and close to zero in the case of artificial radioactive substances, by means of gradual and substantial reductions in discharges, emissions, or radioactive losses, taking account of technical feasibility and the impact on man and the environment.

Furthermore, the waste assessments (periodically updated) produced by the licensees of nuclear facilities must explain the choices made between waste and discharge processing operations (decontamination of solid waste producing liquid effluents, processing of liquid effluents producing solid waste, etc.) and their containment methods, making use of the best available techniques. The primary goals of these processing operations is to reduce the quantity and harmfulness of the waste, and to implement an optimised management method. A description of liquid and gaseous effluent processing and the impact of this processing on the production of waste is presented, explaining the relationship between the quantities and qualities of the waste produced and the quantities and qualities of the liquid and gaseous effluents discharged. This study must reflect a comprehensive examination of the question of effluent and waste management.

The impact of a BNI is considered by the BNI regime, which covers the risks of accident, chronic releases, waste production and other detrimental effects. The same applies for a facility subject to the regime of installations classified on environmental protection grounds (ICPE). Within the regulatory frameworks, discharges are thus covered by requirements, set by ASN resolutions for BNIs (the resolution setting the discharge limits is approved by ministerial order) or by ministerial order for INBSs or by order of the Prefect for ICPEs; these requirements strictly limit the discharges for a certain number of substances and regulate in detail the means of processing, purifying and monitoring of effluents resulting from nuclear activities and industrial activities. An impact assessment in particular is systematically required for any activity leading to discharges into the environment. This analysis should guarantee that discharges from nuclear facilities do not compromise the interests protected by Article L.593-1 of the Environment Code. In order to mitigate the environmental impact, the licensees are systematically required to propose and implement techniques to reduce discharges linked to these activities to the extent made possible by the available techniques and in economically acceptable conditions.

At the national or regional level, some topics are also the subject of specialised planning, for example through water development and management plans for water intake and liquid effluent discharges, or through the PNGMDR for radioactive waste. In this respect, the present plan concerns waste and the management of radioactive materials, but does not cover effluent discharges.

In accordance with the OPECST's recommendations, this Plan mentions any major discrepancies. The following boxes present the position expressed by the ACRO (Association for the Control of Radioactivity in the West) which is that adopted by FNE (France Nature Environnement), the ANCCLI (National Association of Local Information Committees and Commissions), Robin des Bois and Greenpeace, as well as the initial response from the State services.

Discharges and waste – position expressed by ACRO, supported by FNE, ANCCLI, Robin des Bois and Greenpeace

During the discussions held for drafting of the 2013-2015 PNGMDR, the ACRO constantly emphasised the need to include effluent discharges in this document.
The regulatory frameworks associated with waste and with discharges are clearly distinct but these two aspects are inseparable, because radioactive discharges can frequently be the result of the decision to deal with radioactive waste by means of a "clearance" process. In other words, these two problems are inextricably intertwined.
There are in fact technical solutions for retention of virtually all radionuclides (beginning with iodine 129, carbon 14, but also tritium, krypton 85 and so on) but, for technical and economic reasons, these techniques are not implemented. The most striking example is without doubt that of iodine 129, which is almost entirely discharged at sea, even though retention systems have been developed by France and were used in Japan on certain installations as of the 1970s. Similarly, the ACRO considers that all carbon 14 is discharged as an effluent in our country, but that it is trapped by chemical precipitation in other countries and thus managed as waste.
Although the addition of artificial radionuclides to consumer goods and foodstuffs is prohibited, it is authorised in the environment. This situation is paradoxical to say the least. Therefore, if industry were tomorrow to change its radioactive effluents management strategy, either proactively or if required to do so by the public authorities, some of it would then be reclassified as waste.
For the ACRO, this discharge aspect can no longer be swept under the carpet as it implies radioactive waste being declassified by means of what is referred to as "clearance" in the international texts.
In order to provide the reader with full information, but also to adopt a comprehensive approach to the waste problem, our association would have liked to see a sub-chapter on effluent discharges - mentioning the discharge levels – included in the 2013-2015 PNGMDR.

Discharges and waste – position expressed by ASN and the DGEC

ASN and the Ministry responsible for ecology and energy underline that there are links between the production of waste and discharges and that the main goal of the legal systems governing nuclear activities is to enable the impact of these activities to be considered in an integrated manner.
Basic nuclear installations and ICPEs are only authorised after a public inquiry, more specifically based on an impact assessment which includes discharges and waste. The discharges themselves are governed by requirements specified by the Ministers or the Prefect, depending on the facility, after review by the Departmental Council for the Environment and for Health and Technological Risks (CODERST) and, for a BIN and INBS, by the local information committee.
Every year, ASN's annual report on the state of nuclear safety and radiation protection in France presents data on the production of waste and effluent discharges. Moreover, for each BNI, the discharge levels must notably be included in the annual public information reports produced by the licensee, in compliance with Article L.125-15 of the Environment Code, made public and presented to the local information committees; these reports are available on the websites of the BNI licensees. The information about discharges can also be sent out to anyone requesting it, pursuant to Article L. 125-10 of the Environment Code. For INBS facilities, an annual report on the nuclear safety of the site, the risks of radiological origin and the discharges produced by the facility, along with the steps taken to mitigate the impacts, is presented to the local information committees in compliance with the provisions of Article R.1333-39 of the Defence Code, in accordance with national defence requirements.
In addition to the integrated systems for monitoring the impact of facilities and activities, there are also specialised planning arrangements, for example regarding waste management via the PNGMDR. In order to be easily readable, the documents giving the results of these planning arrangements must not exceed the scope defined for them; thus, the presentation of discharge management and levels is not covered by the provisions of Article L.542-1-2 of the Environment Code, which determines the content of the PNGMDR and would thus be liable to make the Plan less comprehensible. Moreover, directive 2011/70 of 19th July 2011 establishing a community framework for the safe and responsible management of spent fuel and radioactive waste also excludes the authorised discharges from its scope of application (article 2 of the directive).

Waste characterisation

A clear understanding of the radiological and physical-chemical content of the waste package[8] helps optimise its packaging and its management route. Improving precision when determining the natures and quantities of the radionuclides is a permanent goal and progress in this field follows advances in sensor technology and discrimination techniques. They should lead to faster analyses which generate less secondary waste.

Waste containment

The purpose of manufacturing a waste package is generally to contain the waste in a stable, solid monolithic[9] form. Processing prior to packaging is also sometimes necessary to ensure compatibility, especially physical-chemical compatibility, between the waste and the matrix, or

[8] At packaging and if necessary after processing, the radioactive waste is placed in a container, in which it may or may not be immobilised to form a waste package.
[9] If the package does not on its own guarantee sufficient intrinsic safety, it is placed in a "monolithic" structure, in which the voids are filled with concrete.

immobilisation system, chosen for the composition of the package. Glass and cement are the main matrices used industrially and have been employed for many years now. The studies to be carried out in the coming years will aim to improve the industrial performance of these processes, either by boosting their production capacity, or by expanding their scope of application to include new wastes, or to develop new matrices with the aim of optimising the containment properties of certain packages.

In the coming years, the legacy waste currently in storage may need to be processed prior to recovery and packaging. This is for example the case of the waste consisting of fuel cladding from the gas-cooled reactors on the La Hague and Marcoule sites.

Optimisation of a particular technology also means following a safety approach so that the repository can perform its containment function until the radioactivity of the radionuclides contained in the waste has sufficiently diminished. The radiological impact of the management solution adopted must be as low as possible.

<u>Optimisation of doses received by the staff and the general public</u>

At an international level, the leading principles of protection against ionising radiation have been defined by the International Commission on Radiological Protection (ICRP) and concern:

- the justification of activities (technical, economical and ethical);
- the mitigation of consequences (doses);
- the optimisation of protection (doses).

The effectiveness of the optimisation approach (optimising the level of radiation protection) is based on the widespread dissemination of the radiological risk culture. All of these principles are of course also applicable to the field of radioactive waste management.

<u>Anticipating processing and storage facility requirements</u>

The time aspect is crucial when setting up long-term management solutions, in order to strike the right balance between the need for storage, processing/packaging facilities, characterisation and transport, and the means available at any given time. Considering this time factor is also a means of anticipating the risks linked to defaulting by the waste producer. The volume and availability of the long-term management solutions in relation to the overall requirements, must be monitored and anticipated in order to avoid waste production creating needs that exceed the available and authorised capacity.

It would in this respect appear to be essential for the waste producers to define a long and medium-term management strategy for their waste, so that either individually, or together with other producers, they can define their present and future needs at each step in the waste management process and acquire the necessary means for optimal management (storage facilities, transport containers, characterisation resources, etc.). These strategies are regularly examined by the safety regulators, with the support of IRSN.

With regard to future waste, such as waste from operation of the ITER facility or waste that would be produced by a fleet of generation IV reactors, its management must be examined in the light of existing routes in order to demonstrate their compatibility or define the necessary changes to the routes or even the creation of new routes (compatibility with disposal specifications, consequences on the repository footprint, etc.).

1.3 The legal and institutional framework of waste management

At the European level, the Council directive of 19th July 2011 established a community framework for the safe and responsible management of spent fuel and radioactive waste. This directive is to be transposed before 23rd August 2013.

The national framework for the management of radioactive materials and waste is defined by the Programme Act 2006-739 of 28th June 2006, concerning the sustainable management of radioactive materials and waste and dealing with the definition of a management policy for radioactive materials and waste, for improving transparency and democratic oversight and for financing and economic support. The Act specifies that the management of radioactive materials and waste must comply with the following fundamental principles: protection of individual health and the environment; reduction in the quantity and harmfulness of radioactive waste; prevention or mitigation of the burden borne by future generations; the polluter-pays principle, which applies in environmental law.

The PNGMDR organises research and studies into the management of materials and waste, in accordance with the three directions defined by the Act:
- reducing the quantity and harmfulness of the waste, in particular by reprocessing spent fuels and processing and packaging radioactive waste;
- storage as a preliminary step, in particular with a view to fuel and waste reprocessing, or to disposal of the waste;
- after storage, deep geological disposal as a permanent solution for ultimate waste that cannot be disposed of on the surface or at shallow depth, for nuclear safety or radiation protection reasons.

In the field of transparency and democratic oversight, the Act in particular gives the National Review Board the task of assessing research into the management of radioactive materials and waste. It also makes provision for regular information and consultation by the French High Committee for Transparency and Information on Nuclear Security (HCTISN).

1.3.1 The European legislative framework

On 19th July 2011, the Council of the European Union adopted directive 19/2011/Euratom establishing a community framework for the responsible and safe management of spent fuel and radioactive waste

This directive defines a binding legislative framework and in particular requires that each member State set up a regulatory authority with competence for the safe management of radioactive waste and spent fuel, given the financial and human resources necessary for the performance of its duties. It sets safety requirements and requires the creation of a system of authorisations for waste and spent fuel management facilities. It also requires those holding these authorisations to devote adequate financial and human resources to waste management.

Moreover, this directive requires the definition of a national programme to implement the waste and spent fuel management policy. This programme, which is based on a national inventory, must concern all waste, from production up to long-term management and must be periodically revised and notified to the European Commission.

The directive also defines measures concerning public transparency and requires that a system of penalties be set up, in particular by means of suspensions. It formally stipulates the ultimate responsibility of each member State for the management of its radioactive waste and specifies the possibilities regarding the export of this waste for disposal.

Finally, the directive requires regular self-assessment of the national framework, of the competent regulatory authorities and of the national programme and its implementation, supplemented by an international peer review.

This directive is a key factor in helping to strengthen nuclear safety within the European Union, while making the member States more accountable for the management of their radioactive waste and spent fuels.

It must be transposed by each member State before 23rd August 2013 and the national programme concerning the implementation of a waste and spent fuel management policy must be transmitted to the European Commission by 23rd August 2015.

1.3.2 The legislative and regulatory framework in France

<u>Background to and context of the drafting of the 28th June 2006 Act</u>

For many years now, significant steps have been taken to ensure appropriate, long-term management of radioactive waste: 90% of the volume of waste produced[10] is now accepted in surface repositories managed by Andra. These repositories are located at Digulleville in the Manche *département* and at Soulaines-Dhuys and Morvilliers in the Aube *département*. The Manche repository has been covered and entered the oversight phase in 2003, while the Aube repositories are still in operation.

The remaining 10%, which account for 99% of the radioactivity, are stored in surface facilities, in particular at La Hague (Manche), Marcoule (Gard) and Cadarache (Bouches-du-Rhône). These facilities were designed for interim storage of the waste, until such time as a final solution becomes available.

To define the long-term management solutions, France is involved in ambitious study and research programmes, in the same way as the other countries concerned, such as the United States, Finland, Sweden, or Germany. On 30th December 1991, the French Parliament more specifically voted an Act defining three areas for research[11] and stipulated that the Government would present a further Bill before 30th December 2006 based on the results of the research carried out in each of these three areas:

1. the first area aims to reduce the volume and toxicity of the waste by separating out the various products contained in the spent fuels (process known as "separation") and by transforming certain long-lived radioactive elements into radioactive elements with a shorter half-life in new nuclear reactors (process known as "transmutation"). This option implies the development of a new generation of fuel fabrication and reprocessing plants and a new generation of nuclear reactors;
2. the second area was irreversible or reversible disposal of waste in a deep geological repository. Radioactive waste repositories already exist, but they are on the surface and dedicated to low and intermediate level, short lived waste, or very low level waste. The potential of deep geological disposal of long-lived radioactive waste was in particular studied by means of Andra's underground laboratory at the border between

[10] Excluding waste which was the subject of legacy management methods.
[11] Act 91-1381 of 30th December 1991 concerning research on radioactive waste management.

the Meuse and Haute-Marne *départements*, in a stable geological layer about 150 million years old, situated at a depth of about 500 m;

3. the third area concerned the study of long-term waste packaging and storage processes. It aimed to develop facilities which would enable waste to be kept safely on the surface for 100 to 300 years, as against 50 to 100 years for the storage facilities currently in service. As storage is by definition temporary, the waste would eventually have to be removed.

This research was carried out under the auspices of the French Alternative Energies and Atomic Energy Commission (CEA) (for areas 1 and 3) and Andra (for area 2) and led to extensive scientific collaboration, both national (with CNRS and universities) and international. This research was finalised in 2005 and summary reports were drafted and submitted to the Ministers responsible for industry and research.

A number of evaluation and consultation initiatives were launched in 2005 and 2006 on the basis of this research, including:

- the report published in March 2005 by the Parliamentary Office for the Evaluation of Scientific and Technical Choices (OPECST) under the auspices of the Members of Parliament Christian Bataille and Claude Birraux and entitled "For sustainable management of radioactive materials and waste";
- evaluation of the research results carried out by the National Review Board (CNE), comprising independent experts, by ASN and by panels of foreign experts selected by the OECD’s Nuclear Energy Agency (NEA);
- the public debate run from September 2005 to January 2006 by the National Public Debates Commission (CNDP), during which the Commission for the first time held a debate not on a tangible infrastructure project, but on a question of general environmental policy.

These various elements, supplemented by the opinions submitted by the Conseil d’Etat[12] and the Economic and Social Council, enabled the Government to draft a bill in 2006 concerning the management of radioactive materials and waste. The public debate, followed by the Parliamentary review, led to major changes, in particular concerning the inclusion of radioactive materials (rather than just waste), the objectives for the new research phase, the role of Parliament after 2006, the notion of reversibility, the local consultation procedures, the economic support system for the regions concerned and ring-fencing of the financial resources needed for management of radioactive waste and decommissioning of the nuclear facilities. Act 2006-739 of 28th June 2006 on the sustainable management of radioactive materials and waste was published in the Official Journal on 29th June 2006.

<u>The 28th June 2006 Programme Act on the sustainable management of radioactive materials and waste</u>

Under the terms of the 28th June 2006 Act (primarily codified in the Environment Code), the sustainable management of radioactive materials and waste must comply with the following principles: protection of human health, safety and the environment; prevention or limitation of the burden to be borne by future generations, polluter-pays principle.

[12] Council of State – France’s highest administrative court

The Act tackles three main subjects:

1. definition of a radioactive materials and waste management policy;
2. improved transparency and democratic oversight;
3. financing arrangements and economic support.

First of all with regard to the management policy, Article L.542-1-2-II of the Environment Code sets the management guidelines for all radioactive materials and waste, that is:

- reducing the quantity and harmfulness of the radioactive waste, in particular by reprocessing spent fuels and processing and packaging radioactive waste;
- storage in specially designed facilities of radioactive materials pending reprocessing and of ultimate radioactive waste pending disposal;
- after storage, deep geological disposal as a permanent solution for ultimate radioactive waste that cannot be disposed of the surface or at shallow depth, for nuclear safety or radiation protection reasons.

Article L.542-1-2 of the Environment Code also makes provision for the drafting of this document, the National Plan For Radioactive Materials And Waste Management, every three years. This Plan, presented in greater detail in part 1.3.3, aims to:

- inventory the existing radioactive materials and waste management methods;
- identify the foreseeable needs for storage or disposal facilities and clarify the capacity needed, as well as the storage durations;
- determine the goals to be attained for radioactive waste for which there is not as yet a definitive management solution; the plan specifically organises the research and studies to be conducted on the management of these wastes and sets deadlines for implementation of new management solutions and for the creation or modification of facilities.

The Act also defines a programme of research into the management of all radioactive materials and wastes (Articles 3 and 4 of the 28th June 2006 Act). For high level and intermediate level, long-lived waste, three areas are stipulated, consistently with those of the 1991 Act.

1. For the first area, concerning the separation and transmutation of long-lived radioactive elements, an assessment of the industrial prospects for the various transmutation techniques will be drawn up by the end of 2012. Depending on the results obtained, a prototype installation could be built by 2020;
2. For the second area, concerning the possibilities for waste disposal in a deep geological repository, the 2006 Act asks Andra to continue its studies and research pursuant to the 1991 Act and now requires that the concept developed by Andra be reversible. With regard to the milestones set by the act, the requirement is that the disposal authorisation application be available for review in 2015 following a public debate and that, provided the prior review is favourable, commissioning take place in 2025;
3. The third area was revised by the 2006 Act, which preferred complementarity between storage and disposal, instead of the notion of "long duration" for storage. The law now requires that Andra carry out studies and research to create new storage facilities or modify the existing ones, in order to meet the needs identified by the Plan, particularly in terms of capacity and duration, no later than 2015.

The ban on the disposal in France of radioactive waste from abroad, introduced by the 1991 Act, is restated and clarified in Article L.542-2 of the Environment Code. In particular, spent fuels or radioactive waste may only be brought into the country for processing, research or transfer between foreign States. Furthermore, entry for processing purposes must be regulated by an inter-governmental agreement, which must specify a date beyond which the waste resulting from the processed substances can no longer be stored in France. However, this ban does not concern the return and disposal in France of radioactive waste or spent fuels resulting from radioactive substances or equipment shipped abroad from France for processing or research, provided that they did not first originate from abroad. Nor does this concern the return to France of sealed radioactive sources or equipment containing them, when these sources or equipment were supplied by a French company.

For waste from small producers outside the power generating sector, Article L.542-12 of the Environment Code tasks Andra with collecting, transporting and dealing with radioactive waste and rehabilitating radioactive pollution sites at the request of and expense of those responsible, or when requisitioned by the Government when those responsible for this waste or these sites have defaulted.

In the field of transparency and democratic oversight, the National Review Board (CNE), created by the 1991 Act, is responsible for performing an annual assessment of the progress of the research and studies concerning the management of radioactive materials and waste. The 2006 Act revised its operating methods and composition, specified in Article L.542-3 of the Environment Code.

A local information and monitoring committee (CLIS) is also created for Andra's Meuse / Haute-Marne underground laboratory. Following the 2006 Act, its operation was modified, its composition extended and its presidency given to a national or local elected official, as required by Article L.542-13 of the Environment Code.

The law also stipulates new milestones to authorise the creation and closure phases of the future deep geological disposal facility. A first Act must therefore set the reversibility conditions, after a debate and consultation of the local authorities concerned, along with the CNE and ASN, before a decree can authorise creation of the disposal facility. In the longer term, final closure of the disposal facility can only be granted by means of an Act (Article L542-10-1 of the Environment Code).

The High Committee for Transparency and Information on Nuclear Security, created by the 13th June 2006 Act on transparency and security in the nuclear field, is also tasked with periodically organising consultations and debates concerning the sustainable management of radioactive materials and waste.

Finally, the Act makes provision for financing, as well as for modernisation of the local support system (Article L. 542-11 of the Environment Code and Article 43 of the 2000 Budget Act, supplemented by the provisions of Article 21 of the 28th June 2006 Act) for the underground laboratory and the future deep geological disposal facility.

The methods for financing the three areas of research are specified by law (Articles L.542-12-1 and L.542-12-2 of the Environment Code). The research carried out by Andra in areas 2 and 3 are in particular financed by an additional tax on top of the BNI tax.

In order to secure the financing of long-term nuclear costs (Article 20 of the 28th June 2006 Act), the law requires that each licensee of a BNI or defence BNI, with the exception of the State, make a prudent assessment of its future expenses, constitute provisions accordingly, and allocate these ring-fenced assets to coverage of these costs before mid-2016. Pursuant to Act 2010-1488 of 7th December 2010, a five-year postponement (as of 30th June 2011) may however be granted, in certain conditions, for implementation of the plan to create the defined assets. These expenses include the cost of managing spent fuels and radioactive waste, as well as the cost of decommissioning BNIs (or, for radioactive waste disposal facilities, the expenses involved in final shutdown, maintenance and oversight/monitoring) and defence BNIs. A system regulating licensee practices in this field is created, more specifically implementing direct State oversight of the methods for assessing and covering these costs. Three-yearly reports, plus annual updates must be submitted for review by the administrative authority (see § 1.5). The law also creates a national financing review board (CNEF) for the long-term nuclear costs in order to evaluate the controls put into place by the State. The CNEF presented its first report to Parliament on 24th July 2012.

Finally, in order to support the areas situated close to an underground research laboratory or deep geological disposal facility, the public interest groupings (which already existed in the Meuse and Haute-Marne *départements*) now have three duties: (i) the management of equipment such as to encourage the siting of a research laboratory or disposal facility; (ii) within the boundaries of the *département,* the implementation of land use planning and economic development measures, particularly in a "proximity area" around the facility; (iii) support for training actions and for actions to develop, promote and disseminate scientific and technical knowledge, more specifically in the fields studied in the laboratory and the new energy technologies. These duties are financed by additional taxes to the existing BNI tax.

<u>Implementation of the 28th June 2006 Act</u>

All of the decrees that were to be issued since the publication of the 28th June 2006 Act have been or are in the process of being published, as mentioned in the following table.

	Subject of the decree	Article	Date
National radioactive materials and waste management policy	Definition of the provisions of the National Plan For Radioactive Materials And Waste Management	Art. L 542-1-2 of the CE	23 April 2012
	Management of foreign waste and processing contracts	Art. L. 542-2 à L. 542-2-1 of the CE	3 March 2008
	Appointment of members to the CNE	Art. L 542-3 of the CE	5 April 2007 20 July 2010
	Nature of information to be submitted for the national inventory and the PNGMDR	Art. L 542-13-1 of the CE	29 August 2008
Support for research conducted in the Meuse / Haute-Marne underground laboratory	CLIS	Art. L 542-13 of the CE	7 May 2007
	GIPs – Generic decree	Art. L 542-11 of the CE	14 December 2006
	Definition of proximity area - GIP Meuse and Haute-Marne	Art. L 542-11 of the CE	5 February 2007
	"Support" tax: fraction paid by the GIPs to the communes within the 10 km zone	Art. 43 2000 Budget Act	7 May 2007

	Coefficient of “support” and “technological dissemination” taxes	Art. 43 2000 Budget Act	26 April 2000 amended
	Consultation area when creating a repository	Art. L 542-10-1 of the CE	Publication to be scheduled in line with the calendar for the deep geological repository project
Financing arrangements	Coefficient of additional “research” tax	Art. 43 2000 Budget Act	26 April 2000 amended
	Ring-fencing of assets to cover long-term nuclear expenses	Art. 20 of 28th June 2006 Act	23 February 2007 amended
	Creation of the CNEF	Art. 20 of 28th June 2006 Act	20 June 2008

Decrees published since publication of the 28th June 2006 Act.

The management policy in particular was clarified by the decrees of 16th April 2008 and then of 23rd April 2012 setting the requirements concerning the National Plan For Radioactive Materials And Waste Management.

The producers of radioactive waste, or those in possession of it, are responsible for its management or for ensuring that it is managed in accordance with the guidelines stipulated in Article L.542-1-2 of the Environment Code. For this purpose:

- attempts must be made to ensure the consistency of the radioactive waste management system and to optimise it both technically and economically;
- the radioactive waste disposal facilities are few in number and of limited capacity and must therefore be optimised by the various players;
- the radioactive waste management sectors take account of the volumes of waste transported and the distances to be covered between the storage and disposal sites.

In this respect, one must remember the principle of the responsibility of the producer of the waste, which must finance its handling by an authorised route.

<u>The other texts</u>

The management of radioactive materials and waste is also governed by other laws and international agreements.

Planning Act 2005-781 of 13th July 2005 setting energy policy guidelines thus defines several priorities, including that of keeping the nuclear option open until the 2020 time-frame. It also specifies that research in the energy sector must be developed, more specifically for the future nuclear reactor technologies (fission or fusion), in particular with the support of the ITER programme, as well as technologies necessary for the sustainable management of radioactive waste.

Act 2006-686 of 13th June 2006 concerning transparency and security in the nuclear field (known as the TSN Act, codified) applies to the management of radioactive materials and waste, in particular certain provisions concerning BNIs (category of most of the radioactive waste disposal facilities), or the transport of radioactive substances.

The management of radioactive waste from BNIs is based on a strict regulatory framework, stipulated by the order of 31st December 1999, setting the general technical regulations intended to prevent and mitigate off-site detrimental effects and risks resulting from the operation of BNIs; This order recalls the need for the licensee to take all necessary steps in the design and operation of its facilities to ensure optimum management of the waste produced, in particular taking account of the subsequent management routes. It requires the drafting of a study specifying the management methods for the waste produced in the BNIs. One part of this study is submitted to ASN for approval. Within the framework of the overhaul of the BNI regulatory system following the "TSN" Act, this order was revised and the requirements concerning waste management in BNIs were grouped within the order of 7th February 2012. An ASN resolution will supplement the provisions concerning the management methods for waste produced in BNIs.

The management of radioactive waste from defence BNIs (INBS) is regulated by the order of 26th September 2007, setting the general technical regulations intended to prevent and mitigate off-site detrimental effects and risks resulting from the operation of defence BNIs; Part VI of this order recalls the need for the licensee to take all necessary steps to reduce the volume, radiological, chemical and biological toxicity of the waste produced in its facilities and to optimise its management, with preference being given to reuse and processing over final disposal, which is reserved for ultimate waste. It requires the drafting of a summary document specifying the management methods for the waste produced in the defence BNIs. This document is submitted to the Defence Nuclear Safety Authority (ASND) for approval and acts as the baseline for optimised management of the waste produced in the INBS.

With regard to the radioactive waste produced outside BNIs and outside defence BNIs, Article R.1333-12 of the Public Health Code requires that the management of effluents and waste actually or potentially contaminated by radioactive substances as a result of a nuclear activity, of whatsoever nature, comprising a risk of exposure to ionising radiation, must be reviewed and approved by the public authorities. The ASN resolution dated 29th January 2008, approved by the Ministers responsible for the environment and for health, issued pursuant to the provisions of Article R.1333-12 of the Public Health Code, sets out the technical rules applicable to the disposal of effluents and waste contaminated by radionuclides, or liable to be so contaminated as a result of a nuclear activity.

With regard to used sealed sources, the principle of prevention and reduction of waste requires that as a priority they be considered as radioactive materials (and therefore potentially reusable). They are managed in compliance with the provisions of the Public Health Code: return to the supplier, which must itself organise their management and can have then retrieved by its own supplier, have them recycled by a manufacturer, store them pending a decision or decide to manage them as radioactive waste.

At an international level, the radioactive waste issue is covered by safety and radiation protection safety baselines (in particular those drafted by IAEA) and by various working groups, within the NEA (OECD's Nuclear Energy Agency) or the working groups of the European Nuclear Energy Forum and the European Nuclear Safety Regulators Group (ENSREG). Furthermore, an international convention (Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, approved by France on 22nd February 2000, entry into force on 18th June 2001), to which 65 States are contracting parties, promotes the relevant principles and organises peer reviews every three years. Bilateral agreements can also be signed, in compliance with Article L. 542-2-1 of the Environment Code, to regulate the import for processing of radioactive waste or spent fuels, in order to comply with the ban on the disposal in France of radioactive waste from abroad.

1.3.3 The National Plan For Radioactive Materials And Waste Management

As mentioned in section 1.3.2, Article L 542-1-2 of the Environment Code defines the objectives of the National Plan For Radioactive Materials And Waste Management:

- inventory the existing radioactive materials and waste management methods;
- identify the foreseeable needs for storage or disposal facilities and clarify the capacity needed, as well as the storage durations;
- determine the objectives to be met for radioactive waste for which there is as yet no final management solution. The Plan specifically organises the research and studies to be conducted on the management of these radioactive wastes and sets deadlines for implementation of new management methods and for the creation or modification of facilities.

The Environment Code requires that the PNGMDR be published every three years and that a decree set the resulting regulatory requirements (Article L.542-1-2). An appendix to the Plan must comprise a summary of achievements and research conducted abroad. It is transmitted to Parliament, which refers the assessment to the Parliamentary Office for the Evaluation of Scientific and Technical Choices (OPECST), and is made public.

The first version of the PNGMDR was transmitted to Parliament in 2007, on the basis of the work done by a pluralistic working group. It was prepared to a large extent jointly with the Act of 28th June 2006. It was then adjusted and updated to take account of the provisions of the Act. A summary of this Plan was then published.

For the 2010-2012 and 2013-2015 editions of the PNGMDR, the Government chose to continue to rely on the work of a pluralistic working group. This group is co-chaired by the DGEC and by ASN and consists more specifically of the waste producers and managers, associations, administrations, the HCTISN, the ASND and IRSN.

Drafting of the 2013-2015 PNGMDR is also based on the national inventory of radioactive materials and waste, published by Andra in 2012, with a view to forecasting the production of waste over the coming decades and the corresponding storage requirements and capacity. Owing to the large number of waste categories identified in the National Inventory (about a hundred), the PNGMDR groups certain categories together to give a more easily comprehensible overview of the management routes.

The 2013-2015 edition of the Plan is also based on the results of studies initiated under the previous Plan, most of which were included in the decree and the order of 23rd April 2012 setting out requirements concerning the 2010-2012 PNGMDR.

Following on from the previous PNGMDR, the approach adopted for the revision of the PNGMDR thus places much emphasis on pluralism and transparency, consistently with the Grenelle environment summit in France. It should also be noted that the incorporation of these concerns into the PNGMDR in fact predates the Grenelle environment summit.

Finally, the value of a National Plan For Radioactive Materials And Waste Management was confirmed in 2011 at the European level, by Council directive 2011/70/Euratom[13], presented in section 1.3.1.

1.3.4 The steering committee for managing the post-accident phase (CODIRPA)

Context

The interministerial directive of 7th April 2005 on the action of the public authorities in the case of an event leading to a radiological emergency situation tasked ASN, together with the ministerial departments concerned, with establishing the framework and defining, preparing and implementing the necessary measures for responding to post-accident situations. In June 2005, ASN set up a Steering Committee to manage the post-accident phase following a nuclear accident or radiological emergency situation (CODIRPA), responsible for developing the corresponding doctrine and it in particular set up working groups (GT).

The CODIRPA involved three phases: definition of the basic scenarios, creation of 11 expert groups (local information committees, associations, elected officials, health agencies, appraisal organisations, authorities, etc.), responsible for producing recommendations on specifictopics and validating the post-accident management doctrine resulting from this work. All of these elements led to two international seminars organised by ASN in Paris, in 2007 and 2011.

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To support the working groups, the CODIRPA adopted two intermediate severity accident scenarios for an NPP: a scenario with partial reactor core melt and a steam generator tube rupture scenario. The eleven working groups met: each group drafted a final report summarising its conclusions on the assigned topic. In 2010, local experiments were carried out on three NPPs and in several surrounding communes, to test the doctrine currently being constructed.

The purpose of the CODIRPA was to prepare provisions designed to address complex post-accident management problems, in particular those concerning the health management of the populations, the economic consequences, or the rehabilitation of living conditions in the contaminated zones. In the light of these issues, three fundamental objectives were identified for post-accident management of a nuclear accident:

- protect the populations against the dangers of ionising radiation;
- provide support to the population that suffers the consequences of the accident;
- reclaim the territories affected, in economic and social terms.

The approach followed by the CODIRPA led to the drafting of aspects of a post-nuclear accident management doctrine, published on 21st November 2012[14]. This doctrine is based on the international principles of radiation protection, but also on guideline values taken from the work of the CODIRPA. It comprises management objectives and various means of achieving them, in order to deal with a situation that is by its nature complex, owing to the large number of subjects to be dealt with and the number of players involved.

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[13] Council Directive 2011/70/EURATOM of 19th July 2011 establishing a community framework for the safe and responsible management of spent fuels and radioactive waste.

[14] The aspects of the doctrine are available on the ASN website http://www.asn.fr, heading "les dossiers", "la gestion post-accidentelle", "comité directeur de phase post-accidentelle".

Waste management in the post-accident phase

For implementation of contaminated territories reclamation and population protection strategies, waste management in the post-accident phase must follow an overall goal of reducing ambient radiological contamination, by limiting the impact of this management on the public and the response teams, more specifically by limiting the transport of contamination outside the zones contaminated by the accident and therefore, whenever possible, opting for management of this waste within these zones. In a normal situation, the principles of radioactive waste management are defined in the Environment Code and in the National Plan For Radioactive Materials And Waste Management (PNGMDR). In a post-accident phase, the nature and volume of the waste to be managed, the availability of the waste management facilities and the potential radiological impact of the processing or removal of certain contaminated waste are all criteria which must underpin the choice of contamination reduction measures, the choice of population and environmental protection measures and the choice of waste management solutions, with the aim of optimising waste management.

The first step in post-accident management of waste is to differentiate between the contaminated waste and the non-contaminated waste. It would appear to be relatively unrealistic to expect waste management to be organised on the basis of measuring the radiological activity of the waste, in particular because the measurement resources available on exiting the emergency phase will probably be used to characterise the environment or to monitor the radiological activity of foodstuffs. It is therefore proposed that the waste be managed according to its origin (zones of various degrees of contamination). These zones will initially be identified on the basis of the results of predictive assessments. These zones are based on those defined for protection of the populations and for reinforced monitoring of areas determined according to the objectives of management of post-accident consequences:

- **The Population Protection Zone (ZPP) corresponds to the perimeter within which measures to reduce the exposure of the people residing there is justified.** This zone is defined according to a radiation protection objective for the population living in the most heavily contaminated areas;
- **The Tightened Surveillance Zone (ZST)** extends beyond the Population Protection Zone. It is characterised by lower environmental contamination which does not in principle warrant population protection measures, other than a ban on marketing of foodstuffs produced locally and recommendations designed to limit the consumption of certain foods produced locally or derived from hunting, fishing or gathering.

The waste considered as contaminated is waste produced in the ZPP as a result of the accident, barring special cases. This waste will be specially stored in a facility to be gradually brought on-line. However, exceptional provisions may be authorised as of the exit from the emergency phase, when perishable waste cannot be stored (for example: spreading of milk), although account must be taken of the vulnerability of the soil and the water resources.

In and beyond the ZST, all waste is considered to be non-contaminated. It can be processed or disposed of in accordance with the usual practices, subject to certain provisions, for example in facilities equipped with radioactivity detection gantries. In-situ periodic measurement of the radiological activity of certain waste could however be envisaged, in particular for the sludges produced in water treatment plants, in which the radioactivity is liable to become concentrated.

Follow-up of the work already done

The Fukushima Daiichi NPP accident of March 2011 in Japan provided a stark reminder of the importance of an approach such as that adopted by the CODIRPA. The consequences of this event, which the Japanese population is facing today, presents new issues for the CODIRPA. This approach will therefore be continued beyond publication of the document entitled "Doctrine for post-accident management of a nuclear accident" on 21st November 2012, which is the cornerstone of post-accident management of a nuclear accident.

The particular aim will thus be to assist with implementation of this plan nationwide. In addition to this implementation work, some questions are also still on hold, pending the outcome of the first phase of the CODIRPA's work and the thought that has so far been given to intermediate scale accidents must be extended to include the management of severe accidents.

CODIRPA's new programme of work will thus concern:

- periodic updating of the established doctrine, taking account of accident situations incorporated into the government plan and experience feedback from post-accident management of the Fukushima event in Japan;
- support with preparation for management of the post-accident phase nationally and regionally, with the administrations concerned and with all stakeholders.

CODIRPA's work is available on the ASN website[15]. It will be presented to a meeting of the PNGMDR working group.

1.3.5 The stakeholders in radioactive materials and waste management.

The owners of radioactive materials and the producers of radioactive waste can be broken down into five economic sectors (see 1.1.2): nuclear power generating, research, defence, industry unrelated to power generation and medical. For the overall inventory, three main waste producers can be mentioned, as stated in the 2012 national inventory of radioactive materials and waste: in alphabetical order, these are AREVA, CEA and EDF.

Andra (the French national radioactive waste management agency) is a specialised public organisation responsible for the long-term management of radioactive waste. Andra's duties, as detailed in Article L.542-12 of the Environment Code, notably include the design and operation of disposal facilities, the performance of studies and research into storage and deep geological disposal, the collection of radioactive waste from the small producers from outside the nuclear power sector, the rehabilitation of polluted sites, and information of the public. Andra drafts, updates and every three years publishes the inventory of radioactive materials and waste present in France, and where it is located around the country.

In addition to Andra, already mentioned, the main French research institutes concerning the management of radioactive materials and waste are CEA, the BRGM (geological and mining research office), CNRS (which organises its research around the research programme called PACEN – French acronym for back-end cycle and nuclear energy – followed since 2012 by the interdisciplinary NEEDS challenge - Nuclear, Energy, Environment, Waste, Society), INERIS (national institute for the industrial environment and risks), IRSN (institute of radiation protection and nuclear safety), the Carnot MINES Institute (Carnot Institute "innovative

[15] The work of the CODIRPA is available on the website http://www.asn.fr, heading "les dossiers", "la gestion post-accidentelle".

methods for companies and society), plus the universities. The 28th June 2006 Act more specifically entrusted responsibility for research on separation-transmutation to CEA and research on disposal and storage to Andra. IRSN focuses its research primarily on safety and radiation protection issues associated with waste management and geological disposal in particular. It contributes to maintaining a high-level of expertise enabling the Institute to act as a technical support organisation for the safety regulators (ASN and ASND). At the same time, a certain number of R&D actions are performed by industry (EDF and AREVA), partly under agreements linking them to CEA or Andra. A committee for the monitoring of research on the cycle back-end (COSRAC) aims to ensure that these research programmes are consistent and coherent. Finally, the National Review Board, whose role was confirmed by the 28th June 2006 Act, annually reviews research being performed into the management of radioactive materials and waste.

Several Ministries are involved in defining, implementing and overseeing radioactive materials and waste management policy. Within the Ministry for Ecology, Sustainable Development and Energy (MEDDE), the General Directorate for Energy and Climate (DGEC) draws up policy and implements Government decisions concerning the civil nuclear sector, while the General Directorate for Risk Prevention (DGPR) and more specifically the Nuclear Safety and Radiation Protection Delegation (MSNR) drafts, coordinates and implements the Government's roles concerning civil nuclear safety and radiation protection, with the exception of the duties entrusted to ASN (see below). Jointly with ASN, this delegation also follows up questions concerning the management of the former uranium mines and sites and soils polluted by radioactive substances. Moreover, still within the DGPR, the waste planning and management office drafts conventional waste management policy, including waste referred to as technologically enhanced naturally occurring radioactive material (TENORM). At the Ministry for Higher Education and Research (MESR), the General Directorate for Research and Innovation (DGRI) coordinates French research efforts.

There are two authorities in France for the oversight of nuclear safety and radiation protection: ASN and ASND. The Nuclear Safety Authority (ASN) regulates nuclear safety and radiation protection for civil nuclear facilities and activities. It is an independent administrative authority, set up by the 13th June 2006 Act on transparency and security in the nuclear field. The regulation of nuclear safety and radiation protection for defence-related activities and facilities is the responsibility of the Defence Nuclear Safety Authority (ASND). The ASND is run by the nuclear safety and radiation protection delegate for defence-related activities and facilities, reporting to the Minister responsible for defence and the Minister responsible for industry.

At Parliament, the Parliamentary Office for the Evaluation of Scientific and Technological Choices (OPECST) can carry out assessments in order to inform Parliament of the consequences of the scientific and technological choices made. These assessments may in particular concern the nuclear energy field. The role of Parliament and its long-term commitment must be underlined with regard to the oversight and definition of national policy for the management of radioactive materials and waste.

In the exchanges organised to promote transparency and consultation, many other stakeholders are required to take part in defining radioactive materials and waste management policy. Representatives of civil society and environmental protection associations thus take part in the PNGMDR working group, such as ACRO (Association for the control of radioactivity in the West), Robin des Bois, the GSIEN (grouping of scientists for information about nuclear energy), WISE-Paris (World Information Service on Energy), Greenpeace and France Nature Environnement. Articles L.125-24 to L.125-40 of the Environment Code also require that the

High Committee for Transparency and Information on Nuclear Security (HCTISN) periodically organise consultations and debates on this topic. Discussions are also held within the local information and monitoring committee (CLIS) for the Andra underground laboratory in Meuse / Haute-Marne, as well as within the local information commissions and committees (CLI) for the basic nuclear installations (BNI and INBS), which are grouped into a national association of CLIs (ANCCLI).

Finally, international organisations are working on harmonising management policies between the various countries: EURATOM (European atomic energy community) at a European level, the NEA (Nuclear Energy Agency) within the OECD and the IAEA (International Atomic Energy Agency) reporting to the UN General Assembly.

1.4 *The social dimension and preservation of memory*

The purpose of research into human and social sciences is to integrate a societal dimension into the various projects concerning both waste management and how it is incorporated into a cross-disciplinary perspective. The studies conducted more particularly concern management of the most highly radioactive waste, which raises complex issues concerning the need to anticipate and make provision for events over very long time-scales. Advance provision must also be made to deal with the question of the long-term preservation and transmission of memory, well after closure of the disposal repositories.

1.4.1 The social dimension / Studies in human and social sciences

The involvement of the Human and Social Sciences (HSS) in the field of radioactive waste and materials management is justified upstream by the desire to make the various recommended solutions more robust. Their acceptability, which in the end is political in nature, is made easier when all the phenomena involved are dealt with in an appropriate framework, without ignoring their socio-economic, environmental, political, cultural, etc. aspects, and the various scientific and technical issues involved are interconnected. One-dimensional, inward-looking R&D has little chance of helping technical projects succeed, as is shown by the history of nuclear waste management in France prior to 1991. The aim of HSS research is thus to integrate the social aspects into the various on-going projects and ensure that they all work together in a cross-disciplinary system. Collaboration with researchers from these diverse backgrounds must from the outset aim to create specialised communities on subjects of common interest with the operators and the stakeholders.

Andra research in the Human and Social Sciences (HSS) focuses on the social dimensions (socio-economic, political, cultural, etc.) of the Agency's projects and thus aims to improve their robustness from a cross-disciplinary perspective. Focus was therefore placed at first on the topic of reversibility, leading to a number of scientific events and publications, as well as a doctoral thesis in economic sciences. Andra is currently looking to develop this approach on a long-term basis by setting up a group of cross-cutting human and social sciences laboratories around the generic topic "transmission between generations and comprehension of long time-scales". The choice of this topic can be justified by the fact that the time-frames involved in the Agency's activities, in particular in the management of the most highly radioactive waste, is indeed unique when compared with other industrial areas. This specificity raises particularly complex questions which notably concern the ability to anticipate events over long periods of time and to ensure that they are managed.

Other research topics, the definition of which is as yet less well-advanced, could be addressed in the near future, more specifically in the fields of long-term economics and environmental assessments, or identified by the new HSS programmes which are currently being set up at CNRS and IRSN.

The CNRS's NEEDS programme (Nuclear, Energy, Environment, Waste, Society) includes HSS in nuclear questions and envisages looking at the question of time in a more general manner, from the risk management and assessment viewpoint. This programme also intends to build on the HSS knowledge acquired on the topic of nuclear waste, more specifically via the considerable work done on this questions at CNRS.

The areas in which work is being done at Andra and the CNRS are detailed in the research part appended to the PNGMDR.

1.4.2 Preservation and transmission of memory

All the currently operated or planned Andra repositories make provision for the preservation of memory, so that a record of these repositories can be handed down after their closure. The question of the long-term preservation and transmission of a recorded trace, after closure of the disposal repositories is quite different from that of managing the knowledge needed for a conventional industrial project. This type of knowledge management system is used by Andra and elsewhere and will necessarily evolve over the coming decades. This evolution cannot however guarantee the transmission of knowledge or even of a trace of the repository over the very long term, especially once no repositories are in activity any more. It is therefore necessary to anticipate what future generations would need in order to preserve a memory of the repository, if the evolution of the knowledge management system were to prove insufficient to keep it operational.

The reference solution implemented by Andra

The problem of maintaining the memory of the disposal facilities was considered as of the 1980s for the Manche repository (CSM). To address this problem, a solution of archival on permanent paper was defined in 1995. In 1996, the Turpin commission confirmed the methods adopted by Andra and recognised further new developments. The reference solution chosen by Andra for the long-term record of its disposal facilities is currently built around three "passive" and two "active" records.

The three "passive" records are:

- the "detailed record" consisting of all the technical documentation necessary for the oversight and monitoring, comprehension and modification of a disposal facility. The creation of the detailed record is based on the selection and ranking of information by means of possible evolution scenarios identified consistently with the long-term safety approach. A set of search instruments (inventories, glossary, index, abstracts, etc.) to ensure accessibility and understanding. The sustainability of the documents is based on an appropriate choice of the "permanent ink/ paper" combination and the conservation of two copies on separate sites, the repository itself and the National Archives. Finally, the validity of the detailed record is updated by additions every five years, until the end of the oversight phase;
- the "summary record" is a single document summarising the technical and historical information, intended for the decision-makers and the public. Updates are planned after each revision of the safety analysis reports. The full informative weight of the final version will depend on its widespread distribution: town halls, lawyers, associations, Conseil Général, Prefect's office, Ministries, national and international institutions, etc.;
- entry on the registry of "institutional controls" ensures that there is an administrative record of the site, giving warning of a potential risk if carrying out works on it.

The two "active" records are:

- the development of communication with the public thanks to the organisation of open days, conferences, exhibitions and interviews, as well as by distributing specific memory communication tools, brochures and website;

- increasing the role of the local information committees (CLI). The question of memory is one of the topics tackled by these committees, which should enable it to be discussed and debated at the local level.

Analysis of this system as a whole, more specifically in the light of experience feedback on the durability of other historical memory systems, leads to the conclusion that there can be a high degree of confidence of it lasting several centuries. This reference solution also complies with regulatory requirements applicable to the various repositories of radioactive waste.

Andra's record project

Without more detailed examination, the reference solution adopted by Andra cannot however be considered as being the best available solution. It in fact comprises certain weak points. It is heavily biased towards the conservation of documents and does not give sufficient consideration to other media. Furthermore, no detailed study of its compatibility with the potential needs of future generations has been carried out. Finally, a record that is guaranteed for "only" a few centuries after closure of the repository is considered to be too short by several stakeholders in this repository, notably for the future local residents. Consequently, in 2010, Andra decided to launch a memory project with a two-fold goal: increase the robustness of the reference solution and look at possibilities for a recorded memory designed to last several thousand years.

The memory project comprises on the one hand work designed to continue to create and improve records about the facilities and, on the other, scientific studies concerning two fields: materials ageing and human and social sciences (also see the research part appended to the PNGMDR).

Recording a memory of legacy disposal situations

Certain radioactive waste was in the past disposed of in conventional waste disposal facilities. In order to retain a record of the presence of such sites, the use of the available regulatory tools such as mentioning this in town planning documents, the implementation of utilisation restrictions or the use of institutional controls is a good practice.

1.5 The cost and financing of waste management

Under the control of the State, the management of radioactive materials and waste is financed by the nuclear licensees, in accordance with the "polluter-pays" principle.

A system to ring-fence the financing of long-term nuclear costs, instituted by the 28th June 2006 Act codified in the Environment Code, provides for the creation of a portfolio of dedicated assets by the nuclear licensees during operation. To do this, the licensees are required to evaluate the long-term costs, which include decommissioning costs and the cost of managing spent fuels and radioactive waste. They must ensure that these future costs are already covered by creating dedicated assets with a high level of security.

These operations are closely monitored by the State, by means of an administrative authority consisting of the Ministers responsible for the economy and for energy. To carry out its oversight duties, the administrative authority notably receives a three-yearly report from the licensees on the methods adopted and choices made for the management of the dedicated assets, plus a quarterly inventory of the dedicated assets. In addition, a non-Parliamentary Commission (the CNEF) reviews the oversight by the administrative authority and submits a three-yearly report of its assessments to Parliament and to the High Committee for Transparency and Information on Nuclear Security (HCTISN).

1.5.1 Legislative and regulatory provisions on the ring-fencing of financing of long-term costs

Application of the "polluter-pays" principle is particularly important in the financing of decommissioning operations and radioactive waste management. It is essential to avoid passing these costs on to future generations or to society as a whole, while it is we who today benefit from nuclear power generation. The 28th June 2006 Act thus introduced an arrangement to ring-fence the financing of long-term nuclear costs[16], supplemented in 2007 by a body of regulations.[17]

This arrangement is based on the creation, as of commissioning of the facility and then gradually during the course of operation, of a portfolio of dedicated assets, managed in such a way that their sale allows the cost of the long-term operations to be financed as and when the time comes. This is done under the control of the State (administrative authority), which analyses the licensees' situation and can prescribe the measures necessary if the identified resources are felt to be insufficient or inadequate. In any case, the nuclear licensees remain fully responsible for the correct financing of the future cost of decommissioning their facilities or managing their waste.

The licensees of BNIs and defence BNIs (INBS) are thus required to make a prudent evaluation of the cost of decommissioning their facilities or, for radioactive waste disposal sites, the cost of final shutdown, upkeep and surveillance[18]. They also evaluate the cost of managing their spent

[16] The provisions of the Act of 28th June 2006, and more particularly its Article 20, are today partly codified (Articles L-594-1 and following of the Environment Code).

[17] Decree n°2007-243 of 23rd February 2007 (modified by decree 2010-1673 of 29th December 2010) and the order of 21st March 2007 concerning secure financing of long-term expenses.

[18] This assessment also includes the cost of retrieving and packaging legacy waste (RCD).

fuel and their radioactive waste. These costs must be covered by updated provisions recorded in the licensees' accounts. These then create a portfolio of dedicated assets, allocated exclusively to covering their provisions, and the sale value of which is at least equal to the amount of the provisions (except for those linked to the operating cycle[19]).

The dedicated assets must offer a sufficient and appropriate level of security, diversification and liquidity. Regulatory provisions therefore specify the asset acceptability rules (more specifically concerning the category of assets and the level of diversification of the portfolio).

The assets allocated to coverage of the provisions may not be used for any other purpose by the licensee and may not be the subject of any claim on the part of a creditor (including if the licensee is experiencing financial difficulties), with the exception of the State in the exercise of its powers, to ensure that the licensees meet their obligations with respect to decommissioning and the management of radioactive materials and waste. The assets must be the subject of an inventory.

Since June 2011, the licensees have been required to permanently maintain the coverage level of their provisions above the 100% threshold. Some nuclear licensees have nonetheless been given a transitional period (until 29th June 2016) in which to reach this level, given the remoteness of the time at which the expenditure will be required, while nonetheless ensuring that a minimum coverage of 75% is met immediately.

1.5.2 Licensee oversight procedures

Licensee compliance with the obligations is monitored by an administrative authority, formed jointly by the Ministers responsible for the economy and for energy. Administratively, this monitoring is performed by the General Directorate for Energy and Climate, which calls on the expertise of the competent nuclear safety regulators (ASN and ASND).

Under the terms of the Environment Code[20], the licensees send the administrative authority a report, every three years, describing their evaluation of the long-term costs, the methods applied to calculate the corresponding provisions and the choices made with regard to the composition and management of the assets allocated to coverage of the provisions. An update of this report must also be transmitted annually, as well as on the occasion of any event entailing a substantial modification of its content. Finally, the licensees send the administrative authority a quarterly inventory of the dedicated assets.

If the administrative authority identifies any insufficiency or inadequacy in the evaluation of costs, the calculation of the provisions or the amount, the composition or management of the assets allocated to these provisions, it may – after hearing the licensee's observations - prescribe the steps necessary for remedying the situation, setting deadlines for compliance. In setting these deadlines, the administrative authority takes account of the current economic conditions and the situation of the financial markets. These deadlines may not exceed three years.

If these requirements are not met within the allotted time, the administrative authority may order the creation of the necessary assets as well as all and any measures concerning their management.

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[19] The order of 21st March 2007 specifies that only the management costs for spent fuels that can be recycled in industrial facilities that either exist or are under construction can be considered as linked to the operating cycle, as defined in Article L.594-2 of the Environment Code (ex-Article 20-II of the Act of 28th June 2006).
[20] Article L-594-1 and following.

The administrative authority may also impose a financial penalty, the amount of which cannot exceed 5% of the difference between the value of the assets put into place by a BNI or INBS licensee and that stipulated by the administrative authority. If the licensee fails to meet its information obligations, the administrative authority may impose a financial penalty of up to €150,000.

Moreover, if the administrative authority observes that there could be an obstacle to the application of the legislative provisions, it may require that the licensee concerned pay the necessary sums into a fund set up with Andra (Article L542-12-2 of the Environment Code).

Moreover, a non-Parliamentary commission (CNEF - the National Review Board for financing the cost of decommissioning of basic nuclear installations and of managing spent fuels and radioactive waste) is tasked with assessing the monitoring by the administrative authority, to check that the updated provisions are adequate for meeting the gross costs, as evaluated by the licensees, and that the dedicated assets are correctly managed. This board also assesses the handling of the two funds managed by Andra – which are separate from the Andra fund created specifically pursuant to Article L.594-2:

- a fund intended for financing of research and studies on the storage and deep geological disposal of radioactive waste (created by Article L.542-12-1 of the Environment Code);
- a fund intended for financing of the construction, operation, final shutdown, upkeep and surveillance of storage or disposal facilities for high or intermediate level, long-lived waste built or operated by the agency (created by Article L.542-12-2 of the Environment Code).

The first meeting of the CNEF was held on 7th June 2011. It receives copies of the three-yearly reports from the licensees and may ask them to submit all and any documents it requires for the performance of its duties.

Every three years, the commission must submit a report presenting its assessment to Parliament and to the High Committee for Transparency and Information on Nuclear Security (HCTISN). The first CNEF report was submitted in July 2012 and is accessible on the website of the Ministry responsible for energy[21].

1.5.3 Amounts of provisions and dedicated assets

For the three main licensees (AREVA, CEA, EDF), as at 31st December 2011, the following table shows:

- the long-term costs in gross value in 2011 economic conditions, *i.e.* the amount that would have to be spent if all the works were carried out in 2011;
- the corresponding provisions updated according to the anticipated expenditure schedules;
- the amount of the share of these provisions which must be covered by dedicated assets as required by law;
- the value of the assets already in place.

[21] http://www.developpement-durable.gouv.fr/Financement-des-charges-nucleaires.html

<table>
<tr><th colspan="2">In billions of euros, and as at 31.12.2011</th><th>Gross costs, in 2011 economic conditions</th><th>Updated provisions</th><th>Provisions to be covered by dedicated assets [22]</th><th>Value of coverage assets portfolio</th><th>Coverage percentage</th></tr>
<tr><td rowspan="4">EDF</td><td>Decommissioning</td><td>21</td><td>11</td><td>11</td><td rowspan="3">-</td><td rowspan="3">-</td></tr>
<tr><td>Fuels</td><td>15</td><td>9</td><td>-</td></tr>
<tr><td>Waste</td><td>25</td><td>7</td><td>7</td></tr>
<tr><td>TOTAL</td><td>61</td><td>27</td><td>18</td><td>16</td><td>85%</td></tr>
<tr><td rowspan="5">CEA</td><td>Decommissioning</td><td>9</td><td>6</td><td>6</td><td rowspan="4">-</td><td rowspan="4">-</td></tr>
<tr><td>Fuels</td><td>1</td><td>1</td><td>1</td></tr>
<tr><td>RCD*</td><td>3</td><td>2</td><td>2</td></tr>
<tr><td>Waste</td><td>3</td><td>1</td><td>1</td></tr>
<tr><td>TOTAL</td><td>16</td><td>10</td><td>10</td><td>10</td><td>100%</td></tr>
<tr><td rowspan="4">AREVA and subsidiaries</td><td>Decommissioning</td><td>8</td><td>4</td><td>4</td><td rowspan="3">-</td><td rowspan="3">-</td></tr>
<tr><td>RCD*</td><td>2</td><td>1</td><td>1</td></tr>
<tr><td>Waste</td><td>2</td><td>1</td><td>1</td></tr>
<tr><td>TOTAL</td><td>12</td><td>6</td><td>6</td><td>5</td><td>95%</td></tr>
<tr><td rowspan="5">Total Licensees</td><td>Decommissioning</td><td>38</td><td>21</td><td>21</td><td rowspan="4">-</td><td rowspan="4">-</td></tr>
<tr><td>Fuels</td><td>16</td><td>10</td><td>1</td></tr>
<tr><td>RCD*</td><td>5</td><td>3</td><td>3</td></tr>
<tr><td>Waste</td><td>30</td><td>9</td><td>9</td></tr>
<tr><td>TOTAL</td><td>89</td><td>43</td><td>34</td><td>31</td><td>91%</td></tr>
</table>

Dedicated assets of licensees AREVA, EDF and CEA as at 31st December 2011[23].

** RCD: recovery and packaging of legacy waste*

[22] Only the provisions for decommissioning and the provisions for radioactive waste management must be covered by dedicated assets. The provisions for management of spent fuel that can be recycled in industrial installations that exist or are under construction are excluded from the coverage basis, even if counted among the provisions under Article L.594-1 of the Environment Code

[23] The data given in this table are taken from the report from the National Review Board for financing the cost of decommissioning of basic nuclear installations and of managing spent fuels and radioactive waste (CNEF) of July 2012 and the DGEC.

2 Existing management routes: Summary and outlook

2.1 Management of legacy situations

Certain radioactive waste may, in the past, have been managed in ways that have since changed, for example disposal within or close to the production sites. In certain cases, it may also have been used as backfill, or handled in routes intended specifically for the management of conventional waste. The term legacy disposal sites is used to describe those places (except for mining residue and waste rock repositories) where radioactive waste is disposed of and is not under the responsibility of Andra and for which, at the time of disposal, the producers or those in possession of the waste did not envisage resorting to existing or planned external routes dedicated to the management of radioactive waste. This concerns:

- thirteen conventional waste disposal facilities which had received VLL waste from the conventional or nuclear industries;
- waste disposed of within or near civil or military BNIs;
- depots of TENORM waste (waste created by the transformation of raw materials naturally containing radionuclides but which are not used for their radioactive properties), which are not covered by the classified installations regulations. This in particular concerns phosphogypsum waste from the production of fertilisers, residues from the production of alumina, coal ash from thermal power plants and residues from activities producing rare earths from monazite.

The 2013-2015 PNGMDR requires carry out investigations to search for legacy disposal sites within or near the perimeter of nuclear facilities and the presentation of management strategies for the disposal sites thus identified. In the case of waste generated by the Comurhex Malvési plant, a distinction must be made between the long-term management of the waste already produced (since 1960) and the management of the waste to be produced between now and 2050. **The 2013-2015 PNGMDR requires that Comurhex continue with feasibility studies concerning the disposal options for the waste already produced**.

2.1.1 Context and issues

The management methods for radioactive waste have changed considerably. The immersion of low and intermediate level radioactive waste which France carried out in the Atlantic in 1967 and 1969, and then in the territorial waters of French Polynesia until 1982 is now a thing of the past and is prohibited by the regulations. The inventory of immersed waste is given in Andra's national inventory[24] but not covered in this chapter.

Certain very low level waste from civil BNIs or defence BNIs (INBS) were in the past disposed of on or near the production sites or in conventional waste disposal sites, when the activity of the waste was felt to be low enough. This practice ceased after the adoption of the order of 31st December 1999[25], setting out the general technical regulations intended to prevent and limit off-site detrimental effects and risks resulting from the operation of basic nuclear installations; This order comprises specific, tightened provisions concerning the management of waste from

[24] The inventory can be consulted on the Andra website, http : //www.andra.fr, heading "les éditions".

[25] This order is replaced by the order of 7th February 2012 setting out the general rules applicable to BNIs as of 1st July 2013.

BNIs. The general requirements for INBS also required application of this text before the adoption of the order of 26th September 2007 comprising provisions similar to those of the above-mentioned order of 31st December 1999. The waste produced in BNIs or INBS has therefore since that date been subject to specific, reinforced management procedures, more specifically disposal in the industrial centre for collection, storage and disposal (Cires) in Morvilliers.
TENORM waste from conventional industry is or has been placed (disposal or transit) close to the production sites and some has even been reused, notably in construction work and road building. It has also been placed in conventional waste disposal facilities subject to the legislation applicable to installations classified on environmental protection grounds, without any management procedures compatible with the nature of this waste being clearly implemented. This waste is the subject of specific management procedures presented in chapter 2.9, based on the requirements of the ministerial order of 25th May 2005 concerning professional activities involving raw materials naturally containing radionuclides not used for their radioactive properties.

These concern:

- disposal in conventional waste disposal facilities which regularly or occasionally received waste with an added radioactivity of about a few Bq/g in many cases;
- waste repositories, generally near to a BNI or INBS, some of which regularly or occasionally received waste with an added radioactivity of about a few Bq/g;
- legacy TENORM waste depots in facilities not subject to the regulations on classified installations.

2.1.2 Legacy waste repositories in conventional waste disposal facilities

Very low level waste (VLL) which may have been disposed of in conventional waste repositories primarily consists of sludges, earth, industrial residues, rubble and scrap metal from certain historical conventional industrial activities or from the civil or military nuclear industry.

The regulations have banned the disposal of radioactive waste in non-hazardous waste disposal facilities, hazardous waste disposal facilities and inert waste disposal facilities since 1997, 1992 and 2004 respectively. These texts require a check on the waste prior to its acceptance, in addition to its characterisation, which is primarily the responsibility of the producer (Article L.541-7-1 of the Environment Code). The checks performed at intake of the waste entail verification of the criteria set in the operating authorisation license. The hazardous waste disposal facilities and the non-hazardous waste disposal facilities must have the means of detecting radioactivity and implement specific waste management procedures if the alarm thresholds of the above-mentioned resources are exceeded.

The national inventory of radioactive waste published in 2012 identifies thirteen conventional waste disposal facilities which have received radioactive waste. For example, it identifies the disposal facility at Vif (38) which received manufacturing residues from the Cézus plant, the facility at Menneville (62) where phosphates transformation residues were disposed of, or the facilities at Pontailler-sur-Saône (21) and Monteux (84) which received treatment sludges from the Valduc research centre and from zirconium oxide production waste respectively. The Solérieux (26) facility contains fluorides form the Comurhex plant.

Radiological checks were carried out on representative sites listed in the national inventory. Those which received the most radioactive waste are placed under surveillance, in particular with radiological monitoring of the groundwater.

Special case of the Pierrelatte fluorites:

The Comurhex plant at Pierrelatte (installation classified on environmental protection grounds) is the origin of the liquid by-products processed to recover the uranium and the fluoride resulting from the natural uranium hexafluoride production unit. The effluents are thus processed with lime, the fluorine precipitates in the form of calcium fluoride (CaF_2), a chemically stable compound which is also naturally occurring. These fluorides have very low activity levels (about 4 Bq/g) and are disposed of in the hazardous waste repository in Bellegarde (30), in compliance with the 10th June 2003 circular on hazardous waste disposal facilities.

2.1.3 Legacy waste repositories within or close to a BNI or INBS

2.1.3.1 Investigation programmes

For the 2010-2012 PNGMDR, AREVA, CEA and EDF presented a programme of investigations to check that there were no legacy waste repositories within any BNI or INBS not mentioned in the declarations to Andra for the 2009 inventory of radioactive materials and waste. The licensees presented the target perimeters and objectives, the surveying, analysis and transmission phases, as well as the deadlines for the intermediate stages. The main principles common to the methodologies adopted by the licensees are based on:

- a survey based on waste management documentation, historical investigations and (historical) surveillance of the environment;
- an analysis and audit step (interviews) with *in situ* measurements if necessary;
- definition of the management strategy if legacy waste repositories are discovered.

The search for legacy waste repositories was extended to cover a broader perimeter than the request, which concerned only the BNI and INBS perimeter. The geographical perimeter adopted corresponds at least to the fencing around the site (and any annexes and dependencies of these sites for CEA) and the investigations sometimes went beyond this limit if so justified by information collected during the inquiries. EDF stated that with regard to the structures outside the sites (embankments, etc.) built before the BNIs were commissioned, it was possible to rule out the presence of any radioactive substances.

With regard to the nature of the waste looked for and the investigative methods used, the licensees considered radioactive waste to be as defined by Article L.542-1-1 of the Environment Code, that is *"waste containing radionuclides for which the activity level or the concentration warrants radiation protection monitoring"*. However, even though the waste zoning approach was only made mandatory by the order of 31st December 1999 (setting the deadline of 15th February 2001 for definition of zoning in the summary document for the new waste studies), the licensees stated that the waste "liable to be contaminated or activated" was not in principle ruled out of the investigations conducted, given that the available data did not generally make it possible to determine either the origin of the waste or the radiological characteristics. The licensees thus stated that all the waste disposed of was initially considered and that more detailed investigations have determined or will determine whether or not it is radioactive. If there is any doubt, measurements will be taken and if no radioactivity is detected, the waste will be considered to be conventional. This practice will be supplemented by environmental monitoring measures. The licensees also recalled that the complementary verification and control approach based on interviews with individuals who were not involved in the initial survey, could constitute a second, independent line of defence.

2.1.3.2 Progress reports

AREVA and CEA, as well as EDF for sites being decommissioned and six facilities in operation, have completed the search for legacy waste repositories. However, additional investigations are required in certain areas in which legacy repositories are suspected.

The main legacy waste repositories identified are presented below. The case of the waste contained in the ponds on the Malvési site is presented in § 2.1.3.4 below.

The Bugey mound

About 130 m^3 of ion exchange resins (not radioactive according to the criteria used at the time) were buried between 1979 and 1984 on an artificial mound of about 1 million m^3 of backfill and were revealed in 2005 during the initial siting studies for the ICEDA facility to the south of the Bugey site. This mound consists of various natural excavated materials and non-radioactive waste produced by the construction of the various production units. The quality of the groundwater in this area is monitored by eleven observation wells distributed around the mound.

The Pierrelatte mound

The mound, with a surface area of about 37,000 m^2, was created in the early 1960s. Between 1964 and 1977, trenches were dug for disposal of technological waste, including diffusion barriers and filters, fluorites produced by the processing of uranium and chromated sludges. A groundwater monitoring plan has been in place since 1998 and the integrity of the structure is also monitored.

The Pierrelatte north zone

Waste from demolition of an old building which housed the experimental chemical enrichment process was contained in twelve retention trenches in the north area, between 1964 and 1977. This waste represents a volume of 15,000 m^3. A clean-out pilot project was conducted in late 2010 and early 2011 on two pits, to confirm the nature of the buried waste (rubble, earth and pebbles). AREVA is examining the possibility of retrieving this waste for management in routes dedicated to radioactive waste.

The inert waste storage zone (ZEDI) on the Cadarache site

The ZEDI is a waste burial zone created when the centre was opened, and in which 192,000 m^3 of inert waste were placed between 1961 and 2007, including 1650 m^3 of contaminated waste (4,600 MBq) buried between 1963 and 1991. The observation wells network was completed in 2002 and is used for surveillance of the groundwater.

Building 133 on the Saclay site

VLL waste backfill (17 m^3 of old earthenware piping debris and 57 m^3 of rubble and earth) was used in the North and South foundations of building 133 on the Saclay centre. Possible retrieval could be eventually envisaged when building 133 is dismantled (not currently scheduled).

Concreted pond in the Marcoule former cladding removal pilot unit

This is an old pond in the STEL which was equipped for underwater fuel cladding removal for a few months, before the cladding removal facility came on-line in 1959. This semi-buried pond was then entirely filled with concrete, with some of the machines and equipment used for the cladding removal process left inside it. The total volume of this pond is 1,116 m^3 and it is entirely isolated from the process, as all the piping was removed and the upper part was sealed. A quarterly surface contamination check is carried out by the radiation protection department during its periodic inspections. No anomaly has yet been detected. The STEL is also bounded by a "slurry wall", which contains the groundwater in this zone, and two boreholes to the South of the mechanical cladding removal pilot unit, to extract the groundwater and thus lower the water level.

The experimentation shafts in the Moronvilliers experimentation centre (PEM)

There are about a hundred shafts containing residues from the experiments conducted in them. These shafts have been filled in and capped. During the survey of polluted sites and soils, CEA declared the PEM site in the BASOL database in May 1997. The entire site, including the hundred or so shafts, is the subject of reinforced environmental surveillance, the results of which are regularly transmitted by ASND to the Prefect. Finally, the radiometric map of the site produced by HELINUC confirmed compliance with the radiological baseline requirements of this site.

The first six conventional waste repositories at Valduc

Until the early 1990s, owing to the isolated nature of the centre, household and general industrial waste, along with rubble, were dumped at six locations on the centre, in compliance with the standards of the period and the practices of all French communes. These sites mainly concerned general, non-hazardous materials, dumped in hollows in the ground. The waste and rubble was thus used to level out the areas in question. Radiological contamination cannot be totally ruled out, owing to the old decontamination practices. The volumes concerned were considerable (estimated at from 100,000 to 150,000 m^3) and their level of radioactive contamination was estimated at from zero to very low by CEA, which does not envisage any retrieval operation. These disposal areas are however under surveillance, more specifically by means of observation wells situated downstream of them, which ensures that there is no escape of any radioactive element liable to pollute the groundwater.

The Valduc 045 disposal area

This area mainly received contaminated earth from the remediation work in the "Au tilleul" valley carried out in 1995. It consists of a silo, the bottom and sides of which are lined with a sealed PEHD membrane, sandwiched between two layers of geotextile fabric, with the assembly then covered with sand. Containment is thus guaranteed. The activity of this earth is low level (average of 1 Bq/g and at most less than 10 Bq/g). The volume concerned is 8,990 m^3. This disposal area is however under surveillance, more specifically by means of observation wells situated downstream, which ensures that there is no escape of any radioactive element liable to pollute the groundwater.

2.1.3.3 Management options

EDF states that the legacy repositories contained in pools A1 (rubble) and A2 (biological shielding) at Chinon and Chooz A (rubble in area HN041) will be removed and sent to routes dedicated to radioactive waste when the facilities are dismantled.

AREVA, CEA and EDF (apart from the above case) stated that to date, for the identified legacy repositories, no external management route has been envisaged, given the absence of environmental contamination. The legacy repositories considered are monitored as part of the more general programmes of site environmental surveillance and steps are taken to retain a record of the presence of waste (definition of specific institutional controls taking account of the nature of the activity, its history and any residual risks) as and when necessary.

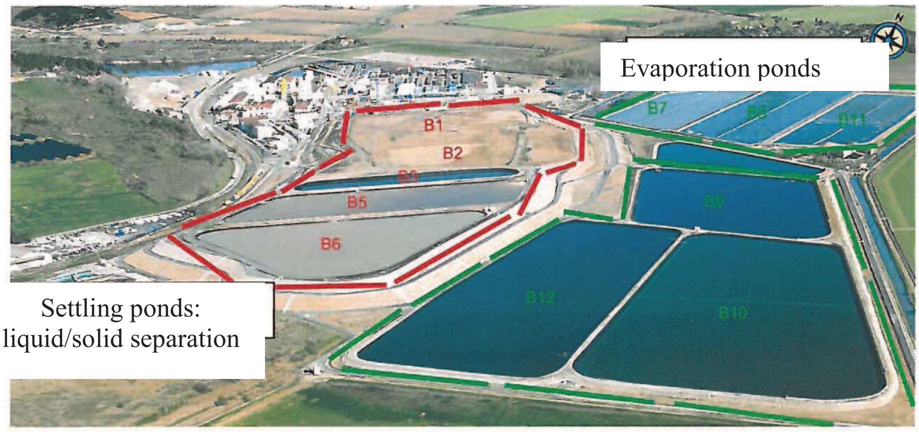
In the event of significant environmental contamination attributable to a legacy waste repository, AREVA, CEA and EDF state that management solutions would be identified on a case by case basis, according to a "cost/benefit" analysis factoring in all the environmental impacts.

2.1.3.4 Particular case of the Comurhex Malvési plant

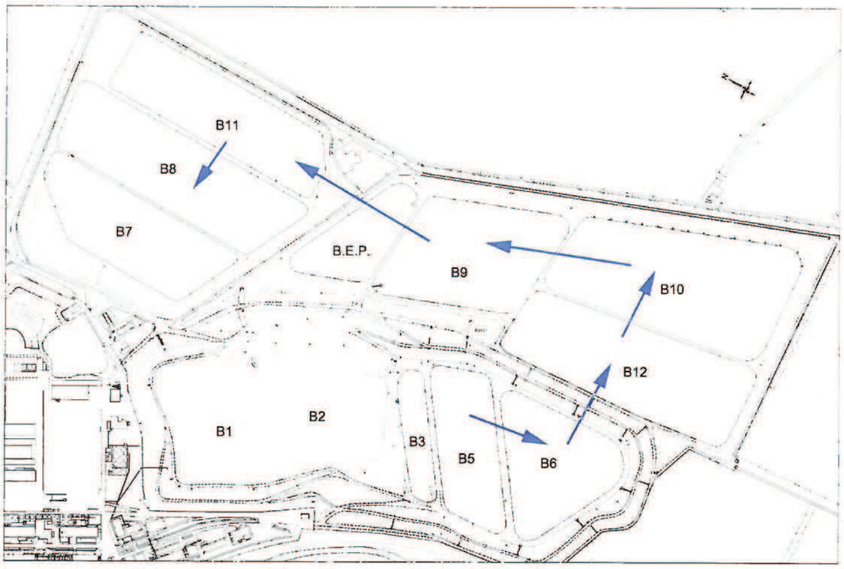
Since 1960, the Comurhex Malvési plant has been converting natural uranium from the mines into uranium tetrafluoride (UF_4). The residues and effluents from the process are managed in ponds after neutralisation with lime: settling of the solid fraction (sludges) in settling ponds B1 to B6 with natural evaporation and concentration of the liquid fraction (nitrated liquids) in evaporation ponds B7 to B12. Pond B4 no longer exists since its inclusion in B5. Pond B3 is used to manage the site's water. It should be noted that from 1960 to 1983, the facility also converted reprocessed uranium (URT) into UF_4.

The waste inventory for the Comurhex Malvési plant mentions about 280,000 m^3 of sludges produced since 1960 and stored in ponds B1 and B2 the filling of which has been suspended since the collapse of the ponds' East embankment in 2004. This volume includes the earth and materials retrieved following collapse of the embankment, as well as the covering materials. At the end of 2011, the inventory also includes:

- 300,000 m^3 of mining rock and residues (linked to the exploitation of sulphur on the site prior to its industrial reconversion by Comurhex) under ponds B1 and B2, contaminated by the infiltration of substances from the sludges in B1 and B2;
- 100,000 m^3 of waste from reprocessing of nitrated liquids present in the evaporation ponds;
- 40,000 m^3 of sludges contained in settling ponds B5 and B6;
- 20,000 m^3 of miscellaneous waste present under pond B3, currently the water management pond.



Situation of the ponds



Effluent circulation diagram

Uranium conversion on the Comurhex Malvési site generates waste every year, which could represent an additional volume of 200 to 300,000 m^3 of sludge by 2050. In order to optimise the volumes, AREVA is currently working on a project to reduce the volume of solid waste to be stored: the settling sludges would thus be dried (by filtration) until a dryness of about 60% is achieved. By 2030, the additional volumes to be managed as a result of the future operation of the facilities (for unchanged process and capacity) are thus evaluated at 88,000 m^3 of solid filtered process residues.

This range of waste represents a wide variety of specific activity levels, appreciably lower than 100 Bq/g for the sludge/materials mixtures, the solid waste from heat treatment of nitrates and contaminated earths, and above this threshold (up to 500 Bq/g total mean activity) for sludges, currently present in the settling ponds, or to be produced by dewatering.

In total, a volume of close to one million m^3 of waste will eventually need to be managed. It should be noted that the artificial radionuclides resulting from URT conversion represent 1% of the total activity of the sludges stored in B1 and B2. Apart from the presence of artificial radioactivity, the sludges in ponds B1 and B2 are radiologically contaminated, primarily by uranium but also by thorium, at levels that are incompatible with the acceptance specifications for disposal in an industrial collection, storage and disposal centre (Cires).

Most of the solid waste is stored in four ponds:

- the former B1 and B2 ponds, currently dried out, for which the administrative situation is currently being regularised given the presence of artificial radioactivity of a level that requires their compliance with the BNI regime (review of the creation authorisation application file currently under way for a BNI called ECRIN (confined storage of conversion residues);
- the current settling ponds B5 and B6, in operation and subject to the ICPE regime (section 1735).

For the 2010-2012 PNGMDR, Comurhex Malvési transmitted an interim report in late 2011 concerning a long and short-term safe management solution for the waste stored on the site.

Short-term management

In its report, Comurhex Malvési plans to modify the location of the current ponds B3, B5 and B6 to create a series of cells to accept the waste that will be produced by the facility. This modification first requires that the sludges already produced be emptied to ponds B1 and B2. Comurhex Malvési thus plans to create a storage cell in the ECRIN storage facility for the filtered sludges from B5 and B6. The ECRIN storage facility will also be covered to prevent transfers by water infiltration.

Long-term management

Given the volumes of waste, its physico-chemical and radiological diversity and the absence of any management route for long-lived waste, the interim report from Comurhex Malvési presents a common long-term management approach for the waste already stored and that to be produced, and is looking to assess several long-term disposal concepts for the site. Comurhex Malvési also presents an investigation programme designed to complement the assessments of the feasibility of such a repository on the site and specifying the characterisation of the waste and the geology of the Malvési site.

2.1.4 The legacy repositories of technologically enhanced naturally occurring radioactive material (TENORM) waste

Several tens of storage sites for TENORM waste are present around France. This concerns phosphogypsum waste from the production of fertiliser, alumina production residues and coal ashes from thermal power plants, some of which can still be reused. Some urban construction work also in the past used materials from conventional industry as backfill, even though it comprises slight radiological activity. This is the case with the La Rochelle port areas, in which the installations were in-filled with residues from the former activities to produce rare earths from monazite ore.

An initial inventory of the situation of these sites, supplemented by an inventory of the deposits of ash from the thermal power plants and phosphogypsum plants was performed for ASN by the Robin des Bois[26] association in 2005 and 2009, which identified about fifty sites (46 ash depots

[26] The reports drawn up by Robin des Bois are available on the website www.robindesbois.org heading, “la radioactivité naturelle technologiquement renforcée” and “les cendres de charbon et les phosphogypses”.

and 5 phosphogypsum depots). Of the sites identified, most are no longer in service and have either been redeveloped, or cleared out (reuse): they are thus considered to be legacy repositories of technologically enhanced naturally occurring radioactive material (TENORM) waste About ten are still in operation for reuse of ashes. Concerning the sites holding residues resulting from the production of alumina from bauxite, five sites are identified in the Provence-Alpes Côte d'Azur region in Vitrolles, Aygalades, La Barasse Saint Cyr, La Barasse Montgrand and Gardanne (the only site still in activity and operated by the RIO TINTO ALCAN company).

The order of 25th May 2005 requires that the licensees of facilities performing a category of professional activity mentioned in its appendix 1 (utilising raw materials naturally containing radionuclides not used for their radioactive properties) perform a study to measure the exposure to ionising radiation of natural origin and evaluate the doses to which the population and workers are liable to be subjected. In 2007, EDF and E.ON France (formerly SNET) each submitted a generic study on the radiological impact of the thermal power plant combustion ash disposal sites, showing that for both the general public and the workers no dose exceeded the 1 mSv/year limit set by Article R.1333-8[27] of the Public Health Code, concerning exposure of the population as a result of nuclear activities in France.

In its 7th November 2008 opinion on radioecological monitoring of the water around nuclear facilities and the management of former radioactive waste storage sites, the High Committee for Transparency and Information on Nuclear Security recommends "encouraging familiarity with contamination types" and that "information about the surveillance of the groundwater around BNIs, INBS and sites where waste was stored, should focus on chemical as well as radiological substances".

The 18th June 2009 circular concerning implementation of the recommendations from the High Committee for Transparency and Information on Nuclear Security requires a guarantee that on radioactive waste disposal or storage sites (outside BNIs and INBS), the environmental surveillance is adequate, with appropriate measures being taken (notably by means of targeted measurement campaigns).

Therefore, in order to confirm the results of the generic studies (showing that the dose for the general public and workers remains below 1 mSv/year) the public authorities asked the current licensees of coal-fired power plants to carry out radiological checks around the combustion ash depots. These checks include the performance of two groundwater sampling campaigns (one at high water and the other in a period of low water) representative of possible contamination by radionuclides resulting from the storage of combustion ashes from thermal power plants on the various sites. Finally, with regard to the atmospheric exposure route, this inspection provides for an analysis of airborne dust on those sites where this risk is liable to occur.

For application of these requirements, the progress of the sampling programmes by the various licensees is:

- for the former coal ash depots under EDF responsibility, split into three categories:
- one category comprising the priority sites (Blénod, La Maxe, Bouchain, Le Havre and Cordemais) for which a groundwater analysis is required in the very short term. The conclusions were released and show no particular anomaly;
- one category comprising the sites on which operation has ceased (Champagne-sur-Oise, Vaires-sur-Marne, Allennes-les-Marais, Pont-sur-Sambre, Beautor, Saint-Leu-d'Esserent). For these sites, the analyses are performed later than the five previous

[27] Article R.1333-8 of the Public Health Code "the total effective dose received by any individual not in the categories mentioned in article R. 1333-9, as a result of nuclear activities, must not exceed 1 mSv/year."

sites. The conclusions were released to the DREAL/DRIEE in mid-2012 and are currently being analysed;

- one category comprising the other sites (no ash, licensee no longer the owner, etc.) for which the analyses will be conducted subsequently;
- for the former coal ash depots belonging to the licensee E-ON/SNET, the measurement campaigns were performed and show no particular anomaly;
- for the other licensees, Azko-Nobel, Colas, Surchiste, Vermeulen-SA, some of whom reuse the ash, they were asked to use the same approach (some studies are in progress, while others have already been transmitted and are being reviewed).

For the fourteen former sites which were the responsibility of Charbonnages de France, five sites in the Nord-Pas-de-Calais (Choques, Dechy, Fouquereuil, Fouquières-les-Lenz, Hallicourt) were covered by a study conducted by the BRGM and IRSN in September 2010. The study assessed the radiological impact of the ash repositories (BRGM/RP-58941-FR[28]). It established that the radiological impact of the ash repositories on the sites examined is negligible and indicated no transfer of pollution to the groundwater. Finally, for the other nine sites, for which no licensee can be identified, the DGPR asked the BRGM and IRSN on 1st March 2012 to take steps to look for possible radioactive pollution linked to the presence of coal ash. An initial ranking of the sites was proposed by the BRGM and IRSN, according to the potential implications, the volume of the depot and the transfer modes. Five priority sites were identified (Viviez, Susville, Montceau-les-Mines, Sarreguemines, Laval-Pradel) for which a land characterisation programme is under way. The conclusions of the study should be transmitted to the administration in early 2013.

Similar measures are also under way for the phosphogypsum depots resulting from the production of fertiliser (five depots have so far been identified: Anneville-Ambourville, Saint-Etienne du Rouvray, Rogerville, Douvrin, Wattrelos), for which radiological surveillance is currently under way, notably for the groundwater.

2.1.5 Outlook[29]

The management of radioactive waste has changed over the years and some waste was disposed of on or near to the production sites. In certain cases, it may also have been used as backfill, or handled in routes specifically for the management of conventional waste.

The inventory of radioactive waste disposal sites was consolidated thanks to the action taken by the public authorities and by Andra for the update of the national inventory, which lists the legacy repositories of radioactive waste. **This inventory needs to be updated and supplemented by the on-going investigations being conducted by AREVA, CEA and EDF on or near certain facilities. A summary of the investigations carried out will be produced for 31st December 2014 and the legacy repositories discovered will be notified in the national inventory. The management strategies adopted will also be specified.** They will opt preferentially for management in existing or planned routes, more specifically when the quantities and nature of the waste so allow. For *in situ* management methods, similarly to the steps taken for the identified legacy repositories, temporary surveillance shall be put into place, or

[28] The report is available on the website http://www.nord-pas-de-calais.developpement-durable.gouv.fr, heading "la dreal", "actualités", "dépôt des cendres de charbon".

[29] Opinion 2012-AV-0166 of 4th October 2012 on the management of temporary or legacy situations is available from the website http://www.asn.fr, heading "les actions de l'ASN", "la réglementation", "bulletin officiel de l'ASN", "avis de l'ASN"

maintained as applicable, in order to detect any environmental contamination. Moreover, steps shall be taken to retain a record of the presence of this waste.

For the Comurhex Malvési plant, it would be premature to select a single solution for management of the waste generated by the plant **and long-term management of the waste already produced must be differentiated from management of the waste to be produced between now and 2050**. **Comurhex will therefore need to continue:**

- **to characterise the waste already produced and the mining residues present under settling tanks B3 to B6 in order to fine-tune the radiological and chemical inventory of the waste to be managed;**
- **the feasibility studies on the sub-surface disposal options for the waste already produced, notably ensuring the availability of a sufficient depth and volume to envisage disposal of this waste in satisfactory conditions. The calendar of studies and investigations necessary will be presented before 31st December 2013.**

2.2 *Management of mining processing residues and waste rock*

In France, the uranium mines were exploited from 1948 to 2001. The exploration, extraction and processing activities concerned about 250 sites in 25 French *départements*. Ore was processed in eight plants. Exploration uranium mines workings generated:

- "processing residues" consisting of products remaining after processing to extract the uranium;
- "waste rock" consisting of soil and rock excavated to access economically useful ores.

Given the large quantities of waste produced, the management method currently adopted for these substances is *in situ* management, including a check on the steps taken to limit the current and long-term impact to a level that is as low as reasonably achievable. These disposal sites are subject to the regulations on installations classified on environmental protection grounds, under the responsibility of Areva.

The studies conducted by Areva mean that it is now possible to assess some of the long-term impacts of the processing residue disposal sites. Those included in the 2010-2012 PNGMDR in particular led to:

- acquisition of the basic knowledge needed to assess the stability of the retention embankments around the treatment residue disposal sites and to define new associated requirement levels;
- improved knowledge concerning modelling of radon for scenarios involving housing on the disposal site and, more generally, confirming the pertinence of the assessments made following modelling, by comparing its results with surveillance measurements.

For the mining waste rock, Areva is continuing work started in 2009, to identify where it has been reused in the public domain, in order to define any incompatibilities between the presence of this waste rock and the public use of these locations. Additionally, the long-term dose impact of the waste rock heaps has been reassessed to take greater account of the radon concentrations.

In the case of water treatment and the impact of discharges from the former mining sites, the studies enabled the impacts associated with the various substances discharged to be assessed. The consideration of chemical and radiological impacts on man and the environment show the need to look for processing solutions which would limit discharges of uranium and barium.

These various studies need to be continued within the framework of the 2013-2015 PNGMDR in order to complete the long-term assessment of the exposure risks for the population and the strength of the embankments, to study the possibility of upgrading or shutting down the water treatment stations and in the end propose concrete steps to mitigate the risks and the impacts on the various sites. With regard to the waste rock heaps, the new approach adopted pursuant to the circular from the Ministry for sustainable development and ASN, dated 22nd July 2009, must be continued.

The consultation approach must also be continued with the corresponding stakeholders.

2.2.1 Context and issues

2.2.1.1 General context

In France, the uranium mines were exploited from 1948 to 2001, leading to the production of 76,000 tons of uranium. The exploration, extraction and processing activities concerned about 250 sites in France, varying widely in size and distributed among some 25 *départements* (from simple survey work to large-scale exploitation workings). Ore was mainly processed in eight plants.

All these sites are described in the national inventory of uranium mining sites called "MIMAUSA" (French acronym for record and impact of the uranium mines: summary and archives) produced by IRSN[30] at the request of the Ministry responsible for the environment. The first inventory was published in April 2004 and updated in September 2007. It was added to more recently by a cartographic database placed on line, giving more complete information.

There are two categories of products resulting from exploitation of the uranium mines:

- **mining waste rock** which refers to the soils and rock overburden excavated to access the useful ore. A distinction must be made between the waste rock which has an average uranium content corresponding to the characteristic ambient background level (for example between 15 and 100 ppm in the Limousin region), and the sub-grade ore consisting of mineralised rock excavated during exploitation of a vein but with uranium content that is insufficient to justify economic processing (less than 300 ppm). The volume of mining waste rock extracted can be evaluated at 167 million ton;
- **processing residues** which cover the products remaining after static or dynamic extraction of the uranium from the ore. The residues are therefore in fact process residues (as defined by the Environment Code) and their volume can be evaluated at 50 million tons.

Most of the mining waste rock has remained on the production site, used to fill in open-pit mines or underground mining structures such as shafts, used for reclamation cover over residue disposal sites or placed in spoil heaps. Mining waste rock with a content of less than 100 ppm may have been used as backfill, in earthworks or for road beds on locations near the mining sites. Their volume is estimated at from 1 to 2% of the volumes of waste rock extracted from the sites, or about 2 million tons.

The processing residues are disposed of on seventeen sites. This is VLL or LLW-LL type waste. There are two types of ore processing residues, characterised by their specific activity levels:

- **low-content ore processing residues** (about 300 to 600 ppm of uranium) with a total average specific activity of 44 Bq/g (including about 4 Bq/g of radium 226). These residues, resulting from static leaching (about 20 million tons) are placed either in spoil heaps, or in open-pit mines, or used as the first covering layer for dynamic leaching processing residue heaps;
- **processing residues for ore with high average content** (about 1,000 to 10,000 ppm, or 0.1 to 1% of the uranium in the French mines) with an average specific activity of 312 Bq/g (including about 29 Bq/g of radium 226). These residues, resulting from dynamic leaching (about 30 million tons) are disposed of in former open-pit mines, sometimes with an additional embankment, or in ponds closed by a surrounding embankment, or behind a dyke blocking a thalweg.

30 The MIMAUSA base is accessible on the IRSN website http://www.irsn.fr, heading "accueil", "base documentaire", "environnement, la surveillance de l'environnement", "les sites miniers d'uranium"

The mining processing residue disposal sites were all located close to the uranium ore processing installations. These disposal sites, covering a surface area of from one to several tens of hectares, contain several thousand to several million tons of residues.

With the gradual closure of the mining sites, a number of steps were taken by the public authorities to define and apply a disposal site reclamation doctrine. First of all, in 1986, a technical instruction notice concerning uranium ore processing installations defined the operating procedures for these installations. In 1993 the Barthélémy – Combes report, entitled "Low level waste – Part 1": disposal of uranium ore processing residues", produced at the request of the Minister responsible for the environment, defines the objectives and technical conditions for reclamation of the disposal sites. In December 1998, the French institute for protection and nuclear safety (IPSN, which since then has become IRSN) drafted residue disposal site reclamation doctrine and in 2001 the methodology for assessing the radiological impact of the uranium ore processing residue disposal sites[31]. In 2001, this document was supplemented by a methodology for assessing the stability of the embankments surrounding certain residue disposal sites, produced by the BRGM. In 2003, a report from the French High Public Health Council reviewed the history of the mining sites and issued recommendations concerning the performance of health surveys of people living around the uranium mining sites[32]. In 2005, a pluralistic expert group (GEP) was set up, with the goal of "providing a critical assessment of the various technical and environmental studies and formulating recommendations such as to reduce the impact of the mining sites on the populations and the environment". Its report was submitted in 2010[33].

Reclamation of the residue disposal sites thus consisted in placing a solid cover over the residues, to provide a geomechanical and radiological protection barrier designed to mitigate the risks of intrusion, erosion, dispersion of products in place and the risks related to external and internal (radon) exposure of the surrounding populations. The aim of reducing the exposure level, notably required as a result of the optimisation principle, was taken into account during the reclamation phase. Public access to these residue disposal sites is nonetheless prohibited.

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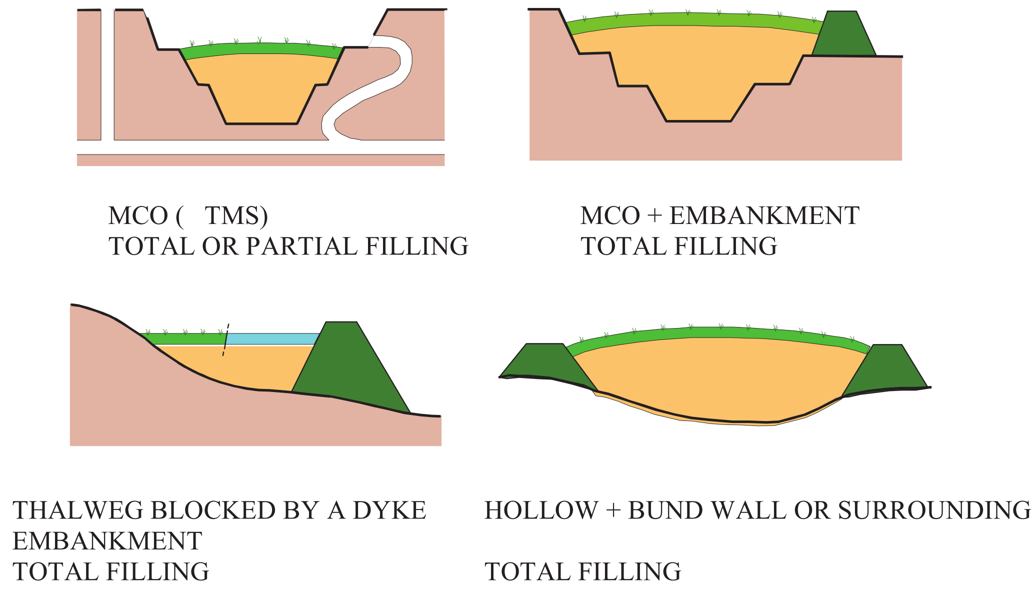
[31] The report IPSN-DPRE/SERGD/01-53 of November 2001 is available on the website www.irsn.fr, heading "base de connaissance", "environnement", la surveillance de l'environnement, "les sites miniers d'uranium" is available on the GEP website http://www.gep-nucleaire.org, heading "fonds documentaire".
[32] The report from the French High Public Health Council, "Les sites miniers d'uranium", June 2003 is available on the website www.hcsp.fr, heading "avis et rapport"
[33] GEP report "Recommandations pour la gestion des anciens sites miniers d'uranium en France. Des sites du Limousin aux autres sites, du court aux moyen et long termes" June 2010 is available on the GEP website http://www.gep-nucleaire.org, heading "fonds documentaire".



ferent types of mining residue disposal sites
(MCO: open-pit mine, TMS: underground mine-workings)

The other mining extraction sites were also reclaimed in accordance with the same optimisation objective. Unlike the residue disposal sites, most of the land concerned was restored to its uses prior to exploitation of the mines, or the land was sometimes reclaimed for new uses. For example, an open-pit mine flooded after closure, was redeveloped as an aquatic leisure centre.

2.2.1.2 Regulatory framework

The uranium mines and their annexes are covered by the mining code. The mines police focuses primarily on exploitation and conventional mining risks. It also deals with the site closure conditions. Exploitation of the French mines is based on a system of concessions and operating permits. The concessions awarded for the uranium mines are recent and of "limited" duration set by their creation decrees. Only the concessions awarded for uranium mines between 1955 and 1977 are of a duration that expires on 31st December 2018[34].

The mining processing residues are considered to be industrial residues subject to the provisions of title I of book V and in particular Article R 511-9 of the Environment Code concerning installations classified on environmental protection grounds. The mining residue disposal sites are thus subject to authorisation as installations classified on environmental protection grounds.

The mining risks prevention plans (Article L. 174-5 of the Mining Code), introduced by the 30th March 1999 Act, are a tool for controlling urban development in communes affected by the residual consequences of former mine-workings. Their aim is to define the areas exposed to mining risks, taking account of the nature and intensity of the risk. To date, no former uranium mining site is covered by a mining risks prevention plan.

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[34] Examples of the concessions granted for uranium the duration of which is set by their creation degree: La Varenne (2035), Champ Grenier (2039), Blanchetierre (2041) concessions and La Maillerie (2041) concession. For information, the concessions granted for uranium between 1919-1955 are of limited duration, while those granted between 1955-1977 are perpetual concessions and those granted after the Act of 16/06/1977 amending the mining code, are of limited duration.

Pursuant to decree 2008-357 of 16th April 2008 and decree 2012-542 of 23rd April 2012, implementing Article L. 542-1-2 of the Environment Code and establishing the requirements of the PNGMDR, AREVA submitted studies on the long-term health and environmental impacts of the mining processing residue disposal sites (physicochemical characterisation of the residues, geo-mechanical strength of the embankments and long-term radiological impact of the disposal sites) as well as of the former mining extraction sites (management of diffuse releases and water treatment, long-term impact of mining waste rock).

Finally, concerning the management and regulation of former uranium mining sites, the Ministry responsible for ecology and ASN defined an action plan, in a circular of 22nd July 2009, comprising the following four areas of work:

1. monitor the former mining sites (inspections);
2. improve understanding of the environmental and health impact of the former uranium mines, plus surveillance (in particular requirement for environmental reports);
3. manage the waste rock (better understanding of its uses, survey the places where reused in the public domain, assess its dose impact and reduce impacts if necessary and if incompatible with the usage);
4. reinforce information and consultation.

2.2.1.3 The Limousin pluralistic expert group

A pluralistic expert group for the Limousin uranium mines (GEP)[35] was set up in 2006 by the Ministry responsible for ecology, the Ministry responsible for health and ASN, in order to intensify the dialogue and consultation process around the former uranium mining sites in the Limousin region. The GEP approach was initiated on the basis of a detailed analysis of certain sites in the Limousin region, in order to develop a broader vision of the situation of the former uranium mining sites in France. On 15th September 2010, the GEP Limousin submitted its final report and its recommendations to the Minister responsible for ecology and to the ASN Chairman, concerning the short, medium and long term management of former uranium mining sites in France.

In its report, the GEP indicates that it is aware of the problems arising from the legacy management of the mining sites and noted the considerable progress achieved in recent years on this point, both in the Limousin and nationwide. The GEP considers that this progress must be continued and built on, so that a clear outlook for the sustainable management of these sites is available within about ten years. The strategy to be adopted for achieving this shall include all the various aspects (technical, institutional, social) of the problem and shall be shared with local stakeholders, incorporating regional specificities.

In a letter of 25th April 2012 to the GEP President, the Director General for the prevention of risks and the Director General of ASN clarified how the recommendations made were to be taken into account. The public authorities decided to divide the GEP's fifteen main recommendations into the following four topics:

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[35] The role of the GEP Limousin is to regularly monitor the third party appraisal of the operating results transmitted in December 2004 by AREVA NC and to take part in its oversight. Its role thus consisted in casting a critical eye over the situation of the forming uranium mining sites, in advising the administration and the licensee concerning the medium to long-term management prospects, as well as informing the local stakeholders and the public via its work and its conclusions. Four working sub-groups were defined: source term and discharges, environmental and health impacts, long-term regulatory framework, measurements.

- modernising and clarifying the institutional, regulatory and doctrine-related framework;
- improving the knowledge and management of the sites;
- improving scientific and technical knowledge;
- taking account of the other various recommendations.

2.2.1.4 The issues involved in reclamation of the former uranium mining sites

Several issues are involved in reclamation of the former uranium mining sites[36]:
- surveillance of the redeveloped former mining sites;
- management of diffuse releases and water treatment;
- mitigation of the impacts on people and the environment;
- management of reuse of the redeveloped former mining sites;

It is nonetheless worth recalling that although some sites have been redeveloped, it may be necessary to continue to treat the water collected before it is discharged. The reduction of diffuse releases and the improvements in water treatment, with preference being given to "green techniques[37]", are in this respect key issues, particularly with regard to the impact of the surrounding environment. In this respect, the analysis of current water treatment practices on the mining sites and of the associated liquid discharges, must take account of all chemical and radiological risks and must analyse their environmental impact, particularly the contamination of sedimentary build-ups in rivers and lakes.

2.2.1.5 The long-term management issues for ore processing residue disposal sites

Processing residues have been disposed of on seventeen sites. The method adopted is in-situ management given the large quantities of waste produced and following verification that the steps taken can limit the long-term impact for as long as reasonably achievable. This is why it would appear vital to assess the long-term impact of the mining residue disposal sites.

As was underlined by the GEP and the public authorities, it would also appear necessary to have an institutional, regulatory and doctrine-related framework appropriate for this long-term management perspective for the mining residue disposal sites and certain former mining sites. The Ministry for Ecology, Sustainable Development and Energy and ASN therefore set up a working group in 2012 to consider these recommendations.

2.2.1.6 reuse of mining waste rock in the public domain

Although since 1984 the transfer of waste rock to the public domain has been traced for the sites operated by Cogema and sometimes carried out in compliance with orders from the Prefect to the quarry operators, the picture remains incomplete with regard to transfers prior to 1984.

The lack of past traceability of waste rock transfers requires a precise survey of the mining waste rock reused in the public domain, in order to guarantee the compatibility of usages and mitigate the impacts if necessary. This issue is the subject of a firm request from the public authorities,

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[36] The term "former uranium mining sites" is used in the broadest sense and encompasses all types of extraction facilities which were operated on the site during its lifetime: open pit mines, underground mine works, but also if transformed into a mining processing residue disposal site.
[37] Technique with no addition of chemicals or with low impact on the surrounding environment.

detailed in the circular from ASN and the ministry responsible for ecology dated 22nd July 2009. The corresponding measures in progress are detailed in § 2.2.4.1.

The generic radiological exposure assessments performed by AREVA concerning the most frequently observed cases of mining waste reuse (farmyards, tracks, etc.) lead to added doses which do not in principle exceed the limit of 1 mSv/year[38] for the scenarios and uranium content hypotheses adopted. However, these results do not take account of exposure to radon in constructions built on land backfilled by mining waste rock. Nonetheless, the assessments need to be verified on the basis of the data collected during the on-going nationwide survey of places in which mining waste rock has been reused and any usage incompatibility situations will have to be dealt with as and when necessary.

2.2.2 Management of former mining sites

2.2.2.1 Surveillance of former mining sites

As part of the mines action plan set up in 2009 by the Ministry responsible for ecology and ASN, and in order to improve understanding of the environmental and health impact of the former uranium mines and their surveillance, AREVA is required to produce environmental summaries for each *département,* in order to reassess the environmental surveillance of all the mining sites (including their annexes such as effluent treatment plants and residues and waste rock disposal sites, etc.), so that it can be improved and adapted to the current context if need be. AREVA shall also continue with rehabilitation of the former sites which so require, in order to ensure that they are perfectly integrated into their local environment over the long-term. The on-going inventory studies as part of the MIMAUSA project and the analysis of the residual risks within the framework of the mines action plan may, if necessary, lead to additional site reclamation work by AREVA, with additional institutional controls wherever applicable.

2.2.2.2 Management of diffuse releases and water treatment

Diffuse releases must be reduced and discharge treatment improved (with preference being given to "green techniques"), in particular in terms of the impact on the surrounding environment. The National Review Board for research into the management of radioactive waste nonetheless in its June 2009 report stresses that it would appear unlikely to be able to improve the effectiveness of the water treatment as currently performed in each of the treatment stations. This physico-chemical treatment requires the use of chemical substances leading to discharges into the aquatic environment (barium, aluminium). The impact of this input must be examined, even if these chemical substances already occur naturally in the aquatic environment and even if aluminium naturally occurs in relatively high quantities in watercourses in granite areas.

Within the framework of the 2010-2012 PNGMDR, an assessment of current water treatment practices on mining sites and of the associated liquid discharges was produced by AREVA in January 2012, taking account of all the chemical and radiological risks (aluminium, barium, radium, uranium) and their impacts.

[38] For information, within the context of "nuclear activities", the regulation value set by the Public Health Code is 1 mSv/year added to the natural background radioactivity.

Fourteen water treatment stations are currently in operation on the former uranium mining sites in France. Ten[39] use physico-chemical precipitation-settling processes in normal operation. An assessment of the effectiveness of this treatment shows satisfactory efficiency and an ability to adapt to variations in the flow rates and geochemical properties of the water, which means that it is in fact the only truly dynamic treatment process. In addition to the human and financial requirements they require for operation and maintenance, the main drawback of these treatment techniques lies in the production of large volumes of sludges, which then require appropriate disposal management. Only one site[40], on which the uranium levels in the water are far higher, uses an active treatment process involving ion exchange resin columns, allowing recovery and reuse of the uranium.

The results provided by AREVA for the period 2001 to 2010, thus confirm that the concentrations of dissolved radium 226 and uranium obtained in the water discharged from the seven physico-chemical treatment stations studied allow compliance with the limits set by order of the Prefect, some of which are more restrictive than the limit values stipulated by the General Regulations of the Mining Industries (RGIE). In certain cases, compliance with certain requirements is obtained before implementation of the treatments. However, the strategy to be adopted for upgrading, shutting down or maintaining water treatment needs to be defined and justified.

AREVA has also initiated research into alternative "passive" processes. Some have allowed a continuous improvement in the treatment processes, notably with installation of limestone drains in three treatment stations[41]. They however require regular maintenance, which is a constraint for the long-term management of former mining sites and further underscores the importance of understanding the potential for the natural improvement of water quality over time.

Two other alternative processes are still being studied. The artificial wetland type process, based on natural mechanisms, is showing promising test results. However, it does have drawbacks, such as performance that is susceptible to external conditions, to the hydraulic load, to the composition of the water to be treated, which means discontinuous operation, uncertainties as to the fate of the waste at the end of treatment and it would in the end appear to be unsuitable for lowering radium 226 levels. The "sorption on wood bark type biomaterials" process will however require the definition of a solution for disposal of this bark in the event of the process being industrialised. It is therefore useful to continue the research into the development of operational alternative processes minimising human intervention.

AREVA also carried out various assessments of the impact of the discharge on the environment and on man. Therefore, based on the assessment of the radiological risks on the aquatic ecosystems of two watercourses in the Limousin region, using the ERICA[42] method, AREVA obtained risk indices[43] of less than 1, corresponding to an acceptable radiological impact by the discharge of water from the former uranium mines. The quantitative evaluation of the radiological exposure of the population as a result of the discharges into the five watercourses selected by AREVA in its study as receiving the largest volumes of treated water from the mining sites, shows compliance with the above-mentioned dose limit of 1 mSv/year, for the five

[39] Six in the Limousin region (industrial site of Bessines, Bellezane, Le Fraisse, Augèree, Silord, le Bernardan), two in Pays de Loire (l'Ecarpière, La Baconnière), one in Rhône-Alpes (Les Bois Noirs) and one in Midi-Pyrénées (Bertholène).

[40] Lodève in Languedoc Roussillon.

[41] One in Languedoc Roussillon (Le Cellier), one in Auvergne (Cérilly) and one in Pays de Loire (Beaurepaire)

[42] ERICA: Environmental Risk for Ionising Contaminants: Assessment and management

[43] The risk index is calculated as the sum of the ratios of the internal dose received by each type of organism to the dose considered to be protective for these organisms (PNEDr)

scenarios studied (residence, time spent on the banks, market gardening, aquatic activities leisure centre and site upkeep close to the watercourse) taking account of external and internal exposure routes by ingestion and inhalation and for the various age categories. Only the "residence" scenario in the event of cessation of treatment of the water discharged into the Ritord leads to a higher assessment (1.2 to 5.2 mSv per year) for children of different age categories.

The quantitative evaluation of the risks to which the populations are exposed as a result of the chemical substances (aluminium, barium and uranium) in the effluents discharged into the Gartempe and Le Ritord (Haute Vienne *département*) from two mining sites, indicate that all the risk indices are lower than 1 for the radiological component, thus demonstrating an acceptable toxic risk for these watercourses. The evaluation takes account of a reference group using the river water for consumption, drinking by animals or market gardening irrigation. The added concentrations of barium and uranium are however higher than values of 58 μg/l[44] for barium and 5 μg/l for uranium, specified by AREVA in order to assess the existence of a possible impact on aquatic ecosystems in the case of the Ritord.

The various impact assessments conducted by AREVA provide important lessons for assessing the impacts associated with the various substances discharged and thus for defining the optimisation possibilities on which to focus. The contribution of the chemical component of the discharges would in particular appear to be predominant in estimating their possible environmental impact. This contribution is especially significant for uranium but also for the barium added to the discharges by the treatment. At this stage of the risk assessment, the results obtained by AREVA for these two elements cannot rule out the existence of a potential risk for the aquatic ecosystems in the receiving watercourses. The assessments must therefore be continued.

In parallel with these studies, it would appear necessary to improve our understanding of the contamination of sediments as related to the quantities of uranium and radium discharged into the hydrographic network after treatment. To do this, AREVA must clarify the relationship between the flows discharged and the build-up of contaminated sediments in the rivers and above all the lakes, notably by means of a study on a site of the speciation of uranium in the water and the detailed radiological characterisation of the sediments according to their granulometry and to the hydraulic conditions of the watercourse. The modelling section of this study will be completed as part of the 2013-2015 PNGMDR.

2.2.3 Long-term management of ore processing residue disposal sites

2.2.3.1 Management of long-term dose impact

A system of surveillance was set up, based on analysis of all transfer and exposure routes and on identification of the population groups liable to be the most exposed. This verification comes up against a practical problem of evaluating the added dose received by a member of the public, in particular because of the natural radioactivity already present locally and the absence of any baseline benchmark identified when the mines were opened.

Generally speaking, the studies initiated by AREVA are recent and utilise the measurements and observations taken during the course of surveillance of its sites. Data must nonetheless be acquired over a sufficient time-scale and on a representative number of sites. The long-term

[44] Value corresponding to the provisional environmental quality standard (NQE) defined for this element

research work could therefore last until 2020, with interim reviews being conducted every three years, when the PNGMDR is updated.

The first studies submitted by AREVA within the context of implementation of the 2007-2009 PNGMDR were a decisive milestone in verifying the safety of the uranium ore residue disposal sites. They were able to assess the current state of knowledge on two key points, that is characterisation of the residues and the strength of the embankments surrounding certain disposal facilities, providing initial information about the expected impacts of a normal evolution scenario for the disposal sites, as well as for a range of deterioration scenarios.

It is worth clarifying that the modelling methodology developed by AREVA to assess the long-term dose impacts of the residue disposal sites comprises one normal evolution scenario and four degraded evolution scenarios, that is: loss of integrity of the embankment and covering, construction of housing above the disposal site, with or without a covering, construction of a road, presence of children playing on the excavated residues. The assessment submitted in 2008 corresponds to an initial tangible application of the approach formally laid out in the circular from the Minister responsible for the Environment, dated 7th May 1999, concerning the rehabilitation of uranium ore processing residue disposal sites. The method developed is also consistent with the approach adopted for the other surface disposal sites, such as those of Andra, notably via degraded scenario studies such as road-building or home construction on the site. This modelling methodology was applied by AREVA to nine disposal sites for ore processing residues, of different sizes and geological contexts. According to the results of the AREVA studies, the impacts in terms of the dose liable to be received by the population in a normal evolution situation remain below 1 mSv/year in the active oversight phase and those conceivable for the major degraded scenarios remain below a few tens of millisieverts per year.

The complementary studies submitted in January 2012 within the framework of the 2010-2012 PNGMDR improved the assessment of the scenarios involving residence on the disposal site, thanks to changes in modelling of the radon flux penetrating a building and originating in the foundation materials, and examined specific cases of the pertinence of the modelling by means of comparison of the results it produces with the surveillance measurement data.

With regard to the modelling of how radon enters a building, the study confirms the significant influence of advection on the results of the added dose calculation for the scenario of residence on the disposal site, by comparison with the diffusion phenomenon. The sensitivity analysis also conducted by AREVA on certain parameters underlined the primary influence of the permeability of the soil underlying the building. These studies lead to a significant rise in the assessed level of exposure by radon inhalation in an enclosed space built over the residues (or waste rock).

With regard the comparison between the modelling results and the surveillance data, AREVA concludes on the one hand that the modelling choices made generally lead to over-estimation of exposure and are therefore conservative and, on the other, that the various contributions measured at the surveillance points primarily reflect the variability in the natural background noise. AREVA thus specifies that the analysis and interpretation of the data do not reveal any significant impact from the residue disposal site and indicate a natural origin for the various contributions to the total dose.

It is nonetheless worth clarifying that the AREVA approach is based on measurements which were mainly taken by the surveillance network on the population groups residing in near proximity to the sites, a distance of generally between 500 m and more than 2000 m.

The reinforced quality of the coverings which, in the light of the long-term impact assessments[45], would appear to be a potentially effective solution on several sites, was not the subject of an additional AREVA study to assess the feasibility and pertinence of this possible strengthening on all the ore processing residues disposals sites. In the light of the results of the complementary studies submitted in January 2012, AREVA considers that these studies concluded that the various contributions to the total dose were of natural origin and that the site coverings are thus effective enough.

With regard to evolution of the long-term physico-chemical characteristics of the ore processing residues, the study submitted by AREVA for the 2007-2009 PNGMDR, indicates that the residues evolve naturally towards a mineralogical and chemical form which significantly limits the mobility of the uranium and radium. The studies requested for the 2010-2012 PNGMDR and expected for late 2012, aim to consolidate the methodology used, by complementing the data already transmitted, including the static processing residue disposal sites, in order to confirm the possibility of extrapolating them to the disposal sites not yet studied. They will also be able to allow more detailed examination of the hydrogeochemical modelling performed on the Bellezane site, to simulate normal functioning and some of the disruptions conceivable as the disposal site evolves. The possibility of extending these results to all the ore processing residue disposal sites which have not yet undergone a characterisation study, will need to be analysed.

2.2.3.2 The long-term strength of the retention embankments around the mining processing residue disposal sites

The embankment strength assessment carried out by AREVA in 2009 as part of the 2007-2009 PNGMDR is consistent with the methodological framework defined by the BRGM and would initially appear to indicate good stability of these structures.

In the additional studies submitted by AREVA in January 2012 within the framework of implementation of the 2010-2012 PNGMDR, AREVA in particular specifies that the actual construction of the structures offers good long-term stability, owing to their shallow slopes, the absence of a layer of water[46], the gradual drying of the residues and their resulting consolidation. AREVA now explicitly takes account of the impact of the cessation of upkeep of the sites on the clogging of the drainage networks and on the evolution of the hydraulic conditions inside the embankments. The unfavourable situations liable to result from this scenario are studied in the generic evaluation via the critical hydraulic conditions. AREVA also significantly reinforced the level of seismic hazard adopted in order to take account of the lifetime of the residue disposal facilities. AREVA sets the disposal facility lifetime to be considered for the long-term stability studies at 1,000 years and the target return period for a seismic hazard[47] at 30,000 years. AREVA also raised the safety factor adopted from 1 to 1.2 to estimate the strength of the embankments in the event of external loadings.

Thus the various elements produced by AREVA now provide the bases for formally defining the doctrine to be adopted for assessing the long-term geomechanical stability of the embankments

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[45] Studies conducted within the framework of the 2007-2009 PNGMDR.

[46] Only the Bois Noirs Limouzat site comprises a layer of water and a project for replacement by a solid cover is currently being studied by AREVA

[47] The seismic movements derived from the methods applicable to so-called special risk installations are given a higher value. The regulations for the design of installations classified on environmental protection grounds, sets 3,000 years for the chosen hazard. AREVA specifies that the increase adopted for its disposal facilities would be to consider a seismic hazard with a return period of 30,000 years.

and defining the associated levels of requirements to guarantee the long-term safety of these disposal facilities.

2.2.4 Management of mining waste rock

2.2.4.1 Reuse of mining waste rock in the public domain

The waste rock from the former uranium mines comprises radioactive elements with varying levels which are generally low but which may sometimes be significant. Over the years, reuse of this waste rock in the environment may lead to usage of the soil no longer being compatible with the presence of this waste rock (for example if homes are built directly above such backfill). Without in principle systematically calling into question all past uses, it is nonetheless important to identify those places where waste rock with a significantly higher level of radioactivity than the natural background level has been reused and to check the compatibility of the usages directly above and in the immediate vicinity of the areas in which the waste rock has been used.

In accordance with the circular of 22nd July 2009 and the letter sent by the AREVA Chairman to the Minister responsible for the Environment on 12th July 2009, a survey of this type is currently underway at AREVA and should be completed in 2013. All usage incompatibility situations should be identified and managed. AREVA thus defined a methodology for assessing usage compatibility, in order to identify the areas in which waste rock is used in the public domain and which of them require corrective measures. This methodology, which is based on an assessment of the added effective doses associated with seven generic exposure scenarios[48], was reviewed by ASN and was the subject of a letter dated 14th May 2012. This method would on the whole appear to be pertinent and appropriate for the large-scale survey approach. The methods for managing any waste rock removed will be examined by the Ministry responsible for ecology.

AREVA thus carried our helicopter-borne measurement campaigns around the former mining sites in France between November 2009 and late 2010. The regions overflown are the *départements* of Creuse, Haute Vienne, Corrèze, Saône et Loire, Allier, Puy de Dôme, Lozère, Loire, Nièvre, Morbihan, Loire Atlantique, Maine et Loire and Vendée. The data were then statistically processed to identify the geographical areas requiring verification on the ground. No situation requiring emergency intervention was identified by this processing. The reconnaissance and ground verification phase began in 2011 and will run until mid-2013. The survey of waste rock in France will only be available once all ground reconnaissance actions have been completed. In each *département* concerned, the survey maps obtained will be presented to the CLIS and then made available to the public in the communes concerned by this reuse.

At the same time, a check will be run to ensure that soil usage is acceptable in the light of the health impact. Finally, in the event of any usage incompatibility, AREVA shall take steps jointly with the public authorities.

2.2.4.2 Management of long-term dose impact

The methodology to assess the long-term dose impact of the residue disposal sites was adapted by AREVA to the case of reuse of mining waste rock in the public domain and to spoil heaps, during studies submitted in 2009. If mining waste rock is reused as backfill in the public domain,

[48] Two indoor scenarios (home, company) and five outdoor scenarios (road or path, residential courtyard, farmyard, leisure area, public areas such as a car park).

the scenarios considered correspond to the cases encountered, that is paths and tracks, farmyards, schoolyards and company platforms. In the case of spoil heaps, four scenarios are studied; the normal evolution scenario and three altered evolution scenarios, that is residential construction on the spoil heap, a road-building construction site and a children's playground. Regardless of the scenario, the assessments made by AREVA identify a uranium level of 30, 60 and 100 ppm in the waste rock. The calculated doses thus obtained all remain below 1 mSv/year.

The content and exposure ranges actually encountered during the survey campaign conducted pursuant to the circular of 22nd July 2009, should be able to confirm or, as necessary, redefine, the exposure scenarios.

The main change introduced by AREVA into the document concerning the use of mining waste rock in the public domain, produced pursuant to the 2010-2012 PNGMDR, concerns the means of calculating radon concentrations in enclosed premises assumed to be built on waste rock, for the "company platform" scenario. This change, which is identical to that introduced into the method for assessing the long-term dose impact of residue disposal sites, leads to a significant rise in the assessment of exposure by inhalation of radon.

The CNE report of June 2009 specifies that there is very little chemical change in the waste rock used in the coverings or spoil heaps. The differences between the disposal sites for processing residues and for waste rock, require the definition of a specific model to characterise the waste rock, which will be applied to the spoil heaps chosen as being most representative. AREVA will then continue to study the phenomena of uranium transport from the spoil heaps to the environment, using the corresponding geochemical modelling simulating the various foreseeable disturbances as the disposal site evolves and which were developed for residue disposal sites.

2.2.5 Outlook[49]

2.2.5.1 Management of releases from former mining sites and water treatment

The new data transmitted by AREVA concerning water treatment and the impact of discharges constitute an inventory of the treatment practices and their radiological and chemical impacts on man and the environment. The results obtained, for both radiological and chemical substances (uranium, barium and aluminium) show the need to look at the pertinence of the current treatment processes for each of the former mining sites and to examine how they could be modified or shut down.

AREVA will consequently need to continue and expand on the approach already underway on all the treatment stations, in order to define and justify the development strategy adopted (shut down, maintain, modify or implement new processes) for processing of the water collected on the former mining sites under its responsibility. The strategy adopted should in particular be justified with respect to:

a. foreseeable natural changes in the quality of the water on each of the sites, in the light of the geochemical mechanisms involved and the trends observed to date;

[49] Opinion 2012-AV-0168 of 11th October 2012 on the assessment of the uranium mining residues impact and former uranium mining sites management report, is available on the website http ://www.asn.fr, heading "les actions de l'ASN", "la réglementation", "bulletin officiel de l'ASN", "avis de l'ASN".

b. the aim of reducing the overall impact of discharges on man and the environment. The chemical impact associated with the various substances discharged, including those linked to the water treatment processes, will more specifically be taken into account;
c. the constraints involved in managing and eliminating sludges and waste associated with the various processes used or studied;
d. maintenance constraints for the processes envisaged and the priorities regarding the deployment of possible alternative solutions.

An interim report will be transmitted by AREVA in late 2014. It will provide answers to points c and d (above) for all the sites and to points a and b (above) for a number of chosen sites. The final study will be submitted with the 2016-2018 PNGMDR.

It would also appear necessary for the PNGMDR working group to begin to look at the choice of continuing with treatment (while improving it whenever necessary) or putting an end to treatment in the light of the various criteria, especially the assessment of the overall impact (radiological and chemical) of the discharges on the receiving environment. This choice requires prior definition of foreseeable development scenarios for the characteristics of the water collected, taking account of the geochemical processes involved and the trends observed. This review could be jointly run by ASN and the DGPR.

By 31st December 2013, AREVA must also clarify the relationship between the flows discharged and the build-up of contaminated sediments in the rivers and, above all, the lakes, notably by means of a site study of the speciation of uranium in the water and the detailed radiological characterisation of the sediments according to their granulometry and to the hydraulic conditions of the watercourse.

2.2.5.2 Long-term management of uranium mining treatment residue disposal sites

<u>The long-term dose impact of the sites</u>

The studies submitted improved understanding of the "residence" on disposal site scenario, based on modelling of the radon flux penetrating a building and originating in the foundation materials, and allowed specific checks on the pertinence of the modelling in relation to the surveillance measurement data.

The variability of the natural background levels (dose rate or radon anomaly) is an element of interpretation that can explain differences between the modelling results and the surveillance data. The predominant influence of this variability in the situations encountered on the sites will need to be confirmed. **By 30th November 2014, AREVA will thus have to continue and ensure widespread adoption of the comparison approach between field data and modelling results, in order to improve the pertinence and effectiveness of its site monitoring system and increase confidence in its calculation results.**

AREVA has not submitted any data on the need to reinforce the covering of certain residue disposal sites. **Without prejudice to any local actions, AREVA will need to review this point by 30th November 2014, more specifically in the light of new knowledge obtained concerning the long-term behaviour of the disposal sites and the interpretation of the surveillance results.**

<u>Long-term strength of embankments</u>

The new data transmitted constitute a significant improvement in the assessment approach adopted by AREVA. This approach must be continued and **by 30th November 2014, AREVA will need to supplement its assessment with an embankment sensitivity study according to the target return period (that is 30,000 years) and the associated seismic hazard level.**

At the same time, ASN and the DGPR will run a working group to define the assessment doctrine for the long-term strength of the embankments surrounding the uranium ore processing residue disposal sites, on the basis of studies already transmitted on this subject by AREVA.

In addition to the work needed to formally define the doctrine, it would appear necessary to begin to create a complete geotechnical file (geological, geotechnical, hydrogeological) immediately for each embankment on the mining processing residue disposal sites, in order to gain a more precise assessment of the potential vulnerabilities highlighted by the generic assessment produced by AREVA. **AREVA will send the files that concern them to the DREALs and to the ASN divisions.**

This review, together with the environmental summary, should enable AREVA to produce a picture of the physical condition of the structures and indicate the countermeasures and compensatory measures required, on the basis of the evolution of the phenomena liable to affect the required safety level of each embankment.

AREVA must therefore define an action plan so that it can initially draw up the geotechnical files associated with each embankment and then assess their stability and determine any need to reinforce them, in particular in the light of their vulnerability to extreme hydraulic conditions, the seismic situations considered, or a combination of various natural hazards (period of high rainfall, high groundwater level, flooding, hundred-year or thousand-year flood levels, etc.). The action plan will more specifically, by 31st December 2013, specify the envisaged calendar and the list of sites today considered to be priorities.

<u>The dose impact of mining waste rock</u>

It is important for AREVA to base its approach to managing the risks associated with waste rock on as detailed an analysis as possible of the various situations encountered on and around the former mining sites.

Management of the risks **linked to the use of waste rock in the public domain** must therefore focus on measures to identify and characterise the land by comparison with the theoretical assessments made on the basis of generic situations. **In this respect, the approach implemented by AREVA pursuant to the circular from the Ministry responsible for the Environment and from ASN, dated 22nd July 2009, must be continued, taking account of the comments identified in the ASN requests expressed in its letter of 14th May 2012. Based on the observations made, an analysis of the results of this national survey campaign could confirm or, as necessary, redefine the exposure scenarios to be considered when assessing the impact associated with the presence of waste rock, based on the ranges of content and exposure levels actually encountered. This report will be submitted by 30th November 2014.**

With regard to the dose impact associated with the presence of a spoil heap, AREVA must utilise its knowledge of the sites, more specifically by means of the environmental assessment reports[50] required by the circular of 22nd July 2009, to identify the presence of spoil heaps and, by 30th November 2013, identify those with the most significant uranium levels, as well as the exposure levels with which they could be associated in the various foreseeable scenarios. These scenarios must take account of the utilisation or rehabilitation situations identified on these sites.

[50] The environmental reports correspond to the operating reports mentioned in the circular of 22nd July 2009.

2.3 Management of radioactive materials

A radioactive material is defined as being a radioactive substance for which subsequent use is planned or envisaged, if necessary after processing. These materials can be used in France or abroad. The 2013-2015 PNGMDR presents the circulation and stocks of the materials produced at the various stages in the fuel cycle, as well as where these materials are reused. The main ones are:

- **spent fuel**: Most of the spent fuel present on French soil is intended to be reprocessed. The uranium (reprocessed) and the extracted plutonium can be reused;
- **uranium:**
    - ***natural uranium***, Used by the enrichment plants to produce two types of substances: enriched uranium and depleted uranium;
    - ***enriched uranium,*** intended for the fabrication of fuels, to produce electricity of nuclear origin;
    - ***depleted uranium,*** which:
        - is used for the fabrication of the MOX fuel utilised in some EDF reactors;
        - can be re-enriched to higher levels, which can be economically advantageous if the price of natural uranium rises or in the event of developments in enrichment techniques;
        - in the longer term, it could be reusable on a large scale in generation IV fast neutron reactors;
    - ***recycled uranium from reprocessed spent fuel (URT),*** extracted from spent fuels, re-enriched to produce **enriched recycled uranium (URE)** used in the fabrication of fuel;
- **plutonium:** contained in spent fuel assemblies, extracted when they are reprocessed;
- **thorium**: could be used in various types of reactors as a fuel in a thorium cycle, but not for several decades yet, given the amount of research and development work still necessary. Other applications are currently under development, in particular for the treatment of certain cancers.

Within the framework of the studies requested by the 2010-2012 PNGMDR, the radioactive material owners supplied studies on the management options in the eventuality of the materials being classified as waste at some time in the future. The depth of the repositories envisaged by the producers in these studies makes them vulnerable to human intrusion and to the natural phenomena liable to occur over the long term. **The producers, together with Andra, must conduct further studies, taking account of the geological conditions most favourable to the confinement and isolation of the radionuclides in various normal and altered development scenarios**.

2.3.1 Context and issues

As specified in paragraph 1.1.1, Article L. 542-1-1 of the Environment Code, a radioactive material is defined as being a radioactive substance for which subsequent use is planned or envisaged, if necessary after processing.

Radioactive materials are exchanged with foreign countries at several steps in the fuel cycle, even when it is being managed by France. Exchanges of natural, depleted and enriched uranium in particular take place, because diversification of uranium supplies is a means of mitigating risks. It would not be sensible to entrust France's supplies of enriched uranium to a single plant, even a French one, and the same logic applies to the conversion and fuel fabrication steps. Any industrial facility can experience operational breakdowns which can interrupt production. In order to avoid any interruption of supplies, and while still primarily utilising AREVA, EDF has signed contracts with Urenco (enrichment company located in the Netherlands, Great Britain and Germany) and with Tenex (company located in Russia). Just like AREVA, these two companies have acquired expertise in enrichment techniques over many years. Furthermore, the industrial process used by Urenco and Tenex is known (gas centrifuge process), and was adopted by AREVA for use in the Georges Besse II plant in Pierrelatte.

In 2009, the Minister for Ecology, Energy, Sustainable Development and the Sea, responsible for green technologies and for climate negotiations, and the Parliamentary Office for the Evaluation of Scientific and Technological Choices asked the French High Committee for Transparency and Information on Nuclear Security (HCTISN) to review the question of the international exchanges involved in the processing of uranium and requested its opinion on the transparency of the management of nuclear materials and waste produced at the various stages in the fuel cycle. The HCTISN submitted its report in July 2010[51]. It presents a detailed analysis of the fuel cycle in France. It gives the traffic and stocks of materials and waste produced at the various stages in the fuel cycle, as well as the conditions for storage and transport of depleted uranium and recycled uranium produced by reprocessing of spent fuels, while presenting the issues involved in the supply of uranium and France's policy to ensure the security of these supplies in an international context. It also draws on the 2010-2012 edition of the PNGMDR. The last part of this report is devoted to the quality of the information delivered to the public. Finally, the High Committee makes recommendations for improving the transparency and quality of the information provided to the public.

Pursuant to the recommendations expressed by the HCTISN[52], the Ministry responsible for energy sends the High Committee an annual inventory of traffic and stocks of materials produced at the various stages of the fuel cycle. These data are summarised in the following tables.

Exports

Reusable material	**Reference data (source: HCTISN report on the transparency of the cycle)**		**2010 and 2011 data (source: data transmitted by the DGEC to the HCTISN in 2011 and 2012)**		**Data updated at end 2012**	
	Years 2006-2009: total exports		Total exports in 2010 (in tons)	Total exports in 2011 (in tons)	Years 2006-2011: total exports	
	Total 2006-2009 (in tons)	Average annual traffic (in tons)			Total 2006-2011 (in tons)	Average annual traffic (in tons)
Depleted uranium	34,229	8,557	6,863	4,507,	45,599	7,600
Natural uranium	20,845	5,211	8,321	8,384,	37,550	6,258
Slightly enriched Uranium	7,394	1,849	2,032	1,011	10,437	1,740
Highly enriched uranium	2	ns[53]	0	ns	2	0
Plutonium	9	ns	2	1	12	2
Thorium	12	ns	2	2	16	3

[51] The complete HCTISN report is available on the following link: http://www.hctisn.fr, heading, "les avis et rapports du Haut Comité".
[52] Recommendation n°5 given in the opinion submitted on 12th July 2010 by the HCTISN.
[53] ns = not significant

Imports							
Reusable material	**Reference data (source: HCTISN report on the transparency of the cycle)**		**2010 and 2011 data (source: data transmitted by the DGEC to the HCTISN in 2011 and 2012)**		**Data updated at end 2012**		
	Years 2006-2009: total imports		Total imports in 2010 (in tons)	Total imports in 2011 (in tons)	Years 2006-2011: total imports		
	Total 2006-2009 (in tons)	Average annual traffic (in tons)			Total 2006-2011 (in tons)	Average annual traffic (in tons)	
Depleted uranium	17,465	4,366	7,330	6,924	31,719	5,287	
Natural uranium	44,680	11,170	8,238	11,016	63,934	10,656	
Slightly enriched Uranium	7,194	1,799	1,720	1,415	10,329	1,722	
Highly enriched uranium	0	0	0	0	0	0	
Plutonium	2	ns	0	ns	2	0	
Thorium	0	ns	0	ns	0	0	

Summary of direct exports and imports with all countries concerned by the nuclear industry

2.3.2 Presentation of current management methods, assessment of reusability of materials and reuse solutions

As recalled in chapter 1.1.1, a radioactive material is defined as being a radioactive substance for which subsequent use is planned or envisaged, if necessary after processing. The classification of a radioactive substance in this category is thus decided on by its owner[54], on the basis of its existing utilisation or the foreseeable industrial prospects. If this substance has not yet been used industrially, it is said to be recoverable.

To date, seven main categories[55] of radioactive materials have been identified:

- spent fuels;
- natural uranium ;
- enriched uranium;
- depleted uranium;
- recycled uranium from reprocessed spent fuel;
- plutonium;
- thorium.

Data concerning the storage of radioactive materials are given in the "summary of fuel cycle impact 2007"[56] file, which more specifically presents the fuel cycle monitoring and anticipation approach, with the aim of defining requirements in terms of storage facilities for these materials.

Article 13 of decree 2008-357 of 16th April 2008 implementing Article L. 542-1-2 of the Environment Code and setting requirements concerning the National Plan for Radioactive

[54] The ministers may recategorise materials as waste in accordance with the provisions of Article 9 of the decree of 23rd April, implementing Article L. 542-1-2 of the Environment Code and establishing the stipulations of the National Plan for radioactive materials and waste management.
[55] The case of used sealed sources is dealt with in chapter 3.1.3.
[56] The elements concerning the "impact cycle 2007" file are available on the ASN website http://www.ans.fr, heading "les actions de l'ASN", "les appuis techniques", "les groupes permanents d'experts", " groupe permanent d'experts pour les déchets".

Materials and Waste Management, had asked that the owners of reusable radioactive materials for which the reutilisation processes had never been implemented, submit a summary of the studies on the reuse processes they envisage employing, by the end of 2008. In accordance with this request, AREVA, CEA and EDF submitted studies concerning the uranium-bearing, plutonium-bearing and thorium-bearing materials in their possession. Rhodia also submitted a study concerning the processes for reusing its suspended solids (SS), its raw thorium hydroxides (HBTh) and its thorium nitrates. These studies presented the processes for reuse of materials present in France, some of them being implemented on an industrial scale, while others are waiting for economic opportunities or subsequent developments. Updates were transmitted by the same four licensees at the end of 2010. The summary of the long-term management of reusable materials presented below was more specifically produced on the basis of these studies and their updates. The situation can be concisely described as follows:

Nature of materials		**Approximate quantities in France at end 2010**[57]	**Status of reuse**
Spent fuels	EDF fleet natural uranium fuels	19,000 tons	established reuse on an industrial scale
	EDF fleet MOX and URE fuels		industrial scale feasibility demonstrated for reprocessing for recycling in generation IV reactors
	naval propulsion fuels		feasibility demonstrated for reprocessing for recycling
	research reactor fuels		reprocessing feasibility demonstrated for most of these fuels. However, the management strategy (reprocessing) for the fuels from the OSIRIS, ISIS, ORPHEE reactors must be clarified by CEA and the EL4 spent fuels from Brennilis were classified as radioactive waste by their owner, EDF.
Natural and enriched uranium		19,000 tons	established industrial use
Depleted uranium		272,000 tons	established use as-is (MOX fuel) reuse by re-enrichment achieved, feasibility of use in generation IV reactors demonstrated on industrial scale
Recycled uranium from reprocessed spent fuel (URT)		24,100 tons	established industrial use (recycled as URE fuel)
Plutonium		80,000 tons	established industrial use (recycled as MOX fuel)
Thorium		8,500 tons[58]	significant industrial experience with processing of thorium-bearing materials and electricity production based on a thorium cycle industrial development in progress for medical requirements

Reusable materials, associated quantity and status of reuse

Spent fuels

Most of the spent fuels present in France are from pressurised water reactors (PWR) either undergoing burn-up in the EDF reactors, or removed from these same reactors and stored in a pool (on the reactor sites themselves or in the AREVA plant at La Hague). Little storage capacity

[57] Data of the 2012 edition of the National Inventory.

[58] The thorium value is different from that of the 2012 edition of the National Inventory (IN), Rhodia having declared to the IN the quantities of thorium hydroxide in its possession rather than thorium.

is available and it could run out by 2020, or even earlier if problems arise with the cycle. EDF and AREVA are envisaging increasing this capacity.

Reuse of the spent fuels from the nuclear power generating reactors, through reprocessing in the La Hague plant and recycling of the materials separated in it, is already extensively utilised industrially for the UOX (Uranium OXide) fuels, consisting of pellets of uranium dioxide.

With regard to URE fuels (fuels based on enriched URT), the feasibility of their reprocessing on an industrial scale was demonstrated in 2006. Given the nature of the materials separated out, the benchmark industrial management process for spent URE fuels is recycling in fast neutron generation IV reactors. With this in mind, spent URE fuels are currently stored, awaiting use as and when needed.

With regard to spent MOX (mixed oxides) fuels, consisting of plutonium dioxide and depleted uranium dioxide, about 70 tons of spent MOX fuels from PWRs have already been reprocessed in the La Hague plant since it opened. Several tens of tons of spent MOX fuels from fast neutron reactors (FNR) have been reprocessed at La Hague and in Marcoule. The industrial feasibility of reprocessing MOX PWR and MOX FNR fuels has been established. Given the isotopic characteristics of the plutonium contained and the quantities of spent MOX fuels unloaded from the French nuclear reactors, the industrial management of these fuels today adopted by the French electricity utility is recycling in the generation IV fast neutron reactors. With this in mind, the plutonium is currently being kept within the spent MOX fuels, until needed.

For fuels from research reactors, a special technique is used in the La Hague plant to reprocess the fuels from certain reactors (IN2P3 in Strasbourg and CEA's SILOE, SILOETTE, ULYSSE and SCARABEE reactors). Moreover, the "caramel" type fuels (sandwich of enriched uranium alloy between two metal plates) currently used in other research reactors (OSIRIS, ISIS, ORPHEE) are intended for reprocessing, even if no reprocessing campaign has as yet taken place in the La Hague plant. "Caramel" type fuels from the OSIRIS research reactor have been reprocessed by CEA Marcoule as part of the post-operational clean-out process. 2.3 tons of UO_2 were reprocessed, thus validating the concept. Finally, CEA possesses other types of research reactor spent fuels: fuels from the CABRI and PHEBUS reactors will be reprocessed in the existing facilities.

Spent fuels from nuclear propulsion are similar to the "caramel" fuels, for which reprocessing poses no particular dissolution-related problems, given the abovementioned experience feedback. It should however be noted that industrial scale reprocessing will require new equipment in the La Hague plant.

Thus, most of the spent fuels are reusable materials. Only spent fuels from the Brennilis EL4 reactor, representing only a small volume (27 m^3) and with insufficient potential for reuse, can be considered as waste.

Uranium

Enriched uranium and depleted uranium are produced by the uranium enrichment plants, which produce two types of substances: on the one hand, uranium enriched with uranium isotope 235, generally to between 3 and 5%, which will be used to fabricate fuels and, on the other, uranium depleted in uranium isotope 235 which is present at a level of about 0.4% or less. The enriched uranium is intended for the fabrication of fuels, to produce electricity of nuclear origin.

In practice, the enricher becomes the owner of the depleted uranium. AREVA thus takes possession of the depleted uranium resulting from the uranium it enriches, whether uranium from EDF or from a foreign customer (American, German, Korean, etc.). The depleted uranium[59] is stored on the sites at Pierrelatte (about 170,000 tons) and Bessines-sur-Gartempe (about 100,000 tons) in the form of oxide packaged in containers, with about 6 to 12 tons of oxide per container of about 3 m^3. The storage facilities should not become saturated until about 2016-2020. Storage capacity will need to be increased before then.

There are three possible uses for depleted uranium:

- the depleted uranium has regularly been used for several years as a support matrix for MOX fuel (fuel consisting of a mixture of uranium and plutonium, produced in France in the MELOX plant in Marcoule in the Gard *département*). This traffic represents about a hundred tons per year (given that about 270,000 tons of depleted uranium are currently stored in France);
- it may also be economically interesting to re-enrich depleted uranium to higher levels if the price of natural uranium rises or in the event of developments in enrichment techniques. In concrete terms, in 2008 and 2009, at a time when the natural uranium market so allowed, 7,800 tons of depleted uranium were used to obtain the equivalent of 1,800 tons of natural uranium. In the medium term, one could thus envisage the current stocks of depleted uranium (here referred to as "primary" Uapp) being re-enriched over a time-frame of about 30 to 50 years. New stocks of depleted uranium, secondary Uapp (with enrichment levels of about 0.1 to 0.2%), would then be created. New technologies, such as laser enrichment, could eventually allow even more advanced separation, thus producing tertiary Uapp (with an enrichment objective of less than 0.1%);
- finally, in the longer term, the stocks of depleted uranium could be reusable on a large scale in the fast neutron generation IV reactors, which could come on-stream in the middle of the century. This type of reactor can draw full benefit from the energy potential of uranium isotope 238.

Recycled uranium from reprocessed fuel (URT) is extracted from spent fuels. It constitutes about 95% of the mass of the spent fuel. For the EDF reactor fleet, its energy value is comparable to that of natural uranium, with a fissile 235 isotope content of about 0.8%. At the request of the customers, this reprocessed uranium can be sent to the enrichment plant to produce enriched recycled uranium (URE) used in the fabrication of nuclear fuels. To date, of the 1,000 tons separated annually by reprocessing of spent fuels from the French nuclear reactors, up to 650 tons per year are re-enriched in place of natural uranium, which enables four reactors on the Cruas site to be supplied.

Until now, the French URT has been re-enriched abroad (Russia and Netherlands) because the technology employed in the Eurodif plant in France was unable to do this. The Georges Besse I plant, today shut down, was dedicated to the enrichment of natural uranium. The Georges Besse II plant, which uses a different isotope enrichment technology (gas centrifuge) was authorised by modified decree 2007-631 of 27th April 2007, to enrich URT in one of the North unit's modules. The South enrichment unit at GB II has been in service since 2011 and exclusively enriches natural uranium. Partial commissioning of the North unit, for non-destructive testing of the isotopic control system, was authorised by ASN in October 2012. The industrial commissioning application for the North unit is currently being examined by ASN. Natural uranium will probably be introduced into the first cascade of centrifuges in the North unit in early 2013. The

[59] According to the National Inventory data, the production of depleted uranium (between 2007 and 2010) stands at about 5,500 tons per year.

Société d'Enrichissement du Tricastin will be able to enrich URT, subject to prior authorisation from ASN, but does not anticipate doing so for the next few years. The GBII support facility, REC II, which is scheduled to be commissioned in the second half of 2013, could also receive URT in accordance with the provisions of decree 2011-1949 of 23rd December 2011.

The URT not immediately recycled is stored at Pierrelatte (about 24,000 t) in the form of uranium oxide powder. This is a strategic stock which can be used in various situations: when the economic conditions in its sector are more favourable than the use of natural uranium; when a choice has to be made between present and future uses, for example with regard to security of supply; depending on ASN authorisations for EDF reactors to operate with URT fuel. Without enrichment, URT is qualified for use in heavy water reactors. URT can also be used in the generation IV fast neutron reactors. New URT storage capacity will be needed by about 2020.

Plutonium

In the same way as uranium, the plutonium contained in spent fuel assemblies is extracted during reprocessing. A light water type uranium spent fuel today contains about 1% of plutonium by mass. Once dissolved, extracted and separated from the other materials contained in the spent fuel, the plutonium is purified and packaged at AREVA NC La Hague in a stable PuO2 oxide powder form. Plutonium is at present recycled in MOX fuel. This fuel consists of pellets of (UPu)O2 oxide powder produced from depleted uranium acting as the support, plus plutonium.

In France, the MOX fuel used by EDF in 22 reactors accounts for about 10% of domestic nuclear power generation. The plutonium requirements for fabrication of MOX fuel, standing at about 10 tons per year, determines the annual traffic of EDF spent fuels reprocessed in the La Hague plant by AREVA.

In the longer term, the plutonium could also be used in the generation IV fast neutron reactors.

Thorium

AREVA, CEA and Rhodia are in possession of about 8,500 tons of thorium, in nitrate and hydroxide form. These materials are stored on the sites at La Rochelle (about 6,200 t) and Cadarache (about 2,300 tons).

By neutron capture, thorium can transmute into fissile uranium 233. A "thorium cycle", using thorium as a fuel and based on recycling of thorium and uranium 233 without using uranium 235 or plutonium, could possibly be envisaged but not for the next few decades, in the light of the research and development work still necessary. A gradual introduction of thorium into the reactors, in order to improve the uranium-plutonium cycle, would allow multi-recycling of these fissile materials. This is actually conceivable in the shorter term, depending on needs and on how the natural uranium market develops.

Thorium has no established use on an industrial scale today, but AREVA and Rhodia anticipate considerable potential, justifying the classification of the stored quantities as material.

In their 2008 study, EDF, AREVA and CEA identified the thorium reuse processes, in particular those which have been applied on an industrial scale in the past. In a 2009 opinion, ASN expressed doubts as to the potential use of thorium in power generating reactors, considering that significant developments were necessary and that the benefits of thorium over uranium in the generation IV reactors were anything but certain.

AREVA initiated an R&D programme involving several phases and concerning reactor applications, with a view to confirming the potential industrial uses of thorium. The first phase was finalised in 2010 in order to identify the potential uses of thorium in a light water reactor. The second phase, scheduled to last two years, should analyse the available technologies and the business plans linked to the use of thorium based fuels, in order to confirm and further detail the results obtained during the first phase. The studies conducted by AREVA with various partners in France and elsewhere, focus on the gradual introduction of thorium, and thus of uranium 233, into the fuel assemblies in order to improve the uranium-plutonium cycle and the multi-recycling of these fissile materials. This is in line with AREVA's goal of addressing various aspects of global demand for fissile and fertile material management, more specifically from countries looking to deploy other options in synergy with third and fourth generation reactors. On this basis, AREVA envisages subsequently establishing a more wide-ranging R&D programme, including tests on the behaviour of the fuel under irradiation, fabrication and processing experiments, and analysis of thorium fuel deployment scenarios.

At the same time, AREVA is continuing to develop its TAO project. This project consists in producing radium 224 from thorium, to support the development of an innovative form of tumour targeting therapy, known as "lead 221 alpha radio-immunotherapy". Pre-clinical trials are being run in the United States with radium 224 produced by a pre-industrial pilot in operation at Bessines-sur-Gartempe. An industrial facility to produce radium 224 is also being built on this same site, for clinical trials scheduled to last several years. INSERM, in France, is also interested in this technique.

With regard to raw thorium hydroxide (HBTh) (containing thorium, uranium and rare earth oxides) and thorium nitrate, the Rhodia study more specifically concludes that the main steps in the process have already been used industrially and that the economic results of processing show that a price of a few tens of euros per kg of ThO_2 is enough to make the investment profitable.

<u>Suspended solids (SS)</u>

The Rhodia radioactive materials also comprise suspended solids (containing rare earth oxides and traces of thorium and uranium). The study submitted by Rhodia concludes that reuse of these suspended solids is technically and economically feasible. Rhodia began their recycling in 2010. All of it should be recycled by 2020.

2.3.3 Management options if the materials were in the future to be classified as waste

Article 10 of decree 2012-542 of 23rd April 2012 implementing Article L. 542-1-2 of the Environment Code and establishing the requirements of the National Plan for Radioactive Materials and Waste Management, instructed the owners of materials *"to conduct interim studies before the end of 2010, on the possible management solutions if these materials were in the future to be classified as waste"*, primarily considering the hypothesis of nuclear power programmes being discontinued both in France and abroad. The work submitted by the owners of the materials at the end of 2010 concerned depleted uranium, recycled uranium from reprocessing and thorium. The feasibility of disposal of the spent fuels as-is in a deep geological facility was proven by Andra in

2005, and uranium (natural and enriched) and plutonium are part of the basic inventory of the current nuclear power industry.

Depleted uranium

In France, depleted uranium is primarily the property of AREVA, held on the sites at Pierrelatte (about 170,000 tons) and Bessines-sur-Gartempe (about 100,000 tons). The current specific activity of the depleted uranium is estimated by AREVA at 40,000 Bq/gU. According to AREVA, the presence of fissile uranium, postpones the radiological peak (about 110,000 Bq/gU) which will be reached three million years after its production.

Depleted uranium is not accepted in the existing radioactive waste disposal centres, given the activity level of uranium in 300 years' time (alpha activity). The lifetime of this material means that containment until decay of the source term cannot be adopted as a safe option.

The study conducted by AREVA for the 2010-2012 PNGMDR consisted in determining the environmental conditions in which depleted uranium must be placed in order to minimise the release of radionuclides liable to have an impact on the health of the surrounding populations if they consume drinking water. The work done by the World Health Organisation (WHO) and the French Agency for Food Safety (AFSSA) led AREVA to select the chemical toxicity of depleted uranium as the predominant criterion, which means a uranium concentration limit in drinking water of 15 micrograms per litre at the time of the study. It should be noted that the WHO guideline value was recently revised up to 30 micrograms per litre[60].

The study conducted by AREVA focused on determining the geochemical conditions that are most favourable to long-term confinement of the depleted uranium, and then the hydro-geochemical conditions liable to permanently reduce the flow of uranium into the biosphere to acceptable values.

AREVA thus determines the main hydro-geochemical characteristics to be met for the disposal facility and its geological environment. It identifies disposal:

- in a clay medium more than 40m thick;
- with the presence of calcite in the clay under the disposal facility, to precipitate the residual fluorine in the form of fluorite, thus eliminating the main source of uranium solubility;
- in reducing conditions to trap a large percentage of the radionuclides and maintain the uranium level below 15 g/L;
- in a saturated environment with low permeability and a low hydraulic gradient, in order to limit the movement of the water masses;
- in which the underlying aquifer is at least 10 m below the buffer consumption zone of the fluoride contained in the disposal facility.

According to AREVA, these many conditions are in fact relatively common hydro-geochemical conditions in France, especially, for example, as very low pyrite levels would be enough to meet the necessary reducing condition.

[60] World Health Organization "Guidelines for drinking water quality" – 4th edition, 2011 – table 8.8

AREVA concludes that the depleted uranium disposal facility is thus conceivable, considering that the necessary characteristics of the geological medium for this facility are on the whole less demanding than those being sought for the disposal of LLW-LL waste.

Recycled uranium from reprocessed spent fuel (URT)

EDF and AREVA possess about 13,000 and 7,000 tons of URT respectively. The chemical composition of URT depends on the reprocessing it has undergone. Its isotopic characteristics vary according to the characteristics of the spent fuels from which the URT was extracted. The radiological characteristics of the URT, the specific activity of less than 100,000 Bq/gU and the presence of long-lived radionuclides, led EDF and AREVA to conduct a pre-feasibility study on the basis of a shallow disposal concept under reworked cover, similar to the concept being studied by Andra for low level, long-lived waste.

AREVA and EDF studied the preliminary feasibility of a reworked cover disposal facility situated within a clay formation, at least 35 m above an aquifer formation, which would be covered with materials (clay, backfill, earth) about fifteen metres thick. The packages would be stored in concrete vaults within the clay formation (the licensees specify that no performance specifications have been set for the concrete). This concept is similar to the reworked cover concept selected by Andra for the LLW-LL type waste disposal project. The assessment of the radiological and chemical impact of such a disposal facility presented by the licensees identifies diffusion in a water-saturated reducing environment as being the main transfer mechanism in the host rock and in the covering. The outlet identified for assessment of the impacts is a river.

The radiological impact assessment conducted by EDF and AREVA leads to a dose of less than 0.1 microsieverts per year. This value is far below the limit of 0.25 millisieverts per year which appears in the General Safety Guidelines for the disposal of LLW-LL waste issued by ASN in 2008[61]. The studies also showed that the chemical impact associated with the uranium would be acceptable in the case of sub-surface URT disposal. EDF and AREVA conclude that sub-surface disposal could be a management method for URT, if this material were to become considered as waste.

Thorium

In response to the request in the 2010-2012 PNGMDR, Rhodia and AREVA examined the procedures for long-term management of thorium if it were one day to be classified as radioactive waste.

The materials are stored in various forms: thorium nitrate, thorium hydroxide, etc. They also contain a small proportion of uranium 238. Their radiological characteristics (the specific activity of thorium nitrate is about 5,000 Bq/g and that of thorium hydroxide is about 2000 Bq/g) make them incompatible with surface disposal. AREVA and Rhodia adopted as the reference concept that studied by Andra for the disposal of radium-bearing waste. The calculations made by AREVA and Rhodia showed that the thorium and uranium concentrations in the environment, in normal conditions, would remain below 1 microgram per litre, a value far below the WHO recommendations for drinking water.

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[61] The guide can be consulted on the ASN website http://www.asn.fr, heading “Les actions de l'ASN”, “La réglementation”, “Règles fondamentales de sûreté et guides de l'ASN”, “Guides de l'ASN et RFS relatifs aux INB autre que les réacteurs”.

AREVA and Rhodia therefore conclude that a sub-surface disposal facility accepting "radium-bearing" type waste would be a possible method of thorium management if this material were one day to be considered as waste.

2.3.4 Outlook[62]

Reuse of the majority of radioactive materials presupposes that nuclear power generating programmes are continued in the future in France or abroad. Insofar as there can at present be no guarantee that this condition will continue to be met in the very long term, the owners of radioactive materials examined the possible management options for uranium-bearing materials (depleted uranium and recycled uranium from reprocessing) and thorium-bearing materials (thorium nitrate and thorium hydroxide) if they were one day to be considered as waste.

The specific activity of the materials concerned (from a few kBq/g up to a few hundred kBq/g) and the presence of long-lived radionuclides (up to several billion years) led AREVA, CEA, EDF and Rhodia to opt for sub-surface disposal similar to that envisaged by Andra in 2009 for management of LLW-LL waste. The licensees have identified disposal within a low-permeability clay formation such as to limit the migration of radionuclides. The assessments of the radiological and chemical impact presented by the licensees identify diffusion and convection in a water-saturated reducing medium as being the transfer mechanism and conclude that there is negligible radiological and chemical impact.

Given the relatively specific activity of the uranium-bearing and thorium-bearing materials considered, in particular the depleted uranium and the recycled uranium from reprocessed spent fuel, and the long half-lives of the radionuclides contained in these materials, the depth envisaged by the owners makes the disposal facilities vulnerable to human intrusion and to natural phenomena liable to occur over the long-term, and there can be no guarantee of durably maintaining conditions favourable to limiting the release of radionuclides.

Consequently, **AREVA, CEA, EDF and Rhodia, together with Andra, will need to conduct further studies into the disposal of these materials if at some time in the future they are categorised as waste, providing assessments of the radiological and chemical impact, taking account of the water, air and soil transfer paths for normal evolution scenarios, plus impact assessments for altered evolution scenarios.** Compliance with the safety guidelines stipulated by ASN for this type of disposal centre must also be justified in these studies.

With regard to the disposal of depleted uranium and recycled uranium from reprocessed spent fuel, AREVA, CEA and EDF will need to conduct further studies, taking account of geological conditions more specifically favourable to the confinement and isolation of materials if they were to be categorised as waste at some time in the future, over a period that is as long as possible and by assessing the consequences of geodynamic and climatic phenomena on these conditions.

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[62] Opinion 2012-AV-0156 of 26th June 2012 concerning radioactive waste management routes if they were in the future to be qualified as waste is available on the website http://www.asn.fr, heading "les actions de l'ASN", "la réglementation", "bulletin officiel de l'ASN", "avis de l'ASN".

For thorium-bearing materials, AREVA and Rhodia identify the principle of a reworked cover disposal facility. The lower specific activity of thorium nitrate, and more particularly thorium hydroxide, means that this type of disposal could be envisaged, according to the complete inventory of waste which would ultimately depend on the repository and the site conditions.

In order to situate the management of thorium hydroxide and, more generally, the uranium and thorium-bearing materials in an overall waste management system, if they were at some time to be re-categorised as waste, AREVA, CEA, EDF and Rhodia must keep a detailed radiological and chemical inventory of the materials concerned at the disposal of Andra and ASN, in particular so that their acceptability in the planned disposal routes can be analysed.

The completion deadlines for the above-mentioned studies will be determined according to the status of these materials and in particular those which could be re-categorised as waste.

2.4 Waste management by radioactive decay

Waste management by radioactive decay is for waste in which the radionuclides have a half-life of less than 100 days, originating mainly in nuclear medicine departments and research laboratories and sent to routes dedicated to conventional waste management. This is also an intermediate step in the management of certain radioactive waste produced in the basic nuclear installations, in particular waste containing tritium. This management method requires the construction of appropriate storage facilities.

2.4.1 Management of radioactive waste with a half-life of less than 100 days

The management of radioactive waste by radioactive decay on the production site is reserved for waste in which the radionuclides have a radioactive half-life of less than 100 days[63]. The aim is to wait for the activity of the waste to have decreased enough for it to be sent to a conventional management route. The main establishments concerned by this type of management are nuclear medicine departments and research laboratories.

The changes to the regulations made in 2008 confirm this approach. The management of this waste is regulated by ASN resolution 2008-DC-0095[64] of 29th January 2008 setting the technical rules for the elimination of effluents and waste contaminated by radionuclides, or liable to be so contaminated as the result of a nuclear activity, implementing the provisions of Article R.1333-12 of the Public Health Code. It can only be sent to a conventional waste route after a period ten times longer than the half-life of the radionuclide (if several radionuclides are present, then the longest radioactive half-life is considered). This resolution in particular makes provision for the implementation of a contaminated waste management plan comprising the management procedures within the establishment concerned, the identification of the places intended for storage of the contaminated waste and the provisions for ensuring their disposal in the appropriate routes and the associated oversight and monitoring procedures.

This resolution was the subject of ASN guide n°18[65] concerning the disposal of effluents and waste contaminated by radionuclides, produced in facilities notified or authorised under Article L.1333-12 the Public Health Code. This guide explains the requirements and more specifically the technical measures to be taken for the storage of waste as well as the specific procedures applicable to nuclear medicine departments.

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[63] Or when daughter products of these radionuclides are not themselves radionuclides with a half-life of more than 100 days, but for which the ratio between the parent nuclide half-life and that of the daughter radionuclide is less than the coefficient 10^{-7}.

[64] Resolution 2008-DC-0095 of 29th January 2008 can be consulted on the ASN website at the following address: www.asn.fr, heading "la réglementation", "bulletin officiel de l'ASN".

[65] Guide n°18 is available on the ASN website at the following address: www.asn.fr, heading "publication", "guide pour les professionnels".

2.4.2 Management of radioactive waste by decay prior to disposal

Some wastes need to be stored for several half-lives before being sent to routes dedicated to long-term radioactive waste management, because they do not meet the acceptance criteria for these routes or the transport criteria. This is in particular the case of waste containing tritium (for which the half-life is 12.3 years) in significant quantities, which will have to be stored for about fifty years before being sent to existing or planned disposal routes. These aspects are also described further in § 3.1.2.

Similarly, the management of certain high level or low and intermediate level, short-lived waste consists in decay storage before acceptance in an existing or planned disposal facility. CEA's currently operational DIAM facility, and the planned DIADEM facility at CEA and ICEDA at EDF, will allow the storage of solid waste which, after radioactive decay, will join the LLW/ILW-SL disposal route. In the case of ICEDA, the criterion which requires storage of EDF's ILW-SL waste for decay is primarily linked to the transport regulations applicable to type IP-2 packages (dose rate less than 10 mSv/h at 3 m from the bare waste).

Appendix [3] concerning waste storage describes the above-mentioned facilities.

2.5 Reusing radioactive waste

Recycling reusable materials extracted from waste is an area for particular focus, pursuant to the fundamental principles laid down in Article L.541-1 of the Environment Code. In France, there are two recycling alternatives for reusing radioactive waste in the nuclear facilities, which were commissioned in the 2000s:

- Centraco melting facility for recycling ferrous metal waste in the form of radiation shielding materials integrated into concrete containers for the fabrication of radioactive waste packages;
- the lead recycling route which, after decontamination, enables the lead to be made into shielding material.

However, there are questions over the future operation of these solutions: the Centraco melting facility has been stopped since the 12th September 2011 accident, while closure of the lead recycling unit is scheduled for 2013.

The studies conducted in compliance with the 2010-2012 PNGMDR identified two reuse options, one for recycling VLL rubble as infill material for the disposal facility at the industrial centre for collection, storage and disposal (Cires), the other for recycling of VLL ferrous metal waste to construct cast iron radioactive waste packages. However, the industrial feasibility of this was not confirmed by the studies conducted, in particular with regard to the recycling of ferrous metal waste.

Given the rising VLL waste management needs created by the forthcoming decommissioning and post-operational clean-out operations, confirmed by the 2012 edition of the national inventory, the 2013-2015 PNGMDR requires the continuation of studies into the implementation of these reuse solutions, in order to preserve scarce disposal site resources.

2.5.1 Context and issues

With a view to ensuring sustainable development, primarily to preserve the use of resources and improve how efficiently they are used, preference must be given to recycling the reusable materials extracted from the waste. This objective is one of the fundamental principles of the Environment Code, contained in Article L.541-1 which ranks the preferred waste treatment modes (preparation for reuse, recycling, reuse and disposal). This practice is common for conventional waste, especially metal waste: in France, about 10 million tons of metal waste is recycled every year, to be compared with the 15 million tons produced.

At a European level, waste from the nuclear facilities can be recycled after clearance in the conventional domain, pursuant to Directive 96/29[66] and the associated technical recommendations[67]. Two melting technologies dedicated to radioactive metal waste (Studsvik in

[66] Council Directive 96/29/Euratom of 13th May 1996 setting basic standards for health protection of the population and workers against the dangers of ionising radiation.

[67] The technical recommendations published by the European Commission are available on the website: http://ec.europa.eu/atoz_en.htm, heading "energy", "nuclear energy", "radiation protection". This in particular concerns the following reports:

Sweden and Siempelkamp in Germany) are operational and recycle metal waste in the nuclear or conventional fields, based on the provisions of Directive 96/29/Euratom.

In France, as mentioned in part 1 of this report, the waste management rules within BNIs and INBS require the zoning of waste, differentiating between conventional waste zones and nuclear waste zones, in which the waste produced is contaminated, activated, or liable to be so. Conventional waste from the conventional waste zones is managed in conventional waste management routes, while waste from the nuclear waste zones must be the subject of special, reinforced management in duly authorised establishments.

Radioactive waste industrial recycling technologies have been developed by AREVA, CEA or EDF, sometimes via subsidiaries of these companies. They are described in section 2.5.2. They concern very low level or low level waste. However, the volumes recycled within these facilities are marginal and represent no significant gain in terms of preserving disposal resources in the industrial collection, storage and disposal centre (Cires).

The increasing VLL waste management needs, more specifically linked to decommissioning / post-operational clean-out operations, justified the study of new means of optimisation. Consequently, the examination of possible ways of reducing the quantities of waste delivered to the disposal facilities, by reusing a part of it, becomes of paramount importance. It justifies the fact that, in accordance with Article 5 of the order of 23rd April 2012, implementing decree 2012-542 of 23rd April 2012 concerning the PNGMDR, a request was made for "a joint study on the benefits and technical-economic feasibility of reuse in the nuclear sector of very low level waste, more specifically metal waste and crushed materials".

2.5.2 Presentation of existing reuse routes and those being studied

In France, there are currently only two operational industrial solutions for recycling radioactive waste:

- a reuse technology reserved for low level ferrous metals, in the CENTRACO melting facility;
- a reuse technology for lead, combining a melting furnace in the Marcoule centre and shaping installations temporarily dedicated to radioactive waste and operated in conventional installations classified on environmental protection grounds.

The materials recycled in these facilities are reused in the nuclear sector. Other attempts to create metal waste reuse solutions, during particular decommissioning worksites, have been conducted, but without leading to any operational industrial options.

- Radiation Protection 89 – recycling of metals, which proposes clearance thresholds for the recycling of metals from dismantling of nuclear facilities;
- Radiation Protection 122 which proposes thresholds for unconditional clearance of solid materials;
- Radiation Protection 113, which proposes mass and surface clearance thresholds for the demolition or reuse of buildings.

2.5.2.1 Recycling of metals

Existing technologies

The melting unit in the CENTRACO plant in Marcoule, operated by SOCODEI, processes VLL and LL metal waste: metal structures, valves, pumps, stainless steel, steel and non-ferrous metal tools used in maintenance and decommissioning of nuclear facilities. The metal waste is sorted and prepared (sorting, cutting to size, etc.) then melted at 1600°C in an electric induction furnace with a capacity of four tons. The furnace is designed to process 4,500 t/year and the average production since 1999 is about 1,700 t/year.

Depending on the physical-chemical characteristics of the metal waste, the melting unit can:

- reduce the volume by melting and moulding of low level metal waste into ingots that can be accepted by surface disposal facilities. The volume of ultimate radioactive waste is significantly reduced and, by segregation of radionuclides in the different products and by-products of melting, melting also enables some of the LL category waste to be downgraded to VLL;
- recycle reusable ferrous metals, corresponding to precise metallurgical criteria, by manufacturing integrated radiological shielding (PRI) inside concrete shells intended for the production of intermediate level waste packages used in the NPPs.

From 1999 to 2011, about 21,700 tons of metal waste were processed in the melting unit, including 600 tons recycled in the form of PRI between 2002 and 2010 (55 t/year on average and nearly 140 tons in 2011). The melting furnace has been shut down since the 12th September 2011 accident[68], when it was probably damaged and in any case it has been under legal seal since then. There is at present no information concerning the date of its possible restart, which would require ASN authorisation.

Experience feedback from the successful manufacture of integrated radiological shielding (PRI) in CENTRACO for the past 10 years demonstrates the feasibility of metal waste recycling for reuse in the nuclear sector. However, the significant metallurgical requirements concerning the PRI steel limits the level of recycling, because they entail extensive preliminary sorting operations, thereby reducing the economic attractiveness of this technology.

Recycling by melting of low level lead was set up in June 2003 in a facility on the Marcoule site (CEA). It allows extremely effective decontamination of low level lead by melting: the activity of the lead leaving the furnace does not exceed an average of 0.5 Bq/g (the ingots produced by melting are thus less radioactive than "natural" lead resulting from ore extraction). The industrial processing capacity is 400 t/year and average production is about 100 t/year. The lead thus decontaminated is then shaped in three authorised conventional foundries (one operated by D'HUART INDUSTRIE and the other two by ROBATEL INDUSTRIE). The physical separation of the streams is verified by means of audits and is based on traceability, on the organisation of the facilities concerned and on the compliance with the specifications. Specialised firms then assemble the lead parts into radiological shielding for the nuclear industry: transport packagings, cell shielding, etc.

[68] The incident notification is available on the ASN website: http//www.asn.fr, heading "action de l'ASN", "contrôle", "actualités du contrôle", "avis d'incident des INB/2011".

Preparation of the VLL-SL lead waste ahead of its introduction into the melting furnace is nonetheless expensive (manual cutting, planning, separation of materials), which means that economic profitability is borderline. The shutdown of the line is scheduled for 2013.

Options being studied

Within the framework of the 2010-2012 PNGMDR, AREVA, CEA and EDF assessed the stocks of VLL ferrous metal waste for the period 2012-2041. The inventory is estimated at between 250,000 and 375,000 tons, or about 10,000 tons per year. This corresponds to one thousandth of the tonnage of steel recycled in France by conventional industry.

The possible outlets in the French nuclear sector were surveyed by Andra, AREVA, CEA and EDF. Two main product categories were identified: waste packages for disposal of LLW/ILW radioactive waste and metal reinforcements, notably in disposal facility structures. They offer a potential tonnage equivalent to the estimated inventory (potential of about 300,000 tons over 30 years).

To recycle these materials in the form of metal reinforcements, a steel mill and a rolling mill will be necessary. The creation of a dedicated facility would not be economically justifiable and the use of a conventional facility would entail considerable traceability constraints for marginal volumes when compared with the usual quantities that pass through such facilities. Moreover, the specific management of reinforcements on construction sites and the subsequent management of these items in the event of subsequent dismantling are issues that would need to be addressed. This type of recycling was therefore ruled out.

Volumes of about 10,000 tons per year, as mentioned before, are more compatible with the use of a foundry. This could be dedicated to the processing of radioactive waste (the throughput of a conventional foundry is about 2,000 to 50,000 tons per year). The development of cast iron packages for disposal of low and intermediate level waste was envisaged, in place of the concrete packages currently used. The use of cast iron packages would however require changing the baseline safety requirements of the LLW/ILW waste disposal centre and the facilities in which the waste would be packaged. These would probably require technical modifications to the waste management units present in the existing BNIs. It would therefore be necessary to make provision for a significant amount of time in which to set up the technology, more than five years and probably about ten, subject to the economic viability of this technology being confirmed.

2.5.2.2 Recycling of rubble

A similar study was conducted as part of the work requested by the 2010-2012 PNGMDR to examine the benefits of recycling crushed rubble. This study, conducted jointly by Andra, AREVA, CEA and EDF, clarified the nature of the stock of rubble waste to be produced and sketched out the feasibility of a means of recycling this waste as in-fill in the disposal areas of the industrial centre for collection, storage and disposal (Cires).

The inventory of VLL rubble was clarified and updated. Production apparently stand at about 10,000 t/year (between 6,000 and 10,000 m^3/year), for a total of 243,000 tons (160,000 to 243,000 m^3) over about 20 years (2012 to 2033). On the basis of experience feedback form waste stored in the Cires and the specific activity declarations by the producers, it was estimated that the level of radioactivity of 38% of this waste was compatible with its reuse[69] in the Cires disposal vaults, with no extra operational radiological constraints. Taking account of the typology of the waste and the operations needed to obtain the required granulometry characteristics, the usable part is estimated at 20%, or about 2,000 tons per year (2000 m^3). It should be underlined that these assessments are simply working guidelines and that there are uncertainties surrounding the actual future decommissioning operations, regarding both the quantities and the radiological aspects.

Concerning utilisation outlets, the potential represented by infill for the Cires disposal zones (more than 10,000 m^3 per year inert material infill) would easily meet the needs for waste compatible with the reuse constraints. Replacing untreated gravel by crushed low-level rubble would be able to increase the density of the repository by increasing the volume of waste per vault by about 7.5%. However, its impact on the consumption of the regulation disposal capacity at Cires is limited, because most of the volumes and activity levels emplaced in the disposal vaults must be accounted in the inventory of waste disposed of in Cires. However, it would contribute to improved use of the repository footprint (by increasing the volume emplaced for the same footprint on the ground).

This recycling option would therefore appear to be technically feasible. The required characteristics for the infill materials are obtained through the use of confirmed techniques. The operational provisions for treatment of this rubble waste and the conditions for reuse in Cires would need to be adapted to take account of the specific risks associated with the VLL rubble used (dust returned to suspension, specific activity and surface contamination levels required, operational constraints, etc.). The most complex point would seem to be to identify the actual level of radioactivity from this waste. Additional studies will be required to substantiate the radiological acceptability of this waste for recycling in Cires. It should therefore be pointed out that in any case, a significant part of the VLL rubble (50 to 75%) will still need to be disposed of in the repository as packaged waste, given its level of radioactivity or the associated uncertainties.

Economic feasibility will also have to be further examined, defining an overview of the solution from waste production up to utilisation, after treatment, in the disposal vaults. It will also be necessary to define and situate the industrial tool used to achieve the granulometry required for reuse of the rubble. Costing of this facility and the economic analysis of the solution as a whole will make it possible reach a conclusion as to the economic viability of such a project.

2.5.3 Outlook[70]

The existing recycling solutions, the CENTRACO melting facility and the Marcoule lead recycling unit, are able to reuse limited quantities of ferrous and non-ferrous metal waste respectively. However, the CENTRACO melting facility has been stopped since the 12th September 2011 accident and closure of the lead recycling unit is scheduled for 2013. **AREVA, CEA and EDF, together with Andra, will by the end of 2014 be assessing the impact of the shutdown of the lead recycling unit. Before the end of 2014, AREVA, CEA and EDF will also be examining whether or not it is opportune to set up a new lead recycling solution.**

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[69] The assessment is based on an average preliminary arbitrary value of 1 Bq/g.

[70] Opinion 2012- AV-0158 of 26th June 2012 on the management of very low level and low and intermediate level, short-lived waste is available on the website http://www.asn.fr, heading “les actions de l’ASN”, “la réglementation”, “bulletin officiel de l’ASN”, “avis de l’ASN”.

The studies conducted within the framework of the 2010-2012 PNGMDR concerning recycling of VLL rubble and metal waste show that recycling of this type of waste in the nuclear sector could be technically possible and would have advantages in reducing the volume of disposal space consumed in the Cires. It is also in line with a sustainable development approach, which is all the more important given the growing need for VLL waste management, as confirmed by the 2012 edition of the national inventory, which anticipates the production of 750,000 m^3 of VLL waste in 2020 and 1,300,000 m^3 in 2030 (or about twice the initial anticipated inventory of the VLL repository). However, the industrial feasibility has not yet been confirmed, in particular with regard to the recycling of ferrous metal waste.

It would be hard to imagine the materials passing through a conventional steel mill, given the traceability requirements arising from application of the order of 31st December 1999[71]: the stock of about 10,000 tons per year is actually quite small when compared with the quantities handled by steel mills. Using a dedicated foundry, for example to produce disposal packages, would be more appropriate, but an economic balance would be hard to guarantee and would in any case be borderline; moreover, the current industrial system and the baseline requirements for the existing BNIs would need to be modified to allow the introduction of new components (baseline safety requirements for the Aube repository or nuclear installations). This can lead to outlets in new projects being preferred (new nuclear facilities, waste packages for future repositories). **Andra, AREVA, CEA and EDF will assess the methods for establishing a metals reuse system and will present a summary of the various work performed, by 31st December 2014.**

The study concerning the reuse of very low level gravel presents a recycling scenario consisting of using finely crushed materials compatible with the Cires infill requirements. **Andra, together with AREVA, CEA and EDF will continue studying the creation of a recycling system such as this in the repositories in operation and will present the results by 30th June 2014.**

For the particular case of the decommissioning of Eurodif's Georges Besse I plant, which should produce 130,000 tons of metal waste as of 2021, reuse of the waste must be preferred, in accordance with the provisions of Article L.541-1 of the Environment Code, provided that its characteristics are compatible with treatment in the existing installations or those being studied. **At a meeting of the PNGMDR working group, AREVA will present the management solutions for the waste produced by decommissioning of Eurodif's Georges Besse I plant and more particularly the inventory of waste liable to be reused**.

Opening new recycling routes requires significant deployment lead-times, in particular for metal waste, which means that saturation of the Cires disposal capacity cannot be put off by more than a few years. The adoption of this type of solution must be considered within a context that goes beyond the framework of the current Cires, as this will be a means of saving space resources in the future repositories for this type of waste.

Moreover, the very low radiological impact observed when handling VLL waste, led AREVA, CEA and EDF to propose new alternatives. The waste producers thus underlined the restrictive absence of clearance levels in the French regulations and mentioned the possibility of recycling

[71] The order of 31st December 1999 stipulating the general technical regulations for preventing and mitigating detrimental effects and external risks resulting from BNI operations, will be abrogated by the order of 7th February 2012, setting out the general rules applicable to BNIs, on 1st July 2013 if no major change is made to the regulatory framework associated with waste management.

outside the nuclear industry, exploiting the possibilities offered by the European regulatory framework. It should be pointed out that the studies mentioned in this chapter will be carried out within the French context, on the basis of recycling in BNIs, within a system allowing management of radioactive waste such as to comply with the associated traceability requirements.

2.6 Incinerating radioactive waste

Waste incineration provides a real solution for the management of a broad spectrum of VLL and LL radioactive waste. It is for example, a way of preserving disposal site resources by reducing the volume of ultimate radioactive waste by a factor of 10 to 20. The CENTRACO incinerator, which entered service in 1999, can thus process solid waste (gloves, overshoes, plastics, etc.) and aqueous liquid waste, in particular oils and solvents, resulting from the day to day operation of nuclear facilities or from small producers of waste outside the nuclear power generating sector (hospitals, etc.).

Incineration is an important aspect of radioactive waste management route.

The shutdown of the Centraco incinerator for nearly a year, in 2011-2012, revealed the vulnerability of this management solution. **The 2013-2015 PNGMDR requires that experience feedback be established and that steps be taken to secure the incinerable radioactive waste management route.**

2.6.1 Context and issues

Incineration is the thermal process most frequently used in the nuclear industry around the world to process low and intermediate level radioactive waste. This is a mature and industrially proven technology, based on processes employed for decades to process conventional waste. Its benefits are a significant reduction in the volume of the waste prior to disposal and the ability to process a broad spectrum of waste: solid waste, organic liquid waste and aqueous liquid waste. The final waste is stable, chemically inert, non-dispersible and packaged in a format suitable for disposal.

Virtually all the nuclear countries thus have one or more operational radioactive waste incinerators. In France, the radioactive waste incineration sector is based on the CENTRACO facility, operated by the SOCODEI company on the Marcoule site and started up in 1999. SOCODEI (Société de Conditionnement des Déchets et Effluents Industriels – *company for industrial effluent and waste treatment*), created by EDF and AREVA in 1990, is now a wholly-owned subsidiary of EDF. The CENTRACO facility is today used by all the radioactive waste producers: AREVA, CEA and EDF as well as Andra, which collects waste from small producers outside the nuclear power sector (mainly from hospitals and research laboratories).

Since its start-up in 1999, the CENTRACO incinerator has been a vital link in the management of the radioactive waste produced by the EDF NPPs in operation and now in the dismantling programme for the reactors shut down, which represents more than 90% by mass of the annual volumes delivered. Most of the very low level and low level technological waste, such as gloves, overshoes, work suits, plastic film, paper, rubber, etc., is solid incinerable waste and is processed in CENTRACO. Liquids such as oils, solvents and sludges from the facilities are also incinerated in CENTRACO. The aim is on the one hand to preserve the space resource in the Andra repositories in the Aube *département* (Cires and CSA) by reducing the volume of ultimate waste (by a factor of more than ten) and, on the other, to allow simultaneous incineration of liquid waste: oils, solvents and aqueous effluents. CENTRACO is thus a vital solution for oils and solvents and its commissioning allowed the processing of certain waste for which there was no management route and which, in certain cases, remained stored on the site. There are alternative solutions for the other waste (incinerable solids, aqueous effluents), for example packaging in metal drums or encapsulation in concrete shells for the Aube repository, but incineration generally remains the preferred option because of the significant reduction in volume it offers.

As soon as the CENTRACO incinerator was started up, Andra incorporated it into its management system for waste from small producers outside the nuclear power sector. It was thus possible to process waste for which there was no industrial solution, such as perishable waste or solvents. Generally speaking, Andra has opted to use the CENTRACO incinerator for all VLL or LL waste liable to be processed in it.

2.6.2 Presentation of the technology

The CENTRACO incinerator entered service in 1999. It is used to process very low or low level radioactive waste (regardless of the half-life of the radionuclides) from nuclear activities, primarily those of AREVA, CEA, EDF and Andra, on behalf of the small producers from outside the nuclear power sector. Waste from European producers is also processed in CENTRACO, with the combustion residues in this case being returned to the original producer of the waste.

- **Incinerable Solid Waste (DSI)** with a capacity of 3,000 tons per year[72]: clothing worn by the personnel intervening in nuclear facilities (gloves, oversuits, etc.), combustible waste from operation and maintenance (packaging, vinyl, rags, etc.), along with waste from hospitals and laboratories using radioactive products;
- **Incinerable Liquid Waste (DSI) with a capacity of 2,000 tons per year[73]:** liquid effluents (washing solutions, oils, solvents), resins and sludges from nuclear facilities, along with waste from hospitals and laboratories using radioactive products.

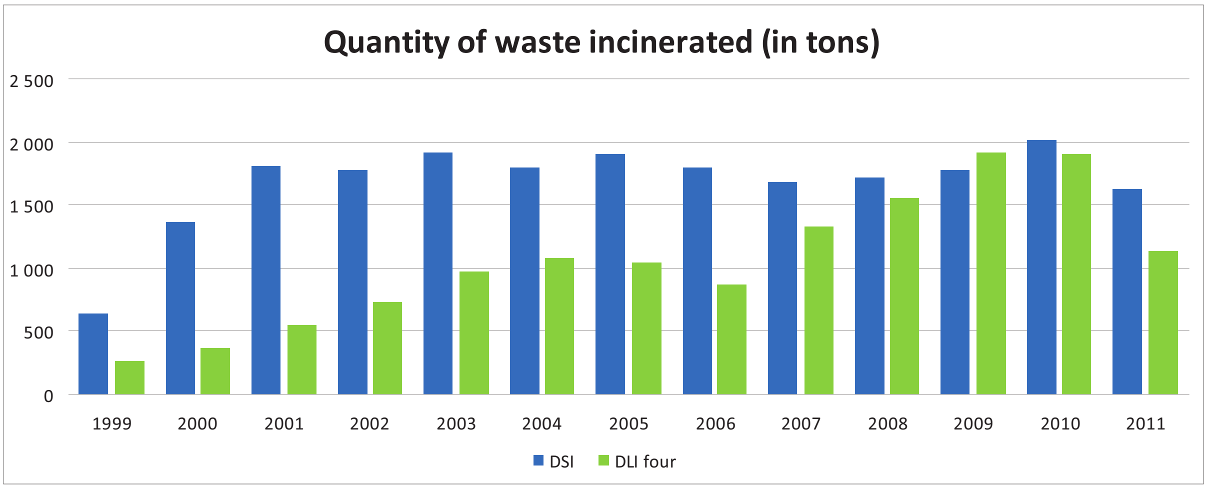
The incineration furnace is a three-chamber static furnace, comparable to an incinerator used in conventional industry. It design has been adapted to the requirements of nuclear facilities, notably with regard to the confinement of radioactivity (negative pressure area inside a building which is itself depressurised).

Processing of the smoke and fumes, utilising the best available techniques, meets both chemical and nuclear requirements. The ashes and clinker from incineration are blocked in a hydraulic binder and packaged in 400-litre shielded metal drums. They are shipped as ultimate waste to the Andra repositories in the Aube *département* for disposal.

From its start-up until late 2011, the incineration furnace processed nearly 22,000 tons of DSI and 14,000 tons of DLI. The following graph presents the quantities of waste processed since 1999.

[72] Maximum specific activity: 370 Bq/g alpha and 20,000 Bq/g beta/gamma (maximum average activity over the year: 1 Bq/g alpha and 1,000Bq/g Beta/gamma).

[73] Maximum specific activity: 50 Bq/g alpha and 20,000 Bq/g beta/gamma (maximum average activity over the year: 1 Bq/g alpha and 2,500Bq/g Beta/gamma).



Quantity of waste incinerated in the CENTRACO facility (source: SOCODEI)

The CENTRACO facility was shut down on 12th September 2011, following the industrial accident which occurred in the melting unit. In a resolution dated 27th September 2011, ASN made restart of the melting and incineration furnaces in the CENTRACO facility dependent on prior authorisation. Pursuant to the provisions of this resolution, the licensee submitted information to ASN to prove that restart of the incineration furnace would take place in satisfactory conditions of safety. In a letter of 29th June 2012, ASN authorised this furnace to restart and CENTRACO re-introduced waste for the first time in mid-July 2012.

2.6.3 Outlook

Incineration is today able to process certain types of very low and low level waste, in particular solid or liquid waste that cannot be directly accepted in Andra's repositories owing to their physical state. It also reduces the volume of waste by a factor of between 10 and 20. It is thus an essential aspect of radioactive waste management.

Andra, the waste producers and SOCODEI will need to look at how to secure the incinerable waste management systems. Aspects of this review will be presented at a meeting of the PNGMDR working group.

SOCODEI envisages expanding the operating envelope of the CENTRACO incineration furnace. The annual processing capacity of the incineration furnace and the activity of the waste processed could be increased. This extension would make it possible to incinerate a large volume of waste, more specifically liquid waste, and process a broader spectrum of waste. The modifications to the CENTRACO operating envelope are subject to authorisation by ASN.

2.7 Disposal of very low level (VLL) waste

The management policy for the VLL waste produced by nuclear facilities in France is not based on clearance or exemption levels, but on the origin of the waste within the facility. All contaminated or activated waste, or waste that is liable to be so, is considered to be radioactive waste and must follow specific, rigorous management, including disposal in a facility dedicated to radioactive waste. A disposal facility, located in the industrial centre for collection, storage and disposal (Cires) operated by Andra, has been accepting this type of waste since 2003.

At the end of 2011, the total volume placed in Cires was about 203,000 m^3, or 30% of the authorised regulation capacity (650,000 m^3) and the latest production estimates indicate that needs will be approximately double the inventory of waste to that originally identified to be disposed of d in this facility.

In order to preserve scare disposal site resources, solutions to reduce the flow of ultimate radioactive waste, such as waste compaction or reuse, were studied and the efforts made must be continued. However, the facility should reach full capacity within 20 to 25 years, instead of the 30 years initially anticipated, requiring the construction of another disposal facility or extension of the authorised capacity of the current facility in around 2025.

The 2013-2015 PNGMDR requires that Andra, together with AREVA, CEA and EDF, draw up a forecast schedule for filling the VLL waste repository at the industrial centre for collection, storage and disposal (Cires) in Morvilliers and propose an overall industrial system to meet the need for new VLL waste disposal capacity. Overall optimisation must also be sought, in particular for the decommissioning operations producing large quantities of VLL waste. The producers will thus be required to work with various public stakeholders, Andra in particular. The 2013-2015 PNGMDR requires that AREVA, CEA, EDF and Andra produce an analysis of experience feedback concerning management of this waste.

2.7.1 Context and issues

Very low level waste comes primarily from the decommissioning of facilities and, to a lesser extent, from operating waste from nuclear facilities. It mainly consists of inert waste (rubble, earth, sand) and metal waste[74]. Some categories of waste, more specifically liquid waste and certain solid wastes, are incinerated, but most VLL waste is today considered to be ultimate radioactive waste and sent to a dedicated disposal facility, the industrial centre for collection, storage and disposal (Cires) in Morvilliers.

VLL waste corresponds to a category of radioactive waste for which the activity level is low enough to require only light shielding of the operators who handle it. It in particular includes all waste liable to be contaminated or activated and coming from nuclear waste zones (see §1.2), which means that some VLL waste is only potentially radioactive. As mentioned in part 1.1 of this report, French regulations make no provision for exemption of waste based on universal clearance levels, as in the majority of countries, so disposal of such waste in a dedicated repository is thus a specifically French feature.

[74] Reuse solutions described in section 2.4 of this report are being studied.

A VLL waste disposal solution, such as to prevent the dangers and drawbacks mentioned in Article L.511-1 of the Environment Code, is dedicated to this waste, with the creation of a very low level waste repository. Andra's VLL waste repository in the Aube *département* has been operational since the summer of 2003. With a disposal capacity of 650,000 m^3 of waste, it initially corresponded to the requirement identified for a period of thirty years. The latest waste production estimates indicate requirements approximately twice those on which the initial waste inventory for this repository was based.

The availability of this solution is important for taking charge of waste generated by the development of the decommissioning operations, even if most of this waste will only be produced after the gradual shutdown of the 58 PWRs currently in service, in other words in principle after 2030. France is however already faced with the decommissioning of nine NPP reactors (including six gas-cooled reactors), the first spent fuel reprocessing plant in Marcoule and other facilities, notably CEA reactors and laboratories and the future decommissioning of Eurodif's Georges Besse I plant.

2.7.2 Presentation of management procedures

The VLL waste is allocated to the repository according to its acceptance specifications. These specifications define the characteristics and performance as well as the administrative conditions and controls with which the waste packages accepted in the facility must comply. Each batch of waste and each waste package must therefore comply with a radioactivity indicator[75], calculated on the basis of the concentrations of the radionuclides present in the waste. The waste producers generally opt for packaging in standard containers, as defined jointly by Andra and its customers when the VLL waste repository was opened. They present Andra with a disposal acceptance application by means of a file in which they provide all the data enabling Andra to assess the compatibility of the waste with the repository acceptance specifications.

The waste is thus packaged on the producers' sites in order to reduce its volume and increase its density. Waste packaging is optimised (production of detailed packaging plans or non-standard containers for specific waste, possibility of grouping waste types within packages in order to maximise use of the available space, etc.) and processing tools are used on some sites (compacting, dewatering of sludges, etc.). The Cires also has a compacting press.

Within the framework of the 2010-2012 PNGMDR, AREVA, CEA and EDF examined the technical possibilities and the economic opportuneness of improving waste compacting by using a new processing system. With this in mind, a centralised facility for mechanical compacting of metal waste was studied. This study shows that in addition to the technical drawbacks linked to this additional processing step (additional dose, environmental discharges, etc.) and the numerous limits and uncertainties (size and availability of the stock of waste, actual compacting gains, etc.) a solution such as this would, according to the producers, entail prohibitive additional treatment costs by comparison with the cost of a VLL repository, for minimal gain.

[75] This indicator, called IRAS, standing for radiological indicator for disposal acceptance, is a linear combination of the specific activity of the radionuclides present in the waste. It is notably derived from an assessment of the impact on the operators of the disposal centre. It must be less than 1 for a batch of waste, with the value for an individual package not exceeding 10.

The study however shows that using press type equipment on the site could be possible (this is the case with decommissioning of the gaseous diffusion plants at Pierrelatte) after a technical-economic assessment which can only be performed on a case by case basis, to take account of the specific nature of the stock to be processed (nature of waste, quantity, availability) and the installation possibilities.

2.7.3 Waste disposal in the industrial centre for collection, storage and disposal (Cires)

Given that for VLL waste, the radioactive risk is very low, whereas the chemical hazard from certain VLL waste can be high, the disposal methods are based on the technical concepts used in hazardous waste disposal facilities. This entails surface disposal in vaults excavated from clay, with the base modified to collect any water infiltration. The waste is thus isolated from the environment by a system comprising:

- a synthetic membrane surrounding the waste and allowing drainage of the leachates to the treatment installations;
- a thick clay layer under and around the sides of the disposal vaults;
- a covering, also of clay, over the waste.

During operation of the centre, when being emplaced, the waste is protected from rain by means of mobile roofs. The long-term containment of long-lived radioactive elements and chemical substances will be ensured by the retention properties of the clay formation, which constitutes a passive barrier.

At the end of 2011, the total volume in the disposal centre was about 203,000 m^3, or 30% of the capacity authorised by the Regulator (650,000 m^3). Cires also has a total activity limit for the disposal facility, which is differentiated according to the radionuclides. These limits were determined on the basis of forecast inventories, by checking their compatibility with the centre's safety targets, in both normal and altered evolution scenarios. Management of the radiological capacity of Cires poses no particular problems, and at the end of 2011, the radiological inventory for each radionuclide has been less than 5% of the authorised limit, except for thorium 232 which stands at 22%. With regard to the management of toxic chemicals, in order to present a coherent approach to the safety of its centre, Andra also assessed their long-term impact, in the same way as for radionuclides, to ensure that they are acceptable, given the forecast inventory of the centre. The results of these assessments did not lead to the definition of any acceptance limit for toxic chemicals insofar as the impacts assessed for the forecast inventories comply with the centre's safety targets, except for asbestos, for which a study is in progress in order to clarify the maximum allowable quantities in the centre.

Since the centre was commissioned, the requirements expressed by the waste producers and the objective of absorbing larger volumes and optimising the filling of the centre led Andra to change the operating conditions and the geometry of the disposal vaults. The annual disposal capacity proposed by Andra to the producers has risen from 26,000 m^3 in 2009 to 32,000 m^3 per year[76]. Thus, in 2006, the dimension of the vaults was doubled, in 2010 the slopes were made steeper and the vaults made deeper. Increasing the height of the top (dome) of the vaults is also being envisaged. These geometrical modifications have made the disposal facility more compact, in other words able to contain a larger volume of waste within the same surface footprint. Thus for the same ground footprint, the changes to the vaults should allow a real disposal capacity[77]

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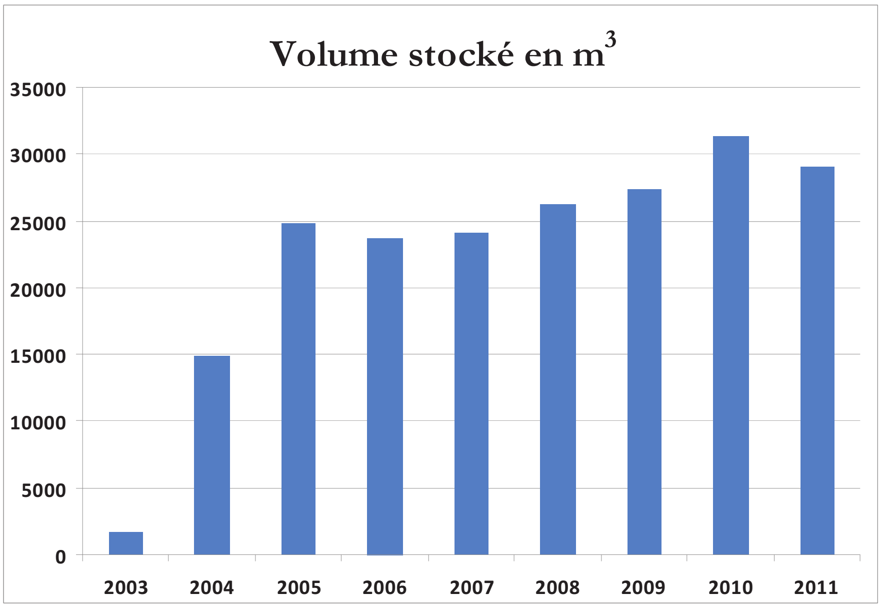
[76] The annual capacity authorised by the Prefect's order is 50,000 tons, enabling about 50,000 m^3 of waste to be accepted.

[77] For information, the useful filling ratio (volume of waste/useful volume) is about 85%.

increase per vault of 50 to 60%. Consequently, subject to the results of the studies to be conducted and changes to the authorisation following a public inquiry, the Cires could accept a greater volume of waste than the capacity currently authorised, within the same perimeter, thus offering greater capacity.



Volume of waste delivered to the VLL waste centre (source: Andra).

An optimisation approach was also undertaken jointly by Andra and the waste producers for management of "outsize" waste, for which the question of the pertinence of packaging in a standard container arises. For the period 2012-2030 a potential stock of about 350 packages of this type representing 17,000 tons was identified. These are mainly transport containers for spent fuels and gas-cooled reactor decommissioning waste (blocks of reinforced concrete). For a volume such as this, Andra considers that it is opportune to develop a dedicated disposal vault able to take this waste in industrial conditions. The gaps between the outsize waste will be filled with standard packages. These management methods were incorporated into the Prefect's order for operation of the centre, dated 9th February 2012. A vault of this type, with a capacity of 40,000 m^3, adequate for the stock identified, could be commissioned as early as 2016.

A search for overall optimisation is considered to be necessary by all the stakeholders involved in waste management, more specifically for the decommissioning operations. The study[78] published by the Nuclear Energy Agency in 2012 on the management of outsize waste presents possibilities for this optimisation that could be more broadly applied to all waste. This study underlines the fact that the best management method must be chosen with a view to the overall optimisation of activities: decommissioning, processing if necessary, transport and disposal. All of the issues involved (regulatory, technical and operational, safety, economics, planning and acceptance by the public and the stakeholders) must be considered transparently by all the stakeholders. An optimisation approach such as this implies mobilisation of all those involved well ahead of the decommissioning work and must be carried out on a case by case basis, taking account of the specificities of each decommissioning project and the waste to be managed.

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[78] This report, entitled "Report on the Management of Large Components from Decommissioning to Storage and Disposal" can be consulted at the following address: http://www.oecd-nea.org, heading "works areas", "radioactive waste and decommissioning", "RWM documents"

2.7.4 Outlook[79]

The density of the waste disposed of in Cires (about 1 since 2010) is below the average density initially considered at the design of the centre (1.45). In addition, the waste traffic delivered began to rise in 2009, owing to the decommissioning and post-operational clean-out operations, a trend which is bound to continue. The VLL waste production forecasts, as notified by the producers to the national inventory, indicate an average annual waste traffic of 55,000 m^3 between 2020 and 2030 and the waste inventory for the 2030 time-frame has doubled by comparison with that used for the dimensioning of the VLL waste repository in 2003. The possible options for reducing the traffic of VLL waste described in chapter 2.5 and the efforts to be continued in order to increase waste density will simply postpone saturation of the Cires repository by a few years. The producers proposed that disposal of certain VLL waste on the producers' sites be explored, if the quantities so warrant. However, on-site disposal of waste within a specially designed facility with performance equivalent to the current disposal of VLL waste could only be envisaged case by case and subject to its acceptability to all the stakeholders, further to an opinion from the safety regulators, because the reference solution for the disposal of VLL waste is the Cires repository.

The authorised disposal capacity in Cires should therefore be reached in 20 to 25 years, instead of the initially estimated 30 years. The acceptance of VLL waste should require the construction of another disposal facility or extension of the authorised capacity of the current facility in around 2025.

Andra, together with AREVA, CEA and EDF will by the end of 2014 draw up a forecast schedule for filling of the Cires repository. This schedule will take account of the capacities handled in terms of annual traffic and the modifications envisaged as a result of the studies initiated into the reuse of waste. It will also present the anticipated evolution in the occupation of the repository's radiological capacity. Based on this information, Andra will in mid-2015 propose an overall industrial arrangement meeting the need for new VLL waste disposal capacity.

The search for overall optimisation based on a multi-criterion assessment involving regulatory constraints, technical, safety and economic issues, as well as the preservation of scarce disposal space resources, involving all the stakeholders involved in waste management, will also need to be sought, more specifically for the decommissioning operations which produce large quantities of VLL waste. **The producers will need to attach particular importance to the decommissioning waste management options and in their decommissioning files must justify the conclusions of the search for optimisation, notably together with Andra. Within this framework, they will also need to approach other public stakeholders. A summary of experience feedback from management of decommissioning waste will be presented to the PNGMDR working group by AREVA, CEA, EDF and Andra.**

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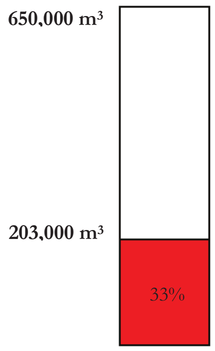
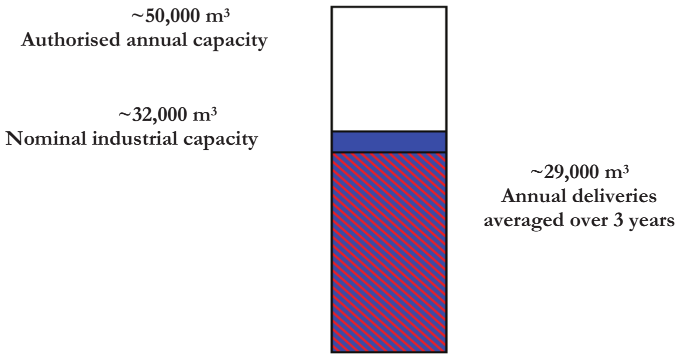
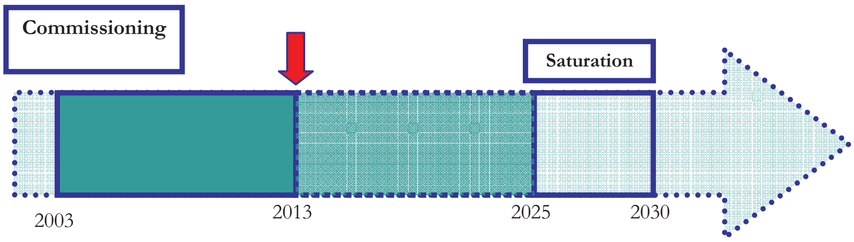
[79] Opinion 2012- AV-0158 of 26th June 2012 on the management of very low level and low and intermediate level, short-lived waste is available on the website http://www.asn.fr, heading "les actions de l'ASN", "la réglementation", "bulletin officiel de l'ASN", "avis de l'ASN".



Occupation of Cires disposal capacity (data as at 31/12/11)



Annual waste traffic



Forecast date for saturation of the Cires repository

2.8 Disposal of low and intermediate level, short-lived (LLW/ILW-SL) waste

"Short-lived" low and intermediate level waste (in which the radioactivity mainly comes from radionuclides with a half-life of less than 31 years) has, since 1969, been disposed of in dedicated surface repositories.

Between 1969 and 1994, the Manche waste disposal facility received 527,000 m^3 of waste packages and entered the oversight phase in 2003. The leaktightness of the facility is based on the installation of a cover, the long-term stability of which requires consolidation (reducing the steepness of the slopes) which will take place over a period of about fifty years. Steps have been taken to ensure that records and a memory of the facility and the waste emplaced in it are kept for future generations. **The 2013-2015 PNGMDR requires that Andra present the steps taken to ensure that records and a memory of the Manche waste disposal facility are maintained.**

At the end of 2011, the total volume of packages emplaced in the Aube waste disposal facility (CSA) was about 255,000 m^3, or 25% of the capacity authorised by the Regulator (1,000,000 m^3). Efforts made at source to reduce the amount of LLW/ILW-SL waste produced, and the commissioning of a VLL waste repository in the Cires, plus the melting and incineration routes, have enabled the lifetime of the waste disposal facility to be extended. **The 2013-2015 PNGMDR requires that the evolution of the remaining radiological capacity of the centre be monitored, taking account of the forecast waste inventory.**

2.8.1 Context and issues

Low and intermediate level, short-lived waste (half-life shorter than 31 years) consists mainly of waste linked to maintenance (clothing, tools, filters, etc.) and to the operation of nuclear facilities (processing of liquid effluents or filtration of gaseous effluents). It can also come from post-operational clean-out and decommissioning of such facilities. Given its radiological content, it has been disposed of on the surface since 1969: In France, there are two repositories of this type: the Manche waste disposal facility which has no longer been accepting any waste since 1994 and the Aube waste disposal facility (CSA) which has been in operation since 1992. Surface or sub-surface repositories for low and intermediate level waste are also in service or planned in many countries. The design and the depth of the repositories determines the type of waste that can be accepted, more specifically with regard to lifetime.

The low and intermediate level, short-lived waste repositories are monitored during what is referred to as the oversight phase, conventionally set at 300 years. The periodically updated facility safety reports, including during the oversight phase, must be able to check that the activity contained in the waste reaches a residual level such that human and environmental exposure is not unacceptable, even in the event of a significant loss of the containment properties of the facility.

The latest production estimates for low and intermediate level, short-lived waste, indicate that the CSA should be able to take all the waste produced by the operation and the decommissioning of the current fleet.

2.8.2 Presentation of management procedures

2.8.2.1 Past management methods: the Manche waste disposal facility

Historically, the low and intermediate level waste management solution was created in 1969 with the opening of the Manche waste disposal facility (CSM) authorised by decree of 19th June 1969. The experience acquired in the first years of operation allowed the definition of a safety concept associated with this type of radioactive waste surface disposal and acceptance rules were also clarified in the acceptance specifications. These standards were given formal status in the basic safety rules[80] (RFS): RFS I.2 of 1984 and RFS III.2.e of 1985 revised in 1995.

The CSM was operated for 25 years, from 1969 to 1994, during which the disposal conditions were continuously improved. In total, 527,000 m^3 of waste packages have been emplaced in it. The covering work took place from 1991 to 1997. In the meantime, the Government had set up the "Turpin" investigatory board to rule on the conditions for transition to the facility oversight phase. This board in particular recommended that Andra produce a concise record file designed to preserve essential information about the Manche waste disposal facility for future generations.

During the period the CSM was in operation, the groundwater circulating under the centre was contaminated by tritium in 1976. The waste which caused this contamination was removed, but the contamination of the groundwater is still significant, even if it is falling regularly. The evolution of this contamination is being closely monitored. The impact of the centre is however extremely slight (the impact on a hypothetical critical group was estimated in 2011 at 0.3 µSv/year).

The cover consists of a bituminous membrane to ensure leaktightness. The presence of this membrane and the slopes of the peripheral mounds built within a very small perimeter, means that a degree of instability has been observed. A programme of gradual consolidation of these mounds was defined by Andra and should take about fifty years. It comprises phases consisting of securing, consolidating and then reducing the steepness of the slopes down to the natural ground, interspersed with observation phases. This programme would require an extension of the centre's footprint, for which Andra has initiated the necessary land management process. Based on the file presented to the CSM's 2009 safety review, ASN requested an additional file in 2015 to enable it to rule on the final configuration of the centre.

The practices used in the Aube waste disposal facility and described below, benefited from the experience feedback from the Manche waste disposal facility.

2.8.2.2 Current management methods

As with the acceptance of waste for disposal in Cires, the producers must obtain approval from Andra confirming the conformity of the packages with the disposal acceptance specifications prior to delivery of the packages. These specifications define the characteristics and performance as well as the administrative conditions and controls with which the waste packages accepted in the facility must comply. The producers must take quality management steps to ensure that the packages they manufacture industrially do actually correspond to the model which was approved. Andra also monitors the quality of the packages produced by means of inspections on the producer sites. A second level of control is also carried out on the packages delivered to the

80 The basic safety rules can be consulted on the ASN website: http://www.asn.fr heading "la réglementation", "règles fondamentales de sûreté et guides de l'ASN".

waste disposal facility. Packages are sampled and assessed in approved laboratories on behalf of Andra. Andra decided to augment its second-level control system by setting up an inspection facility in the Aube waste disposal facility, which it itself operates. Commissioning of this facility is planned for 2015, provided that the necessary authorisations are obtained.

Before it can be emplaced in the waste disposal facility, the waste must be characterised and packaged. The waste can either be directly packaged on the producers' sites for disposal, or may require transit through a packaging facility operated by Andra on the CSA site (injection of containers, compacting of drums) or by another firm on another site (for example SOCODEI for incineration or melting of metals). As with VLL waste, with the aim of preserving disposal space, steps (sorting, processing, etc.) are taken to reduce the volume of ultimate radioactive waste.

2.8.3 Waste disposal in the Aube waste disposal facility (CSA)

The principle of disposal in the Aube waste disposal facility (CSA) is to confine the radioactivity in the packages and the disposal structures, to allow decay over a period of several hundred years. The long-lived radionuclides content of the waste must be low enough for the impact of the waste disposal facility to be acceptable after 300 years of oversight, even if the structures and packages become degraded. The safety analysis must also demonstrate that the impact of any toxic chemicals is acceptable.

The packages are emplaced in concrete bunkers, sheltered from rain by mobile metal structures which can be moved as the facility is utilised. The packages are immobilised by gravel or concreted into the structures, depending on whether the package outer containers are durable (concrete shell package) or perishable (metal drums and containers). The facility will subsequently be closed by a slab, made tight to rainwater by spraying of a plastic material. The structures will then be protected by a cover with very low permeability.

In late 2011, about 255,000 m^3 of waste packages were emplaced in the CSA, or about 25% of its regulation capacity. The volume of annual deliveries (12 to 13,000 m^3) is well below the design-basis volume (30,000 m^3).

Within the framework of the 2010-2012 PNGMDR, the situation of the LLW/ILW disposed of in repositories was analysed. Based on the 2009 edition of the national inventory, this waste (excluding tritiated waste) represents 54,500 m^3. Waste that is untreated or not packaged in the form of containers acceptable in the CSA, accounts for about 85% of this volume. This is mainly legacy waste which needs to undergo waste retrieval and packaging (RCD) operations. The remaining 15% corresponds to disposal packages at various stages of acceptance and collection for disposal.

With regard to the radiological capacity, limits are applied to 19 radionuclides. These limits were determined on the basis of forecast inventories, by checking their compatibility with the centre's safety targets, in both normal and altered evolution scenarios. Similarly, the total alpha activity counted after 300 years is also limited. The centre's consumption of its radiological capacity is at present less than its volume occupation (less than 15%), except for chlorine 36 (the half-life of which is 300,000 years) because the inventory emplaced corresponds to nearly 90% of the authorised capacity. This situation is not unusual in that ASN had issued the authorisation to allow the disposal of a limited number of graphite sleeve containers stored on the Bugey site, which has been completed. Andra has also begun revision of the specifications for the acceptance of tritiated waste in order to ensure more prudent management of tritium-bearing waste, given the mobility of this radionuclide.

The new CSA periodic safety review is scheduled for 2016. On this occasion, the forecast inventory of the waste to be accepted will be updated on the basis of the new data available at that time. The latest data from the national inventory indicate that this regulation capacity will become saturated well after 2030 and that the CSA will be able to take all the waste produced by the operation and decommissioning of the current fleet. The decommissioning programmes could however lead to a gradual rise in traffic in the coming years, with a forecast annual volume of 20,000 m^3 of waste packages per year, between 2020 and 2030. This update of the forecast inventory will also make it possible to check that the centre's radiological capacity is compatible with the volumes to be accepted.

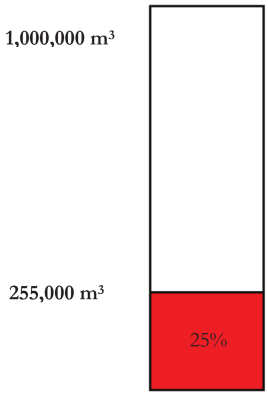
Outsize waste is disposed of in the CSA: reactor vessel heads and other equipment items are emplaced without first being cut up for packaging in standard packages. However, unlike the disposal of such waste in Cires, the forecast inventory for this waste does not justify the construction of new dedicated structures. The conventional structure can be used provided that certain operational measures are taken. The outsized packages concerned are in particular the following: for EDF, the vessel head from the Chooz A reactor and the 49 lateral neutron shielding packages from Creys-Malville (disposal planned from 2012 to 2014); for AREVA, possibly nine spent fuel transport containers and, for CEA, six vessel packages and twelve steam generator packages produced by the decommissioning of submarines. CEA would be unable to take charge of the packages before another 10 to 15 years, to allow for the necessary radioactive decay.

2.8.4 Outlook[81]

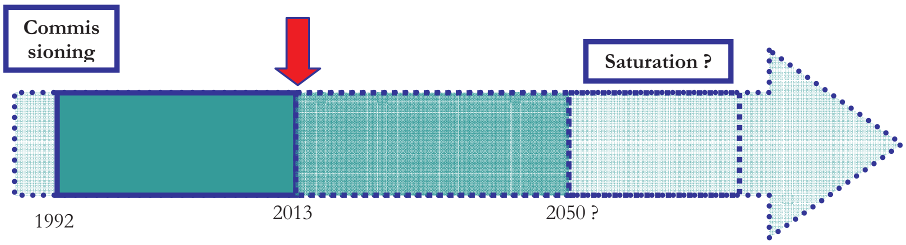
At the end of 2011, the volume of 255,000 m^3 of waste emplaced in the Aube waste disposal facility, represents 25% of its capacity as authorised by the Regulator. The reduction in the production of packages intended for the CSA, as a result of the reduction at source efforts and the opening of disposal capacity at Cires and CENTRACO, means that a significant increase in its operating life, initially estimated at 30 years, can be envisaged. Consumption of the centre's radiological capacity does however require particularly close monitoring.
Andra, together with AREVA, CEA and EDF will, by 31st March 2015, draw up a forecast schedule for the filling of the Aube repository, more specifically presenting the forecast evolution of the consumption of the centre's radiological capacity.

The Manche waste disposal facility is the first radioactive waste repository to have entered the oversight phase. **During a meeting of the PNGMDR working group, Andra will present the steps taken to ensure that a record of the Manche waste disposal facility is maintained.**

[81] Opinion 2012- AV-0158 of 26th June 2012 on the management of very low level and low and intermediate level, short-lived waste is available on the website http://www.asn.fr, heading "les actions de l'ASN", "la réglementation", "bulletin officiel de l'ASN", "avis de l'ASN".



Occupation of CSA disposal volume capacity
(data as at 31/12/11)



Forecast date for saturation of the CSA

2.9 Management of waste containing enhanced natural radioactivity

Waste containing enhanced natural radioactivity (TENORM) is waste created by the transformation of raw materials naturally containing radionuclides but which are not used for their radioactive properties. This is low level, or even very low level, long-lived waste.

TENORM waste is managed *in situ*, or is either disposed of in conventional waste disposal facilities (four facilities are authorised to receive TENORM), or sent to Andra's disposal facilities dedicated to the management of radioactive waste, according to their radiological characteristics. Combustion ashes can also be reused in the production of cement, owing to their very low added radiological activity when compared with the natural radioactivity already present in the concrete.

Improving the management of TENORM waste requires greater knowledge of the quantities concerned and improved traceability. **The 2013-2015 PNGMDR requires a presentation of the regulatory provisions implemented to reinforce the management of TENORM waste.**

2.9.1 Context and issues

Waste containing enhanced natural radioactivity (TENORM) is waste created by the transformation of raw materials naturally containing radionuclides but which are not used for their radioactive properties. Their radioactivity is due to the presence of natural radionuclides, such as potassium 40, radionuclides of the uranium 238, uranium 235 or thorium 232 family. These radionuclides may be concentrated in the waste by transformation processes. The many activity sectors generating this type of waste and the numerous establishments concerned explain the uncertainty that still surrounds the assessments of the quantities of waste produced and the radiological activity of some of them.

Since the order of 25th May 2005 concerning professional activities using raw materials naturally containing radionuclides and not used for their radioactive properties, TENORM waste has been subject to specific management procedures as presented in the following section (the historical management of this waste is presented in chapter 2.1). This order draws up a list of activities liable to generate TENORM waste. This waste falls into the following categories:

- very low level, long-lived waste (for example, foundry sand waste, waste from zirconium-based refractory materials, notably used in the glass-making industry, etc.);
- low level, long-lived waste (for example, certain waste from processing of monazite, certain waste from the manufacture of zirconium sponges, certain waste from the past or future decommissioning of industrial installations, for example phosphoric acid production, titanium dioxide processing, zircon flour processing facilities and former monazite treatment activities).

2.9.2 Management of waste containing enhanced natural radioactivity

TENORM waste is managed *in situ*, or is either disposed of in conventional waste disposal facilities (four facilities are authorised to receive TENORM), or sent to Andra's disposal facilities dedicated to the management of radioactive waste, according to their radiological characteristics.

2.9.2.1 Reuse of TENORM waste

Combustion ashes can be reused in the construction sector, because they can be included in the composition of cement and therefore concrete.

A study concerning the radioactivity added by the presence of ash was carried out by the hydraulic binders industry technical association in July 2010. According to this estimation, the addition of ashes leads to a slight increase in the radiological activity by comparison with the natural radioactivity already present in the components of the concrete (sand, gravel, limestone, basalt, granite). This varies widely and is heavily influenced by the origin of the components of the concrete.

2.9.2.2 TENORM waste disposed of in conventional waste disposal facilities

The regulations make provision for the possibility of disposing of TENORM waste in disposal facilities for hazardous waste (ISDD), for non-hazardous waste (ISDnD) or for inert waste (ISDI).

The order of 25th May 2005 concerning professional activities involving raw materials naturally containing radionuclides not used for their radioactive properties and the circular of 25th July 2006 concerning the conditions for acceptance of TENORM waste in waste disposal centres, offer a strict framework for the management of this waste. This circular is not automatically binding on the licensees concerned but it does urge the Prefects to reinforce the operational requirements applicable to the facilities receiving or wishing to receive this type of waste.

The circular thus stipulates that a specific assessment, supplementing the initial impact assessment, must be transmitted to the Prefect. This assessment aims to prove that the disposal of TENORM waste does not jeopardise the protection of the interests mentioned in Article L.511-1 of the Environment Code, particularly from the radiation protection standpoint, for both the operating personnel and the neighbouring population, including over the long term. It must be produced in reference to a technical guide published by the Ministry responsible for ecology and IRSN in 2006[82]. This circular specifies the procedures applicable to the licensee for the acceptance and control of waste in waste disposal facilities, the conditions for monitoring the radiological impact on the environment of accepting TENORM and the procedures for information about the inspection of classified installations by means of an annual operating report. This report is presented to the local information and monitoring committees (CLIS) for the disposal facilities and, as applicable, to the Departmental Council for the Environment and for Health and Technological Risks (CODERST).

[82] This guide can be consulted on the IRSN website: http ://www.irsn.fr, heading "actualités et presse", actualités".

The above-mentioned circular of 25th July 2006 also recalls that in accordance with the polluter-pays principle, any producer of waste is required to provide a technical justification of the validity of the management solution for its waste. It is on the basis of this principle that the above-mentioned circular recalls the performance of a study for each batch of waste to be managed, in order to confirm its acceptability in the host facilities.

In late 2011, a summary of the facilities accepting TENORM waste was produced by the Ministry responsible for ecology, based on the declarations from the Regional Directorates for the Environment, Planning and Housing. Only the following two hazardous waste disposal facilities were identified as authorised to receive TENORM waste:

- Villeparisis in the Ile-de-France region, licensed until 31st December 2020, for an annual capacity of 250,000 t/year;
- Bellegarde in the Languedoc-Roussillon region, licensed until 4th February 2029, for an annual capacity of 250,000 t/year until 2018 and 105,000 t/year thereafter.

It was also found that two other facilities not identified in the first report, are also authorised to take TENORM waste, but in smaller quantities. They also performed assessments enabling them to receive TENORM waste in accordance with the circular of 25th July 2006. They are:

- Champteussé-sur-Baconne in the Pays de la Loire region, licensed until 2049, for an annual capacity of 55,000 t/year;
- Argences in Basse-Normandie, licensed until 2023, for an annual capacity of 30,000 t/year.

The operational experience feedback from the Bellegarde and Villeparisis facilities shows there is no contamination of the groundwater linked to the presence of TENORM waste in the sludges. Oversight of the acceptance of TENORM waste in these centres has been reinforced by orders of the Prefect imposing particular requirements relative to:

- the implementation of an extended prior acceptance procedure (identification of the naturally occurring radionuclides, evaluation of the cumulative doses over one year);
- radiological surveillance (measurement of naturally occurring and artificial radionuclides in the groundwater, leachates and sludge from leachate ponds;
- monitoring of air quality (activity concentration of dust in the air);
- the monitoring of personnel exposure (Labour Code).

The quantities of TENORM waste received in these facilities is well below their capacity (less than 10% of total capacity). The Villeparisis and Bellegarde facilities in fact received 25,509 tons and 94,680 tons respectively, between 2000 and 2010 (or 10% and 30% respectively of their annual capacity). The Champteussé-sur-Baconne facility received 1,808 tons between 2002 and 2009 (or 3% of its annual capacity) and the Argences facilities received about 1,530 tons between late 2010 and 2011 (or 5% of its annual capacity), which would seem to confirm that there is no risk of a shortage of disposal capacity. However, there is a question over the share of the total stock accepted by these four authorised facilities. Not all the industrial firms liable to produce TENORM waste are actually customers of these facilities.

The work done by the Ministry aims to increase the degree of involvement by the producers of this type of waste: this led to the ordinance of 17th December 2010 modifying the Environment Code, Article L.541-2 of which specifies the responsibilities of the producers or the owners of the waste and requires that they characterise it (Article L.541-7). The purpose of this characterisation is to identify the substances contained in the waste and measure the concentration, including when these substances can be the origin of ionising radiation. This provision should help clarify the inventory of TENORM waste and improve management of the processing circuits.

2.9.2.3 TENORM waste disposed of in radioactive waste disposal facilities

Very low level TENORM waste which cannot be accepted in conventional waste disposal facilities is placed in the Morvilliers industrial centre for collection, storage and disposal (Cires). The 2012 edition of the national inventory identifies 7,800 m^3 of TENORM waste in this category at the end of 2010, excluding the waste generated by spas, paper-mills and biomass combustion. The 2015 edition of the national inventory should be expanded and identify the quantities of waste produced by these three types of industries.

Low level, long-lived TENORM waste is incorporated into the low level, long-lived waste management industrial systems being studied by Andra (see §3.2). The 2012 edition of the national inventory identifies 17,000 m^3 of TENORM waste in this category (excluding waste generated by spas, paper-mills and biomass combustion). Pending the arrival of such a repository, this waste is stored on certain production sites.

2.9.3 Outlook

Improving the management of TENORM waste requires greater knowledge of the quantities concerned and improved traceability. To achieve, this, **modifications to the regulations are needed**, in order to:

- clarify the provisions concerning the waste characterisation obligation contained in Article L.541-7.1 of the Environment Code, which should lead to the drafting of a waste identification sheet, entailing radiological characterisation of the waste for activities subject to the order of 25th May 2005, or any activity classified as generating TENORM waste;
- modify the model of the waste monitoring notice stipulated in the ministerial order of 29th July 2005 as amended, to include a reference to the waste identification sheet corresponding to waste which is being shipped;
- improve knowledge about the stocks, which will first of all require inclusion of the TENORM waste identification in the European list of "waste" codes, at least for the industrial sectors covered by the ministerial order of 25th May 2005. The French authorities could propose this to the European Commission;
- impose an annual declaration of TENORM waste from classified installations, regardless of their administrative classification (modification of the ministerial order of 31st January 2008 concerning the register and annual declaration of polluting emissions and waste).

Finally, the ministerial order of 30th December 2002, amended, concerning hazardous waste disposal facilities and the ministerial order of 9th September 1997, amended, concerning non-hazardous waste disposal facilities, should be modified in order to require the implementation of monitoring of the radiological quality of the groundwater, leachates and sludges from the facilities receiving TENORM waste. These modifications should also require monitoring of the activity concentration of the dust in the air.

A report on the implementation of these recommendations will be presented to a meeting of the PNGMDR working group by the General Directorate for Risk Prevention.

3 The management routes to be implemented: Current needs and future outlook

3.1 Waste management requiring specific work

3.1.1 The industrial system for management of waste from small producers outside the nuclear power sector

Because of their properties, certain categories of radioactive waste require special management routes. This is, for example, the case with waste containing tritium (tritiated waste) and used sealed sources, as well as radioactive waste from small producers outside the nuclear power generating sector, which represents very small quantities.

Most of the tritiated waste cannot be directly accepted in the surface repositories owing to the considerable environmental mobility of tritium. The creation of new storage facilities by CEA over a period of about forty years offers a satisfactory solution in terms of short to medium term safety, pending its future transfer to disposal facilities. The work done to identify management solutions, initiated by the 2010-2012 PNGMDR, must be continued with regard to liquid and gaseous tritiated waste from the small producers not in the nuclear power generating sector. **The 2013-2015 PNGMDR requires that Andra, together with AREVA, CEA and SOCODEI, continue the studies concerning the processing of liquid and gaseous tritiated waste from the small producers outside the nuclear power sector.**

Most used sealed sources are currently stored pending a final management solution. Because of their concentrated activity and their potential attractiveness, only a small part of the used sealed sources can be disposed of in the Aube waste disposal facility. **The 2013-2015 PNGMDR requires the continuation of the work initiated in the previous edition concerning the drafting of a management scheme for used sealed sources.**

Finally, dealing with the waste from small producers outside the nuclear power sector needs to be optimised, owing to the nature and volumes of waste to be processed. **The 2013-2015 PNGMDR requires that Andra identify the investments needed to guarantee the continuity of the management solutions for the waste generated by this sector.**

Since 1979, Andra (part of CEA at the time) has taken charge of waste from the small producers. This sector covers activities outside the nuclear power sector which handle radioactive materials and consequently produce radioactive waste. They more specifically concern the health and biological research sectors and non-nuclear industries. Andra generally collects unpackaged waste and performs sorting, packaging, storage and disposal operations, or has them carried out on its behalf.

When ANDRA was created in 1991 as a public organisation independent of CEA, it did not have its own facilities for managing waste from small-scale nuclear activities. Consequently, most of the services are carried out by Andra contractors:

- waste is collected by transport companies under contract with Andra;
- the collected waste is grouped in a facility (building 204 – North grouping centre), belonging to CEA on the Saclay centre;

- waste is sorted in the Socatri facility at Pierrelatte;
- waste is either incinerated in SOCODEI's CENTRACO incinerator, or compacted by the press in the Aube repository, or encapsulated in concrete in the injection facility in the Aube repository prior to disposal;
- waste for which there is as yet no disposal route is stored at Socatri (Pierrelatte) or in various CEA facilities.

ANDRA has started reconfiguring the route because some of the licensees working on behalf of Andra wished to be able to use premises mobilised for other activities (this is the case of CEA and Socatri) and in 2012 created a collection centre and a storage facility for waste from small producers outside the nuclear power sector in the VLL waste disposal centre, now called the industrial centre for collection, storage and disposal (Cires). Andra also considers that emphasis should be placed on facilities which are closely tailored to the nature and volumes of the waste to be processed, in order to minimise the processing costs.

Consequently, before the end of 2013, Andra will identify the investments required in order to guarantee the continuity of the waste management route for small producers and will present the current conclusions of its review during a meeting of the PNGMDR working group.

3.1.2 Management of waste containing tritium

3.1.2.1 Context and issues

Most of the tritiated waste produced in France is operating and decommissioning waste linked to CEA's military applications (98%), with the rest coming from small producers outside the nuclear power sector, primarily as a result of research or the pharmaceutical and hospital sector, but also national defence waste. At the end of 2010, this represented about 4,500 m^3 for an estimated inventory of about 5,000 TBq. This waste is grouped according to its tritium inventory and more specifically its level of gas release.

The routes at present operational for disposal of tritiated waste only concern the least active waste. Liquid waste can be processed in the CENTRACO facility. For the other waste, shipment to the Andra disposal facilities entails major constraints[83].

The planned disposal centres for LLW-LL waste on the one hand and HLW/ILW-LL waste on the other, will be able to take waste containing significant quantities of tritium.

The waste currently produced by CEA and which cannot be sent to Andra's disposal facilities, is stored on its sites, in particular those at Valduc and Marcoule.

Faced with this absence of outlet for most of the French tritiated waste, the 28th June 2006 planning Act, as codified, concerning the sustainable management of radioactive materials and waste, required "*the development by 2008 of storage solutions for tritiated waste containing tritium allowing the reduction of its radioactivity prior to surface or sub-surface disposal*". Pursuant to decree 2008-357 of 16th April 2008 setting the requirements concerning the 2007-2009 PNGMDR, CEA submitted an orientation file in late 2008 for storage of tritiated waste for which there is no disposal route.

[83] For the waste received in the CSA, the acceptance specifications limit the quantity of tritium released per day and per unit of mass to 2 Bq/g/d and the specific activity to 1 MBq/g.

According to the study data, the storage project for tritiated waste with no disposal route concerns solid tritiated waste already produced and to be produced until about 2060, which is the anticipated date for completion of decommissioning of the ITER installations (nuclear fusion research facility). By 2060, this inventory would thus reach a tritiated waste volume of about 30,000 m^3 for a tritium radiological activity of about 35,000 TBq. The study does not concern the solid and liquid waste liable to be processed in the CENTRACO facility, the waste which can be sent to an Andra disposal centre without prior storage and the waste which can in principle be sent without prior storage to the planned disposal centres for LLW-LL waste and HLW/ILW-LL waste (Cigéo).

3.1.2.2 Presentation of solid tritiated waste management procedures

In the orientation file concerning the storage of tritiated waste for which there is no disposal route, submitted in late 2008, CEA identified six main categories of waste without disposal route, on the basis of the established inventory:

- very low level tritiated waste (pure or mixed tritiated waste);
- pure tritiated waste releasing little gas;
- pure tritiated waste releasing gas[84];
- tritiated alpha waste;
- tritiated irradiating waste containing short-lived radionuclides;
- tritiated irradiating waste containing long-lived radionuclides;

Each category of tritiated waste is associated with a storage concept of sufficiently long duration to allow the decay of the tritium activity in the packages and their acceptance in Andra's future disposal centres which will be designed for the characteristics of the tritiated waste leaving storage.

For the design of the storage facilities, the project identifies the following principles:

- reception and unloading of full transport containers and packages:
- storage of packages for a period of fifty years;
- the design of modular structures tailored to each category of waste;
- surveillance of the facility and the site as a whole;
- inspection and monitoring of the packages and containers;
- the construction of storage facilities close to the main production sites.

The waste is sorted and the packages manufactured by the waste producer. The tritium removal operations involving oven baking, heating, or melting of the most active waste, in order to reduce the tritium inventory or its gas release, are also carried out at the producers' premises.

[84] Tritiated waste is considered to release little gas if the unit measurement of tritium gas release from each package is less than 1 GBq/year /package.

General characteristics of the storages envisioned by CEA for the disposal of tritium-bearing waste

	Regulatory status	Structure	Storage principle	Ventilation	Capacity	dimensioning	Annual discharges in tritium	Tritium impact/scenario			Final outlet envisioned
								Internal fire	Earth-quake	Handling (worker dose)	
VLL waste	Authorised ICPE	Module in metal formwork	3-level stacking of packages	Natural ventilation	1 000 packages (1 PBq)	No dimensioning for earthquake	< 1 TBq	< 5 µSv at 500 m	< 2 µSv at 500 m	< 40 µSv	VLL type
Pure tritium-bearing waste hardly degassing	INB	Module in metal formwork	5-level stacking of pallets (4 drums)	Natural ventilation	15 000 drums (10 PBq)	No dimensioning for earthquake	20 TBq	< 3 µSv at 500 m	< 15 µSv at 500 m	< 40 µSv	LL/IL type
Pure tritium-bearing waste degassing	INB	Module in metal formwork	5-level stacking of pallets (4 drums)	Ventilation by extraction with exhaust shaft	7 000 drums (70 PBq)	Dimensioned for earthquake	140 TBq	30 µSv at 500 m	20 µSv at 500 m	0.8 mSv	LL/IL type
Tritium- and uranium-bearing waste	INB	Confining concrete shutters	Stacking	Ventilation by extraction with exhaust shaft	1 000 drums (10 PBq)	Dimensioned for earthquake	20 TBq	30 µSv at 500 m	1 µSv at 500 m	0.8 mSv	LL/IL type
Irradiating short lived waste	INB	Concrete structure (radiological protection)	Stacking (remote handling)	Ventilation by blowing / extraction with exhaust shaft	Equivalent 26 900 drums of 113 l (12 PBq)	Dimensioned for earthquake	100 TBq	Exclu-ded (concrete pack-ages)	20 µSv at 500 m	Excluded (remote handling)	LL/IL type
Irradiating long lived waste	INB	Ventilated shafts	Shaft (7 packages per shaft)	Nuclear ventilation (shaft) w/ tritium strip-ping unit	1 232 packages	Dimensioning for earthquake of bridges and shafts	35 TBq		70 µSv at 500 m		Deep geological repository

The cost of these facilities varies according to the particularities of the waste and more specifically the presence of certain irradiating gamma emitters. It varies from several million euros for the construction of a VLL tritiated waste storage module to several tens of millions of euros for the construction of a storage module for tritiated irradiating waste containing long-lived radionuclides.

3.1.2.3 Storage facility creation programme

The following estimated schedule for the creation of solid tritiated waste storage facilities takes account of the volumes of tritiated waste production by CEA and ITER. Moreover, and in particular owing to the volume (about 60 m^3 for a total inventory of 30,000 m^3 in 2060) and the activity (less than 90 TBq for a total inventory of 35,000 TBq in 2060) of the solid tritiated waste from the small producers, an overall solution for the whole of France was proposed by Andra and CEA further to the study carried out as part of the 2010-2012 PNGMDR. It consists in storing tritiated waste from ITER and waste from small producers within the same facility.

Tritiated waste types	Storage building	Anticipated commissioning date	Location
CEA pure tritiated waste releasing little gas	Unit 1 pure low gas building	2012	Valduc
	Unit 2 pure low gas building	2021	Valduc
	B Unit 1 pure low gas building	2037	Valduc
	B Unit 2 pure low gas building	2041	Valduc
CEA very low level tritiated waste	VLL building	2017	Valduc
CEA pure tritiated waste releasing gas	Pure gas release building	2030	Valduc
CEA alpha/tritiated waste releasing gas	Tritiated Alpha building	2017	Valduc
CEA tritiated irradiating waste with short-lived radionuclides	IR SL building	2020	Marcoule
VLL and LLW/ILW-SL tritiated waste from ITER and small producers	Building Phase 1	2024	Cadarache
ILW-SL tritiated and pure tritiated waste from ITER	Building Phase 2	2055	Cadarache

Estimated schedule for the construction of tritiated waste disposal facilities

3.1.2.4 Management of liquid and gaseous tritiated waste

The inventory of liquid or gaseous tritiated waste from the small producers represents a small part of the total inventory of tritiated waste. This is a closed inventory of limited volume, but high activity. This inventory cannot be accepted according to the specifications of the existing facilities.

For gaseous waste, the reference solution would appear to be decay storage along with the solid waste, as described in the previous paragraph, after stabilisation for the small-sized ampoules.

For the liquid waste, for which the design inventory is 500 litres, representing 50 TBq, solidification on the production sites is necessary prior to storage. This cannot be envisaged without prior repackaging and assembly. This therefore entails a high risk of tritium gas release given the activity levels concerned, risks that are incompatible with local processing. In the light of the studies conducted within the framework of the 2010-2012 PNGMDR the preferred solution would therefore be processing for disposal in a facility with the necessary discharge licenses, assuming that specific licenses could be obtained for disposal of these legacy volumes.

3.1.2.5 Outlook

The storage of solid tritiated waste with no disposal route concerns all solid tritiated waste already produced and to be produced until about 2060, which is the anticipated date for completion of decommissioning of the ITER installations. The inventory of waste concerned comprises six types of waste, for which the goal is to allow safe storage for a period of about fifty years, prior to disposal in Andra's centres. The creation of new storage facilities for a period of about forty years offers a solution guaranteeing short and medium-term safety for management of tritiated waste, pending its future acceptance in the routes identified. **CEA will be presenting an interim report on its storage creation programme to the PNGMDR working group.**

The solution proposed by Andra and CEA for solid tritiated waste from the small producers, consisting of shared storage with the waste from ITER, whose commissioning is envisaged for 2024, is proportionate to the potential risks presented by this waste. Pending commissioning of the planned storage facility, **Andra will, before the end of 2013, examine the possibility of accepting in its own storage facilities the tritiated waste from a defaulting producer or from a producer where the storage conditions are unsatisfactory. In these particular cases, CEA will be involved in this approach by examining the possibility of temporarily accepting such waste in its tritiated waste storage facilities, subject to approval by the competent safety authority.**

The solution adopted, or liable to be adopted by Andra for processing of gaseous tritiated waste, is disposal in the Andra repository, or storage, after stabilisation if necessary. **Andra will present the conditions for stabilisation of gaseous tritiated waste for which there is a risk of package breakage, before the end of 2013. Andra will also clarify the inventory of gaseous waste compatible with the Aube waste disposal facility's safety case and will propose procedures for accepting this waste, before 31st December 2013.**

Liquid tritiated waste from the small producers cannot be stored in satisfactory safety conditions over long periods on the sites of the waste producers. Shared management solutions must be preferred for processing of this waste and an overall safety analysis, including aspects concerning transport of this waste, must be carried out. **Together with CEA, AREVA and SOCODEI,**

Andra will continue with the preliminary studies carried out for processing of liquid tritiated waste from the small producers and will present an interim report before the end of 2013. Andra will in particular clarify the inventory of waste liable to be handled in each of the routes identified, according to the physico-chemical and radiological properties of this waste and its mode of packaging. The licensees concerned will verify how to deal with this waste, by identifying any special licenses that may need to be obtained.

3.1.3 Management of used sealed sources

3.1.3.1 Context and issues

Used sealed sources are covered by specific management provisions under the Public Health Code. This management begins as soon as devices containing sources are placed on the market, so that these sources can then be traced. About 40,000 sources are managed in the national inventory held by IRSN. Article R.1333-52 of the Public Health Code also requires that the supplier of an item containing sealed sources agree to retrieve the sources, at the request of its customer, the user. The supplier is responsible for the subsequent management of these sources: storage, return to manufacturer, disposal by recycling or transfer to a radioactive waste solution compatible with sealed source management.

There is a system of financial guarantees to cover defaulting suppliers for those sources requiring authorisation. This guarantee is in particular provided by a system of risk sharing by an association, the *Ressources* association, which covers the risks of its members. Non-members of this association can deposit a bond with Andra, according to a table defined by Andra, designed to cover the entire source management chain, from retrieval up to disposal.

These sources inventoried in the national file are supplemented by other sources or objects which were not recorded in this file, such as the sources used on smoke detectors (estimated at some 7 million), lightning arresters equipped with americium 241 or radium 226 radioactive sources (50,000 lightning arresters were sold between 1932 and 1986) and other objects used in the past (radium-based radioactive objects for medical uses). Sources exported by French firms, for which the user requests return to the supplier or country of origin, in compliance with international texts (more specifically the joint convention on the safety of spent fuel management and on the safety of radioactive waste management, an international treaty duly ratified by Act 2000-174 of 2nd March 2000) are not inventoried in the national file either.

In late 2011, Andra updated an inventory of the stocks of used sealed sources considered as waste, together with those in possession of them, pursuant to decree 2012-542 of 23rd April 2012 setting out the requirements of the PNGMDR. Andra was notified of 3.5 million used sealed sources. The companies in the French grouping of electronic fire safety industries (GESI) hold about 74% of all used sealed sources (smoke detectors), the armed forces hold about 23% (scrapped military equipment such as compasses, instrument dials, etc.) and industrial and medical sources represent about 3% (including the sources at CEA, the Compagnie pour l'Etude et la Réalisation de Combustibles Atomiques - CERCA, EDF, etc.).

3.1.3.2 Presentation of current management procedures

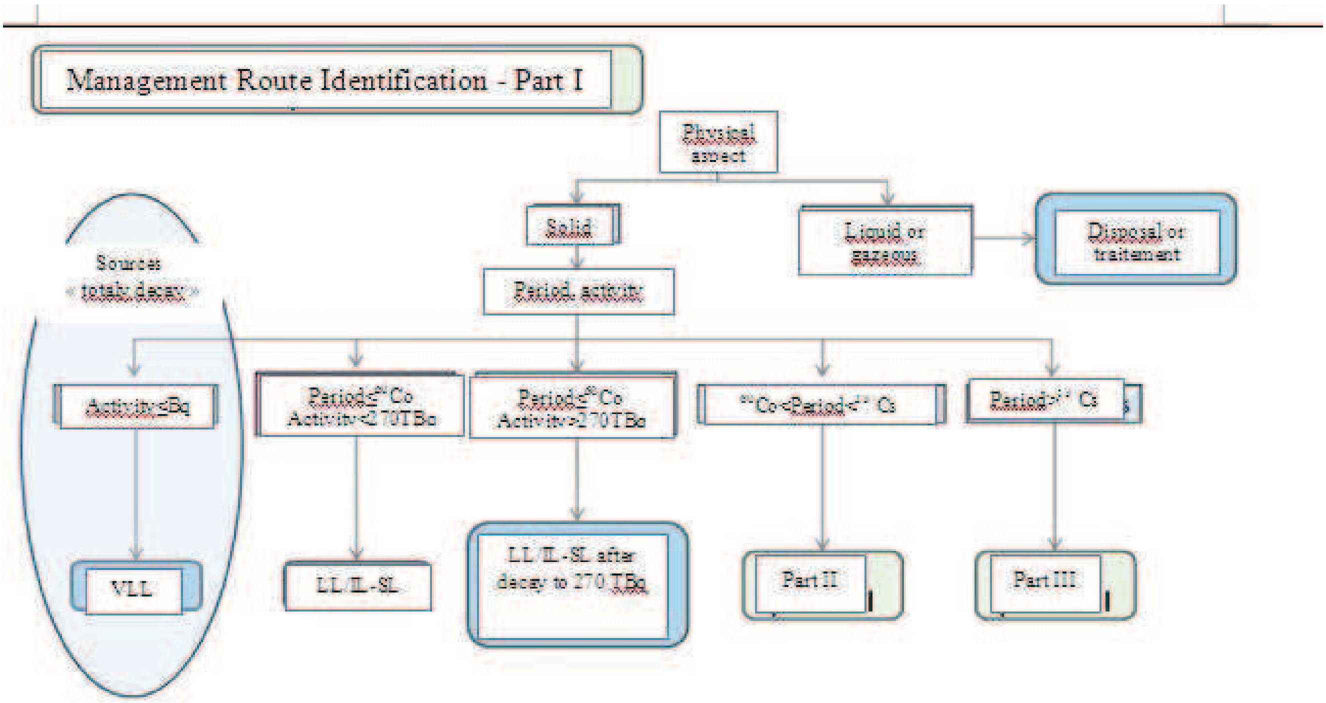
In 2008, Andra published a report on the disposal procedures for sealed sources, if they were to be considered as waste. The specificity of sealed sources is their concentrated activity and their potential attractiveness. In the event of human intrusion following the loss of all recorded memory of a repository, this attractiveness could lead to used sealed sources being retrieved by individuals unaware of the hazards. If the impact that would result from this retrieval were considered to be excessive, the used sealed sources would not be considered acceptable in the repository. The repository acceptance conditions for sources are therefore the subject of specifications, on the one hand with an activity criterion concerning the packages and structures, called "specific activity limit" (LAM) and on the other an activity criterion per radionuclide in each source, called the "source activity limit" (LAS).

In mid-2012, the only available specifications are those concerning the Aube waste disposal facility, which can accept sources with a radioactive half-life less than or equal to that of caesium 137, that is 30 years, with activity levels below certain thresholds defined according to the radionuclide concerned. For the VLL waste repository, the specifications prohibit the disposal of sources, but the regulatory framework of the Cires repository would enable acceptance criteria to be defined.

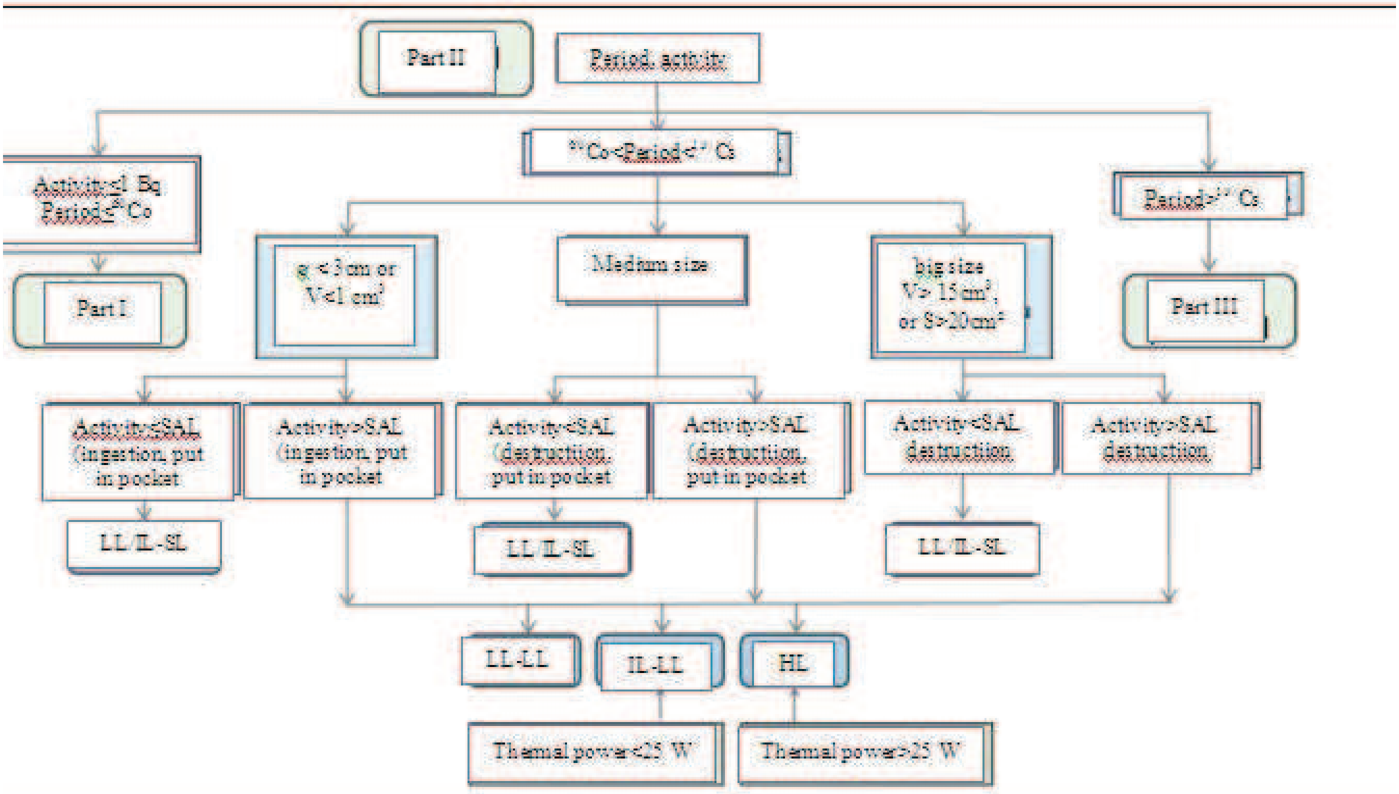
For sources not acceptable in the CSA, Andra examined the possibility of disposal in the LLW-LL route. These acceptance criteria are yet to be defined for a future LLW-LL type waste repository.

Used sealed sources not acceptable in surface or sub-surface repositories were allocated to deep geological disposal, along with ILW-LL waste for used sources having low exothermal properties and with HLW waste for more exothermal used sources.

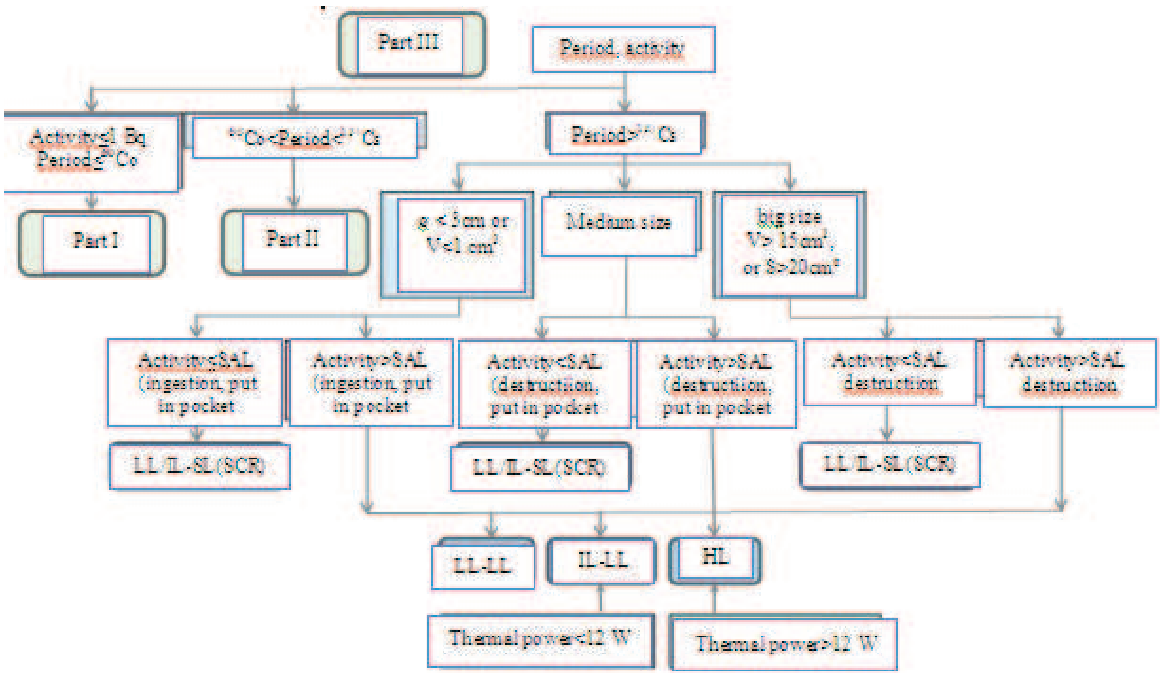
The method for identifying the disposal routes liable to accept each type of source is summarized by the decision-tree presented below, applying the following successive criteria to each sealed source: the form of the radioactive substance, solid, liquid or gaseous, the half-life, short or long, the activity of the disposal package and its compatibility with other repository parameters, primarily the thermal power and chemical nature.



Identification of reference solutions part 1



Identification of reference solutions: part 2



Identification of reference solutions: part 3

An arbitrary threshold of 1 Bq per source was used in the study to identify very low level used sealed sources. This threshold could be met by sources with a very short half-life which have sufficiently decayed (typically for a half-life of less than about 300 days, such as polonium 210, cobalt 57 and germanium 68); it may also concern a few sources with a longer half-life but with very low initial activity. Andra proposes disposing of these used sources in the industrial centre for collection, storage and disposal (Cires) in Morvilliers, with the threshold value to be adopted yet to be defined, on the basis of source acceptability studies for this centre.

In terms of safety, all used sealed sources of ^{60}Co, and the used sealed sources for which the radioactive half-life is between that of ^{60}Co and that of ^{137}Cs and whose activity is less than the LAS, may be disposed of in the Aube waste disposal facility. For used sealed sources with a half-life less than or equal to that of ^{60}Co, the operating safety rules for this waste disposal facility require an activity level per package of less than 270 TBq, which for some of these sources means that they must be placed in decay storage (for up to fifteen years).

The used sealed sources intended for sub-surface disposal are:

- most sources containing long half-life radionuclides;
- neutron sources of radionuclides with a short half-life but with a decay product with a long-half life: ^{233}Pa, ^{244}Cm and ^{252}Cf.

Some used sealed sources consisting of short or intermediate half-life radionuclides, which are not acceptable in the CSA, could be disposed of in sub-surface facilities if their activity remains compatible with the intrusion scenarios to be considered.

However, the effective volumes of sources which could be sent to sub-surface disposal cannot be estimated given that the acceptance specifications for this type of disposal are not available. The forecast volumes can be determined as follows.

Used sources from smoke detectors would be intended for sub-surface disposal. Used sources (or tips) from ^{241}Am lightning arresters, as well as those with ^{226}Ra and combinations thereof, could be disposed of in sub-surface facilities.

Deep geological disposal is proposed for needles and other radium-based objects intended for medical uses. Deep geological disposal is intended for the most active ^{239}Pu, ^{240}Pu, ^{242}Pu, ^{232}Th, neutronic used sources, source rods from EDF reactors and source blocks. The same applies to very active ^{90}Sr or ^{137}Cs sources.

For disposal, the main processes to be used are disassembly of the devices containing the sources and packaging appropriate to each disposal route (most of the used sealed sources are not at present packaged). Particular attention is paid to reducing the volumes to be disposed of, to transport constraints and to storage conditions (available capacity in terms of volume and activity, dose rates).

<u>CEA management strategy for used sealed sources</u>

The source users have the used or expired sources collected by the supplier. Current practice is for this supplier to be either the original supplier, or a supplier who has taken over the activities of this original supplier, or the supplier of new sources (reloading of a device or an industrial irradiator) or a replacement supplier appointed by ASN if the original supplier were to disappear or default.

Suppliers have their own storage capacity and usually have their sources or source batches collected by a manufacturer. Current practice is for this manufacturer to be either the original manufacturer, or the current manufacturer of the sources distributed by the supplier in question. In the event of return to a foreign manufacturer, or transfer to a foreign manufacturer for recycling, an export license is required, as stipulated in Article R.1333-17 of the Public Health Code.

In the past, CEA and CIS-BIO were major manufacturers and suppliers of sealed sources. CEA transferred is calibration sealed source manufacturing activity to CERCA and gradually ceased its sealed source manufacturing activities: only two very small volume activities remain (apart from a few highly specific activities related to national defence): the manufacture of fission chambers used for in-reactor measurements and the recycling or replacement of Pu ^{13}C sources used in the EDF reactor fleet. Maintaining specific recovery routes for this very small number of sources is problematic for the long-term. In the future, CEA hopes that the users of these sources will themselves manage them as radioactive waste once used or expired.

The public interest grouping for high-level radioactive sealed sources (*GIP Sources HA*) was set up to be able to recover, package and store high-level sealed sources (in particular caesium 137 and cobalt 60) manufactured and distributed in France by CEA until 1984 and by CIS-BIO until 2006, along with orphan sources of the same type.

The management strategy for CEA's used sealed sources was presented to the Advisory Committee for BNIs other than nuclear reactors, except for installations intended for the long-term disposal of radioactive waste, at the session of 15th February 2012, as part of CEA's overall waste management strategy.

The CEA source collection strategy, including the actions of the *GIP Sources HA*, is as follows:

- irradiating sources of Cobalt-60, Caesium-137 and some of the rarer isotopes such as europium, are collected by BNI 29 operated by CIS-BIO;
- isotope generator strontium sources were in the past collected and stored on the CEA sites of Fontenay aux Roses (BNI 165) and Saclay (BNI 72);
- neutron sources and high-level sources based on alpha isotopes are collected by the ATALANTE facility (BNI 148) at CEA/Marcoule;
- the other sources would be collected by the CERISE facility (installation classified on environmental protection grounds) at CEA/Saclay (or exceptionally by BNI 72 at CEA/Saclay in the event of incompatibility with the CERISE facility baseline requirements),

The strategy presented by CEA comprises several possible disposal routes:
- possible recovery of the source by a manufacturer (generally the supplier of CEA or of CIS-BIO);
- the possibility of source recycling by a manufacturer;
- the destruction of certain sources (gas, liquid, tritium and incinerable sources);
- management in routes dedicated to radioactive waste, taking account of the acceptance specifications of the disposal facility to which they would be sent.

The Advisory committee noted that CEA has defined recovery and disposal solutions for all the sources under its responsibility, a situation which is felt to be satisfactory. However, the Advisory Committee points out that the disposal routes will only be effectively created once certain authorisations are obtained for the use of packaging or storage facilities, and subject to the availability of certain transport containers.

3.1.3.3 Outlook

Used sealed sources are potentially recoverable. In accordance with the provisions of the Environment Code, preference should be given to reuse, then recycling and recovery. **CEA, together with other owners and Andra, will present the used sealed source recovery solutions to a meeting of the PNGMDR working group, along with the foreseeable recovery prospects for the various source categories inventoried.**

The management methods for used sealed sources, if they are considered as waste, are specified by a working group, which will draw on the work done for the 2010-2012 PNGMDR on the basis of the inventory updated in late 2011.

More specifically, the work of this working group will be organised around the following three points:
- processing and packaging: liquid and gaseous used sealed sources cannot be disposed of directly. They require prior processing in facilities with appropriate effluent handling and processing resources. The working group will on the one hand continue with the preliminary studies conducted into processing of liquid and gaseous sealed sources and, on the other, the packaging process studies for all types of sources. The potential for pooling management resources will be investigated, drawing on the recovery and disposal routes defined by CEA for all used sealed sources under its responsibility;
- acceptance in the disposal facilities: Andra will define acceptance criteria for the disposal of decayed or very low level sources in the industrial centre for collection, storage and disposal (Cires), consistently with the authorised operating perimeter of

this centre. On the basis of these criteria and the principles concerning the acceptance in other active or planned centres, the working group will specify the inventory of used sealed sources liable to be processed in each of the routes identified;

- the system of financial guarantees: the working group will finalise the work started for the 2010-2012 PNGMDR in order to update and if necessary supplement the existing system of financial guarantees. The Ressources association will be involved in this work.

At the end of 2014, the working group will submit a proposed optimised overall system for management of used sealed sources considered to be waste, incorporating the conclusions of the working group on the three points detailed above, and clarifying the resulting anticipated inventories per route.

At a meeting of the PNGMDR working group, CEA will also propose an interim report on any modification applications and any authorisations for used sealed source packaging and storage facilities, as it envisages them for the sources under its responsibility.

3.2 Management of LLW-LL waste

Low level, long-lived radioactive waste (LLW-LL) requires specific management, appropriate to its long lifetime, which rules out disposal in Andra's industrial facilities in the Aube. This waste in particular comprises graphite waste from the operation and future decommissioning of EDF's graphite moderated, gas-cooled, natural uranium type reactors, radium-bearing waste, mainly from the processing of ores containing rare earths, some of the bitumen packages on the Marcoule nuclear site and uranium conversion residues from the Comurhex plant in Malvési. Pending their disposal, possibly following reprocessing, the packages of LLW-LL waste are currently stored in facilities on the sites of the producers.

In 2008, the search for a disposal site capable of taking LLW-LL type waste was unsuccessful. Therefore, in order to learn the experience feedback lessons, working groups were set up, in particular in the HCTISN and within the framework of the Aarhus Convention. They submitted their recommendations in order improve preparations for the search for the future LLW-LL disposal site.

Two concepts are being envisaged for final disposal of LLW-LL waste: Reworked Cover Disposal (SCR) in an outcropping geological layer by excavation and then backfill, and Intact Cover Disposal (SCI), excavated underground in a layer of clay at a greater depth. Various scenarios are being studied for each type of LLW-LL waste. In particular, the possibility of processing and sorting of part of this waste (graphite waste and drums of bitumen packages) is being examined in order to optimise LLW-LL waste management. Without in any way anticipating the assessment to be transmitted by Andra in late 2012, **the 2013-2015 PNGMDR requests that a report on the feasibility of reworked cover disposal be submitted now, specifying the associated waste perimeter and a summary of the management strategy for the lowest level graphite waste and bitumen packages in Marcoule, drawing on the work conducted during the siting search process and the possibility of sorting/reprocessing.** In the meantime, the Cigéo deep geological repository project inventory presented in 2012 with a view to submitting a creation authorisation application file, makes provision for possible acceptance of LLW-LL type bitumen packages drums and waste from sorting/reprocessing of graphites.

3.2.1 Context, issues and experience feedback

Context and issues

Article 4 of Act 2006-739 of 28th June 2006 requires "the development of disposal solutions for graphite waste and radium-bearing waste". This waste is part of the low level, long-lived waste category (LLW-LL) as indicated in section 1.1.3.

The graphite waste is produced mainly by the nuclear power sector (graphite sleeves, stacks), while the radium-bearing waste comes mainly from the chemical industry (more specifically including TENORM waste), old radioactive objects (radium fountains, etc.) and certain used sealed sources (lightning arresters, fire detectors, etc.). It requires specific management, appropriate to its long lifetime, which rules out disposal in Andra's industrial facilities in the

Aube. Their low level of radioactivity however means that they do not need to be disposed of at great depth.

In June 2008, the Government tasked the Andra chairman with issuing a call for candidate local authorities for the siting of a LLW-LL type waste repository. The Government underlined the primary objective which was exemplary nuclear safety, radiation protection and environmental protection. It also underlined the need for extensive consultation with the regional authorities and the local populations concerned, to allow high-quality dialogue on the impact of such a project, in both environmental and socio-economic terms. Andra thus contacted 3,115 communes on land geologically favourable to siting of the repository, and presented the project to them. These communes had until the end of October 2008 to express an interest. Following this call for candidates and the subsequent deliberations by the municipal councils Andra received about forty favourable responses.

In late 2008, Andra submitted an analysis report (geological, environmental and socio-economic) on these candidates to the Government. The Government then started a process of consultation, more specifically with elected officials, ASN and the National Review Board (in charge of assessing the research into management of radioactive materials and waste).

In June 2009, Andra announced the Government's decision to carry out in-depth investigations on two communes and thus check the feasibility of siting the repository in these areas. However, the two communes chosen following the assessment then withdrew their candidacy. The Government and Andra, who had undertaken to allow the communes to withdraw from the process, duly noted these decisions, while regretting them.

The 2010-2012 PNGMDR required that the search for a LLW-LL type waste disposal site be continued, with the goal of exemplary safety, consultation and transparency, to guarantee the principle of selection on a voluntary basis. To allow sufficient time for consultation, the Government thus decided to lift the calendar constraints on the LLW-LL disposal project and stated that a public debate would be held at the appropriate time. At the same time, the Government asked Andra to re-open the various options, examining the possibility of managing radium-bearing and graphic waste separately, and continuing the discussions with the regional authorities or communes which had expressed their interest in 2008.

Experience feedback

After analysing the failure of the search for a site for the LLW-LL repository in 2008-2009, several working groups were set up.

In particular, the French High Committee for Transparency and Information on Nuclear Security decided, during its 8th October 2009 meeting, to create a LLW-LL working group (*GT FAVL*). Alongside this, the HCTISN and ANCCLI participated in the WG1 ACN working group of the Aarhus Convention and Nuclear, which is tasked with diagnosing application of the Aarhus Convention in the context of the LLW-LL disposal project.

These working groups met on many occasions in 2010 and 2011. Their work in particular consisted in conducting joint hearings of the various project stakeholders (players and local or national institutions). Each working group subsequently issued a summary of the hearings and a report.

With regard to the HCTISN's LLW-LL working group, the report was adopted at the HCTISN plenary session on 15th September 2011.
The HCTISN's main findings are as follows:

- there were initially too many communes (3,115) to allow satisfactory prior information;
- the duration of the siting process was too short and too restrictive to allow satisfactory dialogue;
- the lack of State involvement was obvious;
- the commune level chosen is not appropriate;
- the announcement of the 2 communes selected was considered to be political and occurred too late. The 6-month gap between January and June 2009 also contributed to blocking the consultation process, creating fertile ground for conflict.

Furthermore, the High Committee issued a number of recommendations, which can be summarised as follows:

- on the **selection of a site**: safety must be the primary factor in choosing a site. A small number of localities must be selected by the Government, following recommendations by Andra. Localities already housing nuclear facilities should be preferred, for sociological reasons;
- on the **calendar constraints**: time is needed for a satisfactory process, with definition of a realistic calendar and to make provision for a certain number of steps and interim milestones, along with scope for readjustments;
- on **responsibilities**: the Government must clearly make a commitment and must promote the public service and national utility aspect of the LLW-LL waste repository;
- on **the local special point of contact**: the level must at least be intermunicipal, with the support of the State and the large communities;
- on **public information**: extensive information must be given to the population, presenting technical information (nuclear, risks, waste, inventory, financial, social and economic aspects), the planned scheduling of the process, the interim milestones, the presentation of the roles of all those concerned, the procedures for modifications to the project;
- on **consultation**: in order to be effective, this has to be real: the project must be robust enough, while still enabling it to adapt to the locality and undergo certain modifications. Consultation must have a local guarantor;
- on **support for the project**: the repository project must be accompanied by a certain number of real economic and local development advantages and not just promises, which implies discussions with the localities and developing their own industrial, cultural or other projects with them. The support measures must be fairly distributed among the communes located in the vicinity of the actual site. Administrative boundaries must not constitute an obstacle.

The work of the ACN working group was completed in September 2011. The main recommendations of the WG1 ACN are as follows:

- give clearer explanations of the issues and problems (Aarhus Convention: Article 6.2 – *"The public concerned shall be informed, either by public notice or individually as appropriate, early in an environmental decision-making procedure, and in an adequate, timely and effective manner."* ;
- set up permanent coordinating entities (national and local);
- regularly inform about the progress of the project (an estimated calendar specifying the stages and financial resources associated with the project. This calendar is revisable according to technical, legislative and economic changes);
- substantiate the decisions at each step in the process (envisage the possibility of members of the public having recourse to a neutral and objective third-party institution);

- sign a long-term multipartite convention between the various national, regional, departmental and local players (project follow-up is based on the principle of long-term governance).

3.2.2 Storage of LLW-LL type waste

3.2.2.1 Radium-bearing waste

The inventory of radium-bearing waste currently comprises

- legacy waste resulting from the extraction of rare earths from monazite ore: radium-bearing residues and general solid residues (RSB) from Rhodia;
- the raw thorium hydroxide processing residues, which will be produced subsequently if Rhodia opts for recovery of the thorium, uranium and rare earths;
- the waste from manufacturing of zirconium sponges, zirconium and hafnium salts, from zircon ore (Cézus plant belonging to the AREVA group);
- some of the waste from the Itteville landfill (former settling pond and disposal area, annexe to the Bouchet plant) consisting of tailings and hydroxides (CEA);
- the waste from the post-operational clean-out operations on sites polluted with radium, uranium and thorium, managed by Andra as part of its public service duties.

The radium-bearing residues, packaged in about 26,000 drums of 220 litres, are stored in ICPEs 420 and 465 on the CEA site in Cadarache, on behalf of their owner, Rhodia.

The general solid residues are stored on the Rhodia site at la Rochelle. This waste is placed in bulk on a leaktight area under a tarpaulin. It represents a mass of 8,400 tons.

The Cézus waste is stored in a dedicated building on the Jarrie site. At the end of 2010, the mass of carbochlorination residues was estimated at 1,880 tons, that of sublimation residues at 780 tons, giving 2,900 tons of waste after mixing and stabilisation. By 2030, the quantity of waste produced is estimated at about 9,000 tons after stabilisation. Since 2005, the non-stabilised waste has been packaged in 220 litre reinforced steel drums. The Cézus storage building covers a surface area of 6,000 m² and has a capacity of 4,500 tons. Based on current production volumes, its storage capacity will be sufficient until 2023.
Waste from CEA's Itteville landfill is stored under a clay covering on a site adjoining that of the former Bouchet plant. The total mass of waste stored amounts to 40,000 tons, part of which could fall into the VLLW category.

Other types of LLW-LL waste from sectors outside the nuclear power industry will be stored by Andra in its new LLW-LL type waste storage building, authorised by order of 9th February 2012 and situated on the site of the Cires disposal centre. This facility has a storage capacity of 5,000 m^3. As at the end of 2008, the volume of waste to be stored was estimated at 2,200 m^3. The total volume of this waste is uncertain owing to the number of sites to be cleaned out and the level of clean-out required.

The future requirement in terms of storage capacity for radium-bearing waste is linked to the date of availability of a disposal solution for this waste. The assessment of this requirement will be kept up to date, in line with the progress of the studies concerning the management of this waste.

3.2.2.2 Graphite waste

The mass of graphite waste represents about 23,000 tons. Most of it will be produced during dismantling of the old gas-cooled reactors.

The graphite sleeves from the Saint-Laurent A reactor (EDF) are stored on-site in semi-buried silos. They represent a mass of 2,000 tons.

The graphite sleeves from the Chinon A2 and A3 reactors are stored on the CEA Marcoule site in the pits of the MAR 400 facility and the cladding removal facility. They represent a mass of about 750 tons. The retrieval and packaging of this waste is scheduled as part of the Marcoule post-operational clean-up and decommissioning programme to be completed by 2035.

The AREVA graphite and magnesium waste comes from processing of gas-cooled reactor fuels. It represents a mass of about 1,100 tons and is stored in silos 115 and 130 at la Hague. The programme to retrieve waste stored in these silos makes provision for packaging in special packages for storage in a dedicated building on the La Hague site, pending the opening of a disposal route.

The future storage capacity requirement for the graphite waste already produced and that to be produced in the future by decommissioning operations, is linked to:

- the results of the long-term management scenario studies (see section 3.2.3);
- the gas-cooled reactor decommissioning schedule;
- the safety constraints on existing facilities.

The graphite waste to be generated by the decommissioning operations on reactors G1, G2, G3 at Marcoule represents 3,800 tons. CEA's current schedule stipulates retrieval starting in 2030 (pending the availability of a LLW-LL disposal facility, in order to carry out regular removal of waste packages).

3.2.2.3 Drums of LLW-LL type bitumen encapsulated drums

The drums of bitumen packages (LLW-LL and ILW-LL) are stored in bunkers in the STEL at Marcoule, built between 1966 and 1994. All of the bunkers are to be gradually emptied. Some of these drums were thus extracted and reconditioned for storage in the EIP (multi-purpose interim storage facility commissioned in 2000). CEA made a commitment to ASN to retrieve 32,500 drums before 2027. All of the 60,000 drums should be collected by the end of 2035, depending on the conditions of availability and acceptance by the final disposal routes.

Several new waste retrieval, packaging and storage scenarios are being studied by CEA, together with Andra, while maintaining the above-mentioned reference objectives.

3.2.2.4 Waste from the settling ponds in the Comurhex plant in Malvési

The case of the waste already produced and stored on the Malvési site is dealt with in chapter 2.1.

The waste to be produced by the ICPE during the course of the installations modernisation project will be stored in the location of the current settling ponds B5 and B6. In its study, Comurhex plans in the short term to construct storage vaults or racks on the site of these two settling ponds.

Concerning the long-term management of the waste to be produced, Comurhex presented a joint long-term management solution for the waste already produced and that to be produced in the future: disposal on-site, using several disposal concepts. In its opinion 2012-AV-0166 of 4th October 2012[85], ASN indicated that a difference needed to be made between management of the waste already produced and that of the waste to be produced and that this latter should be managed in appropriate routes meeting the requirements in force for the management of radioactive waste.

3.2.3 LLW-LL type waste management scenarios

3.2.3.1 Summary of design options for a sub-surface repository

Two design options differing more specifically in their depth and the thickness of the host clay layer are being considered by Andra for disposal of LLW-LL type waste.

The reworked cover disposal option (SCR) is based on siting of the repository in a predominantly clay or marl outcropping geological layer of low permeability. At this stage, open-pit excavation down to the disposal level is being considered. The disposal vaults are excavated from the clay itself. The waste is packaged in containers and emplaced in the vaults. Once filled, the vaults are covered by a compacted clay layer taken from the site's excavated material, followed by a planted protection layer reconstituting the site's natural level.

In an intact cover disposal (SCI) option, the disposal vaults excavated underground are located in a thick clay formation at a depth of up to 200 m. Access is by a ramp and longitudinal drifts. Once the repository is filled, it is sealed and the drifts and ramps filled in with material excavated from the site.

3.2.3.2 Radium-bearing waste

The low mobility of the radionuclides contained in radium-bearing waste[86] and the significant decay in their radiological activity over a period of several tens of thousands of years[87] enables them to be disposed of in an SCR. Particular attention must nonetheless be given to the inventory of very long lived radionuclides (uranium, thorium) and the chemical content of the waste. The report expected form Andra in late 2012 will present a summary of the performance of SCR disposal of this waste. For the particular case of general solid residues (RSB), Rhodia is conducting studies to explore other management solutions, more specifically:

- the extraction of the rare earths and thorium content as part of the thorium materials recovery process (raw thorium hydroxide and thorium nitrates). The development of this new recovery process could reduce the anticipated volume of waste from the recovery of

[85] Opinion 2012-AV-0166 of 4th October 2012 on the management of temporary or legacy situations will be available on the website http://www.asn.fr, heading "les actions de l'ASN", "la réglementation", "bulletin officiel de l'ASN", "avis de l'ASN"

[86] Subject to favourable physico-chemical and hydraulic conditions being maintained.

[87] This time scale corresponds to preservation of the integrity of the repository as related to the times characteristic of geodynamic evolutions (external, glaciation).

raw thorium hydroxides by a factor of four. This processing could also lead to RSB being downgraded to the VLL category;
- incineration of general solid residues and the use of the ashes as a material for filling the containers of radium-bearing residue drums (provided that an over-pack is being considered for the radium-bearing residues);
- disposal in uranium ore dynamic processing residue ponds.

3.2.3.3 Graphite waste

The studies and research carried out by Andra up until 2009 showed that an SCR would not be able to achieve sufficient performance in terms of delaying and attenuating releases from graphite waste, owing to its presumed inventory of mobile long-lived activation products. The long-term safety of the repository therefore requires a thick, effective and durable clay barrier between the waste and the environment, which leads Andra to propose intact cover disposal (SCI) for this waste[88]. However, the assessments of the radiological inventories used as the basis for these studies have since then been considerably lowered by the producers, as a result of additional characterisation work.

Since 2010, recent developments and processing and changes in radiological characterisation have led Andra, EDF and CEA to envisage other management scenarios, based on upstream processing and sorting operations, which broadens the choice of potential disposal routes. In addition to the scenario which considers disposal of all the graphite waste in an SCI, the following alternative industrial scenarios are also being examined for EDF and CEA graphite waste:
- waste sorting for disposal of the stacks in SCR and of the sleeves in the planned Cigéo repository;
- extraction of radionuclides (^{36}Cl, ^{14}C, ^{3}H, etc.) with disposal of the partially decontaminated graphite in an SCR and the concentrated residues in the planned Cigéo repository;
- total destruction (by gasification) of the graphite after decontamination with disposal of the processing residues in the planned Cigéo repository.

These scenarios are being assessed by Andra, together with EDF and CEA, with regard to safety, technical-economic aspects and the technical and regulatory risks. The Andra report expected in late 2012 will present an interim summary of this work.

From the safety viewpoint, the acceptability in the SCR of the stacks or the partially decontaminated graphite will in particular depend on the following:
- the source term of this waste: residual radiological activity of ^{36}Cl and, as applicable, ^{14}C in organic form, release kinetics of the radionuclides;
- site characteristics;
- the ultimate total radiological inventory to be considered for this repository.

From the technical-economic viewpoint, the assessment will only be conclusive as of about 2015.

AREVA's graphite and magnesium waste comes from the reprocessing of GCR fuels, representing a mass of about 1,100 tons and is stored in silos 115 and 130 at la Hague. The

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88 The clay barrier is able to confine low mobility radionuclides and limit the dispersal of long-lived mobile radionuclides.

programme to recover waste stored in these silos makes provision for packaging in special packages for storage on the La Hague site in a dedicated building, pending the opening of a disposal route. The option to process this legacy waste was ruled out by AREVA owing to its heterogeneity (graphite / magnesium /etc. mixtures) which would require sorting in order to separate and extract substances of different types. According to AREVA, these sorting operations would not be able to guarantee the absence of substances incompatible with the processing of such waste.

Given that no LLW-LL repository consistent with the recovery and packaging of graphite/magnesium waste is available at La Hague, AREVA envisages on-site storage.

In the meantime, the Cigéo deep geological repository project inventory presented in 2012 with a view to submitting a creation authorisation application file, makes provision for possible space for graphite waste. The total radiological activity is considered. This activity is assumed to be concentrated in a small volume of disposal packages, partly resulting from the sorting or processing operations.

3.2.3.4 Drums of bitumen encapsulated LLW-LL

At the request of CEA, Andra conducted an SCR acceptability assessment in 2009 of the safety of the least active bitumen packages at Marcoule. The legacy drums of packages considered in the assessment are those which meet the following two criteria simultaneously:

- alpha activity per drum $\leq$ 10.72 GBq after 300 years.
- dose rate on contact with the drum $\leq$ 60 mGy/h.

On these bases, CEA today evaluates the number of drums concerned at about 40,000. The bitumen packages drums in the LLW-LL category will be identified and characterised during the retrieval and sorting operations.

From the safety viewpoint, the results obtained show that the acceptability of bitumen packages drums in an SCR depends on:

- the radiological activity, more specifically the mobile radionuclides (^{129}I, ^{36}Cl);
- the site characteristics;
- the ultimate total radiological inventory to be considered for this repository.

Andra and CEA defined a number of management scenarios for assessment:

- SCR disposal (CEA reference scenario);
- characterisation and sorting allowing disposal of the least active bitumens in SCR and the other bitumens in the planned Cigéo repository;
- processing (for the time being hypothetical) with disposal of the concentrated residues / ashes in the planned Cigéo repository;
- SCI disposal with the graphite waste (if this solution were to be adopted for the graphite waste);
- disposal in the planned Cigéo repository (as inventory reserves).

As an interim measure, the Cigéo project inventory is supplemented by reserves for the possible disposal of LLW-LL type bitumen packages. A disposal package volume of 39,000 m^3 is considered in the Cigéo project inventory reserves.

3.2.4 Outlook

3.2.4.1 Waste management scenarios

Concerning radium-bearing waste, most of the waste has already been produced and is currently being stored pending the availability of a disposal route. The deployment schedule for this route will affect the need for additional storage capacity as well as the calendar of post-operational clean-out operations on polluted sites and the clearance of the CEA landfill at Itteville (Essonne). **Andra is examining the possibility of taking charge of the radium-bearing waste in a reworked cover disposal (SCR) facility, in a sufficiently thick and primarily clayey outcropping or sub-outcropping geological layer.** Geological investigations are required in order to be able to continue the feasibility studies.

Concerning graphite waste, 20% of the weight by mass is stored pending the availability of a disposal solution. For the other waste to be produced by decommissioning of the gas-cooled reactors, compliance with the decommissioning schedule implies implementing a disposal solution which may or may not be preceded by a storage phase, depending on the date of availability of the disposal facility.

Together with CEA, which has research laboratories for analysing actual active graphite samples, and **Andra, EDF has launched an R&D programme to assess the possibility of using graphite decontamination and/or gasification processes.** This programme, which runs from 2012 to 2014, examines various processing solutions and means of packaging the concentrated residues. **By the end of 2014 it will have identified the potential performance of the processing options and the corresponding costs**. At the same time, EDF and CEA will be continuing their work to consolidate the inventory of the radiological content of the different types of graphite waste.

Between now and 2015, Andra will be continuing assessments of the acceptability of the graphite waste, produced by the scenarios considered, in the various types of repository, as well as the acceptability of the packaging methods for the waste. In order to rule on the possibility of taking charge of some of the graphite waste (possibly decontaminated) in a reworked cover repository, it will be necessary to conduct geological investigations and obtain additional data on the radiological inventory of this waste. All of these elements will make it possible to define the management route for the various types of graphite waste by 2015. The future storage capacity required will be clarified by the producers, taking account of the results of the assessments for the various management scenarios envisaged for this waste and the data supplied by Andra. In the meantime, the Cigéo project inventory will make provision for possible space for graphite waste. The total radiological activity will be considered and will be assumed to be concentrated in a small volume of disposal packages, partly resulting from the sorting or processing operations.

Concerning the drums of bitumen packages, currently stored on the Marcoule site and the subject of industrial retrieval and repackaging operations by CEA, **Andra will continue the studies to assess the safety of disposal of "LLW-LL" type bituminous packages in a reworked cover repository, according to the site data and the additional characterisation work carried out on the waste by CEA. By the end of 2013, CEA will provide a technical and economic analysis concerning the chemical or thermal processing option for these bitumens, in order to attempt to immobilise the radioactive waste they contain in other matrices.** The future storage capacity required will be clarified by CEA, taking account of the results of the assessments for the various management scenarios envisaged for this waste and the data supplied by Andra. All of these elements will make it possible to define a management route for this waste and will contribute to the decisions CEA is required to make in 2014 concerning the industrial investments required on the Marcoule site. In the meantime, the Cigéo project inventory will make provision for possible space for this bitumen waste.

Concerning the waste to be produced by the Comurhex plant in Malvési, it would be premature to opt for just one solution for management of the waste generated by the Comurhex plant and **a distinction must be made between the long-term management of the waste produced since 1960 and the management of the waste to be produced between now and 2050**. This latter must be managed in appropriate routes complying with current radioactive waste management requirements. Therefore, with regard to the waste to be produced, Comurhex shall contact Andra to study the management conditions for this waste, the possible synergies with certain uranium-bearing or thorium-bearing waste and its impact on the LLW-LL route as a whole. **By 31st December 2013 at the latest, Andra and Comurhex shall submit an interim report specifying the envisaged orientations and the optimised routes.**

The work carried out by the optimisation work group within the framework of the 2010-2012 PNGMDR identified the fact that the use of sorting strategies would enable certain waste primarily containing short half-life radionuclides to be redirected to an SCR type repository. The technical feasibility of such sorting scenarios will be examined and, as necessary, confirmed by Andra, on the basis of geological investigations, site impact modelling and the associated physico-chemical hypotheses adopted, as well as the waste characterisations supplied by the producers.

3.2.4.2 Siting process

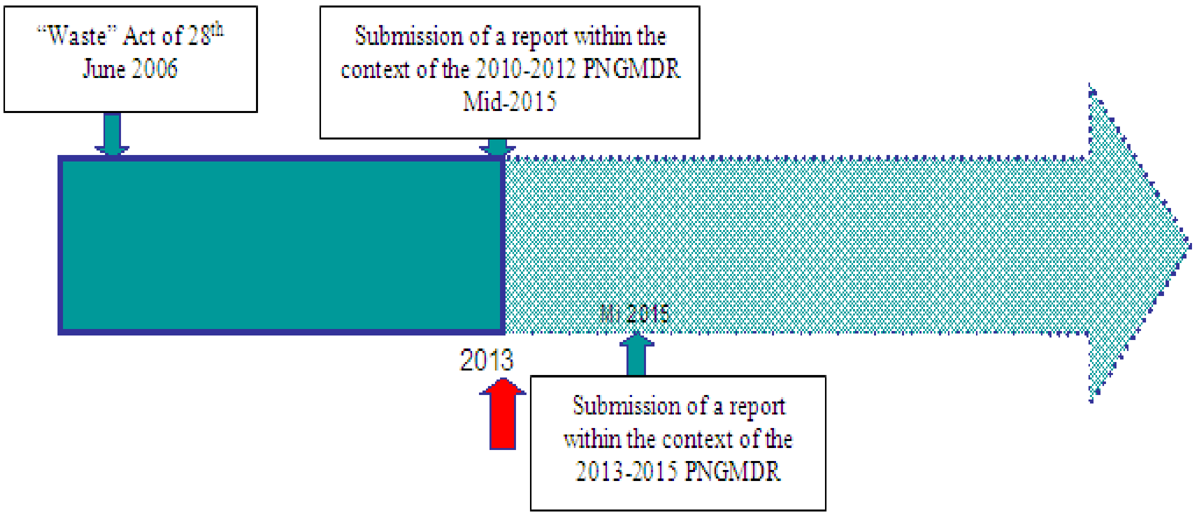
The report submitted by the High Committee for Transparency and Information on Nuclear Security stresses the public service nature of the LLW-LL waste repository and recalls that safety is the principal criterion in the choice of site. It recommends that the State select a small number of localities following recommendations by Andra, using the results of the 2008 call for candidates. It proposes that preference be given to localities already hosting nuclear facilities.
The High Committee recommends that enough time be given for completion of the process, with interim milestones in the calendar being defined. These recommendations will be considered in the rest of the process.

3.2.4.3 2015 milestone

Based on the geological investigations which could be carried out over the period 2013-2015, the continued waste characterisation and the specific R&D into waste processing, Andra will submit a report to the Government in mid-2015, comprising:

- **proposed choices of management scenarios for graphite and bituminous waste, notably with the possibility of relaunching the search for an Intact Cover Disposal site or not;**
- **a feasibility file for a reworked cover disposal site project, the scope of the waste to be emplaced in it and the calendar for implementing it.**

The provision of this technical and economic data by 2015 will enable the Government to establish decision-making milestones for the continued management of LLW-LL type waste.



Main milestones in the management of LLW-LL waste

3.3 Management of HLW/ILW-LL waste

The management of HLW/ILW-LL waste is studied according to the three complementary focal points identified in the 28th June 2006 Act on the sustainable management of radioactive materials and waste, now codified in the Environment Code: reversible disposal in a deep geological layer, storage and the separation and transmutation of long-lived radionuclides. In addition, research is being carried out into the processing and packaging of the waste.

The Environment Code identifies deep geological disposal as the solution for the long-term management of ultimate radioactive waste which cannot be disposed of on the surface or at shallow depth, for nuclear safety or radiation protection reasons. The 28th June 2006 Act requires commissioning of a reversible deep geological disposal facility in 2025. The underground installations in the planned repository, called Cigéo (French acronym for industrial centre for geological disposal) would be located in a layer of clay, at a depth of about 500 m. The research being carried out by Andra in the underground laboratory in the Meuse and Haute Marne départements has produced significant results concerning the feasibility and safety of such a repository. **The law requires that it be possible to review the installation creation authorisation application in 2015 and, subject to this authorisation being granted, that the centre enter service in 2025. Submission of this application shall be preceded by a public debate planned for 2013 and an Act will specify the reversibility conditions for this repository.**

Storage allows the safe management of HLW/ILW-LL waste pending the opening of a long-term management facility. The waste packages are stored in facilities on the sites of the producers. **The storage needs for HLW and ILW-LL packages will have to be analysed by AREVA, CEA and EDF together with Andra, taking account of the schedule of shipments to the planned Cigéo repository and the principle of reversibility**.

The studies being carried out into separation-transmutation, coordinated by CEA, aim to assess the industrial feasibility of the solutions for separating the minor actinides from the ultimate waste. Even if the studies show that separation-transmutation can be considered as a potential means of improving waste management, there are nonetheless a certain number of drawbacks (difficulties with cycle operations, extra costs, etc.). Furthermore, it does not preclude the need for geological disposal. **The forthcoming studies will affect the decisions to be taken following submission by CEA in late 2012 of a study assessing the prospects for the separation-transmutation options.**

3.3.1 Context and issues

<u>The three areas for study and research</u>

Planning Act 2006-739 of 28th June 2006 concerning the sustainable management of radioactive materials and waste requires that high level radioactive waste (HLW) or intermediate level, long-lived waste (ILW-LL) be the subject of three complementary research and study programmes:

- the separation and transmutation of long-lived radionuclides, linked to the research being conducted on the new generations of nuclear reactors, as well as accelerator driven systems dedicated to the transmutation of waste, so that by 2012, an assessment can be made of the industrial prospects for these solutions, with service entry of a prototype facility before 31st December 2020;

- reversible deep geological disposal, with the aim of selecting a site and designing a repository, for examination of the creation authorisation application in 2015 and, subject to this authorisation being granted, commissioning in 2025;
- storage, with the aim of creating new storage facilities or modifying the existing ones in order to address the needs identified by the PNGMDR.

The first area of research, being run by CEA, concerns waste which would be generated by future nuclear power reactors. The other two areas of research, run by Andra, concern ILW-LL and HLW waste already produced (about 60% of the ILW-LL waste and 30% of the HLW waste of the national inventory) or still to be produced by today's installations, those under construction which have obtained their creation authorisation decree as at 31/12/2010 and the ITER installation authorised by decree 2012-1248 of 9th November 2012.

The Environment Code identifies deep geological disposal as the solution for the long-term management of ultimate radioactive waste which cannot be disposed of on the surface or at shallow depth, for nuclear safety or radiation protection reasons, with storage being used to offer the necessary flexibility and implement this solution gradually and in a controlled manner. European directive 2011/70/Euratom of 19th July 2011 also recalls that the storage of radioactive waste is only a temporary solution which cannot be considered an alternative to disposal and that deep geological disposal is currently the safest and most durable solution as the final step in high level waste management.

Andra is carrying out studies into reversible disposal in a deep geological layer, more specifically in the Meuse/Haute-Marne underground laboratory and in the zone of interest for detailed reconnaissance (ZIRA) the perimeter of which was validated by the Government in 2010. This zone is suitable for the location of the underground facilities of the geological disposal industrial centre project: Cigéo. In 2013, Andra will propose a site for the Cigéo surface installations. Following the assessment of the safety, reversibility and design options transmitted to the Ministers responsible for energy, research and the environment, at the end of 2009, Andra in 2012 initiated the Cigéo industrial design phase, by launching the industrial preliminary design studies.

HLW and ILW-LL waste

HLW waste chiefly consists of fission products and minor actinides separated from uranium and plutonium during reprocessing of spent fuels, and then vitrified. The reprocessed plutonium and uranium are materials that can be reused in pressurised water reactors (second and third generation) and then, in the longer term, in fourth generation reactors which could come on-stream as of the middle of the century (§2.2).

In the meantime, the spent fuels are systematically included in the feasibility studies for geological disposal concepts[89]. Studies on the long-term behaviour of spent fuels for disposal without prior processing were continued under a CEA-EDF-Andra partnership. These studies provide a radionuclides release model. On this basis, Andra verifies that the disposal concepts (in particular the design of the ramp and the shafts) remain compatible with the hypothesis of direct disposal of spent fuels. In late 2012, Andra will submit an interim report on this subject, proposing avenues for studies and research for the period 2013-2015.

[89] The spent fuels from the EL4 reactor (Brennilis), stored in Cadarache, do not have sufficient reuse potential and are today slated for direct deep geological disposal.

Most ILW-comes from the nuclear fuel cycle of the first, second and third generation of reactors (metal structures of spent fuels, operating waste and solidified effluents from the spent fuel reprocessing plants and the MOX fuel fabrication plants). Reactor operating and dismantling waste and the waste produced by CEA's activities, also fall into this category.

The public debate on the Cigéo project

The public debate scheduled for 2013 will concern the project to create a reversible geological disposal facility for radioactive waste in the Meuse/Haute-Marne départements: the Cigéo project. The debate could in particular contribute to the discussions on local integration of the Cigéo project and prepare the future Act specifying the reversibility conditions of the disposal facility.

Andra will present the Cigéo project, based in particular on the industrial draft study produced in 2012. The recommendations of the debate could then be taken on board by Andra when preparing the creation authorisation application for the facility. The inter-departmental regional development scheme associated with the Cigéo project is drawn up by the State, in conjunction with the local authorities. The draft scheme will be presented at the public debate. The methods for transporting waste packages from the storage sites up to Cigéo are being studied by AREVA, CEA and EDF. The other stakeholders (assessors, elected officials, HCTISN, CLIs for the Bure Laboratory, associations, etc.) are also involved in preparing this debate.

In compliance with the Environment Code, the public debate calendar will be published by the French National Public Debates Commission (CNDP). As owner of the Cigéo project, Andra will propose a debate dossier to the Chairman of the Special Public Debates Commission (CPDP). This dossier, intended for the public, will be drawn up in compliance with the recommendations of the CNDP. Any documents necessary for the debate may be added as and when requested by the Chairman of the CPDP. Within a period of two months from the date of closure of the public debate, the Chairman of the CPDP shall publish a report on the debate and issue conclusions. Within a period of three months following publication of the conclusions of the public debate, Andra shall publish a decision giving the principle of and conditions for the continuation of the project and, as necessary, shall specify the main modifications made to the project submitted for public debate. Andra shall also state what steps it considers necessary in response to the lessons learned from the public debate.

On 10th October 2012, ANDRA submitted a request to the CNDP (French National Public Debates Commission) asking that this debate take place[90].

Submission of the Cigéo creation authorisation application.

After this public debate, Andra will submit the Cigéo creation authorisation application in 2015, accompanied by the dossier provided for in Article 8 of decree 2007-1557 of 2nd November 2007 concerning BNIs. The requested scope of this authorisation will cover all the waste in the project inventory. The 28th June 2006 Act describes the application examination process. The disposal facility reversibility conditions shall be determined by a new Act. The authorisation decree may only be issued after a procedure including examination of the Andra file by ASN and assessment by the CNE, an opinion from OPECST, consultation of the local authorities and a public inquiry.

[90] This request can be consulted on the CNDP website: http://www.debatpublic.fr, heading "activités de la CNDP", "saisines examinées"

To prepare for this phase, Andra shall present its reversibility proposals in the creation authorisation application back-up file, including governance of the repository and the design principles concerning its operation and the retrievability[91] of the waste packages. The creation authorisation application file shall be updated after enactment of the Act specifying the disposal facility's reversibility conditions.

Studies and research on reprocessing, packaging, storage and transport of waste

The law stipulates that "*reducing the quantity and harmfulness of the waste, is sought, in particular by (….) the processing and packaging of radioactive waste"*. In consultation with Andra, the producers of HLW and ILW-LL waste are continuing with studies concerning the understanding, processing and packaging of these wastes. The aims are to continue to reduce the volume of waste produced, to obtain a physico-chemical form that is as inert as possible with respect to the repository, both during operation and for the longer term, and to reinforce confinement of the waste inside the packages. The new processing and packaging methods must be examined with a view to technical/economic optimisation of the entire management chain for the waste produced, in the best conditions of safety for the sites of the operators producing it and for the safety and reversibility of deep geological disposal. The new processing methods apply mainly to waste that is yet to be packaged.

Interim storage is essential pending the commissioning of the Cigéo disposal project and then to accompany its industrial operation which will develop in stages, associated with an open and gradual decision-making process. The retrievability of the emplaced packages is a key component of the reversibility of the repository, which entails guarantees concerning the feasibility of possible storage of any packages removed from the repository.

Andra is overseeing and coordinating the studies and research concerning interim storage as a complementary aspect of disposal, prior to emplacement of the waste packages in the repository, and then as a factor in their retrievability. It is carrying out studies and research in conjunction with AREVA, CEA and EDF on the basis of their declarations for the production of the national inventory. The collaboration between Andra and the producers will continue to grow in order to build on operating experience feedback concerning the design, construction and working of the facilities and to enhance the complementarity between the plans for storage facilities on the production sites and the future deep geological disposal centre or packaging.

The summary of the results of the interim storage studies and research to be submitted by Andra in late 2012 will in particular propose management scenarios for the waste packages intended for the Cigéo project, along with an identification of the necessary storage capacities. It will go further than the study submitted in late 2009 proposing technical concepts, taking account of the principle of complementarity between storage and disposal. The aim will be more specifically to consolidate the durability of the future storage facilities for a time-frame of a century and increase their versatility with regard to the type of waste packages they can accept.

In addition to extra storage capacity, the long-term management of HLW and ILW-LL waste implies studying new industrial means of processing, packaging, transporting and monitoring long-lived waste with a view to its disposal. This work is being done together with Andra and the waste producers, who are the owners and licensees of the waste management facilities located on their sites.

[91] Retrievability designates the ability to retrieve the waste alone or in the form of packages after emplacement in a repository, regardless of whether or not this option is actually taken up.

Studies and research concerning separation-transmutation

The purpose of separation-transmutation is to remove the minor actinides from the ultimate waste, as they are the main contributors to the long-term radio-toxicity and residual thermal load after the waste decay period.

Research into separation and transmutation (first area of the 28th June 2006 Act) is coordinated by CEA, in order to *"assess the industrial prospects"* of the corresponding technologies (generation IV reactors, accelerator-driven systems) and to prepare for start-up of a prototype facility by about 2020.

In late 2012, CEA is to submit a file presenting the results of the research conducted (together with the other research organisations, particularly the CNRS). This file will notably present current technical progress in the separation and transmutation of the minor actinides, according to the various conceivable methods (homogeneous, heterogeneous, in a dedicated stratum) and for the fabrication of the corresponding fuels or targets. It will also include the results of the technical and economic assessments of the feasibility of the transmutation of the minor actinides[92] as well as its impact on all steps in the nuclear fuel cycle, according to the various criteria to be considered. CEA more specifically assists Andra with assessing the impact on the ultimate waste repository of the implementation of various materials management options. This subject is also dealt with in appendix 4: research aspects.

3.3.2 The processing and packaging of waste

3.3.2.1 Vitrification of HLW and ILW-LL waste

Vitrification, which has been successfully used for several decades with the "hot crucible" technology in the Marcoule and Hague plants, is currently the benchmark industrial process used in France for packaging fission product solutions resulting from the reprocessing of spent fuels (HLW waste). This technology has demonstrated its maturity and robustness, with more than twenty years of experience and the production of more than 16,000 vitrified packages.

Vitrification using an innovative cold crucible induction furnace has also been developed by CEA. This new process makes it possible to vitrify a much wider range of fission products, with higher production rates, in a melting pot that is less susceptible to corrosion than the hot crucible melting pot[93]. In 2010, AREVA thus deployed this process for the production of standard vitrified waste packages of intermediate level rinsing effluents, known as CSD-B, produced by the closure and final shutdown operations for the UP2-400 plant. Owing to the radiological nature of the effluents processed, the CSD-B packages fall into the ILW-LL category. Areva also planned to produce CSD-U packages through the vitrification of fission products rich in Molybdenum. The solutions are the result of processing of spent fuels referred to as "Umo" (consisting of a uranium and molybdenum alloy) used in the gas-cooled reactors (GCR) now shut down. Use of the cold-crucible technology is essential, in particular owing to the high molybdenum content of the solutions. Finally, the cold crucible vitrification process will also be used for fission product solutions produced by processing of spent fuels from pressurised water reactors by 2013.

[92] The research carried out within the framework of the 1991 Act demonstrated that the transmutation of long-lived fission products is not industrially feasible.

[93] The absence of contact between the molten glass and the cold metal guarantees that there is no corrosion of the crucible, despite the aggressiveness and high temperature of the molten glass.

With regard to the long-term behaviour, studies were carried out to estimate the performance of the CSD-B packages and the CSD-U packages in deep disposal. For the existing glasses, and in order to gradually increase the representativeness and robustness of the glass behaviour models Andra needs for the repository studies, a process is on-going to obtain detailed knowledge of the physico-chemical alteration mechanisms, which also take account of the possibilities of changes to the composition of the spent fuels. This research consolidated the vitrified waste package behavioural models, thanks to the development of a mechanistic model for glass alteration. These studies will continue, in order to gain a clearer picture and more specifically improve understanding of the various couplings with the repository environment materials. The glass formulation studies were continued, to take account of changes in the composition of the fission product solutions and process other types of waste, as well as of the glass production technology (cold crucible).

3.3.2.2 Other methods for packaging ILW-LL waste

For the other ILW-LL waste, three packaging methods have been or are still being used: compacting, cement encapsulation and bituminisation. Since the 1991 Act, work has been done to acquire knowledge, and this was formally laid out in the package data files transmitted to Andra. In these files, the producers specify the radiological and chemical properties of the waste and the packages. R&D studies are also being run by the waste producers, in order to determine operational package behaviour models.

One of the main questions which has been more particularly closely examined in recent years, more specifically through joint CEA/AREVA R&D programmes, concerns the hydrogen resulting from the radiolysis of organic materials, this being the main gas released by the packages during the repository operating period. As of 2007, CEA and AREVA initiated a scientific programme to acquire the tools for understanding and quantifying this contaminated polymer waste radiolysis process. The goal is to be able to evaluate the quantity of radiolysis hydrogen, to ensure that storage, transport and disposal are safe.

The overall scientific approach to predicting radiolysis yields was supplemented by direct measurement campaigns on waste packages and drums. All of these studies will lead to the use of a complete tool for modelling the production from waste packages. EDF is also carrying out a programme on the radiolysis of water in the waste encapsulation material, which will be able to quantify the production of hydrogen created by the irradiation of cement-packages packages subjected to a source term of a level comparable with the source term in a package.

For technological waste containing large amounts of alpha emitters, and more specifically within the framework of the 2010-2012 PNGMDR, AREVA submitted a study describing the envisaged packaging method. The particularity of these wastes is that they are mixed, in other words contain both combustible materials (organic waste) and non-combustible materials (metal waste). Two packaging processes are being examined, a cold compacting process and a thermal process.

The file for a package resulting from a compacting process as defined in 2008 does not present sufficient guarantees for long-duration storage or for deep geological disposal, so ASN asked[94] AREVA to examine other packaging methods for this waste.

Since then, AREVA has presented the following in its study:

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[94] ASN resolution 2010-DC-0176 of 23rd February 2010 is available on the website http://www.asn.fr, heading “les actions de l’ASN”, “la réglementation”, “bulletin officiel de l’ASN”, “avis de l’ASN”

- the principle of a cold compacting packaging process, to which the R&D results acquired since 2009 have been added;
- other processing-packaging processes, more specifically thermal. Processing such as to eliminate or reduce the content of organic matter in the waste would be a means of limiting the hydrogen produced by the transformation of this matter under the effect of radiolysis and thus potentially limiting the appearance of possible complexing agents and interactions with micro-organisms.

The best means of complying with the requirements would be to use incineration / melting / vitrification technologies involving plasmas.

The R&D conducted by AREVA for nearly two years now has not yet been able to conclusively determine the industrial feasibility of these technologies in a nuclear environment. In the light of the waste (ILW-LL and HLW) packaging technologies that are most advanced and industrially operational at present, thermal processes imply a number of major technological innovations (use of a plasma torch in a nuclear environment, utilisation of melting and vitrification operations as part of the same process, final packaging comprising two separate glass/metal phases in the same container, use of the final waste container as the melting/incineration crucible.

Furthermore, Article L.542-1-3 of the Environment Code stipulates that the owners of ILW-LL waste produced before 2015 must package it no later than 2030. Several interim targets were the subject of reports submitted by the producers in conformity with the 2010-2012 PNGMDR.

3.3.3 Waste storage and transport

3.3.3.1 Storage studies and research

The studies and research work run and coordinated by Andra on the storage of HLW and ILW-LL waste packages was the subject of an interim report in 2009[95]. Innovative technical concepts had then been proposed by Andra to encourage complementarity with the geological disposal project, from the following standpoints:

- versatility with respect to packages with different characteristics, whether or not placed in their disposal containers: this on the one hand entails offering more flexibility for the operational management of the waste before disposal and, on the other, anticipating management of the waste packages that could be retrieved as a result of disposal reversibility;
- the durability of the storage facilities, with the aim of a century being consistent with the operating life and reversibility of the disposal facility and with the foreseeable decay periods prior to disposal;
- surveillance of the facilities and the stored packages, contributing to the durability objective and allowing improved monitoring of how the packages evolve prior to their disposal;
- modularity, to help storage capacity adapt to changing needs (management flexibility, reversibility).

[95] Andra 2009 interim file C.RP.ADP.08.0038: inventory of existing storage capacity, progress of storage concept studies, proposed options to be studied after 2009.

For the period 2010-2012, Andra was tasked with further examination of these solutions, which could be of benefit to all future storage facility projects, and with examining the storage facilities which could be associated with the planned deep geological disposal centre. A summary of the results of all the studies and research will be submitted in late 2012.

Starting from a number of innovative storage concepts meeting the principle of complementarity with disposal, the studies and research carried out since 2010 aim more specifically to increase the allowable thermal power of the high-level waste packages[96], to optimise the handling processes and to improve the interface with the transport side. Surveillance systems integrated into the facilities are being studied in order to monitor their ageing and that of the stored packages, to measure the thermal power of the packages and the release of gases. At the same time, research into the behaviour of stored materials is being continued with a view to achieving durability of a century. It should be noted that research into surveillance and ageing is being carried out in conjunction with that into the reversibility of disposal, as a large number of the processes involved are comparable.

The concept studies for storage in excavated underground galleries or silos revealed greater complexity than facilities built on the surface or in trenches, in particular with regard to handling, modularity and the cooling of high-level waste packages. Consequently, these concept types are no longer being envisaged.

Following the dialogue between Andra and the Cigéo project stakeholders since 2010, it was decided that Cigéo would comprise no storage facility liable to take the place of those of the waste producers. The decay storage option integrated into the disposal centre[97], the study of which had been proposed in 2009, is no longer being considered. By early 2015, Andra will define limited buffer storage capacity to be provided for in the Cigéo project, looking to optimise the operation of the centre while taking account of the logistics chain as of removal from storage on the producers' sites.

Studies and research into storage, coordinated by Andra, can concern new facilities to be built on the producers' sites, with the latter being the project owners and having subsequent responsibility for nuclear operation. However, the results will also be of use in the design of the buffer storage capacity for the Cigéo project, notably characterised by greater need for versatility.

The studies and research could enable Andra to provide the storage licensees with technical support, to promote complementarity between their creation or extension projects and the disposal facility. As a result of joint work with Andra, AREVA thus added new construction requirements[98] for the vitrified HLW waste storage extension currently under construction at La Hague. This type of exchange could be repeated, notably for the new storage projects on the La Hague site and for EDF's ICEDA project.

3.3.3.2 Adequacy of storage capacity for the forecast waste inventory

The studies and research into storage are being conducted with a view to encouraging complementary between, on the one hand the "system" of current and future storage facilities on the various sites and, on the other, disposal, including the installations for primary package inspection, packaging in the disposal container and transport. In its opinion 2011-AV-129 of

[96] In the storage concepts for high-level waste presented in 2009, the search for versatility leads to a drop in heat removal performance.
[97] This option is being put off until a decision in around 2040.
[98] The formulation of the structural concrete was chosen for lower specific hydration heat and greater resistance to carbonation. The expansion bellows in the storage shafts, which are susceptible to corrosion, were placed at the top, away from all condensation. An inspection shaft was reserved to study ageing of specimen packages containing samples of packaging and structural materials, in the ventilation air and under intense irradiation,.

26th July 2011[99], ASN recommended that the studies be continued between Andra and the waste producers concerned in order acquire the necessary storage capacity for intermediate and high level, long-lived waste, in good time, prior to its disposal.

A coordinated overall approach involving Andra and the waste procures will gradually allow optimisation of the investments and of the operation of the various waste management facilities. This approach must cover the storage and removal from storage facilities, as well as those used for placing in the disposal container, inspection, transport packaging loading/unloading (see below) and repository emplacement. It shall take account of the reversibility of the repository, if necessary and provided that the relevant conditions are defined.

A first step was finalised in January 2012 with the joint establishment by Andra, AREVA, CEA and EDF of the first version of the industrial programme for waste management (PIGD). This programme aims to plan for and ensure consistency between the industrial resources to be implemented for development of the Cigéo project. It describes the inventory of waste to be selected for this project and defines the package delivery scheduling and traffic forecasts (see below, § 3.3.4). The aim of the programme is also to determine the industrial equipment that either exists or is to be implemented by the waste producers on the production sites and concerning transport operations, and to identify that to be implemented on the site of the Cigéo project, demonstrating the overall interfacing in the project's operational context.

Between now and 2025, the storage requirements for HLW and ILW-LL waste packages will be linked to the production of new waste and the retrieval and packaging of legacy waste, which are gradually increasing the volume of packages to be stored, as well as to the closure of the older facilities, which is reducing the available storage capacity.

After 2025, the requirements will also depend on the disposal scheduling and traffic. Disposal is gradually reducing the inventory of stored packages. It is freeing up storage space in the existing warehouses or means that when warehouses are closed they do not need to be replaced.

The work entrusted to Andra, AREVA, CEA and EDF in 2009 concerning the storage-transport-disposal scenarios for waste packages and the resulting storage requirements is still on-going. It is based on the first version of the PIGD.

At this stage, existing storage capacity and the planned creation and extension of facilities by AREVA, CEA and EDF would appear to be sufficient for management of HLW and ILW-LL waste packages until the 2025-2030 time-frame. All the waste packages will be stored on the production or packaging sites until their disposal.

La Hague site.

The existing storage facilities on the La Hague site are as follows:

- R7, T7, E-EV-SE, for vitrified HLW and ILW-LL (CSD-V and CSD-B respectively), representing a combined capacity of 12,420 packages;
- ECC, for compacted structural and technological waste (CSD-C). The current capacity is 20,800 packages, it being understood that reserve land would if necessary allow the construction of up to six modules equivalent to the existing module;

[99] Opinion 2011-AV-129 of 26th July 2011 is available on the ASN website http://www.asn.fr, heading "les actions de l'ASN", "la réglementation", "bulletin officiel de l'ASN", "avis de l'ASN"

- EDS for waste packages encapsulated in asbestos cement (CAC) and fibre-cement (CBF-C'2) containers, with a total capacity of 12,046 packages, adequate to cater for future production until 2040;
- the S and ES buildings for bitumen packages, STE2 sludges and waste contaminated by alpha emitters (packaging currently under study). Their total capacity of 47,000 packages (estimated for drums of bitumen packages) will in principle be sufficient for storage of all planned production between now and 2030.

Extensions are planned for the CSD-V and CSD-C:

- the first extension of E-EV-SE (called E-EV-LH), has entered the construction phase and, according to AREVA, commissioning is scheduled for 2013. It will raise the CSD-V, CSD-U and CSD-B storage capacity to about 20,830 packages. Additional capacity will be required as of 2017 (E-EV-LH2). Andra, AREVA and EDF are working on storage-transport-disposal scenarios for CSD-V which could be presented to the public debate on the Cigéo project. The conclusions will be issued by late 2012.
- an ECC extension could be necessary during the period 2020-2025. AREVA – with the help of Andra and EDF – is analysing scenarios involving removal of CSF-C from storage and transport, as well as the possibilities for optimising this extension, in particular its size, taking account of disposal of the first CSD-C packages in 2025. The removal from storage procedures are assessed with a view to optimising the system as a whole. The conclusions will be submitted by late 2012.

Marcoule site

The existing facilities on the Marcoule site are:

- storage of vitrified waste (SVM) from the Marcoule vitrification unit (AVM). According to CEA, its capacity of 665 m^3 is in theory sufficient to take all planned production at Marcoule. The latest periodic safety review of this facility (2009) showed that reinforcement and refurbishment work is necessary to be able to continue operation of this facility beyond 2014;
- building 213 of the Marcoule Pilot Facility (APM) in which the PIVER glasses have been stored since 1969. As this facility is only temporary, CEA will submit a storage strategy for all the Marcoule glasses;
- storage bunkers for legacy bitumen drums (categories ILW-LL and LLW-LL). These old bunkers needed to be cleared out and operations have been started;
- multi-purpose interim storage facility (EIP), started-up in 2000 to receive legacy waste or packages placed in 380 litre over-packs, primarily those which were removed from the north zone pits and then those currently being retrieved from bunkers 1 and 2. The modular design of the EIP at present comprises two vaults and could be extended if necessary. Saturation of the EIP's current capacity is expected in 2016, with a volume of 4,370 m^3, or 11,500 packages (ILW-LL and LLW-LL), if one considers the announced rate of clear-out of the bunkers.

In accordance with the stipulations of the PNGMDR in 2009, CEA and Andra are jointly studying the option of placing ILW-LL bituminized sludge packages in disposal containers and potentially other solid waste packages, on the Marcoule site (see § 3.3.3.3), paving the way for optimisation of the entire system by minimising the new storage capacity to be created and the volume of packages to be disposed of. For several years, CEA has been envisaging placing bituminised waste packages in disposal containers by about 2017, assuming that bitumen drum disposal packages are shipped to the planned Cigéo repository as of 2025. The scenario studied comprises the creation of a shipment holding facility (IAE) for interim storage of the disposal packages created between now and 2017, with the storage capacity being linked to shipments to the planned Cigéo repository starting in 2025. The date of disposal of the first package of bituminised waste needs to be confirmed on the basis of the following:

- a CEA R&D programme which aims to complement and reinforce the data already acquired concerning the intrinsic behaviour of the packages;
- detailed technical specifications for the concrete disposal package and the impact of the long-term evolution of the packages in the disposal vaults, to be provided by Andra. CEA will continue with the operations to characterise the legacy waste and primary packages (bitumen), and to define the primary packages to be produced for retrieval and packaging of the packages (RCD), the manufacturing scenario for the disposal packages in Marcoule (detailed functional specifications to be drawn up by Andra, manufacturing specifications then to be drawn up by CEA), the storage requirements and the storage removal methods. Furthermore, together with Andra, CEA will identify the package inspection facilities on the Marcoule site;
- the producers will share their R&D programme with Andra concerning the intrinsic behaviour of the waste packages, as well as the performance analyses obtained by comparison with the expected acceptability performance.

CEA also wishes to create the DIADEM facility, notably for storage of heavily irradiating ILW-LL waste packages. The creation authorisation application file, submitted to the Ministers on 27th April 2012, is currently being examined. Subject to it receiving authorisation, CEA envisages commissioning the facility in 2017. In addition to the waste produced by post-operational clean-out and dismantling of the facilities on the Marcoule site (APM, Phénix, etc.), this new facility would allow storage of irradiating waste from other CEA sites (Fontenay, Saclay, Grenoble).

Cadarache site

The existing facilities for waste management on the Cadarache site are:

- the radioactive waste packaging and storage facility (CEDRA), commissioned in 2006. The current buildings n°374 and 375 were designed to take low level waste and have a capacity of 4,450 m^3. Building 376 is dedicated to intermediate level waste, with a capacity of 2,350 m^3.
- BNI 56. The LLW and ILW waste packages stored in this BNI are being gradually retrieved, characterised and transferred to CEDRA. Packages of radium-bearing lead sulphates, solid waste and filtration sludges and large-sized containers (1,000 or 1,800 litres) and “source blocks” are also stored in BNI 56, representing a volume of about 1,275 m^3.
- the “Cascad” (Cadarache dry storage bunker, BNI 22) dry storage facility, which takes EL4 spent fuels from Brennilis intended for the planned Cigéo repository.

Extension of the CEDRA storage capacity is being envisaged by CEA, for commissioning in about 2023 (ILW waste) and 2025 (LLW waste).

The storage requirements for ILW-LL waste containing tritium, produced by the ITER installation, are covered in § 3.1.2.

Bugey site

On the Bugey site, the purpose of the ICEDA facility, authorised by decree 2010-402 of 23rd April 2010 is to store activated waste from operation of the reactors in service and that from the first generation EDF reactors being dismantled (ILW-LL waste packaged by cement encapsulation in C1PG concrete containers). Its envisaged commissioning in 2015 has been called into question by the 6th January 2012 cancellation of the building permit by the administrative court.

Valduc site

The co-precipitation/filtration sludges and concentrates, cement encapsulated until 1995, are initially stored on the Valduc site where they were produced and then sent for interim storage in CEDRA (Cadarache) pending the start-up of the planned Cigéo repository. The 100 litre drums of alpha-rich technological waste are also sent to CEDRA for storage. As of 2032, this waste will be packaged in Valduc. CEA is also planning to vitrify the effluents containing americium, plutonium and uranium as of 2020. These packages of ILW-LL glass produced from highly active effluents will also be stored on the site pending commissioning of the Cigéo repository project.

3.3.3.3 Adequacy of transport means for operation of the repository

The adequacy of the transport means for operation of the repository was the subject of an initial analysis by Andra, AREVA, CEA and EDF, included in the PIGD.

Developing transport packaging, the means of transport and the routing of packaged waste from the production or storage sites to the disposal centre, are the responsibility of AREVA, CEA and EDF. The infrastructures to be built in the Meuse/Haute-Marne départements, will be defined jointly with the inter-departmental regional development plan (SIDT) for Meuse/Haute-Marne produced under the supervision of the State. The organisation of transport operations will be presented at the public debate on the Cigéo project.

Waste shipped from the La Hague, Cadarache, Valduc and Bugey sites will be delivered in the form of primary packages, while that from Marcoule will be shipped, according to CEA's reference scenario, in the form of disposal packages. Alternatively, CEA is also considering the possibility of shipping Marcoule waste in the form of primary packages, with the exception of bituminised waste.

To be able to transport these packages, the producers must have a fleet of appropriate transport containers. The diversity of packages and the level of traffic and scheduling of the deliveries to the planned Cigéo repository, will require a large fleet. The use of existing transport containers should be preferred whenever possible, using existing approvals, but which will nonetheless need to be renewed.

For certain packages, containers are already in use or undergoing approval or design, or can be the subject of realistic extrapolations.

AREVA may seek to optimise the transport containers containing primary packages of large weight and size (for example drums of cement-encapsulated hulls and end-pieces, or CBF-C'2).

The TN 28 is considered by CEA to be the reference for the transport of vitrified waste packages from the AVM, APM and Atalante, provided that a few internal modifications are made and the current approval extended. These aspects will need to be examined to allow use of the TN 28 from Marcoule as of 2025.

For the transport of ILW-LL disposal packages produced on the Marcoule site, CEA considers the hypothesis of IP2 type packaging transport to be the reference. This hypothesis will need to be confirmed by a feasibility study, differentiating between the different package families (bituminous waste, solid, magnesium and powder waste). The corresponding containers will need to be developed consistently with the scheduling of deliveries to the planned Cigéo repository, as defined by the PIGD.

For the other packages for which no transport studies have been carried out, appropriate containers will need to be gradually developed by the producers for approval if they are to be operational within a time-frame compatible with the packages delivery schedule.

The main waste production and storage sites (particularly La Hague and Marcoule) have nearby infrastructures for road and rail transport. EDF's Bugey site has a rail network siding.

AREVA therefore considers that shipments from the La Hague site will be by lorry up to the Valognes transit station and then by rail to the planned Cigéo repository.

Concerning shipments from the Marcoule site, the scenario currently being envisaged by CEA is comparable to that of La Hague via the Orsan rail terminal (scenario assuming upgrading of the installations in this terminal).

Concerning the waste packages shipped from Cadarache, CEA is at present considering two hypotheses: carriage by rail via the nearest terminal and carriage by road.

For EDF, the reference solution would be for shipments from ICEDA, on the Bugey site, to be made preferably by rail. Generally speaking, rail transport is the preferred option for the transport of certain types of packaging, owing to the considerable weights (more than 100 tons for the CSD-C and the CSD-V for example). It is also the preferred solution for long distances, because it enables more containers to be transported in a single operation, thus improving the carbon footprint. In the case of road transport operations, some shipments would need to be classified as an abnormal load owing to the weight of the transport containers, given current hypotheses.

The inadequacy of the river transport infrastructures has ruled out this solution.

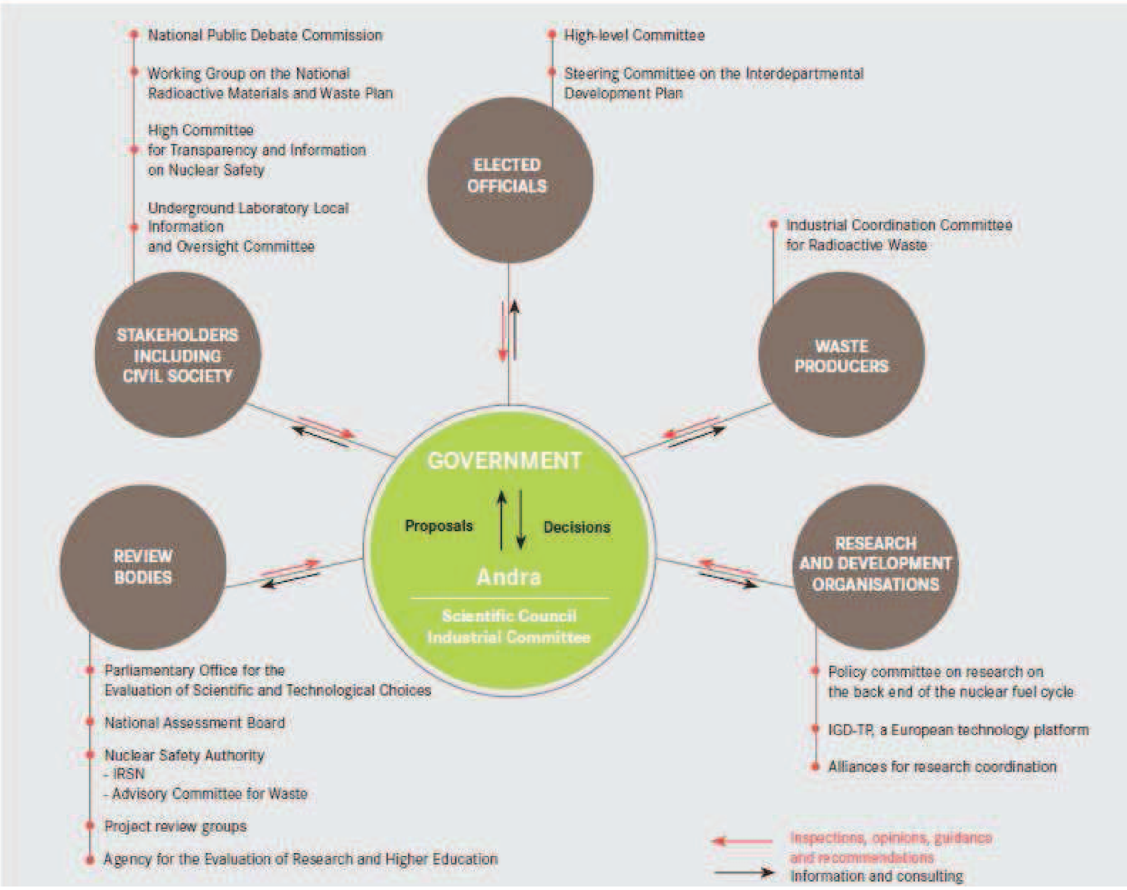
Within the framework of the Cigéo project, the possibility of building a rail siding within the perimeter of the repository is being studied. This option would appear in principle to be interesting from an industrial viewpoint, as it would obviate the need for transhipment and could be put to good use by other traffic to the repository over the hundred-year operating period. If the technical feasibility can be demonstrated, this would imply building a new track up to the planned repository. Several access scenarios are being studied in the Meuse/Haute-Marne interdepartmental regional development plan. The facilities integrated into the repository will be described in the Cigéo creation authorisation application backup file.

3.3.4 Reversible deep geological disposal

The studies and research into reversible deep geological disposal are being carried out in order to choose a site and design the repository, so that the creation authorisation application can be examined in 2015. Provided that authorisation is given, the centre could be operational in 2025. The specific process for examination of the creation authorisation application is defined by the Act of 28th June 2006. The submission of the creation authorisation application for the centre will be preceded by a public debate planned for 2013.

Subject to authorisation, the "Cigéo" industrial geological disposal centre will be built and operated underground for a period of more than a hundred years. This long duration entails the development of successive tranches of disposal structures, which will gradually enter into service, ensuring a degree of flexibility. Subject to authorisation being given, the first construction tranche aims to allow disposal of the first waste packages in about 2025.

The Cigéo project must meet safety, industrial, economic and social challenges. The external governance of the project enables various actors and stakeholders related to these challenges to be involved, under the control of the State and the assessment bodies.



Governance of the Cigéo project (Copyright Andra)

<u>The Cigéo siting approach</u>

The Cigéo creation authorisation application will concern the Callovo-Oxfordian geological clay layers studied in the underground research laboratory operated by Andra, for which the authorisation was renewed until 2030 by decree of 20th December 2011. The underground laboratory enables Andra to test the construction of structures underground, monitor their behaviour over long periods of time, study the disturbances created in the surrounding rock and develop observation-surveillance methods. The data acquired thus help prepare for the repository safety assessments as well as the industrial phase.

In 2009, Andra proposed a zone of interest for in-depth studies (ZIRA) based on geological criteria linked to land use planning and local integration further to discussions with local stakeholders. This geographical area of 30 km² was validated by the State in 2010, after consultation of the stakeholders and on the advice of ASN, for continuation of the siting studies for the underground disposal facilities. In 2010, Andra conducted in-depth geological studies of the ZIRA using 3D seismic means. According to Andra, this survey confirms the absence of any tectonic structure in the Callovo-Oxfordian clay layer and details the geological modelling of the layer and its surrounding rock. On this basis, Andra has a three-dimensional view of the distribution of the properties of the geological medium, to allow precise definition of the location of the structures in the Cigéo underground facility. In its assessment report of November 2011, the National Review Board (CNE) underlined that the new seismic campaign confirms the excellent homogeneity of the ZIRA and that following the initial interpretations of the seismic data, the 3D geological model would appear to be robust enough to be able to exclude the presence of structural discontinuities crossing the Callovo-Oxfordian layer and capable of providing a hydraulic connection with the surrounding aquifers. The CNE also considers that Andra now has a conceptual geological model justifying the transposition to the ZIRA of the data produced from the information acquired in the underground laboratory.

With regard to the surface installations, Andra is studying siting of the entrance to the ramp in the interdepartmental zone, on the Haute-Marne side, adjacent to the Meuse. Several shaft siting scenarios directly in the ZIRA are being studied. Andra will draw up the Cigéo creation authorisation application on the basis of the siting decision in 2013.

The draft interdepartmental regional development plan is produced under the supervision of the Prefect of the Meuse *département,* acting as coordinating Prefect, jointly with the local stakeholders. It will be presented at the public debate on the Cigéo project. The purpose of this plan is to provide answers to the land use planning and development challenges linked to siting of the Cigéo.

<u>The Cigéo industrial design studies</u>

In 2009, Andra presented safety, reversibility and design options. They were examined in 2010 by ASN and its technical support organisation, IRSN, notably identifying the main points requiring further information for the creation authorisation application.
Following this examination, ASN issued opinion 2011-AV-129 on 26th July 2011 in which, while noting that Andra had developed the main design, safety and reversibility provisions for management of risks during operation of the repository, it underlined the need for Andra to provide clarification and further data for the repository creation authorisation application and recommended that Andra remain attentive to satisfactory coordination between the research and experimentation work and the various project development phases, in order to ensure the availability of the data needed to demonstrate the safety of the facility when the time comes.

The performance of nuclear activities underground and their gradual extension as new waste packages are emplaced in the repository, implies defining the best possible baseline requirements for nuclear safety and security.

On these bases, Andra initiated the Cigéo industrial design studies phase. The aim is to prepare the Cigéo creation authorisation application for 2015 and, subject to authorisation being granted, the construction of a first tranche.

The requested perimeter for the creation authorisation will cover all the waste in the Cigéo industrial project inventory and the reserves presented below.

Within this context, the Minister responsible for energy ensured that a dedicated industrial organisation was put into place in early 2011. This in particular includes an external project review process. A first project review was therefore ordered in 2011 by the Minister responsible for energy. It took place in the first half of 2011. Its purpose was a technical and organisational project review, prior to launch of the rough drafts phase.

In mid-2011, Andra issued an initial tender for system lead contractor for the rough drafts phase. Company selection and the purchase order corresponding to lead contractor for the rough drafts phase, took place in early 2012. The end of the rough drafts phase is scheduled for late 2012.

Between now and the public debate, the Advisory Committee for waste will review the documents submitted by Andra since 2010 concerning the industrial waste management programme and its updates, the 3D seismic results produced in 2010 and their integration into the conceptual site model, as well as the long-term behaviour of spent fuels in repository conditions (study submitted within the framework of the 2010-2012 PNGMDR), and IRSN's corresponding analysis. ASN will issue an opinion further to this review.

<u>The Cigéo project inventory and the scheduling of waste package deliveries</u>

The forecast inventory to be adopted for the Cigéo industrial project was updated by Andra, AREVA, CEA and EDF in 2011. Initial hypotheses were also defined for the scheduling and forecast package delivery traffic levels. These elements are the input data for the Cigéo industrial design.

The inventory was based on the "industrial scenario" defined by AREVA, CEA and EDF in 2011. The hypothesis underpinning this scenario is continued nuclear power generation with reprocessing of all the fuels unloaded from the second and third generation reactors and fuels from the Phénix and Superphénix reactors. The operating life conventionally used as the reference for pressurised water reactors, including the Flamanville EPR, is fifty years. This hypothesis in no way prejudges the results of the ten yearly reactor safety review nor any procedures linked to the extension of the operating life of these reactors beyond 50 years. This scenario states that the materials (uranium and plutonium) not reused in the existing 58 PWR reactors and the Flamanville EPR, could be reused in future facilities. The waste produced by a possible future fleet of reactors is not taken into account.

The inventory of the industrial project includes margins for uncertainties. For certain waste not yet packaged, packaging hypotheses were formulated.

For the creation authorisation application, reserves are added to the industrial project's inventory. They aim to take account of uncertainties in the industrial strategies or the implementation of new management solutions for LLW-LL type waste. The reserves considered thus cover the possible creation of a second third-generation reactor, envisaged in the multi-year programme of electricity production investments (period 2009 - 2020). As a precaution, they also cover certain waste resulting from the sorting or processing of graphite waste and certain bituminised waste, for which various management scenarios are being studied, without prejudice to the management scenario that will ultimately be adopted. This approach is in compliance with the provisions of Article L.542-1-2 of the Environment Code, which states that *"after storage, ultimate radioactive waste which, for nuclear safety or radiation protection reasons, cannot be stored on the surface or at shallow depth, shall require deep geological disposal"*.

From the technical standpoint, the Cigéo industrial project should not in any way be such as to rule out a possible change in the inventory, more specifically the reception of spent fuels other than those of Brennilis reactor EL4. Any change to the scope of waste in the creation authorisation shall require a further authorisation.

The design of the first operational tranche of Cigéo, and then the following tranches, requires planning and scheduling of the deliveries of the various waste packages. The characteristics of the first packages delivered determine the functionalities of the surface nuclear units to be commissioned on the repository site in 2025 and the first disposal vaults. The forecast delivery schedule of packages to Cigéo aims to optimise how the producers' shipment needs are handled, while ensuring compatibility in terms of traffic and logistical requirements from storage to the disposal site. It allows gradual ramp-up of Cigéo over the period 2025-2030. Furthermore, the disposal of bituminised waste from Marcoule and La Hague over the period 2025-2029 is an optional scenario being examined (§ 3.3.3). The disposal of heavily exothermic vitrified waste from La Hague is envisaged as of 2075, thus benefiting from a significant decrease in thermal output; the disposal of non-exothermic or only slightly exothermic waste could nonetheless continue beyond that date.

As mentioned in § 3.3.3, the packages delivery schedule is open to adjustment. It will be adjusted by Andra and the producers as part of an iterative optimisation approach consistent with the Cigéo design.

Package checks

A range of checks are and will be carried out on the packages prior to disposal: (i) checks performed under the responsibility of the producers as part of their process to produce and compile the files supplied to Andra, (ii) checks performed on-line under Andra responsibility to check the conformity of the packages with the declarations by the producers and the Cigéo safety baseline requirements, (iii) second-level spot checks also under Andra responsibility, within the framework of package quality control surveillance. If the checks under the responsibility of the producers are performed on their production, packaging and storage sites, the Andra checks may, as applicable, be performed on the sites of the producers or on the Cigéo site. Together with the producers, Andra will look to optimise the entire package inspection chain, in accordance with the respective responsibilities. In the authorisation application file, Andra will present the package check provisions designed to ensure the conformity of the disposal packages with the acceptance specifications.

Reversibility

Discussions on reversibility have continued between Andra and the stakeholders, notably the CLIS. Between 2007 and 2011, the OECD's Nuclear Energy Agency (NEA) ran an international project on reversibility and retrievability, with the aim of providing an objective round-up of the questions and viewpoints expressed in fifteen NEA member countries. An international conference was held in Reims in December 2010. Following the project, the NEA published an international understanding[100] of the notions of reversibility of decisions and retrievability of waste which, at the initiative of Andra, in particular comprised a retrievability scale designed to facilitate dialogue with the stakeholders. The public debate will prolong the dialogue on reversibility already engaged by Andra with the stakeholders.

In its opinion 2011-AV-129 of 26th July 2011, ASN recalled that in principle, reversibility can only be of limited duration. This is because if closure of the repository is deferred for too long, this could compromise the very notion of repository. ASN also recalled that the steps taken for

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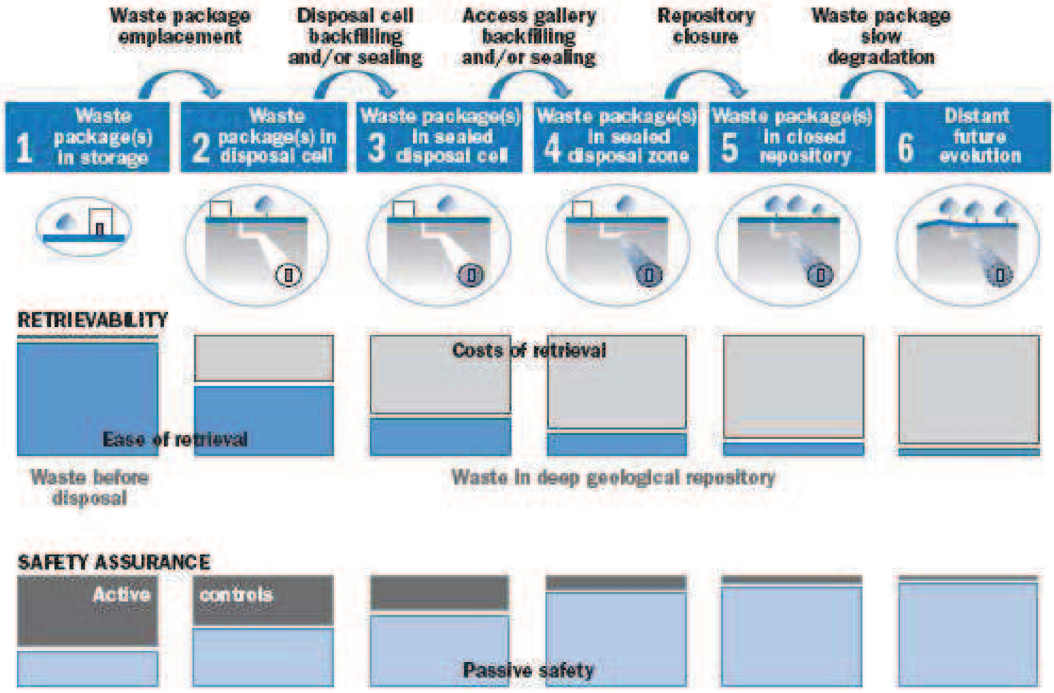
[100] International understanding of the notions of reversibility of decisions and retrievability of waste in a geological repository, NEA November 2011; Reversibility of decisions and retrievability of radioactive waste, NEA n° 7105, OECD 2012.

disposal reversibility should not compromise compliance with the safety and radiation protection objectives, both during operation and after closure of the repository.



Evolution of the ease of retrieval and the passive safety of the facility according to the level on the NEA scale[101]

Monitoring of health and the environment

The methods proposed for environmental and health monitoring will be presented in the public debate support file. Andra will estimate the Cigéo discharges on the basis of the data provided by the producers concerning gaseous releases from the packages and of their storage experience.

In 2007, Andra set up a Permanent Environment Observatory (OPE) in Meuse/Haute-Marne in order to establish an initial benchmark condition for the authorisation application file covering a period of several years and to prepare for the environmental monitoring system that will be put into place for Cigéo. The OPE provides feedback concerning a large number of measurable parameters in various environmental compartments, over an area of about 250 square kilometres and records covering several years. The OPE is not only a tool for monitoring over the very long term, but is also a means of recording environmental parameters. It obtained the "environmental research long-term observation and experimentation system" label in 2010.

In 2013, Andra will start up an eco-library for the long-term conservation of representative samples to allow retrospective analyses. This first French eco-library is currently under construction to meet the needs of Andra. It could be opened to public researchers.

This equipment could be integrated into a "Structure for observing and recording the Earth's environment" and contribute to the creation of a national scientific and technical campus in the Meuse/Haute-Marne region, dedicated to the subject of environment and memory.

[101] Source: NEA

Health monitoring

On several occasions, the local populations expressed their desire to see health monitoring implemented around the project. Andra will propose technical arrangements for this health monitoring, but its governance requires oversight by the public authorities.

The appendix concerning the concepts and plans for the period following closure presents the design provisions adopted to meet the safety objectives and the steps concerning oversight of the planned Cigéo facility.

3.3.5 Research into separation-transmutation

The main lessons learned from the separation-transmutation studies conducted within the framework of the previous PNGMDR include the following:

- although in a normal evolution scenario, the transmutation of minor actinides does not modify the estimated radiological impact of the repository, it does however allow a reduction in its footprint and the long-term radio-toxicity of the ultimate waste;
- separation-transmutation can in this respect be considered a potential avenue of progress for future nuclear systems, subject to confirmation of the feasibility of multi-recycling of plutonium in fast neutron reactors, which are the most appropriate for these operations (the transmutation of elements already packaged in glass would not seem possible);
- it does have drawbacks (notably greater inventory in the cycle, difficulties with cycle operations, extra costs) which define objectives for subsequent research and which also lead to research being focused first of all on the separation-transmutation of americium, which would also help reduce the heat given off by the waste.

The main R&D challenges for the period 2013-2015, presented in the research part of the appendix, are linked to the decisions which could be taken following the 2012 milestone specified in Article 3 of the 28th June Act.

3.3.6 Outlook

3.3.6.1 Storage and transport

Concerning storage studies and research, Andra will until 2015 continue to collect and build on experience feedback from the construction and operation of the existing facilities or those being developed, a process which was started in the period 2010-2012. It will also continue its research on the behaviour of the materials used for the construction of the storage structures and of the package materials as well as oversight techniques. In the period 2013-2015, it will look more closely at the storage concepts linked to reversibility, more specifically taking account of the condition of the packages retrieved from the repository.

Between now and 2015, Andra will draft a set of recommendations for the design of storage facilities to complement the disposal process. The content of the studies and research to be carried out over the period 2013-1015 will be clarified in the results analysis expected for late 2012.

Concerning the adequacy of the storage capacity for the forecast inventories, the work conducted by AREVA, CEA and EDF, jointly with Andra, will continue over the period 2013-

2015, if possible looking to optimise the system as a whole. They will utilise the results of the studies of the facilities on the producer sites, notably the Marcoule site and those of the rough draft studies and then preliminary draft for the Cigéo repository project. On this basis, in a first version for the public debate on the Cigéo repository project and then no later than the creation authorisation application, Andra will present the anticipated waste package management scenarios, incorporating storage and removal from storage, packaging, checks, transport and disposal. **At the same time, the storage requirements for HLW and ILW-LL waste packages will be analysed by AREVA, CEA and EDF - together with Andra - by mid-2015 and will take account of future waste production, packaging, scheduling of shipments to Cigéo and disposal reversibility.**

With regard to transport, for the period 2013-2015, the waste producers will by mid-2015 have produced guideline studies for future containers to be built, in order to ensure compatibility of the surface facilities of the Cigéo disposal project.

Concerning the Marcoule site, CEA will by mid-2015 produce:

- a summary, in line with Andra's waste collection rates, of the scenarios for removal from storage of vitrified waste packages up to shipment to the Cigéo project;
- a feasibility study of the modifications to be made on and near the Marcoule site to allow shipment of waste packages to the Cigéo repository project.

3.3.6.2 Waste packaging[102]

With regard to the processing and packaging processes, the important challenges for the coming years remain:

- decontamination: for liquids, reducing the quantities of reagents and toxic chemicals in line with changing regulations; for solids, reducing the quantities of effluents generated;
- the development of processing solutions for waste containing organic matter, more specifically technological waste (latex or neoprene gloves, other polymers, etc.) or effluents (organic solvents, surfactants, etc.);
- widening the range of waste that can be packaged by cement-encapsulation, compacting or vitrification, while retaining the advantages in terms of cost and implementation; the study of concrete formulations for packaging liquid effluents, the development of specific cement formulations according to the type of waste, the development of processes meeting production safety, volume gains, containment quality and other criteria.

The studies of the long-term behaviour of ILW-LL waste packages will continue to focus on three main topics:

- the behaviour of packages of bituminised waste;
- the production of gas from corrosion of the metals and the production of radioactive gases;
- the study of radiolysis of organic matter (other than bituminised sludges) and cement materials.

With regard to the particular case of technological waste containing organic matter, irradiating matter, or which is rich in alpha emitters:

- **before 31st December 2014, Areva will transmit its studies concerning the development of the process adopted for heat treatment of waste rich in alpha emitters and associated packages and should reach a conclusion on the feasibility of the implementation and nuclearisation of the chosen process. Before 31st December 2014, Areva will transmit a binding calendar for development of the process adopted, with justification of the time-frames planned for performance of the key development steps, allowing compliance with the 2030 deadline set by Article 542-1-3 of the Environment Code;**
- **by 31st December 2014, CEA shall transmit an inventory of the organic matter present as well as the complexing agents it is liable to produce within the alpha waste packages, either to be produced or already being produced. Within the same time-frame, CEA shall transmit the results of the degassing measurement campaigns in progress and the comparisons with the modelling results. It will complete the estimation of the degassing rates from the alpha waste packages, adding an estimation of the hydrogen from radiolysis of the interstitial water in the container concrete**.

[102] Opinion 2012-AV-0167 of 4th October 2012 on the packaging of ILW-LL waste produced before 2015 will be available on the website http://www.asn.fr, heading "les actions de l'ASN", "la réglementation", "bulletin officiel de l'ASN", "avis de l'ASN

Finally, **with regard to the ILW-LL waste produced before 2015 and to be packaged no later than 2030**, or which is packaged using methods that are potentially incompatible with acceptance in the disposal facilities being studied, AREVA, CEA and EDF shall transmit a study presenting the progress of ILW-LL waste characterisation and the consolidated design options for new waste packages appropriate to the disposal solution as envisaged. **By 31st December 2014, the licensees shall transmit the strategies they are adopting for compliance with the 2030 deadline set in Article L.542-1-3 of the Environment Code.**

3.3.6.3 Reversible deep geological disposal

<u>**The public debate and the creation authorisation application file**</u>

Andra shall be presenting a rough draft of the Cigéo industrial project in 2013 with a specific part concerning the first tranche. This will notably comprise several model structures and sealing demonstrators. The flexibility of the project (excavation of the underground drifts as and when needed) enables it to adapt to changes in the French energy systems. These rough studies will in 2013 be the subject of an opinion from ASN and the CNE. **Andra plans to initiate preliminary design studies in October 2013 with a view to submitting the creation authorisation application in September 2015.**

The Cigéo creation authorisation application file will in particular contain a preliminary safety report (RPrS) in which Andra will give a detailed and substantiated demonstration of the safety of all the operations to be performed for the first tranche of repository operations, which will be authorised by the creation authorisation decree. The RPrS shall also contain elements for assessing the overall safety of the operations included in the perimeter requested for the creation authorisation decree. Finally, the support file (RPrS and document dedicated to post-closure safety) will also contain information on the architecture and the feasibility and performance of all components of the repository once completed. It shall in particular contain conclusive evidence for assessing the feasibility of at least one technical solution for the transition to the repository oversight phase (sealing).

Andra will incorporate preliminary specifications for acceptance of waste packages in the authorisation application file. Andra will draw up a list of packages already produced, for which demonstration of compliance with the safety criteria essential for their acceptance in the repository has not yet been produced, along with the associated detailed arguments. It will identify the waste packages and the waste not yet packaged which pose problems in terms of managing the phenomenological behaviour in disposal conditions or compatibility with the planned disposal solutions and which could be the subject of studies on alternative packaging or processing solutions and will provide a detailed explanation of said problems.

In compliance with Article L.542-12 of the Environment Code, Andra will transmit a costing file for the Cigéo project to the Government. After collecting the observations from the waste producers and the opinion of ASN, the Minister responsible for energy will finalise the repository cost assessment and make it public.

During the public debate, Andra will present the reversibility measures it envisages taking. This concerns on the one hand the functionalities and technical provisions incorporated into the design of Cigéo to facilitate the possible retrieval of packages and reinforce the adaptability of the underground architecture during its development, and on the other the governance of the repository. This would include meetings with the stakeholders, following issue of the Centre's creation authorisation, for periodic review of the management options for the waste and the repository and the reversibility conditions. **The governance structures and procedures to be implemented could be discussed prior to Andra submitting proposals in mid-2015, with a view to the drafting of an Act setting out the reversibility conditions.**

Research and development

For the Cigéo project, the R&D work carried out between now and 2015 will have the following aims:

- provide scientific and technical data to supplement the demonstrations, more specifically the safety demonstrations, to back-up the creation authorisation applications. Particular attention is given to the construction techniques and to the performance of the seals on the underground facility;
- develop innovative techniques to be incorporated into the construction and operation of the Centre, as well as the means of observation and surveillance;
- prepare seal demonstrators to be included in the first construction tranche of the Cigéo project, supplementing the experiments carried out in the underground laboratory;
- prepare the development of future tranches of Cigéo.

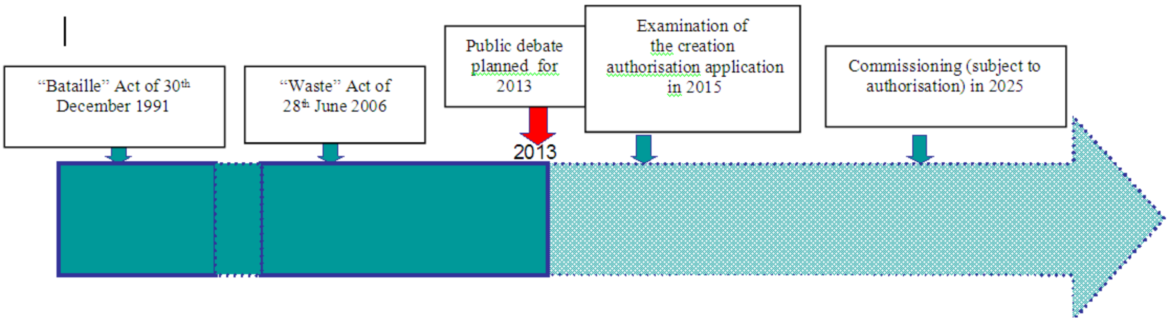
Research will be continued concerning the direct disposal of spent fuels. Between now and 2015, Andra will devote efforts to updating the feasibility assessment it presented in 2005, taking account of subsequent changes to the available knowledge: concepts of containers and disposal vaults, preliminary thermal design, architecture and footprint of disposal areas, thermo-hydromechanical impact and hydraulic functioning of the repository, assessments of criticality and radiological impact. In addition to the option involving disposal of the assemblies as-is, other disposal options could be explored and compared.

More generally, the continued R&D on waste, packaging, inspection and disposal will contribute both to the safety and the technico-economic optimisation of Cigéo - by proposing optimisations as operation of the centre progresses - and to reversibility, by looking at possible developments.

In order to complement the work done in the technical areas (geosciences, materials, instrumentation, simulation, etc.) developments will be continued in the human and social sciences, in order to contribute new data, notably for governance of the repository. Andra and the CNRS, together with the parties concerned, will run these studies.

3.3.6.4 Separation-transmutation

With regard to separation-transmutation, the studies for the period 2013-2015 will be linked to the decisions to be taken following submission of the files by CEA in late 2012.



Main milestones in the management of HLW - LLW-LL waste

3.4 Improving the overall consistency of radioactive materials and waste management

In order to improve the overall management of radioactive waste, working groups were set up under the 2010-2012 PNGMDR in order, on the one hand, to propose solutions for optimisation between management routes and, on the other, to study the waste for which there is currently no disposal route. **The work done by these working groups will be continued and the reports drafted in 2011 will be updated in 2014.**

Optimising the breakdown between the management routes is desirable. Thus, waste and package sorting has been identified as one of the main areas for the optimisation process. At the same time, possible processing hypotheses were examined for waste categories representing large volumes.

On the basis of the 2009 edition of the National Inventory, it was revealed that about 0.1% of the volume of waste produced could not be sent to any existing or planned disposal solutions. Five categories of waste have been identified as having "no disposal route": non-incinerable organic oils and liquids, certain asbestos waste liable to release fibres, waste containing potentially hydrosoluble mercury compounds, sodium pins from Phénix and Superphénix control rods and activated parts of accelerators. Studies are under way for these categories of waste for which there is at present no route, with a view to determining the steps to be taken to ensure that their management is compatible with solutions that exist or are to be created.

3.4.1 Optimisation of waste distribution between management routes

The 2010-2012 PNGMDR noted that for reasons of simplification, the current waste classification focuses on two major criteria, the activity level and the half-life of the radionuclides it contains, whereas in reality, additional criteria must be considered in order to determine whether a given type of waste can be accepted in a given management route. Each of the current or future waste disposal centres will apply acceptance criteria concerning the physical form, the chemical toxicity, the heating power, the release of gas, the radiological capacity, etc., derived in particular from the long-term impact assessments of repositories and their operating constraints. The actual packaging of the waste is also a parameter which has an influence on the route to which it should be sent.

The situation is made complex by the number of criteria to be taken into account, but also by the fact that these criteria can change, notably following the commissioning of a new repository, or the emergence of a new waste processing, packaging or disposal technology. Optimisation of the breakdown of waste between the management routes is therefore desirable and should be periodically updated. The 2010-2012 PNGMDR thus asked that a working group be set up to deal with the subject of optimising the distribution of waste between the existing or planned management routes.

The group concentrated on identifying all the waste worth examining with regard to the possible optimisation of its management mode, from the standpoint of the correct use of the various disposal resources (surface disposal in the CSA, prospects for subsurface disposal of LLW-LL type waste and disposal in Cigéo of HLW/ILW-LL type waste) and the overall consistency of the management modes. The technical possibilities and the industrial prospects were examined first of all with regard to the operational and long-term safety constraints. The waste given priority for optimisation is therefore:

- waste which could change disposal routes or for which the management solution could change, subject to upstream processing;
- waste for which the characterisation does not allow definition of a final management solution;
- waste intended for a route which cannot be guaranteed or for which the limits are not clearly defined or frozen.

The study conducted thus shows that waste and package sorting (if production has started) is one of the main areas for the optimisation process. If the radiochemical data on the waste are precise enough, it is possible to assess the benefits of this optimisation approach once the acceptance criteria in the various types of disposal facilities have been defined and are known. The spectrum of waste that could be concerned by sorting will increase once all the waste is better characterised in radiochemical terms, which also shows all the benefits of continuing the waste characterisation improvement process.

At the same time as looking at the possibilities relating to sorting, the working group also examined the possible processing hypotheses for two categories of waste representing large volumes. For LLW-LL graphite waste, one solution could be to concentrate the radionuclides in a small volume of residues, which could be sent to the planned Cigéo repository, with the remaining graphite then being sent to a reworked cover disposal (SCR) facility, subject to certain processing performance conditions being met. For general solid waste from Rhodia, for which the activity level is similar to that of VLL waste but in which the thorium content means that it must be sent to the LLW-LL route, two types of processing could be of interest, either for economic reuse of certain products, or to reduce the volume of waste, or for routing to surface disposal facilities (hazardous waste disposal facility, VLL waste disposal in Cires).

For all categories of waste that are potential candidates for optimisation, the following table summarises the measures identified for possible implementation of this optimisation (the IN category represents the long-term management solution currently designated for the waste in the national inventory).

Producer	Description of waste	IN Category	Actions identified in search for optimisation
RHODIA	General solid residues (RSB)	LLW-LL	Processing
AREVA	Packages of bitumen-encapsulated waste from effluents treated in STE3	ILW-LL	Package sorting
AREVA	Packages of STE2 sludges packaged in a bituminous matrix	ILW-LL	Package sorting
AREVA	Packages of solid operating waste cemented in CBF-C2 fibre concrete containers	ILW-LL	Package sorting
EDF	Waste from dismantling of EDF's first generation reactors	LLW/ILW-SL	Characterisation, sorting
EDF	Process waste packages from dismantling of EDF's GCR reactors	LLW-LL	Sorting
EDF	Graphite waste from EDF's GCR reactors	LLW-LL	Sorting, processing
CEA	Graphite waste from CEA's reactors	LLW-LL	Sorting, processing
CEA	Drums of bitumen encapsulated waste produced in Marcoule	ILW-LL	Characterisation, package sorting

CEA	Solid operating waste packages cemented in metal drums in Cadarache	ILW-LL	Characterisation, sorting
CEA	Metallic structural waste in 380l drums from Marcoule	ILW-LL	Characterisation, sorting
CEA	Magnesium waste in 223l drums from Marcoule	ILW-LL	Characterisation, sorting
CEA	Process waste in 380l drums from Marcoule	ILW-LL	Characterisation, sorting
CEA	Technological waste in 380l drums from Marcoule	ILW-LL	Characterisation, sorting

Optimisation solutions selected for each waste category by the working group

Using this typology, several overall scenarios for the breakdown of waste between the various routes were envisaged. To ensure that the examination is as wide-ranging as possible, the work took account of all the solutions that are today conceivable for disposal. The group in particular considered four hypotheses concerning the existence and characteristics of the disposal method which could be used for LLW-LL[103] type waste, given that as yet no definitive choice has been made. If no type of disposal is available, the waste is sent to the disposal solution for which the acceptance criteria are compatible. For example, in the absence of a disposal solution for LLW-LL waste, all waste that would be sent to such a solution would be redirected to Cigéo.

The overall results of the exercise show that the various avenues for optimisation would lead to an increase in the inventory of LLW-LL type waste intended for reworked cover disposal (SCR) and a reduction in the amount of LLW-LL type waste intended for intact cover disposal (SCI), notably owing to sorting or processing of the graphites, which have most influence over the results. Over and above these initial findings, the results reveal the effects of the various optimisation avenues which, as applicable, could be deployed industrially. However, this is simply an interim step and in order to be able to confirm the benefits of these scenarios, progress would be needed in several areas:

- improved understanding of the distribution of the radiological content of the waste (graphite waste in particular) will make it possible to make a more accurate identification of the optimisation potential of waste sorting;
- the report that Andra will be submitted in late 2012 concerning LLW-LL type waste management scenarios, should lead to proposals for adaptation of the siting process launched in 2008 and the definition of a new calendar for the project. The technical feasibility of the sorting/processing scenarios for graphite and other waste categories potentially compatible with SCR disposal, could then be confirmed or ruled out, on the basis of the site characteristics acquired in situ;
- test and study results are expected by 2014 concerning the industrial feasibility of graphite processing.

For the packages already produced and characterised, defining criteria for waste acceptance in SCR and SCI disposal facilities will enable the optimisation studies to be continued (example of certain solid waste packages encapsulated in CBF-C'2 fibre concrete containers for which the radiological characteristics do not meet the acceptance criteria of the Aube repository leading to default disposal in Cigéo being envisaged, whereas SCR type disposal could also be studied).

[103] Availability of one or two types of disposal for LLW-LL type waste (under intact cover SCI, under reworked cover SCR; see §3.2.3) and absence of disposal available for LLW-LL waste.

The interim report drawn up by Andra, AREVA, CEA, EDF and Rhodia for the 2010-2012 PNGMDR will have to be updated before the end of 2014, to present more complete industrial scenarios, including the operations to be performed (sorting, processing, etc.) upstream of waste disposal, as well as the main acceptance principles for SCR and SCI type disposal facilities.

In particular, AREVA, Rhodia, EDF and CEA must continue with their work to characterise the waste and waste packages already produced, so that they can be sent towards the most appropriate disposal route. Moreover, for bituminised waste packages, graphite waste and general solid residues, the licensees concerned will be required to present alternative management solutions to disposal, along with the strategy they propose to adopt.

3.4.2 Management of waste with no disposal route at present

The work done within the framework of the 2010-2012 PNGMDR confirmed that the vast majority of radioactive waste has an existing management solution or is covered by a disposal route project currently being studied. On the basis of the 2009 National Inventory, it was shown that only 0.1% of the volume of the waste so far produced could not be linked to existing or planned disposal routes.

The 2010-2012 PNGMDR therefore requested that a working group be set up to define appropriate management procedures for this waste and that a study therefore be submitted providing an inventory and proposing a programme of work to define management procedures appropriate to this waste for which there is as yet no available disposal route.

The group adopted an exhaustive approach, with the first aim of consolidating the inventory based on the list of waste identified in the 2010-2012 PNGMDR, the producer declarations for the National inventory and an additional examination of all waste for which no disposal route has been defined.

This inventory analysis found that the general characteristics of most of the waste initially identified falls within the spectrum of existing or planned management routes. Four waste categories considered to be "priorities" were identified by the working group as effectively corresponding to the definition of waste with no disposal solution. They thus require specific programmes to determine the steps to be taken in order to make them compatible with existing or planned routes. As shown in the following table, this concerns:

- non-incinerable organic oils and liquids, owing to their physico-chemical specifications and their activity;
- certain asbestos waste liable to release fibres (loose asbestos);
- waste containing potentially hydrosoluble mercury compounds;
- sodium pins from Phénix and Superphénix control rods.

Waste identified in the -2010-2012 PNGMDR		
Certain used solvents and oils	Certain asbestos waste	Tritiated incinerable waste
Irradiated beryllium reflectors	Reactor control rod	BF3 detectors
Irradiated lead	Irradiated aluminium	Irradiated cadmium
Uranyl nitrate	Waste with boric acid	Silica (ISOTOPCHIM) ^{14}C
Effluent treatment sludges	Special ashes	Contaminated mercury
Lead casks	Cobalt waste	Hafnium waste
Ampoule containing UF6	Tritiated distillates	NaK coolant
Blowing agent and lubricant		
Waste identified in the supplementary survey		
Materials reacting with concrete (aluminium, magnesium, etc.)	Batteries and WEEE (waste electrical and electronic equipment)	Sodium pins from Phénix and Superphénix control rods.

Defined disposal route

Waste with defined disposal route (additional characterisation required)

Waste with no disposal route

Summary of waste with no disposal route identified in the 2010-2012 PNGMDR

Used solvents and oils

Used oil and solvent type waste is generally processed by incineration, mainly in SOCODEI's CENTRACO facility. However, not all of it meets the acceptance specifications for this facility owing to its radiological activity or its chemical composition.

The firms that produce this waste are therefore required to develop specific processes. A number of solutions are today being examined and are liable to be transposed to an industrial scale. CEA is developing a process (DELOS) for washing and evaporation treatment of uniform batches of non-halogenated solvents, allowing mineralisation of the non-incinerable residues by means of a hydrothermal oxidation treatment process. CEA is also developing a second process (IDHOL) specifically for halogenated solvents, based on treatment to destroy organic compounds in an oxygen plasma in order to produce an effluent that can then be sent to effluent treatment plants. For its part, to process mixtures of chlorinated oils and solvents, AREVA investigated bacteriological treatment, but does not as yet have sufficient data to guarantee the destruction of certain oil components.

These processes have currently reached differing stages of maturity and the challenge is to develop and industrialise the most effective processes for treating the solvents and oils produced by basic nuclear installations and by small producers of waste outside the nuclear power sector.

Asbestos waste

The chemical toxicity risk from loose asbestos means that waste containing loose asbestos is today classified as having “no disposal route”. The toxicity risk is associated with the potential suspension of fibres in operating conditions as related to the long-term safety scenarios for disposal facilities.

Loose asbestos is prohibited in the Aube repositories for these reasons and a treatment solution able to eliminate it without risk is therefore needed. Processes have been identified, such as cement encapsulation, thermal destruction or vitrification. Studies into the possible treatment processes must be continued in order to obtain the technical and economic data needed for selection of the processing solution(s) to be adopted.

In the Cires repository, waste containing bound asbestos is authorised for direct disposal. However the Prefect's order requires that an "asbestos inventory, both total and per vault" be kept up to date and that Andra strictly limit asbestos acceptance. Similarly, in the Aube repository, a limited volume of bound asbestos is authorised for disposal (and prohibited in the compacting unit). Andra has tasked the ANSES[104] with a study into the acceptance of asbestos in order to precisely define the acceptable values, in particular for the Cires repository. At the same time, the producers will be required to improve their knowledge of the actual quantities of asbestos contained in the asbestos waste already produced and yet to be produced.

<u>Contaminated mercury</u>

The toxicity of the waste containing mercury is mainly linked to the chemical toxicity of mercury, classified as hazardous waste in Article R.541-8 of the Environment Code.

The aim is to develop physico-chemical stabilisation treatments such as to avoid any volatilisation of the mercury into the atmosphere, or leaching into the ground. The various processes are tending towards stabilisation in the form of mercuric sulphide, classified as non-hazardous.

The currently on-going studies are aiming for commissioning of a facility by 2014.

<u>Sodium pins from Phénix and Superphénix control rods.</u>

Sodium nuclear waste consisting of reactor control rod pins comes from the sodium-cooled fast neutron reactors: Rapsodie, Phénix and Superphénix. These pins can contain variable quantities of sodium, which are at present hard to quantify with any degree of reliability.

The risks arising from these pins containing sodium are linked to the latter's reactivity in contact with water, generating a release of dihydrogen (explosive gas), producing soda and giving off heat (ignition source) potentially leading to complete consumption of the sodium. The kinetics of the reaction generating the production of dihydrogen is variable (from slow to sudden) and must be controlled taking account of the conditions in which it occurs (quantity of sodium, rate at which water arrives, geometry of the cavities containing the sodium, etc.).

In order to manage these risks, several arrangements are currently being envisaged as likely to limit the release of hydrogen and the quantity of heat produced by the sodium-water reaction. The purpose of these arrangements is either to prevent the reaction by managing the sources of the reaction (sodium removal from waste), or to encourage slow production kinetics (ensuring favourable physico-chemical conditions as close as possible to the waste, limiting the water reaching the waste, the disposal package and/or the vault). Given the current level of knowledge and in order to justify management of the risks associated with disposal of this sodium waste, no particular solution can at present be preferred to another.

A working group consisting of Andra, CEA and EDF has initiated a review of the specific problem posed by sodium waste, with a view to sending this waste to the Cigéo deep geological repository.

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[104] French Agency for Food, Environmental and Occupational Health and Safety.

Three areas of research will in particular require more in-depth investigation:
- characterisation of the sodium/water reaction (liquid and steam) in repository conditions, with a view to defining a possible acceptance threshold for a limited quantity of sodium in the waste packages;
- the search for processes allowing sodium removal in order to eliminate the sodium from this waste or, as applicable, limit the quantity;
- the search for a durable leaktight container concept able to slow down the sodium/water reaction.

Activated accelerator parts

In addition to the work done for the four waste categories above, activated parts from accelerators (about thirty accelerators will be dismantled within about ten years, in addition to the activated parts already produced) require the definition of a management solution. The difficulty lies in the characterisation of these long-lived pure beta emitter wastes, which require measurements on samples and extremely costly modelling by the small producers. This waste is therefore stored in situ in safety conditions that are sometimes unsatisfactory.

The study of an overall management solution, including a system for generic characterisation of the waste, is to be conducted by Andra to allow the acceptance of activated parts in its repositories.

The working group set up under the 2010-2012 PNGMDR on the topic of waste with no disposal route will continue its activities beyond 2012. Its role will be to monitor the projects and ensure that they are progressing, with the aim of defining industrial solutions allowing the creation of effective routes for the four categories of waste identified. In order to monitor the progress of the measures initiated, the list of waste with no disposal route will need to be updated, more specifically taking account of the producers' declarations for the National Inventory.

With a view to integrating the acquired results, the working group will submit a report on the progress of the projects in late 2014.

Conclusion

In accordance with the provisions of the Environment Code, radioactive materials and waste must be managed sustainably, to protect the individual health, security and the environment. It is the responsibility of the producers and the French national radioactive waste management agency (Andra), under the control of the French nuclear safety authority (ASN) and the defence nuclear safety authority (ASND).

Radioactive wastes differ widely in their radioactivity, the lifetime of the radionuclides or chemical substances they contain, their volume and even their nature. From production, through sorting, packaging, interim storage and up until final disposal, each type of waste requires the implementation of a management solution appropriate to the nature of the waste, in order to manage all the inherent risks, more specifically the radiological risk.

In France, 90% of the of radioactive waste now has access to operational long-term management routes, while the other waste is temporarily stored pending the commissioning of such routes. However, the remaining 10% accounts for most of the radioactivity. Although radioactive materials and waste are now safely managed, the recommendations presented in this PNGMDR are essential. These recommendations aim to continue to improve the existing management solutions and to implement new solutions for all waste. They are in line with the objectives of reducing the quantity and harmfulness of waste and the creation of disposals facilities, in particular deep geological disposal as a solution for ultimate waste that cannot be disposed of on the surface or at shallow depth, for nuclear safety or radiation protection reasons.

The 2013-2015 edition of the Plan was drawn up in the light of the results of the studies initiated under the previous Plan, most of which were incorporated into the decree and order of 23rd April 2012 setting out the requirements concerning the 2010-2012 PNGMDR. It is also based on the National Inventory of radioactive materials and waste published by Andra in June 2012, which assesses the prospects for the production of waste in the coming decades and the storage capacity requirements.

The 2013-2015 PNGMDR continues and expands on the actions initiated by the previous version and stresses the need to develop overall industrial management systems and management methods for high-level and intermediate level, long-lived waste. It in particular proposes the following measures.

Developing new long-term management modes

The 2013-2015 PNGMDR requires continuation of the studies and research concerning high level and intermediate level long-lived waste and in particular those concerning the planned deep geological disposal repository, Cigéo, which will enter a new phase during the period 2013-2015 with the submission of the creation authorisation application in 2015, preceded by a public debate planned for 2013. It also requires continuation of the studies into the packaging of intermediate level long-lived waste, more specifically to comply with the 2030 goal of packaging of the waste produced before 2015 as laid down in Article L.542-1-3 of the Environment Code.

With regard to low level, long-lived waste, the 2013-2015 PNGMDR requires definition of management scenarios, in particular by continuing the studies into sorting, characterisation and processing of graphite waste and bitumen packages, as well as feasibility studies concerning disposal options for the waste already produced by Comurhex Malvési.

Improving existing management modes

The 2013-2015 PNGMDR requires the implementation of tools allowing monitoring of the volume and radiological capacities of the disposal facilities, thus anticipating any need for new capacity. It also requires the development of reuse and recycling solutions for very low level waste, in order to preserve disposal site resources.

The 2013-2015 PNGMDR also requires continued studies on mining processing residues, in order to propose tangible improvement measures, whether in terms of understanding the exposure risk for the general public, the long-term strength of the embankments or changes in water treatment techniques. It also requires continuation of the process to implement the circular from the Ministry responsible for Ecology and from ASN, dated 22nd July 2009, in order to determine whether the sites on which mining waste rock is reused are compatible with the usages and to reduce their impact when necessary.

Taking account of significant events over the period 2010-2012

Moreover, the 2013-2015 PNGMDR requires identification of the investments needed to guarantee the future of the management routes for the waste generated by small producers outside the nuclear power generating sector and, in particular, to continue with the studies concerning the treatment of liquid and gaseous tritiated waste produced by this sector. It also requires continuation of the work started to define a management system for used sealed sources. Finally, the 2013-2015 PNGMDR requires analysis of the experience feedback from the shutdown for several weeks of the Centraco incineration plant and the proposal of measures to safeguard the incinerable radioactive waste management solutions.

The range of subjects covered by this new edition is even more exhaustive. The 2013-2015 PNGMDR also takes account of the provisions of Council directive 2011/70/Euratom establishing a community framework for the responsible and safe management of spent fuel and radioactive waste adopted on 19th July 2011, Article 12 of which defines the content of the national waste and spent fuel management programmes. It thus provides a description of the financial implications, with information concerning the costs and financing mechanisms, the concepts and plans for the post-closure period, as well as some indicators for assessing the progress made in implementation of the Plan. The 2013-2015 PNGMDR also presents the work done by the Management Committee set up for management of the post-accident phase of a nuclear accident or emergency situation, the importance of which was underscored by the March 2011 accident that struck the Fukushima Daiichi nuclear power plant.

The structure of this third edition of the Plan was revised to ensure easier understanding by making it readable at several levels. The 2013-2015 PNGMDR was transmitted to Parliament in late 2012 and will be evaluated by the OPECST. In accordance with the provisions of Article L.542-1-2 of the Environment Code, a new decree stipulating the provisions of the 2013-2015 PNGMDR will be published in 2013 to give official shape to the requirements and the studies to be conducted. This Plan, and its summary, will also be available for consultation on-line on the ASN and DGEC websites.

Glossary

ACRO: Association for the Control of Radioactivity in the West
AFSSA: French Agency for Food Safety
ANCCLI: National Association of Local Information Committees and Commissions
Andra: French national radioactive waste management agency
ASN: French nuclear safety authority
ASND: Defence nuclear safety authority
BNI: Basic Nuclear Installation
Bq: Becquerel
BRGM: Geological and Mining Research Office
CE: Environment Code
CERCA: Compagnie pour l'Etude et la Réalisation de Combustibles Atomiques
Cigéo: Industrial centre for geological disposal
Cires: Industrial centre for collection, storage and disposal
CLI: Local Information Committees and Commissions
CLIS: Local Information and Monitoring Committee
CNAR: National Funding Commission for Radioactive Matters
CNDP: French National Public Debates Commission
CNE: National Review Board
CNEF: National Review Board for financing the cost of decommissioning of basic nuclear installations and of managing spent fuels and radioactive waste
CNRS: French National Centre for Scientific Research
CODERST: Departmental Council for the Environment and for Health and Technological Risks
CODIRPA: Steering committee for managing the post-accident phase of a nuclear accident or radiological emergency situation
COSRAC: Committee for the Monitoring of Research on the Cycle Back-End
CPDP: French Special Public Debates Commission
CSA: Aube waste disposal facility
CSM: Manche waste disposal facility
DGEC: General Directorate for Energy and Climate
DGPR: General Directorate for Risk Prevention
DGRI: General Directorate for Research and Innovation
DGS: General Directorate for Health
DLI: Incinerable liquid waste
DREAL: Regional Directorate for the Environment, Planning and Housing
DRIEE: Regional and Interdepartmental Directorate for the Environment and Energy
DSI: Incinerable solid waste
ENSREG: European Nuclear Safety Regulators Group
FI: Low level
FNE: France Nature Environnement
FNR: Fast Neutron Reactor
Gas-cooled reactor
GCR: GESI: French group of Electronic Fire Safety industries
GIP Sources HA: Public interest grouping for high-level sealed radioactive sources
GIP: Public interest grouping
GSIEN: Group of scientists for information on nuclear energy
HCTISN: High Committee for Transparency and Information on Nuclear Security

HL: High level
ICPE: installation classified on environmental protection grounds
ICRP: International Commission on Radiological Protection
ILW-LL: Intermediate Level Waste, Long-lived
INBS: Defence Basic Nuclear Installation.
INERIS: French National Institute for the Study of Industrial Environments and Risks
INSERM: French National Health and Medical Research Institute
IRSN: French Institute for Radiation Protection and Nuclear Safety
LAS: Source Activity Limit
LLW/ILW-SL Low Level and Intermediate Level Waste, Short-lived
LLW-LL: Low Level Waste, Long-lived
MEDDE: Ministry for Ecology, Sustainable Development and Energy
MESR: Ministry for Higher Education and Research
MI: Intermediate level
MIMAUSA: History and impact of uranium mines: summary and archives**MOX:** Plutonium and uranium oxides based fuel
MSNR: Nuclear Safety and Radiation Protection Mission
NEA: Nuclear Energy Agency
NEEDS: Nuclear, Energy, Environment, Waste, Society
OECD: Organisation for Economic Cooperation and Development
OPE: Long-term Environment Observatory
OPECST: Parliamentary Office for the Evaluation of Scientific and Technical Choices
PACEN: Back-end Cycle and Nuclear Energy Programme
PIGD: Industrial waste management programme
PNGMDR: National Plan for Radioactive Materials and Waste Management
PRI: Integrated Radiological Shielding
RCD: Retrieval and packaging of waste
RFS: Basic safety rule
RGIE: General Regulations for the Mining Industries
RSB: General solid residues
SAL: Specific Activity Limit
SCI: Intact cover disposal
SCR: Reworked cover disposal
SHS: Human and Social Sciences
SIDT: Interdepartmental Regional Development Plan
SS: Suspended Solids
Sv: Sievert
TENORM (waste): Technologically enhanced naturally occurring radioactive materials (TENORM)
tHM: ton equivalent heavy metal
Uapp: Depleted uranium
UOX: Uranium oxide based fuel
URE: Enriched Reprocessed Uranium
URT: Reprocessed Uranium
VLL: Very low level
WHO: World Health Organisation
WISE-Paris: World Information Service on Energy
ZIRA: Zone of interest for in-depth studies
ZPP: Population protection zone
ZPS: Heightened territorial surveillance zone

Appendices

Contents of appendices

Appendix 1: Studies on preservation of memory

The memory project launched by Andra in 2010 comprises on the one hand work designed to continue to create and improve records about the facilities and, on the other, scientific studies concerning two fields: materials ageing and human and social sciences (HSS).

With regard to creating and improving a memory of the centres, the following work has been started:

- the pertinence of the memory archive for the Manche disposal facility in the light of the needs of future generations and analysed every ten years by a group of international stakeholders, in order to periodically examine its adequacy and its comprehensiveness. An initial exercise of this type was carried out in 2012 and identified a number of improvements that could be made to the system;
- preparations for recording the memory of the Cigéo geological disposal project are under way: creation of the detailed memory of the Meuse / Haute-Marne underground laboratory and other elements concerning preparations for the creation of Cigéo (from among everything produced since the early 1980s select that which needs to be kept as to substantiate the decision to create Cigéo);
- the "technical" utility of the memory must be better explained, on the one hand to specify the benefits of this record for long-term safety and on the other to clarify the reversibility requirement;
- around these sites, Andra proposes setting up think tanks to interest the local populations in this problem, but also to collect their ideas about how they could locally assimilate it;
- collaborations with a number of French and international artists are organised, in various artistic fields, in order to obtain their vision of the problem of recording the memory of the repositories through their art;
- Andra takes part in international work on memory as part of the NEA/RWMC/RK&M working group (benchmark practices in various participating countries, joint definitions and bibliography, and drafting of recommendations);
- the creation of spaces dedicated to this memory is envisaged (in Andra's public visitor centres, study of the creation of a historical archive centre with delegation to the Archives de France).

Scientific studies into the ageing of materials consisted in testing the permanent ink/paper combination by means of standardised tests. Durability studies on other media for the longer term are currently being defined. They will concern non-paper media for writing and engraving, in particular studies of surface markers to be installed on the cover over the centres and the production of sapphire disks as demonstrators for a memory medium, the longevity of which could be up to a million years.

With regard to HSS, a group of laboratories was created to study perception of long time scales. For the other subjects linked to HSS (archives, linguistics, museography, archaeology of techniques and landscapes, etc.), a three-stage approach is planned: a succinct bibliography designed to show whether works already exist and are sufficient, otherwise a detailed bibliography produced with universities in order to determine any research that is to be incorporated into the scientific programme. The work will in particular concern continuity, temporality and vestiges, as well as the social dimension of the problem.

Continuity will in particular be studied through:
- languages and symbols, in order to determine for what reasonable time current or dead languages can be known and what the communication solutions could be once these languages cease to be known;
- institutional conservation of written works, sounds, images, objects, etc. by specialised French and international organisations, to analyse the preventive measures taken to limit deterioration over time and encourage assimilation and transmission by future generations;
- long-term digital archival, more specifically by organising an intelligence watch in this field, which is beginning to become organised and which, within the next few decades, could open up new prospects for the long term.

Temporality and vestiges will more specifically be studied through:
- the archaeology of techniques and landscapes, incorporating man-made changes and geodynamic changes, as well as the possibilities of memory resulting from the permanence of infrastructures created by Man;
- the memory of “legacy” repositories not managed by Andra, which exist in various places in France (uranium mines, nuclear tests, etc.).

The social dimension will in particular be studied through:
- the perception by the public of long time scales (several thousand years and more), within the framework of the grouping of human and social sciences laboratories;
- the three possible directions of social change in science, technology, humanity, etc. (regression, stagnation, progression);
- the integration of preserving the memory of repositories into teaching programmes on nuclear energy, heritage and memory;
- transmission of memory between generations via internet social networks to provide global information about the memory and records of the repositories.

The memory project is marked by the milestones of the Cigéo disposal facility project, first of all the public debate in 2013, for which Andra will be required to provide the elements needed for a debate involving the stakeholders in the broad sense, and then the creation authorisation application in 2015. It will continue in parallel with the development of the repository and its gradual closure.

Appendix 2: Summary of achievements and research in foreign countries

1 SUMMARY OF ACHIEVEMENTS ABROAD

This summary presents achievements abroad concerning the management of radioactive materials and waste (countries concerned: Belgium, Canada, China, Finland, Germany, Japan, Netherlands, Spain, Sweden, Switzerland, United Kingdom and the United States). The notion of "achievement" is considered relatively broadly, including not only the drafting of the legal framework and the definition of a classification of radioactive waste, but also the development of management programmes.

1.1 Drafting of a legal framework

Radioactive waste management plans (similar to the PNGMDR to varying extents) sometimes exist abroad, but with objectives that vary significantly from one country to another. Furthermore, some of these plans are not made public.

In Belgium, the Ondraf published a national high-level waste plan in 2011, which establishes various avenues for long-term management, analyses their environmental impact and submits them to the public for their opinion.

The United States announced its intention to adopt a new approach, following the 2009 decision to abandon the envisaged disposal of high-level waste and spent fuels at Yucca Mountain. The Blue Ribbon Commission set up to review spent fuel and HLW waste management strategy is proposing plans that will constitute the foundations of future Government actions.

In 2006, the United Kingdom published a White Paper entitled "Managing Radioactive Waste Safely - proposals for developing a policy for managing solid radioactive waste in the UK", which announces a waste management plan and organisation.

For several years now, Spain has been periodically publishing a General Radioactive Waste Plan, which is scheduled to be revised on a regular basis. It mainly stipulates overall institutional directives.

With regard to inventory, practices vary widely, particularly concerning scope (the French specificity concerning VLLW, the inclusion of mining waste in the United States), exhaustiveness and the level of detail (less detailed in Germany than in France), distribution to the public (the inventory is not public in Spain; in Japan the producers are free to make their own inventories public or not), the rate of updating and the coverage of waste referred to as "engaged" given the current rate of production (until 2080 for Germany, but in the United States for instance, engaged waste is not included).

Despite the work done by IAEA (which provides a database common to all countries, but with a relatively global approach involving very broad categories), comparisons remain difficult, in particular because the units of reference (volume, weight, etc.) used to measure the quantities of radioactive waste, differ from one country to another.

As in France with Andra, a public organisation is responsible for implementing the management of radioactive materials and waste in Belgium (ONDRAF-NIRAS) and in Spain (ENRESA). There is a public organisation in the Netherlands, COVRA, but it is not really comparable, be it in terms of scope of waste covered, or in terms of activities. However, the waste producers (especially the private ones) are most often directly responsible for practical implementation of waste management. They then create a cooperative to manage certain waste, jointly with the public producers: Canada (NWMO-SGDN), Finland (Posiva Oy, for spent fuel only), Sweden (SKB), Switzerland (CEDRA-NAGRA, which does not manage storage). Sometimes, there is no centralised organisation, notably in Japan, where each type of waste corresponds roughly to its own management route and organisation. It should be noted that these organisations are far from always being the "owners" of the waste they have to manage: in Canada, the producer remains responsible, even after closure of the disposal facility; in the United States, the State is responsible for civil waste as of the transport phase (followed by the disposal after burial and disposal after site closure phases).

The list of organisations in charge of radioactive waste is given in the following table:

With regard to the financing of radioactive waste management, the polluter-pays principle would seem to be universally applied for management of radioactive waste facilities, but not for waste management research.

It should be noted that all the countries mentioned here (except for China, which is currently at the joining stage) are members of the IAEA Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management, which entered into force on 18th June 2001. These countries meet in Vienna every three years under the auspices of IAEA, to present their national reports describing the implementation of their obligations and any developments. The last meeting of this type was held in May 2012.

It should be underlined that European directive 2011/70/Euratom of 19th July 2011 concerning the responsible and safe management of spent fuel and radioactive waste requires the establishment of national programmes specifying how the member States implement their national spent fuel and radioactive waste management policies. The content of the national programmes, set in article 12 of the directive, more specifically comprises an inventory of spent fuels and radioactive waste, the general objectives to be reached, the financing mechanisms and an estimate of the cost of the programme. This directive will thus harmonise the European framework for the management of spent fuel and more specifically the establishment every three years of the national programmes, the first version of which must be transmitted to the European Commission in mid-2015.

1.2 Waste classification

1.2.1 The different types of classifications

There are two main approaches to defining the classification of radioactive waste: one approach based on the waste management route and one on the waste production route (this latter approach being in part inherited from the historical construction of radiation protection, generally built around the individual production routes).

Within the first approach (by management route), the classification abroad, in the same way as in France, often combines the parameters of the activity and lifetime of the radionuclides making up the waste (for example Belgium, Spain).

However, the waste classification is sometimes based solely on the activity level. For example, in Canada, there are only two main categories (LLW/ILW and HLW + spent fuels), except for the specific management of mining waste. In the Netherlands, the classification comprises a larger number of categories, but there is no distinction between short-lived and long-lived waste; consequently, there is no surface disposal project.

Other classifications sometimes exist (leading to categories that are qualitatively comparable but which have quantitatively different thresholds): Germany for example has based its classification primarily on the exothermic nature of the waste.

In countries which adopted the second approach (by production route), the classification is more complex, with routes specific to certain types of waste and combining activity and lifetime: United States, Japan and Sweden (in this last country the two types of approach in fact co-exist). Finally, a category is sometimes added for waste from hospitals, universities, etc., for example in Finland.

In addition, certain categories correspond to specifically national characteristics: Belgium (processing of 50% of the radium sources used worldwide) and Canada (large-scale uranium mining).

Finally, the absence of clearance thresholds in France (for waste which contains or which is liable to contain only very small quantities of radioactive elements) is specific to this country. Such thresholds exist in the other countries studied, but vary considerably both in terms of the threshold itself and the scope of waste considered; the VLLW category therefore rarely exists in its own right and does not correspond to the same waste as in France.

1.2.2 Waste classification adopted by IAEA

In late 2009, IAEA published a wide-ranging revision of the radioactive wastes classification which dated from 1994 (IAEA, 2009). It is used by the member countries for an international presentation of their radioactive waste management and their inventories, such as for example in IAEA's NEWMDB database; the European Union refers to it in the 19th July 2011 directive on the responsible and safe management of waste and spent fuels.

This revision was felt to be necessary because the classification system previously defined by IAEA was not completely exhaustive: it did not cover all types of radioactive waste, nor did it provide a direct link with the disposal and management options for all types of radioactive wastes. These aspects of the former classification proved to be obstacles to its utilisation and its application.

The new 2009 classification system introduces a new category of waste - VLLW (Very Low Level Waste) and uses the LLW (Low Level Waste), ILW (Intermediate Level Waste) and HLW (High

Level Waste) classes. These classes take account of both the level of radioactivity and the half-life of the radionuclides contained in the waste.

In it, the wastes are classified according to the degree of confinement and isolation required to guarantee long-term safety, based on their nature and the risk they represent. This waste classification allows an incremental approach to obtaining the required level of safety, because it is based as much on practices as on the characteristics of the sources, the levels of exposure to which they lead and their occurrence.

2 MANAGEMENT ROUTES THAT EXIST OR ARE UNDER CONSTRUCTION

2.1 Choice of the type of fuel cycle

The decision to reprocess spent fuels was taken in various countries in the 1950s for military purposes and at the end of the 1970s for civil uses. A certain number of countries today have facilities:

- for the complete processing of fuels, as in France, the United Kingdom and Japan (for which industrial start-up has however not yet been declared and which could be postponed indefinitely);
- for reprocessing of fission products in the United States as part of the clean-out of former sites such as Hanford, or for separation in Russia to retrieve reusable materials from spent fuels;
- for research as in China, which also opted for a closed fuel cycle, but which is developing test facility projects, notably with the help of France, and in India which is building a pilot plant for the vitrification of fission products.

Several other countries which do not have dedicated facilities on their own territory had or indeed still have all or part of their spent fuels reprocessed in plants abroad, mainly in the United Kingdom and France, this particularly being the case with Germany, the Netherlands, Switzerland and, to a far lesser extent, Spain. Some of the Eastern European nations also do the same in Russia. Some of these countries have however decided to put an end to reprocessing abroad at some time in the near or not so near future: Germany and Switzerland in particular have made a legislative commitment to this and Belgium has for the time being suspended its waste reprocessing contract with the La Hague plant.

For the time being, South Korea has not made a final decision concerning the reprocessing of its light water reactor fuels, which are currently stored on the production sites.

The other option currently being used is to directly manage the spent fuels without any separation or reprocessing phase. This is used in Canada, Finland and Sweden. This has also been the case in Spain and the United States since the implementation of non-proliferation arrangements in the 1980s (under President Carter).

2.2 Decommissioning activities

The countries which have operated nuclear power generating, research or fuel cycle facilities for half a century have initiated major decommissioning programmes for the older facilities, along with post-operational clean-out of the sites. In recent decades, the United States has been carrying out a programme concerning 108 sites covering a total surface area of 800,000 ha. In 2005, the United Kingdom created the Nuclear Decommissioning Authority for eventual decommissioning of all existing nuclear facilities. Decommissioning generates a considerable volume of waste, mainly VLLW waste but also LLW/ILW waste. Its management requires rigorous technical planning and available financing.

2.3 Management of LLW/ILW waste and LLW-LL waste

In several countries, surface or sub-surface disposal centres for LLW/ILW waste are already in operation. They were created to accompany the production of energy of nuclear origin: in China, the Beilong and Diwopu disposal centres, in Spain that at El Cabril, in the United States the centres in Barnwell, Richland, Clive and Andrews, in Finland those of Olkiluoto and Loviisa excavated into the granite at a depth of 60-100 m, in Japan that of Rokkasho-Mura, in the United Kingdom that of Drigg opened in 1959 (about 1,000,000 m^3 are emplaced in it in trenches and on platforms), and in Sweden the SFR centre in Forsmark located 50 metres under the Baltic Sea.

Others are under construction or design in specific locations and the projects differ widely, in terms of the type of site chosen, the design of the disposal centre and its depth. These factors ultimately determine the type of waste that can be disposed of (particularly with regard to the lifetime). Thus, in Belgium, the Dessel disposal centre, which was to enter service in 2016, will in fact only accept short-lived LLW/ILW waste. Among the other projects currently under construction or planned, we should mention the geological disposal facilities planned for about 2017 in Kincardine, Canada at a depth of 700m near the Bruce reactor (Ontario) and in Germany, as of 2014, for LLW/ILW in the old Konrad iron mine in Salzgitter, at a depth of about 1000 m.

However, several countries have not completed or defined their LLW/ILW waste disposal project, such as Switzerland and the Netherlands, but also Italy, which has undertaken the disposal of the waste from its shutdown facilities. Others are closing old sites and reconsidering new locations, such as in Germany for the Asse and Morsleben sites or in the United Kingdom for the Dounray site, on which the construction of a storage facility is being envisaged.

More specifically with respect to LLW-LL waste, current management abroad consists primarily in storing it on the production sites. Long-term management routes have yet to be defined. The volumes concerned are particularly high in Belgium: radium waste comes from the reprocessing of 50% of the radium sources used worldwide and is currently stored on the Olen site. In Spain, graphite waste from the gas-cooled reactors is currently being stored on a reactor decommissioning site. Neither is there any formal plan in Switzerland, the United Kingdom, Japan, the United States, Russia and Ukraine, which all possess graphite type waste.

2.4 Management of HLW waste

Most countries are moving towards deep geological disposal, but they are all at widely differing stages in the site selection and centre construction process.

Finland and Sweden have already selected their (first) sites, in Olkiluoto and Osthammar (location of the Forsmark NPP) respectively and are initiating the building permit application phase. The commissioning of these centres is scheduled for between 2020 and 2025. In Finland, excavation of the Onkalo underground laboratory to characterise a granite environment for construction of the repository reached the reference depth of 420 m in the summer of 2010. Several in-situ tests are underway in Onkalo in order to examine various local characteristics of the massif. They include studies of the hydrological properties, retention, the mechanical behaviour of the rock and geochemical transformations. Testing of a confinement and installation materials manufacturing pilot in Onkalo has started. In Sweden, the site was selected in June 2009, following several years of detailed studies and investigations and a large-scale experimentation programme in the Aspö laboratory (near the Oskarshamn site which was not selected). The building permit application for a geological disposal facility for spent fuels was submitted in March 2011. If authorised, the fuels disposal facility will be built at a depth of about 500 m, in granite rock. Its construction should begin in 2015 and continue until the early 2020s.

In the United States, after choosing the Yucca Mountain site in 2002, the US-DOE (OCRWM) submitted a disposal facility building permit application in June 2008. The file was considered to be correct and its examination was accepted by the safety regulatory body (NRC). However, the Obama administration decided that "Yucca Mountain was not a feasible option for the long-term disposal of spent fuels". Since 2009, there has been no funding for the disposal facility preparation phase and examination of the creation authorisation application has been suspended.

The "Blue Ribbon" commission, created in January 2010 to examine all the possible strategic options concerning the management of spent fuels and high-level radioactive waste, submitted its final report in January 2012. It reaffirms that the reference solution must be geological disposal and in particular recommends setting up an organisation in charge of the disposal of radioactive waste and a siting process based on acceptance. The report takes account of the opinions and comments of the public, obtained during meetings in October 2011.

Although the site has not yet been chosen, various milestones for the near or more distant future have been set in certain countries. Japan has launched a vitrified waste disposal site selection process, which should enter service in about 2035; however the process has been blocked in phase 1 since July 2007 owing to the absence of candidates; a new information campaign was initiated in 2009 in preparation for the launch of a new call for candidates system. Germany and China have set targets for initial operation of the geological disposal centre at 2030 and 2040 respectively. After the 2010 expiry of the moratorium concerning the Gorleben disposal facility, underground exploration work on the salt dome has resumed. At the same time, a preliminary safety analysis is being carried out on this formation. The results should be published in 2012. The conformity of Gorleben with the most recent international safety standards will be examined by an international group of experts in 2013.

Other countries have chosen to focus on research into geological disposal and in particular to postpone the site selection process. For example, no date has been set in Belgium or Canada (in both countries a gradual process involving the stakeholders has been set up). In the United Kingdom, the Government asked the NDA (Nuclear Decommissioning Authority) to study the possibility of speeding up the geological disposal facility construction programme, to allow operation of this facility as of 2029 (instead of 2040 as initially planned). Similarly, the Netherlands have built a long-term (about a century) storage facility, during which time geological disposal is to be studied. In Spain, the construction of a disposal facility for spent fuel, high-level waste and intermediate level waste which cannot be disposed of in El Cabril, is scheduled for start-up in about 2050. The site of Villar de Cañas (situated in Cuenca province about 130 km south-east of Madrid) was however chosen in 2011 to house the future spent fuel and high-level waste storage centre. The radioactive waste management policy should be clarified in the forthcoming seventh general radioactive waste plan.

3 RESEARCH TO SUPPORT GEOLOGICAL DISPOSAL

In most countries, the reference solution for managing high-level waste and intermediate level, long-lived waste, is deep geological disposal. In the recent 2011 directive, the European Council reaffirmed this: "deep geological disposal is currently the safest and most sustainable solution as the final step in the management of high level waste and spent fuel considered to be waste". The host rock chosen varies according to its confinement properties and the geological possibilities of the countries concerned.

However, no country has yet issued a formal authorisation for disposal of this waste, including spent fuels, except for the United States with regard to waste of military origin. Most countries are experiencing significant delays in the development of their disposal programmes, owing to attempts to identify sites on primarily scientific and technical bases, without sufficient local consultation. Those who learned the lessons of this failure and who have restarted the process from the beginning, with prior debate and consultation, are now the furthest advanced.

In certain cases, scientific and technical feasibility can today be considered as proven and those few countries which have reached the most advanced stage are now in the final site qualification and concepts and engineering optimisation phase.

3.1 The organisation of research

With regard to research programmes concerning radioactive waste for which there is currently no industrial route, the most common option is to entrust oversight to the organisation responsible for management, whether private or public: SKB in Sweden, POSIVA in Finland, ENRESA in Spain or ONDRAF in Belgium.

This configuration nonetheless implies specific technical support, similar to that which Andra received from research organisations such as CEA: POSIVA with VTT, NAGRA with PSI, ENRESA with CIEMAT, ONDRAF with CEN-SCK.

Nonetheless, for historical reasons, R&D may sometimes be run by another organisation, which involves the future waste management operator and other research organisations.

A typical case is that of Germany, where there is considerable involvement by GRS (a research organisation which reports to the waste manager BfS) and the BGR, a German public research institute specialising in earth sciences and natural resources, specifically for geological disposal of exothermic waste. DBE (German company for the construction and operation of waste disposal facilities) has an exclusive contract with BfS for the construction, operation and monitoring of disposal sites.

Another special case is that of Japan, although the situation has become simpler since the merging of the two public research organisations, JNC and JAERI, into JAEA. In addition to JAEA is CRIEPI, financed by the electrical utilities and RWMC, financed by METI.

3.2 The underground laboratory – a possible precursor to the disposal project

For R&D into geological disposal (SF or HLW and ILW-LL waste), the various configurations of the waste management organisation in the various countries considered leads to considerable differences in the status of the underground research laboratory, whether in terms of ownership or objective (methodology[1] or qualification of the site and the host rock).

In Sweden, the Hard Rock Laboratory at Äspö is the property of SKB (methodology and qualification of the granite). Since 1995, it has been carrying out research on its KBS3 concept (vertical vaults and copper container) under three-yearly R&D programmes approved by the Government. Since 2000, proof of concept demonstrators have been operational. Their aim is to acquire expertise in the construction and operation of a deep geological repository, for which authorisation was requested in 2011 concerning the Forsmark site in Östhammar.

In Finland, POSIVA is excavating a qualification laboratory in the granite at Onkalo on the actual site of the future disposal facility. Excavation reached its nominal depth of 455 m in February 2012. The research covers subjects such as geological surveys, instrumented drilling, characterisation niches and mechanical studies of the crystalline massif.

In Belgium, the objective of the Hades research laboratory, situated at a depth of 230 m, is methodological and it is used for qualification of the Boom clay. It is now managed by an IEG of ONDRAF and CEN/SCK, the Belgian counterpart of CEA. This laboratory is demonstrating the possibility of building a geological repository consisting of a network of drifts, with limited disturbances within the host clay formation.

Switzerland with two laboratories of widely differing status:

- GTS (GRIMSEL Test Site), a granite environment methodology laboratory made available to NAGRA using drifts belonging to the electrical utilities; current research concerns the instrumentation and monitoring of the structures. However the Spanish proof of concept demonstrator for spent fuel disposal in drifts, called FEBEX and set up by ENRESA in 1997, is still active.
- Mont Terri, an international consortium initiated in 1996 by NAGRA and Andra and now run by a Swiss federal authority. The laboratory's objectives are methodological. It enables Nagra to qualify the clay in Opalinus (potential host rock).

At Mont Terri, about a hundred experiments have been carried out on different scales since the beginning of the research programme in 1996 and more than about forty were still on-going in 2012. In a clayey material that had hitherto been little studied, rock characterisation methods were established. Thus, with regard to safety, the diffusion of radionuclides in clay was measured and the water contained in the rock was collected. Andra's close involvement in the Mont Terri projects and experiments enabled it to prepare for the experiments in the Meuse/Haute-Marne underground laboratory.

In Japan, JAEA is building two underground laboratories, with purely methodological objectives. In the Mizunami laboratory (crystalline rock), a depth of 460 m, out of the planned 1,000 m, was reached in 2011. Studies concerning hydrology and rock mechanics are on-going. In the Horonobe laboratory (sedimentary rock), hydrological tests and hydrochemical measurements are continuing. A depth of 250 m, out of the planned 500 m, was reached in 2011.

[1] In other words a laboratory designed to develop "in situ" characterisation techniques, but for which its status and geological environment exclude it from a possible geological siting sector.

In Germany, following the experiments which took place in the 1990s in the former salt mine at Asse (for which the initial work dates back to the early 1970s), the Gorleben salt dome intended for disposal of high level radioactive waste is currently the subject of reconnaissance and survey work. This phase should last for seven years, according to the German federal office of radiological protection (BfS) which is responsible for the site.

In the United States, after about twenty years of research and characterisation work performed on the Yucca Mountain site in the State of Nevada, the DOE in June 2008 submitted an authorisation application for disposal of spent fuels in volcanic rock at Yucca Mountain. As mentioned earlier, this approach was queried by the State, which undertook to redefine the disposal strategy for radioactive waste and spent fuels.

3.3 Coordinated research in Europe

Technological research and development work developed under the EU's R&D Framework Programme (FP) focuses on:

- the management and safety of geological disposal of High and Intermediate Level, Long lived waste (HLW/ILW-LL);
- the European dimension of its management and disposal;
- the development of processes enabling their quantity and harmfulness to be reduced (e.g.: separation, transmutation, etc.).

3.3.1 Projects in progress

The following table presents the various projects and research programmes in progress within the framework of Euratom, the scope of which concerns developments in radioactive waste management.

Project	Subject	Period	Leader
ReCoSy	Study of Redox phenomena and their influence on the retention and transport of radionuclides	*7th FP* 2008-2012	FZK (D)
CARBO-WASTE	Reprocessing and disposal of irradiated graphite and other carbon waste	*7th FP* 2008-2012	FZH (D)
PETRUS II	Coordination of teaching and training at the European level	*7th FP* 2009-2011	INPL (F)
MODERN	Development and implementation of disposal facility control and monitoring techniques	*7th FP* 2009-2013	Andra (F)
FORGE	Assessment of the impact of gases in radioactive waste disposal facilities	*7th FP* 2009-2013	NERC (UK)
IGDTP	Technological Platform (following on from CARD) aiming to coordinate resources and actions in the field of geological disposal facilities	*7th FP* 2009-2012	SKB (S)
CATCLAY	Study of cation migration processes in clayey rocks	*7th FP* 2010-2014	CEA (F)
PEBS	Long-term performance of engineered barrier systems	*7th FP* 2010-2014	BGR (D)
SKIN	Very slow kinetic process in fluid-rock interactions	*7th FP* 2011 - 2015	EM-Nantes (F)
LUCOEX	Experiments in 3 underground laboratories to test and confirm the conceptual choices made.	*7th FP* 2011 - 2015	SKB (S)
FIRST Nuclides	Acquisition of data on the IRF (labile fraction of the source term) for high burnup fraction UOX fuels, in particular to reduce the uncertainties associated with certain radionuclides of interest (I^{129}, Se^{79}, Cs^{135}, C^{14})	*7th FP* *2012-2015*	KIT (D)
IGDTP-2	Technological Platform aiming to coordinate resources and actions in the field of geological disposal facility projects	*7th FP* 2012-2015	Andra (F)
DOPAS	Full scale demonstration of the construction and performance of seal structures	*7th FP* 2012-2015	Posiva (Fin)
OFSeSa	Novel and Reliable Optical Fibre Sensor Systems for Future Security and Safety Applications	*COST Action* [2] *2010-2014*	36 European Member States

Projects and research in progress concerning radioactive waste management within the Euratom framework

[2] http://www.enseignementsup-recherche.gouv.fr/cid55959/le-programme-europeen-cost.html

3.3.2 IGD-TP - Implementing Geological Disposal of Radioactive Waste Technology Platform

The IGD-TP technology platform was set up in 2009 with the aim of better targeting research, development and demonstration (RD&D) programmes and ensuring improved research between Member States. The IGD-TP platform is therefore run by the organisations responsible for geological disposal projects in the European countries, but also involves research organisations, design offices, technical support organisations for the safety regulators and all players interested or involved in research programmes. Its role is to boost confidence in the safety and implementation of radioactive waste disposal solutions in deep geological formations. IGD-TP is not only of use for the construction of the first facilities, but also for waste management programmes with tight schedules. Most of the nuclearised countries have developed radioactive waste management programmes, but their progress and their implementation schedules are different.

According to the IGD-TP, Europe will in 2025 have its first geological disposal facilities for spent fuels and highly radioactive long-lived waste, offering safe, long-term management.

The engagement by the platform members consists in:
- reinforcing confidence in the safety of geological disposal solutions among Europe's citizens and decision-makers,
- encouraging the drafting of waste management programmes which include geological disposal as the accepted reference option to ensure the long-term management of high level and intermediate level, long-lived waste,
- facilitating access to expertise and technology to maintain skills in the field of geological disposal, for the benefit of all Member States.

In 2011, IGD-TP published a programme called the "Strategic Research Agenda", which defines the research (RD&D) priorities, with a view to obtaining authorisation for construction of the disposal facilities. Implementation of this programme makes provision for funding included in the 7th Framework Programme for technological R&D – 7th FP.

The main subjects of the IGD-TP concern:
- the safety study;
- packaging and behaviour of waste;
- technical feasibility and long-term performance of disposal facility components;
- disposal facility development strategy;
- the safety of disposal facility construction and operation;
- measurements and monitoring;
- governance and involvement by the stakeholders.

Several cross-cutting subjects are dealt with concerning dialogue with the safety regulator, individual skills, knowledge management and subjects relating to communication and information. In addition, dozens of questions will be studied, ranging from disposal facility oversight to performance assessment methods. All of the subjects selected will be combined in programmes, the objectives of which will be attached to "'Horizon 2020" established jointly with the 8th European Framework Programme.

3.4 OECD-NEA

The role of the Nuclear Energy Agency (NEA) is not to run research programmes but rather to bring together players from the various countries to deal with subjects that need to be shared between countries.

The Radioactive Waste Management Committee (RWMC) assists the member countries with management of radioactive substances and waste, more specifically with regard to developing strategies to guarantee the safe, sustainable and generally acceptable management of all types of radioactive waste, in particular long-lived waste and spent fuel, along with the decommissioning of end-of-life nuclear facilities.

The RWMC's main tasks are:

- to create a forum for the exchange of information and experience on waste management policies and practices in the NEA member countries;
- to develop a common understanding of the fundamental questions involved and to promote the adoption of common philosophies based on the various possible waste management strategies and their alternatives;
- to monitor changes in the technical and scientific state of the art in the management of radioactive materials and waste;
- to contribute to the dissemination of information in this field through the organisation of meetings of specialists and the publication of technical reports and joint opinions summarising the results of joint activities on behalf of the international scientific community, the competent national authorities and other audiences interested by the field;
- to provide a framework for on-demand performance of an international peer review of the activities of a country in the field of radioactive waste management, such as R&D programmes, safety assessments, specific regulations, etc.

Appendix 3: Analysis of the compatibility between storage capacity and anticipated radioactive waste volumes

The storage capacity available for packaged waste is located on the production sites (mainly La Hague, Marcoule and Cadarache for HLW and ILW-LL waste). Each storage facility generally accepts one or more waste families. Certain capacity can be shared between the HLW and ILW-LL, ILW-LL and LLW-LL or even ILW-LL and LLW/ILW-SL routes.

Storage of packages of vitrified HLW/ILW-LL waste in La Hague

The standard packages of vitrified waste (CSD-V and CSD-B packages) produced in the R7 and T7 units of the UP2-800 and UP3 spent fuel reprocessing plants at La Hague, are placed in the warehouses adjoining these facilities and then in the Glass storage Extension – South-East (E-EV-SE), once their thermal power drops below 2,000 watts. The R7 and T7 units entered service in 1989 and 1992 respectively, for an anticipated operating lifetime of 50 years. The E-EV-SE storage facility has been operational since 1996, for an anticipated operating lifetime of 70 years.

The three storage facilities: R7, T7 and E-EV-SE, have a combined capacity of 12,420 packages, which will become saturated in about 2013. In 2006, AREVA undertook the study and construction of an extension to the E-EV-SE (called E-EV-LH) for which service entry is scheduled in 2013 and which will be able to store about 8,420 additional packages.

In late 2010, 10,943 CSD-V and CSD-B packages were stored in the three facilities, including 640 CSD-V packages stored in the E-EV-SE, pending shipment to the facilities of AREVA's foreign customers.

In 2015, the combined production of French vitrified waste will reach a volume of 2,560 m^3. The annual production of vitrified waste packages will be about 800 packages (140 m^3) until 2027 and will rise to 1,180 packages (210 m^3) by about 2030, with the start of dilution reprocessing in UOX and URE fuels of the 2,900 tHM[3] of MOX fuels which will have been accumulated by that date.

Other similar capacity will be required as of 2017 and a new extension of the E-EV-SE is in particular being envisaged (see chapter 3.3.3.2). Production will last until vitrification of the rinsing effluents which will be generated after final closure of the UP2-800 and UP3 units, planned for 2040.

Storage of compacted structural waste at La Hague

Since 2002, fuel assembly structural waste (hulls and end-pieces) from the R1 and T1 shearing units in the UP2-800 and UP3 plants have been compacted with metal technological waste in the Hulls and End-pieces Compaction Facility (ACC) which produces CSD-C standard compacted waste packages (ILW-LL route). Plans are for the production of CSD-C packages to continue beyond 2040 in support of decommissioning of the UP2-800 and UP3 plants.

On the La Hague site, the CSD-C packages are placed in the compacted hulls storage facility (ECC), which has a capacity of 20,800 packages and which entered service in 2002 at the same

[3] Ton heavy metal (tHM): this is the quantity, expressed in metric tons, of the uranium and plutonium contained in the fuel before burnup.

time as the ACC, with an anticipated operating lifetime of 50 years. The packages produced in accordance with contracts with third-party countries are also stored in the ECC.

In 2015, the cumulative production of packages of the French share of compacted structural and technological waste will reach a volume of about 2,300 m^3 and in 2020 a volume of about 3,100 m^3. The ECC has a capacity of 20,800 packages, or about 3,800 m^3 which, given this time-frame, will be sufficient to take both the French and foreign shares of the CSD-C packages. Its design is modular, with land in reserve which would allow construction of up to six modules equivalent to the existing module, if necessary. An extension could be necessary during the period 2020-2025. The extension of the ECC facility will need to be studied in the light of the volumes of CSD-C produced by reprocessing of UOX, MOX and URE, as well as the date of disposal of the packages.

Storage of packages of alpha contaminated sludges and metal and organic technological waste in La Hague

The STE3 effluent treatment station has since 1989 been treating liquid effluents from the La Hague plants. Sludges are encapsulated in bitumen and placed in stainless steel drums of 238 litres. As at the end of 2010, 11,278 packages of this type had been produced (or about 2,500 m^3). They are stored in halls with a capacity of 20,000 packages (about 4,500 m^3) in building S of the STE3 station, which entered service in 1987 and is expected to operate until 2040.

Until 2020, the STE3 station will be used to package the first part of the rinsing effluents resulting from decommissioning of the UP2-400 plant. The STE3 station has also started the bituminisation of the sludges produced from 1966 to 1991 by the STE2 station in the UP2-400 plant, but production was banned in 2008 and AREVA is studying alternative packaging solutions.

AREVA is also examining the definition of a packaging method for alpha technological waste (mainly contaminated by plutonium) from the La Hague and MELOX plants. The production of these waste packages is scheduled to continue until decommissioning of the plants, after 2040.

The Removal from Storage and Bitumen Drums Storage Extension units (D/E EB), built in 1995, have the capacity (about 11,200 m^3) to take the above-mentioned packages, for which the estimated volume in 2030 is 9,500 m^3. Depending on the quantities of packages actually produced, a rearrangement of these storage facilities may prove necessary by about 2017 in order to be able to accept all the alpha technological waste.

The alpha sludges and technological waste packaged at La Hague will not therefore generate any requirement for additional storage capacity before 2030. However, the alpha waste packages should probably be stored for longer, to allow a decrease in their production of hydrogen through radiolysis.

Storage of packages of solid operating waste, powder waste and cemented hulls and end-pieces in La Hague

Since 1990, solid waste: gloves, suits, tools, standard operating and maintenance parts in the UP2-800, UP3 plants or from decommissioning of the UP2-400 plant, have been cement-encapsulated in the AD2 unit, originally in asbestos cement containers (CAC) and, since 1994, in fibre-reinforced concrete containers (CBF-C2). These packages are placed in the solid waste storage facilities: EDS/ADT2 and EDS/EDT - EDC (storage of technological waste and storage of hulls); they are linked to the ILW-LL and LLW/ILW-SL routes. The traffic of LLW/ILW-SL packages passing through these facilities occupies a variable volume of a few hundred cubic metres. The EDS/EDC facility is also used to store stainless steel drums containing cement-encapsulated hulls and end-pieces (ILW-LL) produced until 1995. In the future, it will also be used to store ECE stainless steel drums containing cement-encapsulated powder waste: resulting from the purification and filtration of pool water and fines from the dissolution or cladding removal of spent fuels from the gas-cooled reactors which are awaiting packaging in the facilities of the UP2-400 plant undergoing decommissioning.

The total capacity of the solid waste storage facilities is 14,330 m^3. They are scheduled to operate until 2040. This capacity appears adequate to take the anticipated production within this time frame, which will see the volume of ILW-LL packages rise from 9,012 m^3 in 2009, to about 11,100 m^3 in 2030.

Storage of packages of vitrified HLW/ILW-LL waste in Marcoule

The Marcoule vitrification unit (AVM) has a storage facility. Vitrified waste packages: fission products and minor actinides from past production (HLW route) and rinsing effluents from the circuits in the UP1 plant which has been finally shut down (ILW-LL route) are stored in it along with operating technological waste from the AVM (ILW-LL route) to which could be added a very small number of vitrified waste packages (about five packages, or about 1 m^3) produced in the Atalante laboratories.

The capacity[4] of the AVM warehouse (665 m^3) should be sufficient to take all planned production at Marcoule.

A shipment interface to the stored packages disposal centre will need to be developed by CEA. CEA will identify the technical options and conduct an initial analysis of the transport methods, together with Andra, more specifically in order to present storage-transport-disposal scenarios to the public debate on the Cigéo disposal centre project planned for 2013.

The pilot vitrification unit in Marcoule, PIVER, has produced a small quantity of vitrified waste packages (HLW route) with a total volume of 17 m^3 and which is currently stored in building 213, specially outfitted in the Marcoule pilot unit (APM) which entered service in 1969 and for which an operating life extension is currently being examined.

The periodic safety reviews, supplemented by the stress tests, showed the need to look at the lifetime of the storage facilities for this type of waste, on the basis of the filling time-lines for the Cigéo disposal facility project.

[4] The indicated capacity corresponds to the unit volume of 175 litres per container considered in the 2012 edition of the National Inventory.

Storage of packages of bituminised sludges and solid waste on the Marcoule site

Since 1966, the Marcoule liquid effluents treatment station, STEL, has produced packages of sludge encapsulated in bitumen and then placed in 230 litre steel drums. From 1966 to 1996, the drums made of non-alloy steel, were stored in 35 pits in the site's North zone (about 6,000 drums) and then in bunkers numbered from 1 to 13 in the South zone (about 54,000 drums, to which should be added the 2,200 drums produced since 1996 and stored in bunker 14).

Steps are under way to retrieve and repackage these legacy drums. From 2000 to 2006, all the drums in the North zone pits, most of which were classified LLW-LL, were removed, inspected and placed in 380 litre stainless steel over-packs. Retrieval is continuing with the legacy drums from bunkers 1 to 2 in the South zone. At the same time, in response to the requests from ASND, retrieval is in progress for the drums of salting out product (LLW/ILW-SL type encapsulation process drums, mixed with bitumen encapsulation drums in bunkers 1 to 10) which are considered to represent most of the mobilisable source term.

At present the Marcoule STEL is continuing to produce packages of bituminised sludges. Since 1996, packaging has been in 230 litre stainless steel drums. These packages, which correspond to the LLW/ILW-SL and ILW-LL routes, are stored in bunker 14 which entered service in 1994 with a capacity of about 1,200 m^3. Shutdown of the encapsulation unit is scheduled for 2014.

Retrieval and packaging of the following waste is being envisaged for the 2017-2020 time-frame:

- non-magnesium metal structural waste from fuels reprocessed in the UP1 plant and structural waste from the PHENIX fast neutron reactor;
- powder waste, filters, graphite powder from cladding removal from gas-cooled reactor fuels, settling sludges and solid metal and partially organic operating and maintenance waste from the units or from decommissioning, with beta-gamma spectrum;
- magnesium structural waste from gas-cooled reactor fuels.

The sludges produced by the treatment of liquid effluents in the STEL are encapsulated in a cement matrix, which will replace bituminisation in 2015 (STEMA project). The waste packages produced (380 litre drums) most of which will be of the LLW/ILW-SL type, will be packaged in the centre before being shipped to the LLW/ILW waste disposal facility. Any ILW-LL packages will be managed in the same way as the packages resulting from treatment of powder waste.

A multipurpose interim storage facility (EIP) entered service in 2000 for the storage of packages in 380 litre drums (called EIP drums). Its design is modular and it currently consists of two vaults[5]. The operating lifetime as currently planned is 50 years.

The packages at present stored in the EIP are drums of bituminised sludges produced by the STEL before 1996, retrieved from the North zone pits and bunkers 1 and 2 and repackaged in 380 litre drums. They represent a volume[6] of 2,660 m^3 (about 8,000 packages).

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5 Its extension up to 16 vaults could be envisaged, to raise the total capacity to 33,880 m^3.

6 The capacities and volumes are here expressed for a unit volume of 380 litres considered for an EIP drum in the National Inventory. Considering the overall external volume of this package (441 litres) would lead to greater capacities and volumes for the same number of packages.

The continued retrieval of the waste from the bunkers and its packaging in 380 litre drums will lead to saturation of the EIP's current capacity by 2017, with a volume of 4,370 m^3, or 11,500 packages (linked to the ILW-LL and LLW-LL routes). The need to create additional storage capacity, in line with the operations to retrieve these legacy packages, is being examined by CEA.

Storage of high-dose ILW-LL waste packages on the Marcoule site

The retrieval and packaging operations for the legacy and decommissioning waste will generate high-dose ILW-LL waste packages for which there is no storage facility. For the Marcoule site, the volumes of this category of waste produced by decommissioning of the PHENIX reactor (the most activated waste) and by the retrieval of structural waste from fuels reprocessed in the Marcoule pilot unit (APM) are evaluated at about 250 m^3. In order to meet this need, CEA has decided to create the DIADEM facility, which is scheduled to enter service in 2017, subject to its authorisation being granted (see §3.3.3.2) Furthermore, this new facility will be used to store high-dose waste from other CEA sites (Fontenay-aux-Roses, Saclay, Grenoble).

Storage of low-dose ILW-LL waste packages on the Cadarache site

Since 1970, the effluent treatment station (STE) of BNI 37 has been packaging the centre's filtration sludges and evaporation concentrates by cement encapsulation in 225 litre metal containers, themselves placed in 500 litre concrete containers (with or without immobilisation). Occasionally, the concentrates were packaged in 700 litre drums, which were repackaged in 1,100 litre non-alloy steel containers.

The low-dose ILW-LL solid waste from operation or decommissioning, which came mainly from the Saclay, Fontenay-aux-Roses, Cadarache, Valduc and Grenoble sites, was packaged in the solid waste treatment station in BNI 37 (compactable waste) or ICPE 312 (non-compactable waste). The primary waste is packaged by compaction and/or immobilisation in 870 litre metal containers. As of 2013, this type of packaging will be used for alpha spectrum solid decommissioning waste from Marcoule.

The inventory as at the end of 2011 is 5,014 packages of 870 L and 4,309 concrete containers of 500 L. The forecast inventory for 2030 is for about 6,360 packages of 870 L and 4,870 concrete containers of 500 L.

Until 2006, ILW-LL packages were placed for storage in BNI 56, which today is no longer accepting any new ILW-LL packages. In 2006, CEA commissioned the CEDRA radioactive waste storage facility at Cadarache, with two buildings (n° 374 and 375) each of which offers a capacity of 7,572 packages of low-dose ILW-LL waste packages. CEA has initiated transfer of low-dose ILW-LL packages from BNI 56 to CEDRA. As at 31st December 2011, 737 packages had been shipped to CEDRA

As at the end of 2011, 1,369 low dose packages of 870 litres and 47 low dose packages of 500 litres were present in CEDRA. The current storage capacity of 7,572 packages of low dose ILW-LL waste packages will not be enough to manage all the packages to be retrieved from BNI 56 and that will be produced in the CEA centres. The total number of low dose ILW-LL packages estimated between now and 2030 is 11,230. CEA thus envisages raising CEDRA's storage capacity for ILW-LL waste to 11,358 low dose packages, by building tranche 3. It should be noted that the filling ratio of the CEDRA facility depends to a large extent on the BNI 56 retrieval programmes.

Storage of medium dose ILW-LL waste packages on the Cadarache site

Solid operating or decommissioning waste from the various CEA sites and falling into the low dose ILW-LL category, has since 1970 been packaged in BNI 37 by compacting and then immobilisation in a steel container of 500 litres. Until 2006, the packages were placed in the storage pits in BNI 56. CEA has initiated transfer of some of these packages to CEDRA: at the end of 2011, CEA had removed from storage 183 recent packages produced under quality assurance from pit F6 in BNI I56 and transferred them to CEDRA. Retrieval of the recent stainless steel packages from pit F6 in late 2015 is one of CEA's priority safety objectives.

Since it entered service in 2006, the CEDRA facility has comprised building n°376, with a useful capacity of 1,400 packages, for storage in shafts of ILW-LL packages produced or removed from storage in BNI 56.

As at the end of 2011, the total number of medium dose ILW-LL waste packages stored on the Cadarache site stood at 1,173. It will reach 3,264 packages in 2030. The current capacity of CEDRA (1,400 packages) will not be enough to meet the need. CEA envisages increasing this capacity: after the construction of CEDRA tranche 3 by about 2023, it would be raised to 2,800 packages; an additional tranche 4 would be able to raise the total capacity to 4,200 packages.

Storage of other waste packages on the Cadarache site

Packages of radium-bearing lead sulphates (resulting from treatment between 1958 and 1970 of uranothorianite ore), solid waste and filtration sludges and large-sized containers (1,800 or 1,000 litres) and "source blocks" are currently stored in BNI 56. Production is now completed and represents a volume of about 1,275 m^3.

Storage of ILW-LL waste packages on the CEA Valduc site

Packages of sludges and concentrates immobilised in 220 litre metal drums, produced in the past from 1984 to 1995 by the Valduc liquid effluents treatment station, are stored on the Valduc site. This waste is part of the ILW-LL route and will be stored in CEDRA pending the commissioning of the Cigéo disposal facility project. Transfer of these waste packages is in progress.

Processing of recyclable materials produces effluents containing americium, plutonium and uranium which CEA plans to vitrify in about 2020. These ILW-LL packages will be stored on the Valduc site. In 2030 the total volume of packaged waste will reach about twenty m^3.

Storage of EDF activated waste packages on the site of the Bugey NPP

The ICEDA facility, located on the Bugey site, authorised by decree 2010-402 of 23rd April 2010, is designed to take activated waste produced by the decommissioning of the EDF reactors at Creys-Malville, Brennilis, Chooz A, Bugey 1, Saint-Laurent-des-eaux A1 and A2 and Chinon A1, A2 and A3 as well as the internals removed from the NPP reactors in operation (control rods and poison rods). For short transit period (a few months prior to shipment to disposal centres) this facility should also accept metal and graphite waste resulting from the decommissioning of Bugey 1 and intended for the LLW/ILW-SL and LLW-LL routes respectively.

Commissioning of this facility, envisaged for 2015, has been called into question by the 6th January 2012 cancellation of the building permit by the administrative court.

The hypothesis adopted by EDF is packaging in "C1PG" reinforced concrete containers for ILW-LL or LLW/ILW-SL activated waste.

The ICEDA facility would consist of two storage halls with a unit capacity of 1,000 waste packages, or about 2,000 m^3 of waste packages per hall.

Storage of radium-bearing radioactive waste

On its La Rochelle site, Rhodia stores different types of radioactive waste, resulting from the processing of monazite and then, as of 1994, from processing of rare earth concentrates. The site is authorised as an installation classified on environmental protection grounds. Rhodia possesses about 13,700 t of waste, in the following form:

- radium-bearing waste, known as RRA (about 1850 Bq/g alpha and beta activity in 2002): 160 t in la Rochelle, with most of the RRA being stored in Cadarache (5120 t);
- general solid waste, or RSB (about 75 Bq/g): 8400 t in la Rochelle.

This waste, classified as low level, long-lived, is part of the waste inventory intended for Andra's radium-bearing waste disposal project.

RRA waste is stored on the Cadarache site (ICPE 420 and 465) and RSB waste on the Rhodia site at la Rochelle, in a building (BAT. 135).

Cézus on the Jarrie site stores radium-bearing waste resulting from the processing of zirconium. It is located in a dedicated building consisting of six vaults of 1,000 m² each equipped with retention pans, for a storage capacity of 4,500 t. The storage capacity should be able to cover the needs until 2020 (2015 if the waste is insolubilised beforehand).

Storage of radioactive waste from the small producers

For the management of radioactive waste from the small producers, Andra uses the storage capacity on the AREVA/SOCATRI and CEA (BNI 56) facilities in Cadarache and Saclay (BNI 72), or even on other sites for diffuse nuclear waste, most of which falls into the LLW-LL category. In the light of the planned use of these facilities by the licensees for their own requirements, or the scheduled decommissioning of some of them, Andra has opted for its own storage facility.

In order 2012040-0002 of 9th February 2012, Andra was authorised to operate grouping and storage facilities for radioactive waste from the small producers in the VLL waste disposal centre. The storage facility, intended for waste for which the long-term management routes are currently under development, has a capacity of 6,000 m^3 with Andra having collected about 700 m^3 as at mid-2012. This facility entered service in 2012. It is more specifically accepts fire detectors, lightning arresters and other sources containing radioactive substances, waste from the clean-up of polluted soils (Operation Radium) and so on.

Storage of tritiated waste

At present, most tritiated waste for which there is no disposal solution is generated by Defence activities. Most of this waste is stored on the Valduc site. The volume of waste stored in France represents about 4,600 m^3. Some facilities will soon become saturated. New facilities are currently under construction.

Solid tritiated waste from the small producers will be stored in the ITER tritiated waste storage facility, scheduled to enter service in 2024, subject to authorisation being granted.

The tritiated waste storage capacity requirements are presented in part 3.1.

Appendix 4: Research aspects

1 INTRODUCTION, PLAYERS AND THE MAIN MILESTONES CONCERNING RESEARCH CONDUCTED FOR THE PNGMDR

The Act of 28th June 2006 gives the responsibility for research into separation-transmutation to CEA and research into reversible disposal of HLW/ILW-LL waste and storage to Andra. It organises the roles of the various research players in the field of radioactive waste management. At the same time, a certain number of R&D actions are performed by industry (EDF and AREVA), partly under agreements linking them to CEA or Andra. As necessary, all of these organisms draw on the pool of expertise available at the CNRS, which restructured its research in 2011 around a new cross-cutting research programme called, Nuclear: energy, environment, waste, society (NEEDS), the Universities and other organisations, such as the BRGM or INERIS. Finally, one must mention IRSN, where research aims primarily to provide it with a satisfactory level of nuclear safety and radiation protection expertise enabling it to fully play its role of technical support for ASN and ASND.

The PNGMDR, which has been in place since 2006, describes the management solutions developed for radioactive materials and waste and specifies a certain number of research strategy milestones over a period of three years. The National Review Board (CNE2) regularly assesses the research carried out in the field and in its recommendations proposes a number of orientations for the strategy to be implemented. It should be noted that the Board is in favour of a more international approach to a good part of the research performed by Andra, CEA and the CNRS. It particularly appreciated the importance given to this aspect during the hearings.

To ensure that all these programmes are consistent, a Committee for the Monitoring of Research on the Cycle Back-End (COSRAC) was set up and is chaired alternately by the DGRI and the DGEC. COSRAC, a unique forum for debate among all research players, helps with defining a common research strategy in line with the 28th June 2006 Act.

This document presents a summary of research carried out on the subjects covered in the previous PNGMDR and an outlook on research to be conducted in the coming three years. This document is not exhaustive in that certain long-term prospects can be carried out in parallel.

It is worth reviewing the main milestones of the 28th June 2006 Act:

No later than 31st December 2012: "Andra shall submit to the Ministers responsible for energy, research and the environment the dossier supporting the organisation of the public debate" which will be held before the authorisation application for the creation of a deep geological disposal site is submitted.
No later than 31st December 2012: CEA shall submit to the Ministers responsible for energy, research and the environment a dossier reviewing research carried out on the subject of separation- transmutation.
No later than 31st December 2014: "Andra shall submit the creation authorisation application" for a deep geological disposal site.

2 IMPROVING KNOWLEDGE AND WORKING UPSTREAM ON WASTE PACKAGING AND THE BEHAVIOUR OF THE PACKAGES

The producer is responsible for the production of the waste package and must demonstrate its characteristics by producing a data file and an operational model describing the long-term behaviour of this package. At the request of the producers, CEA on the one hand performs a significant share of the R&D necessary for implementation of the processes and for improving the understanding of the characteristics of the packaged waste.

The process to verify the ability of the packages to perform all the functions necessary for disposal is the responsibility of Andra, which has set up long-term behavioural study programmes for the various package families in the disposal environment. In order to carry out this R&D, Andra realised that it would need to set up dedicated structures such as the "Glass/Iron/Clay" Laboratories Grouping to study glass alteration and overpack corrosion and the "Evolution of cement structures" laboratory to study the long-term behaviour of concrete disposal packages. CEA and EDF take part in these two laboratory groupings.

All those involved agree on the need to continue characterisation work on all waste concerned by disposal, in order to:

- clarify the radiological inventories by developing or improving appropriate analytical and experimental resources;
- evaluating the chemical inventories and the source terms of the chemical and gaseous compounds;
- improving the characterisation of the behaviour of the waste in disposal conditions;
- evaluating release kinetics for waste for which the source terms are currently considered to be too conservative or penalising (labile).

2.1 Graphite waste: Management and processing scenarios

The low specific activity of graphite waste leads to the study of a shallow depth disposal facility (less than 200 metres) but its long-lived radionuclides content, as currently estimated, does not enable surface disposal to be envisaged. In particular, the declared chlorine 36 activity in the graphite waste requires the use of a sufficient thickness of clay to limit the flow into the geological environment.

One alternative could be offered by recent developments in graphite waste treatment processes. Some of these processes could make partial decontamination of the graphite possible in order to make its radiological inventory acceptable for shallow depth disposal (disposal under reworked cover, SCR). The radionuclides extracted would then require appropriate packaging and disposal. Provided that decontamination efficiency were sufficient, it would even be possible to envisage gasification of the decontaminated graphite in the form of carbon dioxide. A first series of heat treatment tests were performed by EDF jointly with the Studsvik company. The results obtained at this stage confirm the potential of heat treatment for removing the chlorine 36, tritium and some of the carbon 14 from the graphite.
At the same time, work is under way to clarify the radiological inventories of graphite waste, in particular thanks to improvements in analytical techniques and to additional sampling from the reactor blocks.

Outlook
One major avenue of the R&D programme will be devoted to heat treatment. This will involve on the one hand completing the pilot tests. On the other hand, a certain number of more exploratory tests will be performed, more specifically at CEA, in order to better understand the decontamination levels obtained and verify their validity on various samples.

The management of treatment effluents and packaging of the resulting waste are essential development points, whichever treatment process is envisaged. The carbon residues resulting from the treatment could thus be cemented or compacted to form a matrix compliant with Andra's disposal acceptance specifications. If not, separation of the radionuclides would be necessary, followed by dedicated treatment for each radionuclide (cement encapsulation, precipitation, etc.). These actions are formally laid out in special agreement between Andra, CEA and EDF.

2.2 ILW-LL Waste

2.3 Bituminised sludges

Over and above studying the release of radionuclides, the studies on bituminised sludges concerned the production of dihydrogen by radiolysis, the chemical source term of waste (complexing or aggressive species) and the swelling of waste packages under water.

The producers carried out R&D work leading to tools for calculating the production of radiolysis hydrogen. The challenges linked to the production of gas by radiolysis concern the design of a disposal package which must allow the gas to escape, and the risk of the explosive limits in the vault being exceeded if the ventilation were to stop for whatever reason. The production of dihydrogen by the bituminised sludge packages remains below 10 L/drum/year and recent estimates for the production of dihydrogen for the packages in the effluent treatment stations in La Hague (STE2 and STE3) give rates of less than 3 L/drum/year. The same conclusion applies for at least half the population of bitumens in the liquid effluent treatment station (STEL) in Marcoule (CEA). The work will continue on all the populations of bitumen encapsulated items produced in Marcoule.

Swelling under water is the result of an osmotic process, the origin of which is the behaviour of the bitumen which acts as a semi-permeable membrane. This process could only occur at resaturation of the vault, in other words within a time-frame of between ten and a hundred thousand years. For this time-frame, the main risk identified is damage to the argilite around the disposal vaults. In order to assess the rock damage process resulting from this phenomenon, preliminary modelling of the mechanical consequences of the pressure levels exerted at swelling of the bitumens was carried out by Andra. To fine-tune these models and validate the associated results, experiments must be carried out specifically to acquire pressure/strain curves.

Andra is currently carrying out studies to consolidate its understanding of complex near field phenomenological interactions of bitumen packages. An assessment will be provided in 2013/2014, in order to gain an approximate understanding of the reactivity of the nitrates and the consequences of the plume of salts on the large-scale migration of radionuclides. Pending the results, the long-term safety assessment is carried out on the basis of conservative hypotheses designed to cover the uncertainties associated with these phenomena. With regard to the complexing species, the analysis of the inventories of TBP (tributylphosphate) will make it possible to decide on the pertinence of setting up a special R&D programme lasting about two years.

2.3.1 Technological waste containing organic matter rich in alpha emitters

Technological waste rich in alpha emitters comes from fuel fabrication and processing facilities. The particularity of this waste is that it contains both metal and organic matter. Studies are required to determine to what extent, in a disposal situation, the radiolysis of the organic matter could lead to the production of gas such as dihydrogen and corrosive gases. The radiolysis and then hydrolysis of the organic materials will release complexing species which could complex actinides such as uranium and plutonium. The aim will be to complete the thermodynamic bases of the main expected complexes in order to verify their stability domain in geological conditions.

In the field of radiolysis, R&D work has been performed by CEA and Areva and led to the development of a database and predictive models allowing quantification of the gas source terms of these packages. With regard to determining water-soluble degradation products (PDH), the work already under way will continue in the form of a joint Andra/producers R&D programme. The aim of this programme is to obtain data to quantify the conservative nature of the assessment of these PDH and their complexing capacity.

All of this work does not however prejudge any significant rise in the mobility of actinide complexes within the Callovo-Oxfordian (COX) clays.

At present, CEA packages this type of waste using compaction and cement encapsulation. Areva has a cement encapsulation packaging mode for some of the alpha contaminated technological waste. All of the waste produced cannot however be packaged using this mode. Areva has also studied a treatment-packaging process using compaction. The S5 package has the advantage of reducing the packaging volume for this type of waste. This packaging was the subject of quadripartite discussions (ASN, Andra, IRSN, AREVA). In 2009, AREVA submitted the draft production specification for the S5 package to ASN and the detailed package data file to Andra. In February 2010, following examination of the package file, ASN asked Areva to study other treatment-packaging scenarios "*considering that the S5 package project developed by Areva, the characteristics of which are described in the above-mentioned letter of 20/01/2009, do not offer sufficient guarantees for long-duration storage and for disposal in deep geological formation, more specifically owing to the presence of organic materials"*. Areva thus initiated studies to find other possible treatment-packaging scenarios (including thermal) as requested by ASN in Article 1 of its resolution 2010-DC-0176 of 23rd February 2010 and is continuing R&D into the S5 package, in order to demonstrate that sufficient guarantees could be provided.

Areva transmitted files in April 2010 and early 2012, notably containing the additional R&D results acquired, which describe the corrosion-resistance of the container and the trapping of radiolysis hydrochloric acid gas by its internal carbon steel sheath, along with studies on the nature of the complexing agents. Furthermore, in response to a prescription of the 2010-2012 PNGMDR, AREVA also submitted a report presenting the characteristics of the S5 package, its manufacturing process, the results of the R&D describing the thermal processes and a forecast implementation schedule for all these avenues.

The orientation studies concerning the thermal processes performed in 2010/2011 revealed the absence of any technology that could be directly transposed to technological waste containing organic matter rich in alpha emitters. The R&D concerned the incineration / melting /

vitrification technologies involving plasmas, which most closely match the process specifications. They consist in heating the metal phase by low-frequency induction and then heating the glass by heat transfer at the metal/glass interface. One or more plasma torches are sufficient to ensure combustion of the organic part of the waste.

These technologies are based on major technological innovations (use of a plasma torch in a nuclear environment, deployment of fusion and vitrification operations within the same process, final packaging comprising two separate glass/metal phases in the same container, etc.) and feasibility has yet to be confirmed. They generally imply the ability to manage specific criticality constraints and the use of a very high temperature process associated with a glove box design.

At this stage, the production of a full-scale prototype was felt to be necessary and this was the R&D subject for the period from 2011 to 2018. It should allow inactive qualification of the process. Funding for this R&D has been requested under the investing in the future programme.

2.3.2 Other ILW-LL waste

In the 2005 File concerning the feasibility of geological disposal in a clay layer, the models and data available for ILW-LL waste mainly concerned the corrosion rates of the metal materials in standard compacted waste packages (CSD-C) and the source terms of bituminised sludge packages. The studies carried out since then by CEA, AREVA and EDF on behalf of Andra, confirm these elements, but also provide additional information concerning the following waste:

- metal waste: determination of corrosion rates of aluminium and magnesium alloys;
- polymer waste: evaluation of radiolytic production rates for different gases and different polymers, determination of the nature and quantity of water-soluble degradation products resulting from the radiolysis and hydrolysis of these polymers;
- ILW-LL glasses: proposal of a glass alteration model.

2.4 Spent fuels

A study programme for PWR fuels was carried out in accordance with the PNGMDR requirement which was to produce a less conservative release model than that adopted for the 2005 File, by 2011. The work done on this project can to a large extent be transposed to other fuels[7] with a UO_2 matrix.

A spent fuels matrix alteration model was developed by CEA. It includes radiolysis, geochemistry and electrochemistry. It is applicable to UOX and MOX fuels and should eventually allow coupling with the materials in the environment. It leads to a fuel lifetime similar to that adopted in the 2005 File (from 50,000 to 100,000 years).

Initial experimental results on the dissolution of UO_2 doped with clayey water would also seem to show control of alteration by a silicate phase (uranium silicate). This could lead to a slower alteration mechanism than that shown by current models.

In the rest of these actions, it will be necessary to include the results obtained by the FIRST-Nuclides programme, conducted at a European level and to continue the experimental and modelling work on the behaviour of UOX and MOX fuels, in order to clarify the available data, more specifically with regard to their respective behaviours in disposal conditions.

[7] Spent fuels from civil reactors (GCR and EL4 heavy water reactor), CEA experimental reactors, land-based or on-board reactors for national defence activities.

2.5 Vitrified waste

The initial and residual dissolution rates of the R7T7 glasses produced at La Hague were assessed throughout a broad range of operating conditions: atmospheric corrosion, alteration in pure or clayey water, alteration in the presence of environmental materials (corrosion products and argilites):

- the atmospheric corrosion of glass (in the presence of water vapour) leads to alteration rates higher than the residual rate in pure water;
- the glass dissolution rates in pure and clayey water were acquired at 30°C, leading to a reduction in glass alteration by comparison with the 50°C values adopted in the 2005 File;
- glass corrosion in clayey water leads to initial rates 5 times higher than those obtained in pure water. The residual rates are also multiplied by a factor of 1 to 5 depending on the magnesium concentration in the vicinity of the glass and the pH.

The effect of magnetite on the increased kinetics of glass alteration was confirmed, even if attenuated in the presence of a diffusion barrier, but the underlying mechanisms could be more complex than those considered in the 2005 File. In addition, the studies confirm very slow development of glass fracturing under mechanical loading in the disposal facility.

At the same time, a mechanistic model of the long-term behaviour of the R7T7 glass, the GRAAL model, is currently under development. This model aims to describe the complete kinetics of glass dissolution as a function of environmental conditions. The current studies aim to expand its scope of application so that this mechanistic description can be integrated into the operational model of vitrified waste package behaviour in a disposal situation.

With regard to "cold"[8] glasses (UMo, PIVER and AVM glasses), a model assuming alteration according to the initial rate of glass dissolution was developed. In the particular case of certain AVM glasses, a model of the same type as that used for the R7T7 glasses was configured, leading to a lifetime estimation that is higher by one to two orders of magnitude.

With regard to the performance and safety calculations needed for the Cigéo creation authorisation application, this work will allow a more precise and robust evaluation of the behaviour of these glass families in a disposal situation. The work to be done in the coming three years should in particular concern:

- glass alteration in atmospheric conditions. This process is at the origin of the measured rate;
- alteration in the water on the site and identification of the effect of magnesium;
- the influence of environmental materials, more specifically of corrosion products;
- interpretation of the results of the underground laboratory experiments.

[8] These are glasses for which the radionuclides content is such that the thermal release is less than that of the R7T7 glasses today produced at La Hague

2.6 Generation IV reactor waste

The aim is to obtain the data necessary to prepare for gradual deployment of generation IV reactors:

- initially supplied with the uranium and plutonium contained in the fuels taken from PWR reactors (more specifically spent MOX fuels);
- then with the implementation of systematic recycling of the uranium and plutonium;
- and, as necessary, the use of transmutation options for certain minor actinides.

It will be necessary to examine the impact of these technology and material management strategy developments on the waste generated. The deployment time-frame for these systems enables innovative R&D to be carried out on these subjects. The following primary research objectives can be mentioned:

- studies designed to limit the generation of waste comprising long-lived elements, as of the design stage;
- the study of alternative treatment and packaging solutions, for example a melting process for metal waste;
- the characterisation and management of waste, in particular secondary waste (cladding, structural elements of fuel assemblies) and reactor operating waste (cold traps, control rods);
- the impact of transmutation options on the design of the disposal facility (for example to assess the impact of a glass thermal load that is significantly reduced over the long term).

3 SUPPORTING DISPOSAL PROJECTS FOR HLW/ILW-LL AND LLW-LL TYPE WASTE, PLUS STORAGE

3.1 Disposal of low level, long-lived waste

The deployment of long-term management routes for LLW-LL type waste requires R&D to clarify the feasibility, acquire the additional data necessary to demonstrate the safety of disposal and draft design requirements for the industrial means that are eventually to be selected. These means concern shallow disposal and the upstream operations such as possible treatment and packaging of the associated residues.

The siting of such a disposal facility will in the coming years imply the search for and characterisation of the site and the production of substantiation data to be provided with a view to creation of the disposal facility.

R&D work on characterising and treating radium-bearing waste aims to improve knowledge of its behaviour in a disposal situation, to reduce physico-chemical disturbances after closure and reduce the overall volumes to be disposed of, thus preserving scarce disposal capacity.

R&D work on characterisation and treatment of graphite waste aims to explore the different processes enabling high decontamination performance to be achieved and the feasibility of total destruction of the decontaminated graphite, along with evaluation of concentrated residue packaging processes (see § 2.1 of this appendix).

For bitumen encapsulation, R&D carried out by CEA aims to confirm the radiological inventories by sampling followed by radiochemical analyses.

3.2 Reversible disposal in deep geological layer for high level, intermediate level long-lived waste, and the Cigéo project

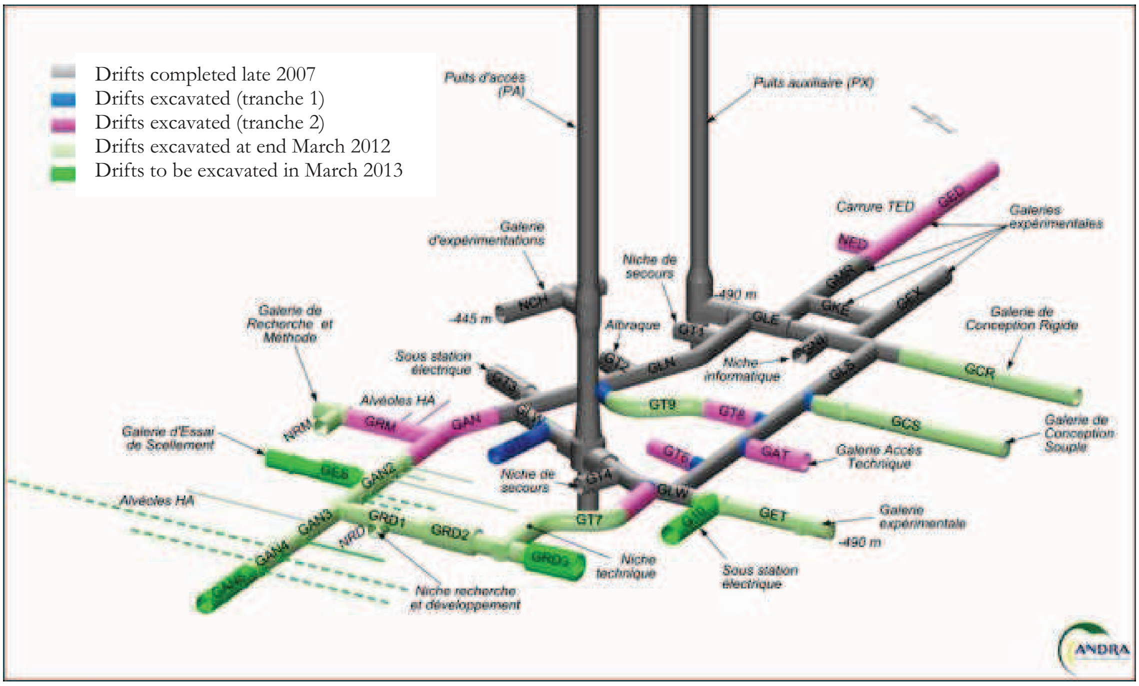
Progress and prospects of the Meuse-Haute-Marne experimental underground laboratory programme

Until 2014, the experimental programme intends to intensify technological testing and the performance of experiments in order to meet the needs of the assessors and to acquire more extensive data for drafting of the creation authorisation application. The work is being organised in three areas:

1. Continue the programme associated with the construction of drifts and HLW vaults;
2. Complete data acquisitions concerning the characteristics of the Callovo-Oxfordian argilites from the geomechanical standpoint (argilite behavioural laws, EDZ[9] transport properties corresponding to a dense 3D network of interconnected fractures), priority for engineering and simulation and transport-retention study programmes (long-duration diffusion experiment);
3. Testing of the drift sealing components, with a view to technological optimisation of the disposal structures, plus performance testing.

To carry out this programme, new drifts were excavated in 2011.

[9] EDZ - Excavation Damaged Zone.



Progress of drift excavation in the underground laboratory as at March 2012

Progress of the experimental programme

<u>Technological tests</u>

Two technological tests on the construction of drifts were performed in order to study the interactions between the rock and the liner according to the time between excavation and application of the liner:

- the construction of the rigid design test drift (GCR) in the direction of the major horizontal stress allowed examination of the conditions for production of a concrete liner poured in-situ about 6 months after excavation. Long-duration monitoring of this drift will provide information on (i) its loading and its behaviour over time, (ii) the development of the initial argilite fracturing after installation of the liner, and in particular the permeability;
- the construction of the GRD2 drift with a thick sprayed concrete liner (45 cm), installed as soon as possible after excavation.

The comparison of the behaviour of these drifts and the flexible design drift (GCS) will make it possible to understand the relationship between deformation of the terrain and the loading of a rigid liner.

In phase 2 of the test programme on the HLW vaults, a 40m vault was excavated, with an instrumented sleeve. The aim is two-fold; (i) to complete the data on the evolution of the argilite/sleeve interface, (ii) to test sleeve examination methods, in particular using optical fibres. The sleeve was made completely watertight, including on the drift wall, in order to see whether the outer ring fills with water and if so at what rate.

Characteristics of the Callovo-Oxfordian argilites

Load measurements on the instrumented tubing in small diameter boreholes continued. After more than a year of testing, the loading tends to become isotropic. A traction test was attempted on one of the tubes to measure the steel/argilite friction coefficient. The friction forces exceeded the strength of the thread, which failed at just over 60t. The heating tube installed for the tube elongation check (TEC) was subjected to a first heating cycle up to 55°C from 7th June to 9th September 2011. After total cooling, a new cycle was launched in January 2012 and reached a temperature of 90°C. With regard to the TED experiment (tube elongation test), the heating phase with three probes is currently being completed. It will be followed by a controlled heat reduction phase. A preliminary feasibility check on the long-duration diffusion test began in the autumn of 2011. The detection capability of the sensors is verified using sealed sources of ^{22}Na. This test will continue into 2012.

Tests on drift sealing components

A complete circumferential groove 30 cm wide and 2.5 m deep was made in an experimental drift parallel to the major horizontal stress (GET drift) in mid-2011. It was the subject of deformation measurements during and after it was excavated. Its walls are still stable and show no signs of change.

Planning of the experimental programme until 2014

Technological tests

The next steps in the technological tests on drift construction will on the one hand concern the excavation of a large diameter structure, the tunnelling machine assembly chamber, for which the excavated diameter is 7.8 m, and on the other, the excavation with the tunnelling machine of 80 m of drifts to test the installation of prefabricated segments. The same structures will be built in the direction of the minor stress between 2014 and late 2015. Experimental phase 3 for the HLW vaults will begin with building of a longer vault, with a new and more powerful machine. In late 2012, full-scale thermo-hydromechanical behaviour (THM) testing will be implemented in and around a HLW vault in its reference configuration.

Characteristics of the Callovo-Oxfordian argilites

The CDZ experiment (mechanical compression of the EDZ) will continue with a new loading cycle after soaking of the fractured zone. This will end with sudden removal of the wall, to observe the reaction of the EDZ. A new experimental arrangement was put into place to acquire date on the chemistry of the pore water of the argilite at 80°C. A sampling system for monitoring the development of the liquid and gaseous phases extracted from a significant volume of argilite raised to a temperature of 85±5°C thanks to four heating boreholes. New diffusion experiments are envisaged as of early 2014, if the feasibility tests prove conclusive. The aim is to have transfer distances of several decimetres, significantly higher than the centimetre scale damage zone around the injection chamber and to assess the anisotropy of the transfer properties due to the stratification of the formation. The tracers used will be ^{36}Cl, ^{22}Na and actinides.

Tests on drift sealing components

With regard to the closure of the underground structures, technological or performance tests on the individual components are scheduled. The performance test on a portion of the swelling argilite seal core (NSC experiment) is scheduled to last several years, in order to monitor resaturation and ultimately estimate the equivalent permeability of the system (core, interfaces, EDZ). In order to finalise the method for interrupting the damaged zone by means of a groove filled with swelling clay, tests will be carried out in 2012 on a mock-up to define the filling method. If these tests prove positive, a complete groove will then be filled with swelling clay, followed by a forced hydration phase. The full-scale implementation test on the complete seals (swelling clay core and low pH concrete support blocks) will be carried out in an installation outside the underground laboratory. It will be part of the European DOPAS project, starting in late 2012.

3.3 Contributions to phenomenological understanding of the disposal centre

Here we look at four topics on which, by means of experimentation and modelling, significant progress has been made in understanding the properties and behaviour of various components of the disposal centre, more specifically the geological medium.

3.3.1 The damaged zone in the argilites around the disposal structures

The initial damaged zone (defined as being that following excavation) is today characterised starting from the structure walls by:

- a zone, referred to as the EDZ, corresponding to a dense 3D network of interconnected fractures;
- a zone, referred to as fractured, characterised by fractures which are only slightly connected, if at all.

The EDZ and the fractured zone are interleaved and have an overall elliptical shape; the dimensions of the small axes of the EDZ and the fractured zone are virtually identical, whereas those of the large axes differ significantly (several radii).

The experimental data as a whole underlines the self-sealing capability by swelling of smectite minerals and mechanical closure of the damaged argilites. These mechanisms intervene very rapidly and lead to the restoration of very low water permeability.

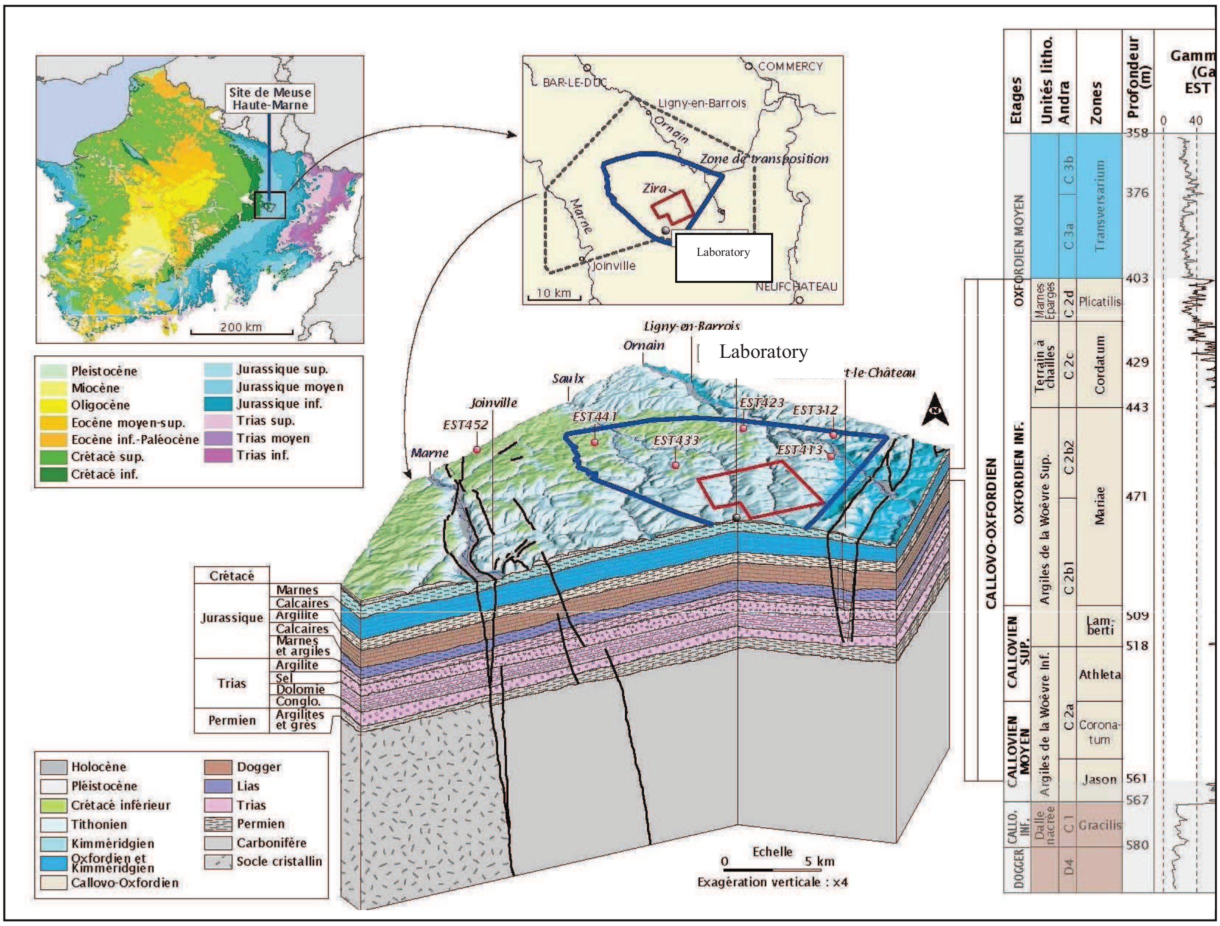
3.3.2 Transfer of solutes to the Callovo-Oxfordian layer

The analysis of the solute transfer conditions in the Callovo-Oxfordian layer of the Transposition Zone, based on the detailed characterisation of the hydro-dispersive parameters of the Callovo-Oxfordian (permeability, anion and cation diffusion coefficients) underlines the predominance of diffusion throughout the Zone.

3.3.3 Flows in the surrounding formations and their evolution over the next million years

From the Zone of Interest for In-Depth studies (ZIRA), and at the scale of the Transposition Zone (see diagram below), the hydraulic trajectories in the Oxfordian and the Dogger are on the whole homogeneous both in terms of direction (North for the Oxfordian and South-East for the Dogger) and rate (about 1 km per 100,000 years for the Oxfordian and 1 km per 20,000 years for the Dogger). The vertical hydraulic load gradients in the

Callovo-Oxfordian on the ZIRA are low, less than 0.1 m/m in absolute terms, and are primarily descending on the ZIRA. The simulations based on the surface terrain erosion rates show very little evolution of the hydraulic load gradients and associated trajectories in the surrounding formations above and underlying the Callovo-Oxfordian over the next million years.



Siting of the Cigéo disposal centre project

The hydraulic-gas transient

The data acquired from the experiments (on samples and in the underground laboratory) carried out by the gas transfer laboratory grouping, as well as via the European Forge programme and numerical simulations based on a more faithful representation of the source terms and gas transfer parameters, led to the identification of new elements. The capacity for easy transfer of gas in the argilite fractures and in the assemblies of clay pellets used to create the seal cores, leads to the consideration of gaseous hydrogen transfer in the central zone and its passage through the access structures to the Oxfordian, as being as probable as its confinement in the waste zones. The hydrogen only migrates in dissolved form and by diffusion through the Callovo-Oxfordian layer to the Carbonaceous Oxfordian and the Dogger. The hydrogen gas pressures in the repository are far lower than those of the argilite fracturing domain.

3.4 Scientific outlook

Here we identify several essential points on which research is being carried out to improve the overall understanding of the behaviour of the repository elements.

Corrosion kinetics of low alloy steel metal components and the coupling with the hydraulic behaviour of the vaults in the production and migration of gases. The degree of uncertainty surrounding the corrosion kinetics of low alloy steel components of HLW vaults (sleeve and over-pack) is still high; more specifically, there is as yet no complete explanation for the evolution of corrosion kinetics which is are first high and then fall or even remain constant over time.

3.4.1 Mechanical behaviour of structures and repository

The representation of the damaged zone as a fractured environment and the long-term deferred argilite deformation rates are two avenues for work involving both experimentation and numerical simulation. The saturation of the seals is today evaluated at full scale, notably using simplified representation models of the hydromechanical behaviour of the pellet assemblies. The specific local and transient effects during saturation must more specifically be evaluated to consolidate the design domain adopted and ultimately the control of the phenomenological evolution of the seals.

3.4.2 Radiological inventories and source terms of certain waste

The ^{129}I inventory is the result of a factory summary and is a maximum of 1% of the total inventory of initial spent fuels. New analytical techniques should be able to show that the ^{129}I levels are lower than those currently being considered in the inventory. The same applies to ^{36}Cl, with a more precise evaluation of the contents in the glasses and the structural elements potentially leading to savings, the scale of which has yet to be determined.

Observation-oversight of the repository
Significant progress has been made in the field of R&D on sensors (optical fibres, spectrometers, miniaturisation, wireless transmission). Efforts must be continued, more specifically to harden the sensors, ensuring their durability and independence, but also to develop means of merging the data that will be acquired during operation of the repository and decision-making systems.

Micro-nano approach to processes
Certain Thermo Hydro Mechanical and Chemical (THMC) processes, notably those taking place at the interfaces, require a small scale approach in order to improve our understanding of them. Work will therefore be needed for modelling and quantifying these processes. This research will in particular be carried out in a project supported by the NEEDS cross-cutting programme.

3.5 Storage research

3.5.1 Studies and research on innovative storage concepts

In order to explore the potential innovations highlighted in the interim report it published in 2009, Andra has initiated a technical and optimisation study to examine in greater detail three concepts for HLW waste and one concept for ILW-LL waste, all assumed to take place on the surface. These concepts were drawn up in particular to ensure greater versatility for the packages accepted. Furthermore, a more detailed study of the oversight systems has been initiated.

HLW in ventilated shafts concept

In the ventilated shafts concept, racks containing six primary packages or three disposal packages make the storage system more versatile. As a counterpart, the heating power of the primary packages is limited to 1,000 W, as opposed to 2,000 W in the glasses storage extension – south-east (E-EV-SE) at La Hague. The racks are stacked vertically in closed stainless steel shafts. Cooling is by means of natural ventilation. This concept is suitable for storage, after an initial thermal decrease period, of a large number of heavily exothermic vitrified waste packages for significant cooling up to 85 years and beyond. Its versatility means that it is able to take HLW packages of various dimensions removed from disposal.

The more detailed study aims notably to increase the unit thermal power of the HLW packages that can be accepted in this storage concept and to improve the oversight possibilities, while looking to reduce the mass of mobile handling equipment which would make it necessary to reinforce the storage vaults.

The HLW on slabs concept

The storage of compacted hulls at La Hague was designed to take standard compacted hulls and end-pieces (CSD-C) waste packages. The average thermal power of these packages is about 15 to 20 W. A storage concept for HLW packages in their primary or disposal form, with a thermal power of less than 500 W was in 2009 derived from that for storage of compacted hulls (ECC). Ventilation is mechanical and horizontal, with partial recycling of the outgoing hot air, in order to regulate the humidity. The storage building responsible for radiation protection and handling is remote-operated. The primary or disposal packages are placed vertically on a base positioned on the building slab(s). This concept is versatile and ensures the accessibility of each package for monitoring. It would be suitable for creating buffer storage capacity or to take packages retrieved from disposal. Additional studies concern improvements to heat removal.

The HLW in concrete modules concept

A storage concept for HLW packages in their primary or disposal format, with a thermal power of less than 1,500 W, was derived in 2009 from that of the existing NUHOMS® concept for spent fuels. The HLW packages are placed in storage containers in groups of 24 to 30 primary packages or 15 to 16 disposal packages. These containers are closed and ensure package confinement and protection. They are placed horizontally in concrete bunkers cooled by circulating natural ventilation in contact with the container. The bunkers provide radiation protection. This storage concept will be optimised for improved interfacing with the transport system.

The ILW-LL storage concept

The primary packages grouped in racks, or the disposal packages, are pre-stacked and then placed in a long row on mobile beams, for adjustment to the various dimensions of the racks and disposal packages.

4 CONTINUED RESEARCH INTO SEPARATION-TRANSMUTATION

4.1 Purpose and implications of the research:

The purpose of separation-transmutation is to remove the minor actinides from the ultimate waste as they are the main contributors to its long-term radiotoxicity and to the residual thermal load after the decay period.

By the end of 2012, in accordance with the requirements of the Act of 28th June 2006, CEA must issue a file on the studies and research carried out into separation and transmutation, jointly with that carried out on the new generations of nuclear reactors and on accelerator driven reactors dedicated to the transmutation of waste, in order to be able to make an assessment of the industrial prospects for these technologies. The following aspects of the studies carried out during the period 2010-2012 should be underlined:

- the transmutation of americium and curium would reduce the long-term radiotoxicity of ultimate waste by a factor of up to 100 at 1000-10,000 years, but without contributing any gains with regard to the radiological impact of the repository. Andra has identified the fact that the significant retention in the Callovo-Oxfordian argilites confined them in the near field and that the activity flux associated with the minor actinides leaving the host formation was negligible);
- transmutation of americium alone would have limited impact on long-term toxicity but would help reduce the repository footprint by a factor of 2 to 5 (a factor 5 reduction would require prior storage for 120 years) for HLW waste alone;
- overall performance is limited by the waste produced beforehand ("initial stock"), as well as by the ability to absorb the inventory at the end of life of the fleet;
- the transmutation of minor actinides only has any sense if multi-recycling of plutonium is utilised;
- transmutation is only effective if it leads to the fission of the actinides: in this respect, fast neutron systems are the most appropriate.

Transmutation is a complex operation, which requires the recovery of elements of interest (separation of the minor actinides), and then their fission recycling in the reactor, with several options being possible: homogeneous or heterogeneous solutions, dedicated stratum.

The research carried out at CEA has validated minor actinide separation processes and certain transmutation systems on real fuels on a laboratory scale; these concepts still need to be consolidated and, prior to industrialisation, would require experimentation on a larger scale.

Implementation would only be possible with the deployment of fast neutron systems in the fleet. The impact of such a strategy, in terms of both reactors and the cycle, was the subject of an initial analysis based on all the criteria to be considered. Industrial implementation cannot be envisaged for the waste generated or engaged by the current NPP fleet.

In the light of current knowledge, retrieval of the minor actinides from vitrified waste for the purposes of transmutation would not seem to be conceivable. The implementation of the transmutation option does not obviate the need for geological disposal of the ultimate waste.

The aim of the research for the period 2013-2015 will mainly be (depending on decisions which could be taken after the 2012 deadline stipulated by the 28th June 2006 Act):

- to consolidate the separation concepts developed for retrieval of the minor actinides;

- to continue with development of fabrication processes of fuels charged with minor actinides;
- to continue with experimental irradiation concerning the various concepts envisaged for the transmutation of the minor actinides, and to clarify the possibilities for demonstrations in the ASTRID prototype and the MYRRHA facility;
- to fine-tune the technico-economic assessments according to the various deployment scenarios;
- to continue with upstream, exploratory or fundamental research in this field.

4.2 The separation of the minor actinides

The research has led to the development of specific extractants and separation processes, successfully tested in the laboratory, for each of the solutions envisaged: Am extraction (EXAm), extraction of Americium and Curium (SANEX) and grouped extraction of all the actinides (GANEX).

For the period 2013-2015 this will mainly involve:

- for the EXAm concept, continuing with "complete experimentation", envisaged for a few kg of spent fuels, from reprocessing up to the production of AmO2 pellets. This experimentation will allow laboratory scale testing of the sequence of the various individual operations (separation, conversion into oxide, fabrication of pellets), but also of the various related operations (in particular the management of effluents and by-products);
- optimising the processes, primarily the EXAm concept for the retrieval of americium. This will essentially involve seeking to reduce the size of unit operations and the flows generated, by looking to work with more concentrated material flows;
- conducting exploratory research on alternatives to the molecules and processes being studied;
- continuing studies to consolidate the processes in order to better examine the conditions for possible industrial deployment (technologies, monitoring and control systems in particular).

4.3 Fabrication of fuels containing minor actinides

The fabrication of compounds containing minor actinides must take account of the specific nature of these actinides, more specifically their radioactivity (alpha, gamma, neutrons), especially for recycling concepts referred to as "heterogeneous", with fuels containing about 10% minor actinides on a UO_2 support.

For the elements of interest, primarily americium, the research concerns:

- the production of oxide powders obtained by means of U-minor actinides co-conversion processes (in particular co-precipitation followed by co-calcination);
- the fabrication of compounds rich in minor actinides, paying particular attention to the production of very fine particles, which are potential sources of irradiation (eradication of crushing steps from the powder mixing processes, exploratory studies concerning alternatives such as gelification processes);
- studies to develop remote-operated technological components (conventional maintenance in gloveboxes impossible), and the development of advanced robotics concepts (which could also benefit the plutonium fuels fabrication studies);

- studies concerning compound cladding materials, taking particular account of the large quantities of helium generated.

The period 2013-2015 will also involve continued study of pilot facilities to prepare for transmutation experiments on fuel cladding pins.

4.4 Experimental burnup

The studies are continuing in order to finalise the transmutation devices and validate the correct behaviour in-reactor of the transmutation fuels with respect to the various concepts envisaged. We will first of all focus on the transmutation of americium:

- ***for homogeneous recycling***, further to the previous experiments in PHENIX which validated the concept up to levels of a few % of minor actinides, this will involve looking at the high burnup fractions (about 100 GWd/t) envisaged for fast reactors;
- ***for the concept of recycling in blankets***, this involves conducting mini-disk scale experiments (less than one gram) while attempting to reproduce the operating characteristics of fast reactor blankets during burnup in experimental reactors (temperature conditions in particular) and the production of gas corresponding to the targeted transmutation ratios. It is on this mode of transmutation, for which the studies are least advanced, that most of the efforts over the period 2013-2015 will focus.
- ***for the concept of recycling in a dedicated layer (Accelerator Driven System - ADS in particular),*** post-burnup analyses of FUTURIX fuel samples irradiated in PHENIX under an international CEA-DOE-ITU programme will be carried out (as of 2013).

The possibility of demonstrating an ADS prototype, such as the European MYRRHA project in Belgium to demonstrate burnup of fuels dedicated to the transmutation of minor actinides, will also be examined. The aim will also be to specify the nature and scale of the demonstrations that could be carried out in the ASTRID prototype or in the MYRRHA facility. During the 2013-2015 phase, efforts will more specifically focus on precisely identifying the domain chosen for the ability to transmute the minor actinides in ASTRID, by in particular identifying any “threshold effects” on the dimensioning of the reactor (domains for which the aim of the demonstration significantly affects the dimensioning of the reactor).

4.5 The scenarios studied

The studies carried out over the period 2006-2012, which will be reported in the file presented by CEA in late 2012, allowed an assessment – on the basis of various criteria (gains in waste management, but also costs and detrimental effects throughout the fuel cycle) – of the impact of the implementation of minor actinide separation-transmutation options.

During the period 2013-2015, CEA will focus on:

- consolidating these assessments: estimation of uncertainties, more detailed quantification of certain criteria, updating of certain technological parameters on the basis of advances in research, whether for critical reactor concepts or for ADS;
- together with the industrial partners and coherently with the decisions to be taken by the public authorities, drafting of possible deployment scenarios for such options in the French NPP fleet. The options will be envisaged for the ASTRID prototype sodium-cooled fast neutron reactor and, at a European level, for the MYRRHA facility.

4.6 Upstream research

Upstream research to support separation and transmutation studies will be carried out:

- for aspects related to radiochemistry and to studies of separation concepts in Atalante at CEA Marcoule, and within the Marcoule Separation Chemistry Institute (ICSM). A contribution to CNRS's NEEDS programme is also being envisaged. Research is looking at identifying and controlling the fundamental phenomena which govern the selective extraction of the elements of interest, the actinides, and the exploration of original concepts;
- for aspects linked to transmutation, through the use of analytical irradiation (PROFIL irradiations) to validate nuclear data at CEA and validate elementary nuclear data within the framework of the CNRS' NEEDS programme and European programmes, the acquisition of nuclear data concerning the elements of interest and the sensitivity studies will be continued.

5 CONDUCTING RESEARCH TO SUPPORT THE SAFETY ANALYSIS FOR DISPOSAL PROJECTS

IRSN has taken the necessary steps to produce the appraisals of the safety files which Andra will be presenting for the creation of new radioactive waste disposal facilities. The areas which warrant major efforts include that of the safety of a deep geological disposal facility. The focus of the research activities at IRSN on this subject is different from those for which Andra is responsible. They can call on far more limited resources and concentrate on a smaller number of subjects, aiming to provide the independent support necessary for the future appraisals. In this respect, the deadline of the 2013-2015 plan coincides with a key date for IRSN, which will be required, on behalf of ASN, to examine the file prepared by Andra to support the application for creation of a deep geological disposal facility. With this in mind, IRSN will utilise all the new knowledge acquired by itself or by the scientific community, more specifically since the examination of the 2005 File on the feasibility of a geological disposal facility in a layer of clay. For the period 2013-2015, the Institute more specifically intends to:

- **increase its research efforts on the behaviour of the repository confinement barriers, more specifically during the transition period,** beginning during the operations phase and continuing after its closure**.** In this respect, the phenomena to be considered are notably**:**
  - **thermal, hydric and mechanical (THM) phenomena** liable to affect the performance of the components of the disposal facility. Over the coming three years, IRSN will focus its efforts on the one hand on continuing the seal experiments in the Tournemire experimental station, in order to assess the key parameters which govern the overall performance of the seals (SEALEX tests), and on their modelling and, on the other, on understanding and modelling the effects of gases inside a repository. On this point, IRSN is a contributor to the European Forge project, more specifically concerning the evaluation of the gas formation mechanisms and the numerical simulations of the expected effects. With regard to the mechanical behaviour of the disposal vaults (appearance of the EDZ, role of supports, etc.), IRSN will supplement the simulations of this behaviour by taking account of the observations made in the various laboratories, notably that in Bure;
  - **the main factors in the physico-chemical evolution of the components of the disposal facility.** The studies initiated aim to clarify the influence on safety of the chemical processes during the various phases in the life of a disposal facility. On this point, the Institute intends to continue the study of the possible effects of bacterial growth on steel corrosion and the study of the phenomena of radiolysis and the

degradation of waste packages, as well as to step up its effort to understand the complex cement/iron/clay chemical interactions. The experiments carried out for this purpose in the Tournemire experimental station will concern the impact and duration of the oxidising transient in the vaults simulating the HLW type vaults (OXITRAN tests) and the durability of the concrete and near field disturbance of the clays under the effect of temperature (CEMTEX tests) or of artificial ventilation. IRSN will also take part in the experimental studies carried out in the Mont Terri underground laboratory (Switzerland) concerning the evolution of clayey materials under the effect of an alkaline pH;

- **complete the study of the characteristics of importance for the confinement capacity of the geological barrier,** over and above the analysis of the field data concerning the site studied by Andra and the assessment of the limits of the survey methods employed, the Institute will continue its efforts to study the differential fracturing of the clays, which notably aim to provide an explanation of the presence or absence of fracturing in various clay formations. At the same time, by means of the FRACTEX programme to be implemented in the Tournemire experimental station, IRSN plans to study the transport properties associated with areas of the medium subject to little disturbance. IRSN will also complete its knowledge of the transfer properties associated with the indurated clay formations, by taking part in the TAPSS2000 national programme associated with deep drilling to 2000 m by Andra on the Meuse /Haute-Marne site and the research programmes associated with the project to drill through the entire clay layer of Mont Terri;
- **to consolidate its ability to perform global modelling of the disposal facility.** In this respect, it will continue its assessment of the influence of the hydraulic schemes of the Meuse/Haute-Marne site by means of specific hydrogeological models, integrating all the new data acquired. IRSN also intends to make a particular effort on simulation of radionuclide transfers in the geological medium, by means of the MELODIE computer code. Finally, IRSN intends to delve more deeply into defining possible very long-term evolution scenarios for the disposal facility and its environment, on the basis of the scientific knowledge available on the subject.

IRSN does not carry out its research in isolation. At the national level, it is involved in numerous cooperative ventures with a network of renowned scientific partners, organisations, schools and universities. Together with CNRS, the Institute will also take part in the NEEDS programme. IRSN has also renewed the MoU established with Andra and allowing joint research work to be carried out in accordance with the provisions allowing compliance with the necessary ethical rules. At an international level, IRSN aims to work on European projects. In addition to the Forge programme in the 7th Framework Programme (FP) in which it is a participant (see above), the Institute is coordinating the SITEX project, a grouping of 15 appraisal organisations and safety regulators, which aims notably to create the conditions, Europe-wide, for interconnecting the research on technological developments, supported by the IGDTP platform, and that concerning the safety of disposal facilities, in which the independent organisations of the nuclear licensees must necessarily take part. International partnerships, notably with Japanese (JAEA, JNES), Canadian (CCSN) and Russian (SEC/NRS, IBRAE) organisations have also been created around particular IRSN research projects.

Finally, the Institute recalls its desired to continue to make its experimental resources available to the French and foreign scientific community, in particular the clay medium experimental station at Tournemire (Aveyron), which since 2007 has been integrated into the IAEA centres of excellence.

6 GAINING A CLEARER UNDERSTANDING OF THE SOCIAL DIMENSION OF WASTE MANAGEMENT

The involvement of the Human and Social Sciences (HSS) in the field of radioactive waste and materials management is justified upstream by the desire to make the various recommended solutions more robust. Their acceptability, which in the end is political in nature, is made easier when all the phenomena involved are dealt with in an appropriate framework, without ignoring their socio-economic, environmental, political, cultural, etc. aspects, and the various scientific and technical issues involved are interconnected. One-dimensional, inward-looking R&D has little chance of helping technical projects succeed, as is shown by the history of nuclear waste management in France prior to 1991. The aim of HSS research is thus to integrate the social aspects into the various on-going projects and ensure that they all work together in a cross-disciplinary system. Collaboration with researchers from these diverse backgrounds must from the outset aim to create specialised communities on subjects of common interest with the operators.

The topic of reversibility was the first to be given this level of priority by Andra. Several scientific events, in particular the organisation of two symposia, resulted from this work, as did the publication of the collective work "Making radioactive waste governable. Deep disposal undergoes the reversibility test". An economic sciences PhD dissertation was also defended on this topic.

Andra also launched the "memory" project in 2010, which on the one hand comprises work designed to continue to create and improve the memory of and records about the facilities and, on the other, scientific studies concerning materials ageing and issues specific to human and social sciences (HSS). Scientific studies into the ageing of materials consisted in testing the permanent ink/paper combination by means of standardised tests. Durability studies on other media for the longer term are currently being defined. They will concern non-paper media for writing and engraving, in particular studies of surface markers to be installed on the cover over the centres and the production of sapphire disks as demonstrators for a memory medium, the longevity of which could be up to a million years. As for HSS studies, an initial bibliographical approach is planned, in order to define a framework for any research to be included in the Agency's scientific programme. The envisaged work is based around the following topics:

- the longevity of languages and symbols, in order to determine for what reasonable time current or dead languages can be known and what the communication solutions could be once these languages cease to be known;
- institutional conservation of written works, sounds, images, objects, etc. by specialised French and international organisations, to analyse the preventive measures taken to limit deterioration over time and encourage assimilation and transmission by future generations;
- long-term digital archival, more specifically by organising an intelligence watch in this field, which is beginning to become organised and which, within the next few decades, could open up new prospects for the long term;
- the archaeology of techniques and landscapes, incorporating man-made changes and geodynamic changes, as well as the possibilities of memory with human creations (use of the in-fill of surface-underground links as a memorisation tool);
- the memory of "legacy" repositories not managed by Andra, which exist in various places in France (uranium mines, nuclear tests, etc.);
- the foreseeable changes in society;
- the inclusion of preserving the memory of repositories in teaching programmes on nuclear energy, heritage and memory;

- transmission of memory between generations via social networks on the internet.

Andra is also taking part in international work on memory within the “Preservation of record knowledge and memory” working group set up by the Nuclear Energy Agency (NEA/RWMC/RK&M).

More recently, in 2011, Andra set up a multidisciplinary steering committee to create a grouping of cross-cutting human and social sciences laboratories (GL-SHS). It comprises researchers from CNRS, SciencesPo Paris, Ecole des Hautes Etudes en Sciences Sociales, the Institut Francilien Recherche Innovation Société, Mines ParisTech and other university institutions. The general central research topic for the grouping is “transmission between generations and understanding of long time scales”. This is because the time-frames involved in the Agency’s activities, in particular in the management of the most highly radioactive waste, is indeed unique when compared with other industrial areas. It raises particularly complex questions which notably concern the ability to anticipate events over long periods of time and to ensure that they are managed. The approach adopted was to look at the practices and tangible arrangements made to produce the Cigéo geological disposal project. The question arises of the transmission to future generations of the means and resources for intervention on the fate of this project. In order to encourage and promote cross-pollination between perspectives and exchanges, the GL-SHS research programme is jointly put together by the members of the steering committee and the pilot Andra. It is built around the following three core subjects: governance, knowledge and memory; socio-economic evaluation.

The first part deals primarily with integrating social aspects and scientific and technical elements into the decision-making processes. Profound changes have been made in the field of radioactive waste governance in recent decades. The question now is to ensure that the long time entailed by research (and by radioactive waste) is in phase with industrial time and political time. These time-related approaches are the responsibility of numerous players with interests that may often conflict, and lead to organisational changes.

The second point above all concerns the robustness of knowledge over the long-term and the transmission between generations of the information, practices and knowledge necessary for the Cigéo project. The specific knowledge production system put into place at Andra to demonstrate the feasibility and safety of reversible geological disposal of radioactive waste, in particular the use of modelling and numerical simulation tools, is of particular interest to science historians. It is the characteristic traits of modern techno-sciences and the complexity of the relationship between science and society that can also be examined from a fresh perspective. The techno-scientific understanding of long time-scales, or more precisely of the future, is an object of research which is currently arousing great hopes on the part of the social sciences. The problem of understanding long time-frames is not independent of that of memory, for which the socio-anthropological perspective preferred by the GL-SHS could prove to be highly instructive. Despite the efforts made since the 1980s by those in charge of managing radioactive waste around the world, in particular the Scandinavian countries with institutional archival and the United States with markers, little academic research has however been carried out on the question of multi-millennia memory in this field.

The third point is based on the study of socio-economic evaluation methodologies and practices applicable to the Cigéo project, as well as directories of the players involved, or liable to be so, in this evaluation. One aspect is to analyse the respective functions and roles of the different types of evaluation (ex ante/ex post/ex nunc, internal/external, etc.), how different forms of knowledge are integrated into them, and the place given to forward planning and scenario writing in them. The comparability with similar projects through the study of "major project" case studies was also given particular attention.

Finally, CNRS's NEEDS places HSS at the heart of nuclear questions and envisages looking at the question of time in a more general manner, from the risk management and assessment viewpoint. This perspective requires adopting an approach that is both retrospective concerning methods of memorising sites entailing a risk (polluted sites, mining residues, etc.) and forward-looking concerning the pooling of data and their transformation into operational knowledge in appropriate information systems (for example, for management of contamination according to differentiated time-scales). This positioning will more broadly enable concrete answers to be given, based on past experience in order to envisage scenarios for assuming public responsibility for the transfer of knowledge. With regard to the management of radioactive waste and materials over long time-frames, the issues are not only the memorisation of data but more generally the actual process of recording a trace by the players involved and the decision-making styles that are most appropriate to such time-scales. Particular attention will be given to the following questions:

- What are the social implications of the various possible long-term management solutions (storage, transmutation, geological disposal, etc.)?
- What decision-making modes are associated with the various long-term management options?
- What are the technical and political consequences of the absolute need for reversibility?
- How can the knowledge necessary for the operation or decommissioning of today's equipment be passed onto future generations?
- How can one ensure justice and fairness across the territory and between the various generations? What are the underlying ethical issues?
- What are the impacts on man and the environment of the various radioactive waste and materials management choices?
- What is the role of the experts and the citizens with regard to the public decision concerning a situation of long-term uncertainty?

In answering these questions, the aim is not social engineering in order to make the envisaged solutions more acceptable, but on the contrary to fuel the debate and the public choice and thus strengthen the ties between science and society. The aim of the NEEDS (nuclear, risk, society) programme is to ensure that knowledge progresses and that programmes, networks and diverse skills are established, while addressing the need for transparency and sustainability which today characterises the public debate. This programme also intends to build on the HSS knowledge acquired on the topic of nuclear waste, more specifically through the considerable work done on this question at CNRS.

Appendix 5: Concepts and plans for the post-closure period

1 CONCEPTS AND PLANS FOR THE PERIOD FOLLOWING THE CLOSURE OF INSTALLATIONS CLASSIFIED ON ENVIRONMENTAL PROTECTION GROUNDS

1.1 Very low level waste disposal centre

The operation of the VLL waste disposal centre is regulated by order 2012040-0002 authorising the operation of this first installation classified on environmental protection grounds dedicated to the disposal of radioactive waste. This order is derived from the regulations applicable to the disposal of hazardous waste (ministerial order of 30th December 2002, amended). Andra also wanted to follow the same methodology for assessment of the long-term impact of the VLL waste disposal centre as that already used for the low and intermediate level waste disposal centres, the Manche disposal centre and the LLW/ILW waste disposal centre in the Aube.

The order thus presents the requirements of resources imposed on the hazardous waste disposal facilities by the regulations as well as the additional requirements specified as a result of the safety assessments carried out for all the phases in the lifetime of the facility, from the construction phase to the post-oversight phase.

In accordance with the authorisation order, Andra will propose a project to the Prefect defining the institutional controls to be applied to all or part of the facility, no later than one year after the end of the operating period. These institutional controls could prohibit the building of constructions and structures liable to impair the conservation and oversight of the site covering. They should also ensure that the means of collecting leachates before sealing of the shafts at the end of the oversight phase are protected and that confinement of the waste emplaced is durably maintained. Moreover, the purpose of the oversight phase will be to monitor the evolution of the disposal facility for a period of at least thirty years after the last waste is emplaced, and its conformity with the forecasts and the order of the Prefect. For this purpose, controls will be maintained, more specifically:

- regular upkeep of the site (ditches, cover, ponds, fencing, etc.);
- geotechnical observations of the site, with regular and at least annual updating of the topographical survey;
- periodic measurement of the quality of the water collected from the centre and discharged into the environment and checks on compartments of the ecosystem in the near environment of the VLL waste disposal centre.

All of these measurements will be to verify the absence of radioactive or chemical pollution in the environment of the centre. They will be able to ensure early detection of any behavioural anomalies and anticipate any remediation measures.

Following this oversight phase, the record of the centre will at the very least consist of institutional controls entered into the land registry.

2 CONCEPTS AND PLANS FOR THE PERIOD FOLLOWING THE CLOSURE OF BASIC NUCLEAR INSTALLATIONS

The legislative framework applicable to Basic Nuclear Installations for the period after closure of the facilities, is more specifically based on:

- the Act on transparency and security in the nuclear field (TSN Act 2006-686 of 13th June 2006 codified) which specifies that the transition of a BNI to the oversight phase is subject to authorisation (Article L.593-25 of the Environment Code) and that the administrative authority can apply institutional controls around this BNI (Article L.593-5 of the Environment Code);
- decree 2007-1557 of 2nd November 2007 which specifies the content of the authorisation application file for transition to the oversight phase. This file in particular contains: the impact assessment, a safety report, a risk management study, the general oversight rules and, as applicable, the institutional controls (see Art. 43)
- the order of 7th February 2012 setting out the general rules for BNIs. Chapter V of this order concerning radioactive waste disposal facilities stipulates that: *"In compliance with the objectives set forth in Article L. 542-1 of the environment code, the choice of the geological environment, the design and the construction of a radioactive waste repository, its operation and its entry into the oversight phase are defined such that protection of the interests mentioned in Article L. 593-1 of the Environment Code is ensured passively against the risks presented by the radioactive or toxic substances contained in the radioactive waste after entry into the oversight phase. This protection must not require intervention beyond a limited oversight period, determined according to the radioactive waste disposed of and the type of disposal repository. The licensee justifies that the chosen design meets these objectives and justifies its technical feasibility."*

2.1 The Manche disposal facility

From the regulatory viewpoint, the Manche disposal facility (CSM) is a basic nuclear installation (BNI n°66) dedicated to the surface disposal of low and intermediate level, short-lived waste. The creation authorisation decree dates from June 1969. The transition of the facility to the oversight phase was authorised by decree 2003-30 of 10th January 2003. This oversight phase is conventionally scheduled to last a period of three hundred years and includes a discharge license dating from 10th January 2003. In 1996, on the basis of the conclusions of the Commission assessing the situation of the Manche disposal facility (known as the "Turpin Commission"), it was decided that the *"site may not be relieved of all controls"* after this oversight period. Andra thus confirmed the need to maintain and eventually transmit the memory of the site and take all necessary steps to limit the nature of the constructions or equipment which could be installed on it.

The concepts and plans for the post-closure period comprise: the design of the facility, oversight and maintaining a recorded memory:

- <u>measures concerning the design were taken by the licensee during the operating phase</u>. Thus, after closure, the disposal facility corresponds to a mound in which the waste packages disposed of in the structures are protected from climatic hazards by a low-permeability cover; an effluents management system recovers water that has infiltrated through the cover and/or into the disposal facility. The water recovered is transferred to the AREVA-La Hague treatment installation, in accordance with the discharges authorisation order;

- decree 2003-30 authorising transition to the oversight phase mentioned that the licensee must ensure oversight appropriate to the facility and its environment. This is defined in the regulatory oversight plan, which includes monitoring of the cover, the confinement of the disposal structures and the discharges from the centre. This plan specifies that the results are regularly sent to ASN (annual report) and to the public (summary of the annual report presented to the CLI). The decree also defines that protection of the facility against the risks of intrusion and malicious acts is guaranteed for the duration of the oversight phase. Furthermore, the decrees stipulates that every ten years, the licensee shall study whether or not the oversight and protection measures applicable to its facility need to be updated;
- in terms of maintaining a memory and record of the facility, three avenues have been identified:
  (i) long-term archival of the information: decree 2003-30 defines the requirements concerning the long-term archival of information:
  - *Detailed memory*: the documents are duplicated on permanent paper and archived in two separate places, in the Manche disposal facility and in the French National Archives. The archive is updated every 5 to 10 years depending on developments in the Centre;
  - *Summary memory:* an initial version of this document of about a hundred pages was submitted to ASN and the CLI in 2008. This document should be revised as and when the safety reviews are carried out, in order to incorporate all the operating experience feedback gained from the oversight phase. When it is considered as having been stabilised, it will be printed on permanent paper and widely distributed, as stipulated by the technical prescriptions;
  (ii) information of the public, in particular during the oversight phase, more specifically via exchanges with the local information committee (CLI) and through communication measures;
  (iii) the draft application for the creation of institutional controls to minimise the risk of intrusion into the disposal facility for as long as possible after the oversight phase. Such institutional controls were suggested by the Turpin Commission and envisaged by Andra, in the 2009 safety report, pursuant to Article 31 of Act 2006-686 of 13th June 2006.

2.2 The Aube waste disposal facility

From a regulatory viewpoint, the Aube LLW/ILW waste disposal facility, which took over from the Manche disposal facility, is also a basic nuclear installation (BNI n°149). The creation authorisation decree of 4th September 1989, was modified by decree 2006-1006 of 10th August 2006 plus its discharge license of 21st August 2006.

With regard to the period following operation, the creation authorisation decree for the LLW/ILW waste disposal centre notably stipulates that: (i) during the oversight phase, "*the structures shall be protected by a cover of very low permeability" and "the facility shall continue to be monitored for a time allowing radioactive decay of the radionuclides with short or intermediate half-lives, to a level presenting no further significant radiological risk.";* (ii) following the oversight phase, *"it shall be possible for the land occupied by the facility to be used normally without any radiological restriction […] no later than 300 years following the end of the operations phase"*.
In addition to the regulatory aspects, Andra also follows the recommendations of RFS I.2 which defines the fundamental safety objectives, the design bases for a repository and the monitoring of the facility during the operations and oversight phases.

In the same way as the CSM, the concepts and plans for the LLW/ILW waste disposal centre's post-closure period comprise: the design of the facility, oversight and maintaining a recorded memory:

- the measures concerning the design were taken by the licensee during the operating phase in accordance with the requirements of RFS I.2:
  (i) the limitation of initial activity: the radioactive waste accepted by the LLW/ILW waste disposal centre is waste with a short or intermediate half-life, with limited quantities of long-lived radionuclides or those with low or intermediate specific activity. The aim is for the activity of the radionuclides disposed of to have significantly decreased during the 300 years of facility oversight;
  (ii) confinement of the waste is ensured by the package and the structure (including the cover and the infiltrated water collection networks) during the operation and oversight phases and by the geological formation on which the disposal facility is sited, notably during the post-oversight phase;
- the provisions concerning oversight of the facility and its environment. At closure of the centre, in accordance with decree 2007-1557, Andra will apply for authorisation to make the transition to the oversight phase and shall propose general oversight rules. A decree will authorise transition to the oversight phase. The monitoring approach currently employed during the operations phase will in principle continue during the oversight phase. This oversight relies on a certain number of measurements (notably radiological, chemical, water table heights, hydrological, climatological) the monitoring of which over a period of time should make it possible to: (1) verify the correct working of the disposal facility, ensuring the absence of any unacceptable dissemination of the radionuclides initially contained in it; (2) detect any abnormal situation or development in order to identify and locate the causes and initiate the necessary remedial measures; (3) gain sufficient understanding of the disposal facility evolution mechanisms; (4) assess the radiological and chemical impact of the disposal facility on the population and the environment and monitor its evolution, in order to verify compliance with the regulatory requirements; (5) ensure protection of the facility against the risks of intrusion and malicious acts;
- provisions concerning maintaining the recorded memory: Andra relies on the reference solution developed for the CSM, for which preparations are made as of the operations phase. The CLI should also continue during the oversight phase and thus allow public information and consultation.

2.3 The Cigéo disposal facility project

The Safety Guide for the final disposal of radioactive waste in a deep geological formation was issued by ASN in 2008. This guide defines:
- the fundamental safety objective: protection of the health of individuals and of the environment is the fundamental safety objective of the disposal facility. After closure of the disposal facility, protection of the health of individuals and the environment should not depend on oversight and institutional controls, which cannot be maintained with any degree of certainty beyond a limited period;
- the design bases and safety principles;
- oversight and maintaining the recorded memory: a facility oversight programme must be put into place during the construction of the disposal structures and until closure of the facility. Certain monitoring provisions could also be maintained after closure of the facility. The need to implement this oversight should be taken into account as of the design of the disposal system. The recorded memory must be maintained after closure of the site.

The Cigéo disposal project will be designed in a deep geological layer, the Callovo-Oxfordian, to allow long-term confinement of the substances contained in the High Level and Intermediate Level long-lived waste. According to Article L542-10-1 of the Environment Code, *"a disposal* *facility in a deep geological formation for radioactive waste is a basic nuclear installation".* The Cigéo project thus falls under the regulations applicable to BNIs, as defined in §1.2.

In accordance with the regulatory framework, notably the order of 7th February 2012, and the above-mentioned ASN Safety Guide, the Cigéo disposal facility project is designed to evolve from active safety to entirely passive safety, where no human intervention will be required. After operation, the facility will be closed and enter the oversight phase.

As with surface facilities, the concepts and plans for the period following closure of the planned Cigéo disposal facility comprise the design of the disposal installations, oversight and maintaining the recorded memory:

- provisions concerning the design: to meet the post-closure safety objectives, the deep geological formation disposal facility is designed to be able to guarantee and demonstrate safety during operations and for a long time following its closure, with regard to both man and the environment, while being reversible for a period of at least 100 years. In accordance with the regulations and the ASN Guide, the underground disposal facility shall, once closed, meet the post-closure safety objectives passively. The safety of the facility is thus based on a range of components for confining the radioactivity and isolating the waste from any possible external hazards;
- the provisions concerning oversight of the facility and its environment. Steps shall be taken to maintain the memory and ensure monitoring for as long as possible. Monitoring of the environment is envisaged prior to construction (initial baseline state), during construction and throughout the operating period. This could be continued after closure of the underground facility and decommissioning and dismantling of its operating installations on the surface. This oversight will meet the regulatory requirements concerning the monitoring of the impacts of the facility. All of these measurements will be to verify the absence of radioactive or chemical pollution in the environment of the centre and ensure that it is functioning correctly. The long-term environment observatory (OPE) offers a framework for monitoring the environment before and during construction and operation. An oversight programme is also designed with respect to post-closure safety to monitor a certain number of parameters in the underground facility during its operating phase. The means implemented for post-closure oversight of the Cigéo project will notably be based on experience feedback from the surface facilities;
- provisions concerning maintaining the recorded memory focus primarily on: transmission to future generations, to inform them of the existence and contents of the facility and provide them with knowledge enabling them to understand their observations, to facilitate any actions or to transform the site. At present the reference solution adopted by Andra to guarantee a memory of its disposal centres is based on five measures: (i) two "active" memory systems to ensure that a recorded memory is preserved in the short and medium terms, and (ii) three "passive" systems for the longer term. This reference arrangement must be implemented for the Cigéo disposal project, with the need for the memory to be maintained following closure of the facility for as long as possible, and at least for five centuries.
At this stage of the project, the reference solution implemented in the Manche disposal facility constitutes the basis for the memory system to be implemented for the Cigéo disposal project.

2.4 Low level, long-lived waste disposal project

Andra is drawing on the "General safety orientations report concerning the search for a site for disposal of long-lived waste of low specific activity" published by ASN in May 2008. It thus stipulates that:

(i) after closure of the disposal facility, protection of the health of individuals and the environment should not depend on oversight and institutional controls, which cannot be maintained with any degree of certainty beyond a limited period;

(ii) with regard to the oversight phase, the designer must consider the means of ensuring this oversight as of the design of the disposal facility.

The concepts and plans for the period following closure of the LLW-LL project are closely linked to the concepts developed, to the site(s) chosen for the disposal facilities and to the nature of the waste disposed of. Steps shall be taken with respect to oversight following closure of the disposal facility. They shall be studied and clarified as the design studies progress. They shall be based on all the operating experience feedback from the licensee, Andra, concerning the other centres.

Appendix 6: Inter-governmental agreements concluded in France for management of spent fuel or radioactive waste (agreements in force, given in chronological order)

1 AGREEMENTS IN FORCE, LISTED IN CHRONOLOGICAL ORDER

1 – Switzerland:
Exchange of letters constituting the agreement between France and Switzerland with regard to the COGEMA reprocessing contract, signed on 11th July 1978.

2 – Netherlands:
a) Agreement in the form of an exchange of letters between the Government of the French Republic and the Government of the Kingdom of the Netherlands, concerning the reprocessing in France of spent fuel elements, signed in Paris on 29th May 1979.
b) Modifying agreement dated 9th February 2009, published by decree 2010-1167 of 30th September 2010

3 – Sweden:
Exchange of letters constituting the agreement between France and Sweden regarding the reprocessing contracts, signed on 10th July 1979. Additional exchange of letters constituting the agreement between France and Sweden, signed on 10th July 1979.

4 – Spain:
Exchange of memoranda constituting the agreement between France and Spain on radioactive waste from spent fuels produced by the Vandellos I NPP, signed on 27th January 1989.

5 – Japan:
Cooperation agreement between the Government of the French Republic and the Government of Japan for the use of nuclear energy for peaceful purposes, signed in Tokyo on 26th February 1972. Protocol modifying this cooperation agreement (set of three annexes, a report and an exchange of letters), signed in Paris on 9th April 1990.

6 – Australia:
Arrangement between the Government of the French Republic and the Government of Australia concerning the implementation of a reprocessing contract concluded between COGEMA and the Australian Nuclear Science and Technology Organisation (ANSTO), in the form of an exchange of letters, signed in Paris on 27th August 1999.

7 – Italy:
Agreement between the Government of the French Republic and the Government of the Italian Republic concerning the reprocessing of 235 tons of Italian spent fuel, signed in Lucca on 24th November 2006, published by decree 2007-742 of 7th May 2007

8 – Germany:
Agreement in the form of an exchange of letters between the Government of the French Republic and the Government of the Federal Republic of Germany concerning the transport from the French Republic to the Federal Republic of Germany of packages of radioactive waste from reprocessing of spent fuel, signed in Paris on 20th and 28th October 2008, published by decree 2008-1369 of 19th December 2008

9 – Monaco:
Agreement between the Government of the French Republic and the Principality of Monaco concerning the entry into French territory of radioactive waste from Monaco, signed in Paris on 9th November 2010.

2 AGREEMENT CURRENTLY ENTERING INTO FORCE

1 – Netherlands:

Agreement between the Government of the French Republic and the Government of the Kingdom of the Netherlands, concerning the reprocessing in France of Dutch spent fuel elements, signed in The Hague on 20th April 2012.

Appendices

Contents of appendices

Appendix 1: Studies on preservation of memory

The memory project launched by Andra in 2010 comprises on the one hand work designed to continue to create and improve records about the facilities and, on the other, scientific studies concerning two fields: materials ageing and human and social sciences (HSS).

With regard to creating and improving a memory of the centres, the following work has been started:

- the pertinence of the memory archive for the Manche disposal facility in the light of the needs of future generations and analysed every ten years by a group of international stakeholders, in order to periodically examine its adequacy and its comprehensiveness. An initial exercise of this type was carried out in 2012 and identified a number of improvements that could be made to the system;
- preparations for recording the memory of the Cigéo geological disposal project are under way: creation of the detailed memory of the Meuse / Haute-Marne underground laboratory and other elements concerning preparations for the creation of Cigéo (from among everything produced since the early 1980s select that which needs to be kept as to substantiate the decision to create Cigéo);
- the "technical" utility of the memory must be better explained, on the one hand to specify the benefits of this record for long-term safety and on the other to clarify the reversibility requirement;
- around these sites, Andra proposes setting up think tanks to interest the local populations in this problem, but also to collect their ideas about how they could locally assimilate it;
- collaborations with a number of French and international artists are organised, in various artistic fields, in order to obtain their vision of the problem of recording the memory of the repositories through their art;
- Andra takes part in international work on memory as part of the NEA/RWMC/RK&M working group (benchmark practices in various participating countries, joint definitions and bibliography, and drafting of recommendations);
- the creation of spaces dedicated to this memory is envisaged (in Andra's public visitor centres, study of the creation of a historical archive centre with delegation to the Archives de France).

Scientific studies into the ageing of materials consisted in testing the permanent ink/paper combination by means of standardised tests. Durability studies on other media for the longer term are currently being defined. They will concern non-paper media for writing and engraving, in particular studies of surface markers to be installed on the cover over the centres and the production of sapphire disks as demonstrators for a memory medium, the longevity of which could be up to a million years.

With regard to HSS, a group of laboratories was created to study perception of long time scales. For the other subjects linked to HSS (archives, linguistics, museography, archaeology of techniques and landscapes, etc.), a three-stage approach is planned: a succinct bibliography designed to show whether works already exist and are sufficient, otherwise a detailed bibliography produced with universities in order to determine any research that is to be incorporated into the scientific programme. The work will in particular concern continuity, temporality and vestiges, as well as the social dimension of the problem.

Continuity will in particular be studied through:

- languages and symbols, in order to determine for what reasonable time current or dead languages can be known and what the communication solutions could be once these languages cease to be known;
- institutional conservation of written works, sounds, images, objects, etc. by specialised French and international organisations, to analyse the preventive measures taken to limit deterioration over time and encourage assimilation and transmission by future generations;
- long-term digital archival, more specifically by organising an intelligence watch in this field, which is beginning to become organised and which, within the next few decades, could open up new prospects for the long term.

Temporality and vestiges will more specifically be studied through:

- the archaeology of techniques and landscapes, incorporating man-made changes and geodynamic changes, as well as the possibilities of memory resulting from the permanence of infrastructures created by Man;
- the memory of "legacy" repositories not managed by Andra, which exist in various places in France (uranium mines, nuclear tests, etc.).

The social dimension will in particular be studied through:

- the perception by the public of long time scales (several thousand years and more), within the framework of the grouping of human and social sciences laboratories;
- the three possible directions of social change in science, technology, humanity, etc. (regression, stagnation, progression);
- the integration of preserving the memory of repositories into teaching programmes on nuclear energy, heritage and memory;
- transmission of memory between generations via internet social networks to provide global information about the memory and records of the repositories.

The memory project is marked by the milestones of the Cigéo disposal facility project, first of all the public debate in 2013, for which Andra will be required to provide the elements needed for a debate involving the stakeholders in the broad sense, and then the creation authorisation application in 2015. It will continue in parallel with the development of the repository and its gradual closure.

Appendix 2: Summary of achievements and research in foreign countries

1 SUMMARY OF ACHIEVEMENTS ABROAD

This summary presents achievements abroad concerning the management of radioactive materials and waste (countries concerned: Belgium, Canada, China, Finland, Germany, Japan, Netherlands, Spain, Sweden, Switzerland, United Kingdom and the United States). The notion of "achievement" is considered relatively broadly, including not only the drafting of the legal framework and the definition of a classification of radioactive waste, but also the development of management programmes.

1.1 Drafting of a legal framework

Radioactive waste management plans (similar to the PNGMDR to varying extents) sometimes exist abroad, but with objectives that vary significantly from one country to another. Furthermore, some of these plans are not made public.

In Belgium, the Ondraf published a national high-level waste plan in 2011, which establishes various avenues for long-term management, analyses their environmental impact and submits them to the public for their opinion.

The United States announced its intention to adopt a new approach, following the 2009 decision to abandon the envisaged disposal of high-level waste and spent fuels at Yucca Mountain. The Blue Ribbon Commission set up to review spent fuel and HLW waste management strategy is proposing plans that will constitute the foundations of future Government actions.

In 2006, the United Kingdom published a White Paper entitled "Managing Radioactive Waste Safely - proposals for developing a policy for managing solid radioactive waste in the UK", which announces a waste management plan and organisation.

For several years now, Spain has been periodically publishing a General Radioactive Waste Plan, which is scheduled to be revised on a regular basis. It mainly stipulates overall institutional directives.

With regard to inventory, practices vary widely, particularly concerning scope (the French specificity concerning VLLW, the inclusion of mining waste in the United States), exhaustiveness and the level of detail (less detailed in Germany than in France), distribution to the public (the inventory is not public in Spain; in Japan the producers are free to make their own inventories public or not), the rate of updating and the coverage of waste referred to as "engaged" given the current rate of production (until 2080 for Germany, but in the United States for instance, engaged waste is not included).

Despite the work done by IAEA (which provides a database common to all countries, but with a relatively global approach involving very broad categories), comparisons remain difficult, in particular because the units of reference (volume, weight, etc.) used to measure the quantities of radioactive waste, differ from one country to another.

As in France with Andra, a public organisation is responsible for implementing the management of radioactive materials and waste in Belgium (ONDRAF-NIRAS) and in Spain (ENRESA). There is a public organisation in the Netherlands, COVRA, but it is not really comparable, be it in terms of scope of waste covered, or in terms of activities. However, the waste producers (especially the private ones) are most often directly responsible for practical implementation of waste management. They then create a cooperative to manage certain waste, jointly with the public producers: Canada (NWMO-SGDN), Finland (Posiva Oy, for spent fuel only), Sweden (SKB), Switzerland (CEDRA-NAGRA, which does not manage storage). Sometimes, there is no centralised organisation, notably in Japan, where each type of waste corresponds roughly to its own management route and organisation. It should be noted that these organisations are far from always being the "owners" of the waste they have to manage: in Canada, the producer remains responsible, even after closure of the disposal facility; in the United States, the State is responsible for civil waste as of the transport phase (followed by the disposal after burial and disposal after site closure phases).

The list of organisations in charge of radioactive waste is given in the following table:

With regard to the financing of radioactive waste management, the polluter-pays principle would seem to be universally applied for management of radioactive waste facilities, but not for waste management research.

It should be noted that all the countries mentioned here (except for China, which is currently at the joining stage) are members of the IAEA Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management, which entered into force on 18th June 2001. These countries meet in Vienna every three years under the auspices of IAEA, to present their national reports describing the implementation of their obligations and any developments. The last meeting of this type was held in May 2012.

It should be underlined that European directive 2011/70/Euratom of 19th July 2011 concerning the responsible and safe management of spent fuel and radioactive waste requires the establishment of national programmes specifying how the member States implement their national spent fuel and radioactive waste management policies. The content of the national programmes, set in article 12 of the directive, more specifically comprises an inventory of spent fuels and radioactive waste, the general objectives to be reached, the financing mechanisms and an estimate of the cost of the programme. This directive will thus harmonise the European framework for the management of spent fuel and more specifically the establishment every three years of the national programmes, the first version of which must be transmitted to the European Commission in mid-2015.

1.2 Waste classification

1.2.1 The different types of classifications

There are two main approaches to defining the classification of radioactive waste: one approach based on the waste management route and one on the waste production route (this latter approach being in part inherited from the historical construction of radiation protection, generally built around the individual production routes).

Within the first approach (by management route), the classification abroad, in the same way as in France, often combines the parameters of the activity and lifetime of the radionuclides making up the waste (for example Belgium, Spain).

However, the waste classification is sometimes based solely on the activity level. For example, in Canada, there are only two main categories (LLW/ILW and HLW + spent fuels), except for the specific management of mining waste. In the Netherlands, the classification comprises a larger number of categories, but there is no distinction between short-lived and long-lived waste; consequently, there is no surface disposal project.

Other classifications sometimes exist (leading to categories that are qualitatively comparable but which have quantitatively different thresholds): Germany for example has based its classification primarily on the exothermic nature of the waste.

In countries which adopted the second approach (by production route), the classification is more complex, with routes specific to certain types of waste and combining activity and lifetime: United States, Japan and Sweden (in this last country the two types of approach in fact co-exist). Finally, a category is sometimes added for waste from hospitals, universities, etc., for example in Finland.

In addition, certain categories correspond to specifically national characteristics: Belgium (processing of 50% of the radium sources used worldwide) and Canada (large-scale uranium mining).

Finally, the absence of clearance thresholds in France (for waste which contains or which is liable to contain only very small quantities of radioactive elements) is specific to this country. Such thresholds exist in the other countries studied, but vary considerably both in terms of the threshold itself and the scope of waste considered; the VLLW category therefore rarely exists in its own right and does not correspond to the same waste as in France.

1.2.2 Waste classification adopted by IAEA

In late 2009, IAEA published a wide-ranging revision of the radioactive wastes classification which dated from 1994 (IAEA, 2009). It is used by the member countries for an international presentation of their radioactive waste management and their inventories, such as for example in IAEA's NEWMDB database; the European Union refers to it in the 19th July 2011 directive on the responsible and safe management of waste and spent fuels.

This revision was felt to be necessary because the classification system previously defined by IAEA was not completely exhaustive: it did not cover all types of radioactive waste, nor did it provide a direct link with the disposal and management options for all types of radioactive wastes. These aspects of the former classification proved to be obstacles to its utilisation and its application.

The new 2009 classification system introduces a new category of waste - VLLW (Very Low Level Waste) and uses the LLW (Low Level Waste), ILW (Intermediate Level Waste) and HLW (High

Level Waste) classes. These classes take account of both the level of radioactivity and the half-life of the radionuclides contained in the waste.

In it, the wastes are classified according to the degree of confinement and isolation required to guarantee long-term safety, based on their nature and the risk they represent. This waste classification allows an incremental approach to obtaining the required level of safety, because it is based as much on practices as on the characteristics of the sources, the levels of exposure to which they lead and their occurrence.

2 MANAGEMENT ROUTES THAT EXIST OR ARE UNDER CONSTRUCTION

2.1 Choice of the type of fuel cycle

The decision to reprocess spent fuels was taken in various countries in the 1950s for military purposes and at the end of the 1970s for civil uses. A certain number of countries today have facilities:

- for the complete processing of fuels, as in France, the United Kingdom and Japan (for which industrial start-up has however not yet been declared and which could be postponed indefinitely);
- for reprocessing of fission products in the United States as part of the clean-out of former sites such as Hanford, or for separation in Russia to retrieve reusable materials from spent fuels;
- for research as in China, which also opted for a closed fuel cycle, but which is developing test facility projects, notably with the help of France, and in India which is building a pilot plant for the vitrification of fission products.

Several other countries which do not have dedicated facilities on their own territory had or indeed still have all or part of their spent fuels reprocessed in plants abroad, mainly in the United Kingdom and France, this particularly being the case with Germany, the Netherlands, Switzerland and, to a far lesser extent, Spain. Some of the Eastern European nations also do the same in Russia. Some of these countries have however decided to put an end to reprocessing abroad at some time in the near or not so near future: Germany and Switzerland in particular have made a legislative commitment to this and Belgium has for the time being suspended its waste reprocessing contract with the La Hague plant.

For the time being, South Korea has not made a final decision concerning the reprocessing of its light water reactor fuels, which are currently stored on the production sites.

The other option currently being used is to directly manage the spent fuels without any separation or reprocessing phase. This is used in Canada, Finland and Sweden. This has also been the case in Spain and the United States since the implementation of non-proliferation arrangements in the 1980s (under President Carter).

2.2 Decommissioning activities

The countries which have operated nuclear power generating, research or fuel cycle facilities for half a century have initiated major decommissioning programmes for the older facilities, along with post-operational clean-out of the sites. In recent decades, the United States has been carrying out a programme concerning 108 sites covering a total surface area of 800,000 ha. In 2005, the United Kingdom created the Nuclear Decommissioning Authority for eventual decommissioning of all existing nuclear facilities. Decommissioning generates a considerable volume of waste, mainly VLLW waste but also LLW/ILW waste. Its management requires rigorous technical planning and available financing.

2.3 Management of LLW/ILW waste and LLW-LL waste

In several countries, surface or sub-surface disposal centres for LLW/ILW waste are already in operation. They were created to accompany the production of energy of nuclear origin: in China, the Beilong and Diwopu disposal centres, in Spain that at El Cabril, in the United States the centres in Barnwell, Richland, Clive and Andrews, in Finland those of Olkiluoto and Loviisa excavated into the granite at a depth of 60-100 m, in Japan that of Rokkasho-Mura, in the United Kingdom that of Drigg opened in 1959 (about 1,000,000 m^3 are emplaced in it in trenches and on platforms), and in Sweden the SFR centre in Forsmark located 50 metres under the Baltic Sea.

Others are under construction or design in specific locations and the projects differ widely, in terms of the type of site chosen, the design of the disposal centre and its depth. These factors ultimately determine the type of waste that can be disposed of (particularly with regard to the lifetime). Thus, in Belgium, the Dessel disposal centre, which was to enter service in 2016, will in fact only accept short-lived LLW/ILW waste. Among the other projects currently under construction or planned, we should mention the geological disposal facilities planned for about 2017 in Kincardine, Canada at a depth of 700m near the Bruce reactor (Ontario) and in Germany, as of 2014, for LLW/ILW in the old Konrad iron mine in Salzgitter, at a depth of about 1000 m.

However, several countries have not completed or defined their LLW/ILW waste disposal project, such as Switzerland and the Netherlands, but also Italy, which has undertaken the disposal of the waste from its shutdown facilities. Others are closing old sites and reconsidering new locations, such as in Germany for the Asse and Morsleben sites or in the United Kingdom for the Dounray site, on which the construction of a storage facility is being envisaged.

More specifically with respect to LLW-LL waste, current management abroad consists primarily in storing it on the production sites. Long-term management routes have yet to be defined. The volumes concerned are particularly high in Belgium: radium waste comes from the reprocessing of 50% of the radium sources used worldwide and is currently stored on the Olen site. In Spain, graphite waste from the gas-cooled reactors is currently being stored on a reactor decommissioning site. Neither is there any formal plan in Switzerland, the United Kingdom, Japan, the United States, Russia and Ukraine, which all possess graphite type waste.

2.4 Management of HLW waste

Most countries are moving towards deep geological disposal, but they are all at widely differing stages in the site selection and centre construction process.

Finland and Sweden have already selected their (first) sites, in Olkiluoto and Osthammar (location of the Forsmark NPP) respectively and are initiating the building permit application phase. The commissioning of these centres is scheduled for between 2020 and 2025. In Finland, excavation of the Onkalo underground laboratory to characterise a granite environment for construction of the repository reached the reference depth of 420 m in the summer of 2010. Several in-situ tests are underway in Onkalo in order to examine various local characteristics of the massif. They include studies of the hydrological properties, retention, the mechanical behaviour of the rock and geochemical transformations. Testing of a confinement and installation materials manufacturing pilot in Onkalo has started. In Sweden, the site was selected in June 2009, following several years of detailed studies and investigations and a large-scale experimentation programme in the Aspö laboratory (near the Oskarshamn site which was not selected). The building permit application for a geological disposal facility for spent fuels was submitted in March 2011. If authorised, the fuels disposal facility will be built at a depth of about 500 m, in granite rock. Its construction should begin in 2015 and continue until the early 2020s.

In the United States, after choosing the Yucca Mountain site in 2002, the US-DOE (OCRWM) submitted a disposal facility building permit application in June 2008. The file was considered to be correct and its examination was accepted by the safety regulatory body (NRC). However, the Obama administration decided that "Yucca Mountain was not a feasible option for the long-term disposal of spent fuels". Since 2009, there has been no funding for the disposal facility preparation phase and examination of the creation authorisation application has been suspended.

The "Blue Ribbon" commission, created in January 2010 to examine all the possible strategic options concerning the management of spent fuels and high-level radioactive waste, submitted its final report in January 2012. It reaffirms that the reference solution must be geological disposal and in particular recommends setting up an organisation in charge of the disposal of radioactive waste and a siting process based on acceptance. The report takes account of the opinions and comments of the public, obtained during meetings in October 2011.

Although the site has not yet been chosen, various milestones for the near or more distant future have been set in certain countries. Japan has launched a vitrified waste disposal site selection process, which should enter service in about 2035; however the process has been blocked in phase 1 since July 2007 owing to the absence of candidates; a new information campaign was initiated in 2009 in preparation for the launch of a new call for candidates system. Germany and China have set targets for initial operation of the geological disposal centre at 2030 and 2040 respectively. After the 2010 expiry of the moratorium concerning the Gorleben disposal facility, underground exploration work on the salt dome has resumed. At the same time, a preliminary safety analysis is being carried out on this formation. The results should be published in 2012. The conformity of Gorleben with the most recent international safety standards will be examined by an international group of experts in 2013.

Other countries have chosen to focus on research into geological disposal and in particular to postpone the site selection process. For example, no date has been set in Belgium or Canada (in both countries a gradual process involving the stakeholders has been set up). In the United Kingdom, the Government asked the NDA (Nuclear Decommissioning Authority) to study the possibility of speeding up the geological disposal facility construction programme, to allow operation of this facility as of 2029 (instead of 2040 as initially planned). Similarly, the Netherlands have built a long-term (about a century) storage facility, during which time geological disposal is to be studied. In Spain, the construction of a disposal facility for spent fuel, high-level waste and intermediate level waste which cannot be disposed of in El Cabril, is scheduled for start-up in about 2050. The site of Villar de Cañas (situated in Cuenca province about 130 km south-east of Madrid) was however chosen in 2011 to house the future spent fuel and high-level waste storage centre. The radioactive waste management policy should be clarified in the forthcoming seventh general radioactive waste plan.

3 RESEARCH TO SUPPORT GEOLOGICAL DISPOSAL

In most countries, the reference solution for managing high-level waste and intermediate level, long-lived waste, is deep geological disposal. In the recent 2011 directive, the European Council reaffirmed this: "deep geological disposal is currently the safest and most sustainable solution as the final step in the management of high level waste and spent fuel considered to be waste". The host rock chosen varies according to its confinement properties and the geological possibilities of the countries concerned.

However, no country has yet issued a formal authorisation for disposal of this waste, including spent fuels, except for the United States with regard to waste of military origin. Most countries are experiencing significant delays in the development of their disposal programmes, owing to attempts to identify sites on primarily scientific and technical bases, without sufficient local consultation. Those who learned the lessons of this failure and who have restarted the process from the beginning, with prior debate and consultation, are now the furthest advanced.

In certain cases, scientific and technical feasibility can today be considered as proven and those few countries which have reached the most advanced stage are now in the final site qualification and concepts and engineering optimisation phase.

3.1 The organisation of research

With regard to research programmes concerning radioactive waste for which there is currently no industrial route, the most common option is to entrust oversight to the organisation responsible for management, whether private or public: SKB in Sweden, POSIVA in Finland, ENRESA in Spain or ONDRAF in Belgium.

This configuration nonetheless implies specific technical support, similar to that which Andra received from research organisations such as CEA: POSIVA with VTT, NAGRA with PSI, ENRESA with CIEMAT, ONDRAF with CEN-SCK.

Nonetheless, for historical reasons, R&D may sometimes be run by another organisation, which involves the future waste management operator and other research organisations.

A typical case is that of Germany, where there is considerable involvement by GRS (a research organisation which reports to the waste manager BfS) and the BGR, a German public research institute specialising in earth sciences and natural resources, specifically for geological disposal of exothermic waste. DBE (German company for the construction and operation of waste disposal facilities) has an exclusive contract with BfS for the construction, operation and monitoring of disposal sites.

Another special case is that of Japan, although the situation has become simpler since the merging of the two public research organisations, JNC and JAERI, into JAEA. In addition to JAEA is CRIEPI, financed by the electrical utilities and RWMC, financed by METI.

3.2 The underground laboratory – a possible precursor to the disposal project

For R&D into geological disposal (SF or HLW and ILW-LL waste), the various configurations of the waste management organisation in the various countries considered leads to considerable differences in the status of the underground research laboratory, whether in terms of ownership or objective (methodology[1] or qualification of the site and the host rock).

In Sweden, the Hard Rock Laboratory at Äspö is the property of SKB (methodology and qualification of the granite). Since 1995, it has been carrying out research on its KBS3 concept (vertical vaults and copper container) under three-yearly R&D programmes approved by the Government. Since 2000, proof of concept demonstrators have been operational. Their aim is to acquire expertise in the construction and operation of a deep geological repository, for which authorisation was requested in 2011 concerning the Forsmark site in Östhammar.

In Finland, POSIVA is excavating a qualification laboratory in the granite at Onkalo on the actual site of the future disposal facility. Excavation reached its nominal depth of 455 m in February 2012. The research covers subjects such as geological surveys, instrumented drilling, characterisation niches and mechanical studies of the crystalline massif.

In Belgium, the objective of the Hades research laboratory, situated at a depth of 230 m, is methodological and it is used for qualification of the Boom clay. It is now managed by an IEG of ONDRAF and CEN/SCK, the Belgian counterpart of CEA. This laboratory is demonstrating the possibility of building a geological repository consisting of a network of drifts, with limited disturbances within the host clay formation.

Switzerland with two laboratories of widely differing status:

- GTS (GRIMSEL Test Site), a granite environment methodology laboratory made available to NAGRA using drifts belonging to the electrical utilities; current research concerns the instrumentation and monitoring of the structures. However the Spanish proof of concept demonstrator for spent fuel disposal in drifts, called FEBEX and set up by ENRESA in 1997, is still active.
- Mont Terri, an international consortium initiated in 1996 by NAGRA and Andra and now run by a Swiss federal authority. The laboratory's objectives are methodological. It enables Nagra to qualify the clay in Opalinus (potential host rock).

At Mont Terri, about a hundred experiments have been carried out on different scales since the beginning of the research programme in 1996 and more than about forty were still on-going in 2012. In a clayey material that had hitherto been little studied, rock characterisation methods were established. Thus, with regard to safety, the diffusion of radionuclides in clay was measured and the water contained in the rock was collected. Andra's close involvement in the Mont Terri projects and experiments enabled it to prepare for the experiments in the Meuse/Haute-Marne underground laboratory.

In Japan, JAEA is building two underground laboratories, with purely methodological objectives. In the Mizunami laboratory (crystalline rock), a depth of 460 m, out of the planned 1,000 m, was reached in 2011. Studies concerning hydrology and rock mechanics are on-going. In the Horonobe laboratory (sedimentary rock), hydrological tests and hydrochemical measurements are continuing. A depth of 250 m, out of the planned 500 m, was reached in 2011.

[1] In other words a laboratory designed to develop "in situ" characterisation techniques, but for which its status and geological environment exclude it from a possible geological siting sector.

In Germany, following the experiments which took place in the 1990s in the former salt mine at Asse (for which the initial work dates back to the early 1970s), the Gorleben salt dome intended for disposal of high level radioactive waste is currently the subject of reconnaissance and survey work. This phase should last for seven years, according to the German federal office of radiological protection (BfS) which is responsible for the site.

In the United States, after about twenty years of research and characterisation work performed on the Yucca Mountain site in the State of Nevada, the DOE in June 2008 submitted an authorisation application for disposal of spent fuels in volcanic rock at Yucca Mountain. As mentioned earlier, this approach was queried by the State, which undertook to redefine the disposal strategy for radioactive waste and spent fuels.

3.3 Coordinated research in Europe

Technological research and development work developed under the EU's R&D Framework Programme (FP) focuses on:

- the management and safety of geological disposal of High and Intermediate Level, Long lived waste (HLW/ILW-LL);
- the European dimension of its management and disposal;
- the development of processes enabling their quantity and harmfulness to be reduced (e.g.: separation, transmutation, etc.).

3.3.1 Projects in progress

The following table presents the various projects and research programmes in progress within the framework of Euratom, the scope of which concerns developments in radioactive waste management.

Project	Subject	Period	Leader
ReCoSy	Study of Redox phenomena and their influence on the retention and transport of radionuclides	*7th FP* 2008-2012	FZK (D)
CARBO-WASTE	Reprocessing and disposal of irradiated graphite and other carbon waste	*7th FP* 2008-2012	FZH (D)
PETRUS II	Coordination of teaching and training at the European level	*7th FP* 2009-2011	INPL (F)
MODERN	Development and implementation of disposal facility control and monitoring techniques	*7th FP* 2009-2013	Andra (F)
FORGE	Assessment of the impact of gases in radioactive waste disposal facilities	*7th FP* 2009-2013	NERC (UK)
IGDTP	Technological Platform (following on from CARD) aiming to coordinate resources and actions in the field of geological disposal facilities	*7th FP* 2009-2012	SKB (S)
CATCLAY	Study of cation migration processes in clayey rocks	*7th FP* 2010-2014	CEA (F)
PEBS	Long-term performance of engineered barrier systems	*7th FP* 2010-2014	BGR (D)
SKIN	Very slow kinetic process in fluid-rock interactions	*7th FP* 2011 - 2015	EM-Nantes (F)
LUCOEX	Experiments in 3 underground laboratories to test and confirm the conceptual choices made.	*7th FP* 2011 - 2015	SKB (S)
FIRST Nuclides	Acquisition of data on the IRF (labile fraction of the source term) for high burnup fraction UOX fuels, in particular to reduce the uncertainties associated with certain radionuclides of interest (I^{129}, Se^{79}, Cs^{135}, C^{14})	*7th FP* *2012-2015*	KIT (D)
IGDTP-2	Technological Platform aiming to coordinate resources and actions in the field of geological disposal facility projects	*7th FP* 2012-2015	Andra (F)
DOPAS	Full scale demonstration of the construction and performance of seal structures	*7th FP* 2012-2015	Posiva (Fin)
OFSeSa	Novel and Reliable Optical Fibre Sensor Systems for Future Security and Safety Applications	*COST Action* [2] *2010-2014*	36 European Member States

Projects and research in progress concerning radioactive waste management within the Euratom framework

[2] http://www.enseignementsup-recherche.gouv.fr/cid55959/le-programme-europeen-cost.html

3.3.2 IGD-TP - Implementing Geological Disposal of Radioactive Waste Technology Platform

The IGD-TP technology platform was set up in 2009 with the aim of better targeting research, development and demonstration (RD&D) programmes and ensuring improved research between Member States. The IGD-TP platform is therefore run by the organisations responsible for geological disposal projects in the European countries, but also involves research organisations, design offices, technical support organisations for the safety regulators and all players interested or involved in research programmes. Its role is to boost confidence in the safety and implementation of radioactive waste disposal solutions in deep geological formations. IGD-TP is not only of use for the construction of the first facilities, but also for waste management programmes with tight schedules. Most of the nuclearised countries have developed radioactive waste management programmes, but their progress and their implementation schedules are different.

According to the IGD-TP, Europe will in 2025 have its first geological disposal facilities for spent fuels and highly radioactive long-lived waste, offering safe, long-term management.

The engagement by the platform members consists in:

- reinforcing confidence in the safety of geological disposal solutions among Europe's citizens and decision-makers,
- encouraging the drafting of waste management programmes which include geological disposal as the accepted reference option to ensure the long-term management of high level and intermediate level, long-lived waste,
- facilitating access to expertise and technology to maintain skills in the field of geological disposal, for the benefit of all Member States.

In 2011, IGD-TP published a programme called the "Strategic Research Agenda", which defines the research (RD&D) priorities, with a view to obtaining authorisation for construction of the disposal facilities. Implementation of this programme makes provision for funding included in the 7th Framework Programme for technological R&D – 7th FP.

The main subjects of the IGD-TP concern:

- the safety study;
- packaging and behaviour of waste;
- technical feasibility and long-term performance of disposal facility components;
- disposal facility development strategy;
- the safety of disposal facility construction and operation;
- measurements and monitoring;
- governance and involvement by the stakeholders.

Several cross-cutting subjects are dealt with concerning dialogue with the safety regulator, individual skills, knowledge management and subjects relating to communication and information. In addition, dozens of questions will be studied, ranging from disposal facility oversight to performance assessment methods. All of the subjects selected will be combined in programmes, the objectives of which will be attached to "'Horizon 2020" established jointly with the 8th European Framework Programme.

3.4 OECD-NEA

The role of the Nuclear Energy Agency (NEA) is not to run research programmes but rather to bring together players from the various countries to deal with subjects that need to be shared between countries.

The Radioactive Waste Management Committee (RWMC) assists the member countries with management of radioactive substances and waste, more specifically with regard to developing strategies to guarantee the safe, sustainable and generally acceptable management of all types of radioactive waste, in particular long-lived waste and spent fuel, along with the decommissioning of end-of-life nuclear facilities.

The RWMC's main tasks are:

- to create a forum for the exchange of information and experience on waste management policies and practices in the NEA member countries;
- to develop a common understanding of the fundamental questions involved and to promote the adoption of common philosophies based on the various possible waste management strategies and their alternatives;
- to monitor changes in the technical and scientific state of the art in the management of radioactive materials and waste;
- to contribute to the dissemination of information in this field through the organisation of meetings of specialists and the publication of technical reports and joint opinions summarising the results of joint activities on behalf of the international scientific community, the competent national authorities and other audiences interested by the field;
- to provide a framework for on-demand performance of an international peer review of the activities of a country in the field of radioactive waste management, such as R&D programmes, safety assessments, specific regulations, etc.

Appendix 3: Analysis of the compatibility between storage capacity and anticipated radioactive waste volumes

The storage capacity available for packaged waste is located on the production sites (mainly La Hague, Marcoule and Cadarache for HLW and ILW-LL waste). Each storage facility generally accepts one or more waste families. Certain capacity can be shared between the HLW and ILW-LL, ILW-LL and LLW-LL or even ILW-LL and LLW/ILW-SL routes.

Storage of packages of vitrified HLW/ILW-LL waste in La Hague

The standard packages of vitrified waste (CSD-V and CSD-B packages) produced in the R7 and T7 units of the UP2-800 and UP3 spent fuel reprocessing plants at La Hague, are placed in the warehouses adjoining these facilities and then in the Glass storage Extension – South-East (E-EV-SE), once their thermal power drops below 2,000 watts. The R7 and T7 units entered service in 1989 and 1992 respectively, for an anticipated operating lifetime of 50 years. The E-EV-SE storage facility has been operational since 1996, for an anticipated operating lifetime of 70 years.

The three storage facilities: R7, T7 and E-EV-SE, have a combined capacity of 12,420 packages, which will become saturated in about 2013. In 2006, AREVA undertook the study and construction of an extension to the E-EV-SE (called E-EV-LH) for which service entry is scheduled in 2013 and which will be able to store about 8,420 additional packages.

In late 2010, 10,943 CSD-V and CSD-B packages were stored in the three facilities, including 640 CSD-V packages stored in the E-EV-SE, pending shipment to the facilities of AREVA's foreign customers.

In 2015, the combined production of French vitrified waste will reach a volume of 2,560 m^3. The annual production of vitrified waste packages will be about 800 packages (140 m^3) until 2027 and will rise to 1,180 packages (210 m^3) by about 2030, with the start of dilution reprocessing in UOX and URE fuels of the 2,900 tHM[3] of MOX fuels which will have been accumulated by that date.

Other similar capacity will be required as of 2017 and a new extension of the E-EV-SE is in particular being envisaged (see chapter 3.3.3.2). Production will last until vitrification of the rinsing effluents which will be generated after final closure of the UP2-800 and UP3 units, planned for 2040.

Storage of compacted structural waste at La Hague

Since 2002, fuel assembly structural waste (hulls and end-pieces) from the R1 and T1 shearing units in the UP2-800 and UP3 plants have been compacted with metal technological waste in the Hulls and End-pieces Compaction Facility (ACC) which produces CSD-C standard compacted waste packages (ILW-LL route). Plans are for the production of CSD-C packages to continue beyond 2040 in support of decommissioning of the UP2-800 and UP3 plants.

On the La Hague site, the CSD-C packages are placed in the compacted hulls storage facility (ECC), which has a capacity of 20,800 packages and which entered service in 2002 at the same

[3] Ton heavy metal (tHM): this is the quantity, expressed in metric tons, of the uranium and plutonium contained in the fuel before burnup.

time as the ACC, with an anticipated operating lifetime of 50 years. The packages produced in accordance with contracts with third-party countries are also stored in the ECC.

In 2015, the cumulative production of packages of the French share of compacted structural and technological waste will reach a volume of about 2,300 m^3 and in 2020 a volume of about 3,100 m^3. The ECC has a capacity of 20,800 packages, or about 3,800 m^3 which, given this time-frame, will be sufficient to take both the French and foreign shares of the CSD-C packages. Its design is modular, with land in reserve which would allow construction of up to six modules equivalent to the existing module, if necessary. An extension could be necessary during the period 2020-2025. The extension of the ECC facility will need to be studied in the light of the volumes of CSD-C produced by reprocessing of UOX, MOX and URE, as well as the date of disposal of the packages.

Storage of packages of alpha contaminated sludges and metal and organic technological waste in La Hague

The STE3 effluent treatment station has since 1989 been treating liquid effluents from the La Hague plants. Sludges are encapsulated in bitumen and placed in stainless steel drums of 238 litres. As at the end of 2010, 11,278 packages of this type had been produced (or about 2,500 m^3). They are stored in halls with a capacity of 20,000 packages (about 4,500 m^3) in building S of the STE3 station, which entered service in 1987 and is expected to operate until 2040.

Until 2020, the STE3 station will be used to package the first part of the rinsing effluents resulting from decommissioning of the UP2-400 plant. The STE3 station has also started the bituminisation of the sludges produced from 1966 to 1991 by the STE2 station in the UP2-400 plant, but production was banned in 2008 and AREVA is studying alternative packaging solutions.

AREVA is also examining the definition of a packaging method for alpha technological waste (mainly contaminated by plutonium) from the La Hague and MELOX plants. The production of these waste packages is scheduled to continue until decommissioning of the plants, after 2040.

The Removal from Storage and Bitumen Drums Storage Extension units (D/E EB), built in 1995, have the capacity (about 11,200 m^3) to take the above-mentioned packages, for which the estimated volume in 2030 is 9,500 m^3. Depending on the quantities of packages actually produced, a rearrangement of these storage facilities may prove necessary by about 2017 in order to be able to accept all the alpha technological waste.

The alpha sludges and technological waste packaged at La Hague will not therefore generate any requirement for additional storage capacity before 2030. However, the alpha waste packages should probably be stored for longer, to allow a decrease in their production of hydrogen through radiolysis.

Storage of packages of solid operating waste, powder waste and cemented hulls and end-pieces in La Hague

Since 1990, solid waste: gloves, suits, tools, standard operating and maintenance parts in the UP2-800, UP3 plants or from decommissioning of the UP2-400 plant, have been cement-encapsulated in the AD2 unit, originally in asbestos cement containers (CAC) and, since 1994, in fibre-reinforced concrete containers (CBF-C2). These packages are placed in the solid waste storage facilities: EDS/ADT2 and EDS/EDT - EDC (storage of technological waste and storage of hulls); they are linked to the ILW-LL and LLW/ILW-SL routes. The traffic of LLW/ILW-SL packages passing through these facilities occupies a variable volume of a few hundred cubic metres. The EDS/EDC facility is also used to store stainless steel drums containing cement-encapsulated hulls and end-pieces (ILW-LL) produced until 1995. In the future, it will also be used to store ECE stainless steel drums containing cement-encapsulated powder waste: resulting from the purification and filtration of pool water and fines from the dissolution or cladding removal of spent fuels from the gas-cooled reactors which are awaiting packaging in the facilities of the UP2-400 plant undergoing decommissioning.

The total capacity of the solid waste storage facilities is 14,330 m^3. They are scheduled to operate until 2040. This capacity appears adequate to take the anticipated production within this time frame, which will see the volume of ILW-LL packages rise from 9,012 m^3 in 2009, to about 11,100 m^3 in 2030.

Storage of packages of vitrified HLW/ILW-LL waste in Marcoule

The Marcoule vitrification unit (AVM) has a storage facility. Vitrified waste packages: fission products and minor actinides from past production (HLW route) and rinsing effluents from the circuits in the UP1 plant which has been finally shut down (ILW-LL route) are stored in it along with operating technological waste from the AVM (ILW-LL route) to which could be added a very small number of vitrified waste packages (about five packages, or about 1 m^3) produced in the Atalante laboratories.

The capacity[4] of the AVM warehouse (665 m^3) should be sufficient to take all planned production at Marcoule.

A shipment interface to the stored packages disposal centre will need to be developed by CEA. CEA will identify the technical options and conduct an initial analysis of the transport methods, together with Andra, more specifically in order to present storage-transport-disposal scenarios to the public debate on the Cigéo disposal centre project planned for 2013.

The pilot vitrification unit in Marcoule, PIVER, has produced a small quantity of vitrified waste packages (HLW route) with a total volume of 17 m^3 and which is currently stored in building 213, specially outfitted in the Marcoule pilot unit (APM) which entered service in 1969 and for which an operating life extension is currently being examined.

The periodic safety reviews, supplemented by the stress tests, showed the need to look at the lifetime of the storage facilities for this type of waste, on the basis of the filling time-lines for the Cigéo disposal facility project.

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4 The indicated capacity corresponds to the unit volume of 175 litres per container considered in the 2012 edition of the National Inventory.

Storage of packages of bituminised sludges and solid waste on the Marcoule site

Since 1966, the Marcoule liquid effluents treatment station, STEL, has produced packages of sludge encapsulated in bitumen and then placed in 230 litre steel drums. From 1966 to 1996, the drums made of non-alloy steel, were stored in 35 pits in the site's North zone (about 6,000 drums) and then in bunkers numbered from 1 to 13 in the South zone (about 54,000 drums, to which should be added the 2,200 drums produced since 1996 and stored in bunker 14).

Steps are under way to retrieve and repackage these legacy drums. From 2000 to 2006, all the drums in the North zone pits, most of which were classified LLW-LL, were removed, inspected and placed in 380 litre stainless steel over-packs. Retrieval is continuing with the legacy drums from bunkers 1 to 2 in the South zone. At the same time, in response to the requests from ASND, retrieval is in progress for the drums of salting out product (LLW/ILW-SL type encapsulation process drums, mixed with bitumen encapsulation drums in bunkers 1 to 10) which are considered to represent most of the mobilisable source term.

At present the Marcoule STEL is continuing to produce packages of bituminised sludges. Since 1996, packaging has been in 230 litre stainless steel drums. These packages, which correspond to the LLW/ILW-SL and ILW-LL routes, are stored in bunker 14 which entered service in 1994 with a capacity of about 1,200 m^3. Shutdown of the encapsulation unit is scheduled for 2014.

Retrieval and packaging of the following waste is being envisaged for the 2017-2020 time-frame:

- non-magnesium metal structural waste from fuels reprocessed in the UP1 plant and structural waste from the PHENIX fast neutron reactor;
- powder waste, filters, graphite powder from cladding removal from gas-cooled reactor fuels, settling sludges and solid metal and partially organic operating and maintenance waste from the units or from decommissioning, with beta-gamma spectrum;
- magnesium structural waste from gas-cooled reactor fuels.

The sludges produced by the treatment of liquid effluents in the STEL are encapsulated in a cement matrix, which will replace bituminisation in 2015 (STEMA project). The waste packages produced (380 litre drums) most of which will be of the LLW/ILW-SL type, will be packaged in the centre before being shipped to the LLW/ILW waste disposal facility. Any ILW-LL packages will be managed in the same way as the packages resulting from treatment of powder waste.

A multipurpose interim storage facility (EIP) entered service in 2000 for the storage of packages in 380 litre drums (called EIP drums). Its design is modular and it currently consists of two vaults[5]. The operating lifetime as currently planned is 50 years.

The packages at present stored in the EIP are drums of bituminised sludges produced by the STEL before 1996, retrieved from the North zone pits and bunkers 1 and 2 and repackaged in 380 litre drums. They represent a volume[6] of 2,660 m^3 (about 8,000 packages).

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5 Its extension up to 16 vaults could be envisaged, to raise the total capacity to 33,880 m^3.

6 The capacities and volumes are here expressed for a unit volume of 380 litres considered for an EIP drum in the National Inventory. Considering the overall external volume of this package (441 litres) would lead to greater capacities and volumes for the same number of packages.

The continued retrieval of the waste from the bunkers and its packaging in 380 litre drums will lead to saturation of the EIP's current capacity by 2017, with a volume of 4,370 m^3, or 11,500 packages (linked to the ILW-LL and LLW-LL routes). The need to create additional storage capacity, in line with the operations to retrieve these legacy packages, is being examined by CEA.

Storage of high-dose ILW-LL waste packages on the Marcoule site

The retrieval and packaging operations for the legacy and decommissioning waste will generate high-dose ILW-LL waste packages for which there is no storage facility. For the Marcoule site, the volumes of this category of waste produced by decommissioning of the PHENIX reactor (the most activated waste) and by the retrieval of structural waste from fuels reprocessed in the Marcoule pilot unit (APM) are evaluated at about 250 m^3. In order to meet this need, CEA has decided to create the DIADEM facility, which is scheduled to enter service in 2017, subject to its authorisation being granted (see §3.3.3.2) Furthermore, this new facility will be used to store high-dose waste from other CEA sites (Fontenay-aux-Roses, Saclay, Grenoble).

Storage of low-dose ILW-LL waste packages on the Cadarache site

Since 1970, the effluent treatment station (STE) of BNI 37 has been packaging the centre's filtration sludges and evaporation concentrates by cement encapsulation in 225 litre metal containers, themselves placed in 500 litre concrete containers (with or without immobilisation). Occasionally, the concentrates were packaged in 700 litre drums, which were repackaged in 1,100 litre non-alloy steel containers.

The low-dose ILW-LL solid waste from operation or decommissioning, which came mainly from the Saclay, Fontenay-aux-Roses, Cadarache, Valduc and Grenoble sites, was packaged in the solid waste treatment station in BNI 37 (compactable waste) or ICPE 312 (non-compactable waste). The primary waste is packaged by compaction and/or immobilisation in 870 litre metal containers. As of 2013, this type of packaging will be used for alpha spectrum solid decommissioning waste from Marcoule.

The inventory as at the end of 2011 is 5,014 packages of 870 L and 4,309 concrete containers of 500 L. The forecast inventory for 2030 is for about 6,360 packages of 870 L and 4,870 concrete containers of 500 L.

Until 2006, ILW-LL packages were placed for storage in BNI 56, which today is no longer accepting any new ILW-LL packages. In 2006, CEA commissioned the CEDRA radioactive waste storage facility at Cadarache, with two buildings (n° 374 and 375) each of which offers a capacity of 7,572 packages of low-dose ILW-LL waste packages. CEA has initiated transfer of low-dose ILW-LL packages from BNI 56 to CEDRA. As at 31st December 2011, 737 packages had been shipped to CEDRA

As at the end of 2011, 1,369 low dose packages of 870 litres and 47 low dose packages of 500 litres were present in CEDRA. The current storage capacity of 7,572 packages of low dose ILW-LL waste packages will not be enough to manage all the packages to be retrieved from BNI 56 and that will be produced in the CEA centres. The total number of low dose ILW-LL packages estimated between now and 2030 is 11,230. CEA thus envisages raising CEDRA's storage capacity for ILW-LL waste to 11,358 low dose packages, by building tranche 3. It should be noted that the filling ratio of the CEDRA facility depends to a large extent on the BNI 56 retrieval programmes.

Storage of medium dose ILW-LL waste packages on the Cadarache site

Solid operating or decommissioning waste from the various CEA sites and falling into the low dose ILW-LL category, has since 1970 been packaged in BNI 37 by compacting and then immobilisation in a steel container of 500 litres. Until 2006, the packages were placed in the storage pits in BNI 56. CEA has initiated transfer of some of these packages to CEDRA: at the end of 2011, CEA had removed from storage 183 recent packages produced under quality assurance from pit F6 in BNI I56 and transferred them to CEDRA. Retrieval of the recent stainless steel packages from pit F6 in late 2015 is one of CEA's priority safety objectives.

Since it entered service in 2006, the CEDRA facility has comprised building n°376, with a useful capacity of 1,400 packages, for storage in shafts of ILW-LL packages produced or removed from storage in BNI 56.

As at the end of 2011, the total number of medium dose ILW-LL waste packages stored on the Cadarache site stood at 1,173. It will reach 3,264 packages in 2030. The current capacity of CEDRA (1,400 packages) will not be enough to meet the need. CEA envisages increasing this capacity: after the construction of CEDRA tranche 3 by about 2023, it would be raised to 2,800 packages; an additional tranche 4 would be able to raise the total capacity to 4,200 packages.

Storage of other waste packages on the Cadarache site

Packages of radium-bearing lead sulphates (resulting from treatment between 1958 and 1970 of uranothorianite ore), solid waste and filtration sludges and large-sized containers (1,800 or 1,000 litres) and "source blocks" are currently stored in BNI 56. Production is now completed and represents a volume of about 1,275 m^3.

Storage of ILW-LL waste packages on the CEA Valduc site

Packages of sludges and concentrates immobilised in 220 litre metal drums, produced in the past from 1984 to 1995 by the Valduc liquid effluents treatment station, are stored on the Valduc site. This waste is part of the ILW-LL route and will be stored in CEDRA pending the commissioning of the Cigéo disposal facility project. Transfer of these waste packages is in progress.

Processing of recyclable materials produces effluents containing americium, plutonium and uranium which CEA plans to vitrify in about 2020. These ILW-LL packages will be stored on the Valduc site. In 2030 the total volume of packaged waste will reach about twenty m^3.

Storage of EDF activated waste packages on the site of the Bugey NPP

The ICEDA facility, located on the Bugey site, authorised by decree 2010-402 of 23rd April 2010, is designed to take activated waste produced by the decommissioning of the EDF reactors at Creys-Malville, Brennilis, Chooz A, Bugey 1, Saint-Laurent-des-eaux A1 and A2 and Chinon A1, A2 and A3 as well as the internals removed from the NPP reactors in operation (control rods and poison rods). For short transit period (a few months prior to shipment to disposal centres) this facility should also accept metal and graphite waste resulting from the decommissioning of Bugey 1 and intended for the LLW/ILW-SL and LLW-LL routes respectively.

Commissioning of this facility, envisaged for 2015, has been called into question by the 6th January 2012 cancellation of the building permit by the administrative court.

The hypothesis adopted by EDF is packaging in "C1PG" reinforced concrete containers for ILW-LL or LLW/ILW-SL activated waste.

The ICEDA facility would consist of two storage halls with a unit capacity of 1,000 waste packages, or about 2,000 m^3 of waste packages per hall.

Storage of radium-bearing radioactive waste

On its La Rochelle site, Rhodia stores different types of radioactive waste, resulting from the processing of monazite and then, as of 1994, from processing of rare earth concentrates. The site is authorised as an installation classified on environmental protection grounds. Rhodia possesses about 13,700 t of waste, in the following form:

- radium-bearing waste, known as RRA (about 1850 Bq/g alpha and beta activity in 2002): 160 t in la Rochelle, with most of the RRA being stored in Cadarache (5120 t);
- general solid waste, or RSB (about 75 Bq/g): 8400 t in la Rochelle.

This waste, classified as low level, long-lived, is part of the waste inventory intended for Andra's radium-bearing waste disposal project.

RRA waste is stored on the Cadarache site (ICPE 420 and 465) and RSB waste on the Rhodia site at la Rochelle, in a building (BAT. 135).

Cézus on the Jarrie site stores radium-bearing waste resulting from the processing of zirconium. It is located in a dedicated building consisting of six vaults of 1,000 m² each equipped with retention pans, for a storage capacity of 4,500 t. The storage capacity should be able to cover the needs until 2020 (2015 if the waste is insolubilised beforehand).

Storage of radioactive waste from the small producers

For the management of radioactive waste from the small producers, Andra uses the storage capacity on the AREVA/SOCATRI and CEA (BNI 56) facilities in Cadarache and Saclay (BNI 72), or even on other sites for diffuse nuclear waste, most of which falls into the LLW-LL category. In the light of the planned use of these facilities by the licensees for their own requirements, or the scheduled decommissioning of some of them, Andra has opted for its own storage facility.

In order 2012040-0002 of 9th February 2012, Andra was authorised to operate grouping and storage facilities for radioactive waste from the small producers in the VLL waste disposal centre. The storage facility, intended for waste for which the long-term management routes are currently under development, has a capacity of 6,000 m^3 with Andra having collected about 700 m^3 as at mid-2012. This facility entered service in 2012. It is more specifically accepts fire detectors, lightning arresters and other sources containing radioactive substances, waste from the clean-up of polluted soils (Operation Radium) and so on.

Storage of tritiated waste

At present, most tritiated waste for which there is no disposal solution is generated by Defence activities. Most of this waste is stored on the Valduc site. The volume of waste stored in France represents about 4,600 m^3. Some facilities will soon become saturated. New facilities are currently under construction.

Solid tritiated waste from the small producers will be stored in the ITER tritiated waste storage facility, scheduled to enter service in 2024, subject to authorisation being granted.

The tritiated waste storage capacity requirements are presented in part 3.1.

Appendix 4: Research aspects

1 INTRODUCTION, PLAYERS AND THE MAIN MILESTONES CONCERNING RESEARCH CONDUCTED FOR THE PNGMDR

The Act of 28th June 2006 gives the responsibility for research into separation-transmutation to CEA and research into reversible disposal of HLW/ILW-LL waste and storage to Andra. It organises the roles of the various research players in the field of radioactive waste management. At the same time, a certain number of R&D actions are performed by industry (EDF and AREVA), partly under agreements linking them to CEA or Andra. As necessary, all of these organisms draw on the pool of expertise available at the CNRS, which restructured its research in 2011 around a new cross-cutting research programme called, Nuclear: energy, environment, waste, society (NEEDS), the Universities and other organisations, such as the BRGM or INERIS. Finally, one must mention IRSN, where research aims primarily to provide it with a satisfactory level of nuclear safety and radiation protection expertise enabling it to fully play its role of technical support for ASN and ASND.

The PNGMDR, which has been in place since 2006, describes the management solutions developed for radioactive materials and waste and specifies a certain number of research strategy milestones over a period of three years. The National Review Board (CNE2) regularly assesses the research carried out in the field and in its recommendations proposes a number of orientations for the strategy to be implemented. It should be noted that the Board is in favour of a more international approach to a good part of the research performed by Andra, CEA and the CNRS. It particularly appreciated the importance given to this aspect during the hearings.

To ensure that all these programmes are consistent, a Committee for the Monitoring of Research on the Cycle Back-End (COSRAC) was set up and is chaired alternately by the DGRI and the DGEC. COSRAC, a unique forum for debate among all research players, helps with defining a common research strategy in line with the 28th June 2006 Act.

This document presents a summary of research carried out on the subjects covered in the previous PNGMDR and an outlook on research to be conducted in the coming three years. This document is not exhaustive in that certain long-term prospects can be carried out in parallel.

It is worth reviewing the main milestones of the 28th June 2006 Act:

No later than 31st December 2012: "Andra shall submit to the Ministers responsible for energy, research and the environment the dossier supporting the organisation of the public debate" which will be held before the authorisation application for the creation of a deep geological disposal site is submitted.

No later than 31st December 2012: CEA shall submit to the Ministers responsible for energy, research and the environment a dossier reviewing research carried out on the subject of separation- transmutation.

No later than 31st December 2014: "Andra shall submit the creation authorisation application" for a deep geological disposal site.

2 IMPROVING KNOWLEDGE AND WORKING UPSTREAM ON WASTE PACKAGING AND THE BEHAVIOUR OF THE PACKAGES

The producer is responsible for the production of the waste package and must demonstrate its characteristics by producing a data file and an operational model describing the long-term behaviour of this package. At the request of the producers, CEA on the one hand performs a significant share of the R&D necessary for implementation of the processes and for improving the understanding of the characteristics of the packaged waste.

The process to verify the ability of the packages to perform all the functions necessary for disposal is the responsibility of Andra, which has set up long-term behavioural study programmes for the various package families in the disposal environment. In order to carry out this R&D, Andra realised that it would need to set up dedicated structures such as the "Glass/Iron/Clay" Laboratories Grouping to study glass alteration and overpack corrosion and the "Evolution of cement structures" laboratory to study the long-term behaviour of concrete disposal packages. CEA and EDF take part in these two laboratory groupings.

All those involved agree on the need to continue characterisation work on all waste concerned by disposal, in order to:

- clarify the radiological inventories by developing or improving appropriate analytical and experimental resources;
- evaluating the chemical inventories and the source terms of the chemical and gaseous compounds;
- improving the characterisation of the behaviour of the waste in disposal conditions;
- evaluating release kinetics for waste for which the source terms are currently considered to be too conservative or penalising (labile).

2.1 Graphite waste: Management and processing scenarios

The low specific activity of graphite waste leads to the study of a shallow depth disposal facility (less than 200 metres) but its long-lived radionuclides content, as currently estimated, does not enable surface disposal to be envisaged. In particular, the declared chlorine 36 activity in the graphite waste requires the use of a sufficient thickness of clay to limit the flow into the geological environment.

One alternative could be offered by recent developments in graphite waste treatment processes. Some of these processes could make partial decontamination of the graphite possible in order to make its radiological inventory acceptable for shallow depth disposal (disposal under reworked cover, SCR). The radionuclides extracted would then require appropriate packaging and disposal. Provided that decontamination efficiency were sufficient, it would even be possible to envisage gasification of the decontaminated graphite in the form of carbon dioxide. A first series of heat treatment tests were performed by EDF jointly with the Studsvik company. The results obtained at this stage confirm the potential of heat treatment for removing the chlorine 36, tritium and some of the carbon 14 from the graphite.
At the same time, work is under way to clarify the radiological inventories of graphite waste, in particular thanks to improvements in analytical techniques and to additional sampling from the reactor blocks.

Outlook
One major avenue of the R&D programme will be devoted to heat treatment. This will involve on the one hand completing the pilot tests. On the other hand, a certain number of more exploratory tests will be performed, more specifically at CEA, in order to better understand the decontamination levels obtained and verify their validity on various samples.

The management of treatment effluents and packaging of the resulting waste are essential development points, whichever treatment process is envisaged. The carbon residues resulting from the treatment could thus be cemented or compacted to form a matrix compliant with Andra's disposal acceptance specifications. If not, separation of the radionuclides would be necessary, followed by dedicated treatment for each radionuclide (cement encapsulation, precipitation, etc.). These actions are formally laid out in special agreement between Andra, CEA and EDF.

2.2 ILW-LL Waste

2.3 Bituminised sludges

Over and above studying the release of radionuclides, the studies on bituminised sludges concerned the production of dihydrogen by radiolysis, the chemical source term of waste (complexing or aggressive species) and the swelling of waste packages under water.

The producers carried out R&D work leading to tools for calculating the production of radiolysis hydrogen. The challenges linked to the production of gas by radiolysis concern the design of a disposal package which must allow the gas to escape, and the risk of the explosive limits in the vault being exceeded if the ventilation were to stop for whatever reason. The production of dihydrogen by the bituminised sludge packages remains below 10 L/drum/year and recent estimates for the production of dihydrogen for the packages in the effluent treatment stations in La Hague (STE2 and STE3) give rates of less than 3 L/drum/year. The same conclusion applies for at least half the population of bitumens in the liquid effluent treatment station (STEL) in Marcoule (CEA). The work will continue on all the populations of bitumen encapsulated items produced in Marcoule.

Swelling under water is the result of an osmotic process, the origin of which is the behaviour of the bitumen which acts as a semi-permeable membrane. This process could only occur at resaturation of the vault, in other words within a time-frame of between ten and a hundred thousand years. For this time-frame, the main risk identified is damage to the argilite around the disposal vaults. In order to assess the rock damage process resulting from this phenomenon, preliminary modelling of the mechanical consequences of the pressure levels exerted at swelling of the bitumens was carried out by Andra. To fine-tune these models and validate the associated results, experiments must be carried out specifically to acquire pressure/strain curves.

Andra is currently carrying out studies to consolidate its understanding of complex near field phenomenological interactions of bitumen packages. An assessment will be provided in 2013/2014, in order to gain an approximate understanding of the reactivity of the nitrates and the consequences of the plume of salts on the large-scale migration of radionuclides. Pending the results, the long-term safety assessment is carried out on the basis of conservative hypotheses designed to cover the uncertainties associated with these phenomena. With regard to the complexing species, the analysis of the inventories of TBP (tributylphosphate) will make it possible to decide on the pertinence of setting up a special R&D programme lasting about two years.

2.3.1 Technological waste containing organic matter rich in alpha emitters

Technological waste rich in alpha emitters comes from fuel fabrication and processing facilities. The particularity of this waste is that it contains both metal and organic matter. Studies are required to determine to what extent, in a disposal situation, the radiolysis of the organic matter could lead to the production of gas such as dihydrogen and corrosive gases. The radiolysis and then hydrolysis of the organic materials will release complexing species which could complex actinides such as uranium and plutonium. The aim will be to complete the thermodynamic bases of the main expected complexes in order to verify their stability domain in geological conditions.

In the field of radiolysis, R&D work has been performed by CEA and Areva and led to the development of a database and predictive models allowing quantification of the gas source terms of these packages. With regard to determining water-soluble degradation products (PDH), the work already under way will continue in the form of a joint Andra/producers R&D programme. The aim of this programme is to obtain data to quantify the conservative nature of the assessment of these PDH and their complexing capacity.

All of this work does not however prejudge any significant rise in the mobility of actinide complexes within the Callovo-Oxfordian (COX) clays.

At present, CEA packages this type of waste using compaction and cement encapsulation. Areva has a cement encapsulation packaging mode for some of the alpha contaminated technological waste. All of the waste produced cannot however be packaged using this mode. Areva has also studied a treatment-packaging process using compaction. The S5 package has the advantage of reducing the packaging volume for this type of waste. This packaging was the subject of quadripartite discussions (ASN, Andra, IRSN, AREVA). In 2009, AREVA submitted the draft production specification for the S5 package to ASN and the detailed package data file to Andra. In February 2010, following examination of the package file, ASN asked Areva to study other treatment-packaging scenarios "*considering that the S5 package project developed by Areva, the characteristics of which are described in the above-mentioned letter of 20/01/2009, do not offer sufficient guarantees for long-duration storage and for disposal in deep geological formation, more specifically owing to the presence of organic materials"*. Areva thus initiated studies to find other possible treatment-packaging scenarios (including thermal) as requested by ASN in Article 1 of its resolution 2010-DC-0176 of 23rd February 2010 and is continuing R&D into the S5 package, in order to demonstrate that sufficient guarantees could be provided.

Areva transmitted files in April 2010 and early 2012, notably containing the additional R&D results acquired, which describe the corrosion-resistance of the container and the trapping of radiolysis hydrochloric acid gas by its internal carbon steel sheath, along with studies on the nature of the complexing agents. Furthermore, in response to a prescription of the 2010-2012 PNGMDR, AREVA also submitted a report presenting the characteristics of the S5 package, its manufacturing process, the results of the R&D describing the thermal processes and a forecast implementation schedule for all these avenues.

The orientation studies concerning the thermal processes performed in 2010/2011 revealed the absence of any technology that could be directly transposed to technological waste containing organic matter rich in alpha emitters. The R&D concerned the incineration / melting /

vitrification technologies involving plasmas, which most closely match the process specifications. They consist in heating the metal phase by low-frequency induction and then heating the glass by heat transfer at the metal/glass interface. One or more plasma torches are sufficient to ensure combustion of the organic part of the waste.

These technologies are based on major technological innovations (use of a plasma torch in a nuclear environment, deployment of fusion and vitrification operations within the same process, final packaging comprising two separate glass/metal phases in the same container, etc.) and feasibility has yet to be confirmed. They generally imply the ability to manage specific criticality constraints and the use of a very high temperature process associated with a glove box design.

At this stage, the production of a full-scale prototype was felt to be necessary and this was the R&D subject for the period from 2011 to 2018. It should allow inactive qualification of the process. Funding for this R&D has been requested under the investing in the future programme.

2.3.2 Other ILW-LL waste

In the 2005 File concerning the feasibility of geological disposal in a clay layer, the models and data available for ILW-LL waste mainly concerned the corrosion rates of the metal materials in standard compacted waste packages (CSD-C) and the source terms of bituminised sludge packages. The studies carried out since then by CEA, AREVA and EDF on behalf of Andra, confirm these elements, but also provide additional information concerning the following waste:

- metal waste: determination of corrosion rates of aluminium and magnesium alloys;
- polymer waste: evaluation of radiolytic production rates for different gases and different polymers, determination of the nature and quantity of water-soluble degradation products resulting from the radiolysis and hydrolysis of these polymers;
- ILW-LL glasses: proposal of a glass alteration model.

2.4 Spent fuels

A study programme for PWR fuels was carried out in accordance with the PNGMDR requirement which was to produce a less conservative release model than that adopted for the 2005 File, by 2011. The work done on this project can to a large extent be transposed to other fuels[7] with a UO_2 matrix.

A spent fuels matrix alteration model was developed by CEA. It includes radiolysis, geochemistry and electrochemistry. It is applicable to UOX and MOX fuels and should eventually allow coupling with the materials in the environment. It leads to a fuel lifetime similar to that adopted in the 2005 File (from 50,000 to 100,000 years).

Initial experimental results on the dissolution of UO_2 doped with clayey water would also seem to show control of alteration by a silicate phase (uranium silicate). This could lead to a slower alteration mechanism than that shown by current models.

In the rest of these actions, it will be necessary to include the results obtained by the FIRST-Nuclides programme, conducted at a European level and to continue the experimental and modelling work on the behaviour of UOX and MOX fuels, in order to clarify the available data, more specifically with regard to their respective behaviours in disposal conditions.

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[7] Spent fuels from civil reactors (GCR and EL4 heavy water reactor), CEA experimental reactors, land-based or on-board reactors for national defence activities.

2.5 Vitrified waste

The initial and residual dissolution rates of the R7T7 glasses produced at La Hague were assessed throughout a broad range of operating conditions: atmospheric corrosion, alteration in pure or clayey water, alteration in the presence of environmental materials (corrosion products and argilites):

- the atmospheric corrosion of glass (in the presence of water vapour) leads to alteration rates higher than the residual rate in pure water;
- the glass dissolution rates in pure and clayey water were acquired at 30°C, leading to a reduction in glass alteration by comparison with the 50°C values adopted in the 2005 File;
- glass corrosion in clayey water leads to initial rates 5 times higher than those obtained in pure water. The residual rates are also multiplied by a factor of 1 to 5 depending on the magnesium concentration in the vicinity of the glass and the pH.

The effect of magnetite on the increased kinetics of glass alteration was confirmed, even if attenuated in the presence of a diffusion barrier, but the underlying mechanisms could be more complex than those considered in the 2005 File. In addition, the studies confirm very slow development of glass fracturing under mechanical loading in the disposal facility.

At the same time, a mechanistic model of the long-term behaviour of the R7T7 glass, the GRAAL model, is currently under development. This model aims to describe the complete kinetics of glass dissolution as a function of environmental conditions. The current studies aim to expand its scope of application so that this mechanistic description can be integrated into the operational model of vitrified waste package behaviour in a disposal situation.

With regard to "cold"[8] glasses (UMo, PIVER and AVM glasses), a model assuming alteration according to the initial rate of glass dissolution was developed. In the particular case of certain AVM glasses, a model of the same type as that used for the R7T7 glasses was configured, leading to a lifetime estimation that is higher by one to two orders of magnitude.

With regard to the performance and safety calculations needed for the Cigéo creation authorisation application, this work will allow a more precise and robust evaluation of the behaviour of these glass families in a disposal situation. The work to be done in the coming three years should in particular concern:

- glass alteration in atmospheric conditions. This process is at the origin of the measured rate;
- alteration in the water on the site and identification of the effect of magnesium;
- the influence of environmental materials, more specifically of corrosion products;
- interpretation of the results of the underground laboratory experiments.

[8] These are glasses for which the radionuclides content is such that the thermal release is less than that of the R7T7 glasses today produced at La Hague

2.6 Generation IV reactor waste

The aim is to obtain the data necessary to prepare for gradual deployment of generation IV reactors:

- initially supplied with the uranium and plutonium contained in the fuels taken from PWR reactors (more specifically spent MOX fuels);
- then with the implementation of systematic recycling of the uranium and plutonium;
- and, as necessary, the use of transmutation options for certain minor actinides.

It will be necessary to examine the impact of these technology and material management strategy developments on the waste generated. The deployment time-frame for these systems enables innovative R&D to be carried out on these subjects. The following primary research objectives can be mentioned:

- studies designed to limit the generation of waste comprising long-lived elements, as of the design stage;
- the study of alternative treatment and packaging solutions, for example a melting process for metal waste;
- the characterisation and management of waste, in particular secondary waste (cladding, structural elements of fuel assemblies) and reactor operating waste (cold traps, control rods);
- the impact of transmutation options on the design of the disposal facility (for example to assess the impact of a glass thermal load that is significantly reduced over the long term).

3 SUPPORTING DISPOSAL PROJECTS FOR HLW/ILW-LL AND LLW-LL TYPE WASTE, PLUS STORAGE

3.1 Disposal of low level, long-lived waste

The deployment of long-term management routes for LLW-LL type waste requires R&D to clarify the feasibility, acquire the additional data necessary to demonstrate the safety of disposal and draft design requirements for the industrial means that are eventually to be selected. These means concern shallow disposal and the upstream operations such as possible treatment and packaging of the associated residues.

The siting of such a disposal facility will in the coming years imply the search for and characterisation of the site and the production of substantiation data to be provided with a view to creation of the disposal facility.

R&D work on characterising and treating radium-bearing waste aims to improve knowledge of its behaviour in a disposal situation, to reduce physico-chemical disturbances after closure and reduce the overall volumes to be disposed of, thus preserving scarce disposal capacity.

R&D work on characterisation and treatment of graphite waste aims to explore the different processes enabling high decontamination performance to be achieved and the feasibility of total destruction of the decontaminated graphite, along with evaluation of concentrated residue packaging processes (see § 2.1 of this appendix).

For bitumen encapsulation, R&D carried out by CEA aims to confirm the radiological inventories by sampling followed by radiochemical analyses.

3.2 Reversible disposal in deep geological layer for high level, intermediate level long-lived waste, and the Cigéo project

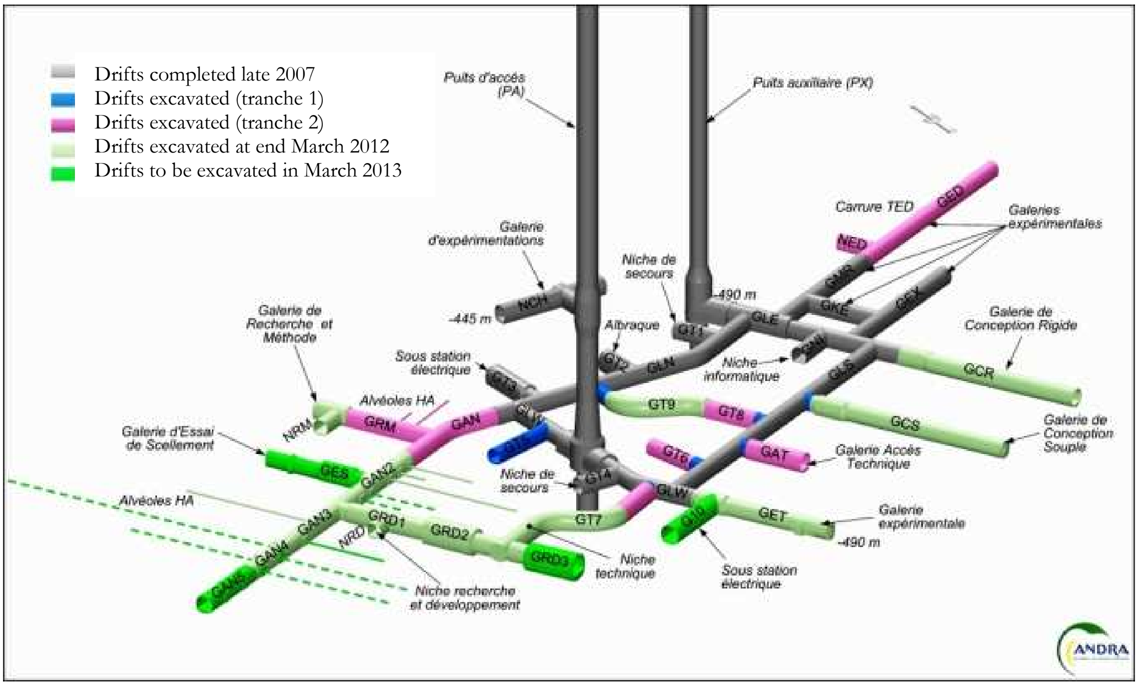
Progress and prospects of the Meuse-Haute-Marne experimental underground laboratory programme

Until 2014, the experimental programme intends to intensify technological testing and the performance of experiments in order to meet the needs of the assessors and to acquire more extensive data for drafting of the creation authorisation application. The work is being organised in three areas:

1. Continue the programme associated with the construction of drifts and HLW vaults;
2. Complete data acquisitions concerning the characteristics of the Callovo-Oxfordian argilites from the geomechanical standpoint (argilite behavioural laws, EDZ[9] transport properties corresponding to a dense 3D network of interconnected fractures), priority for engineering and simulation and transport-retention study programmes (long-duration diffusion experiment);
3. Testing of the drift sealing components, with a view to technological optimisation of the disposal structures, plus performance testing.

To carry out this programme, new drifts were excavated in 2011.

[9] EDZ - Excavation Damaged Zone.



Progress of drift excavation in the underground laboratory as at March 2012

Progress of the experimental programme

Technological tests

Two technological tests on the construction of drifts were performed in order to study the interactions between the rock and the liner according to the time between excavation and application of the liner:

- the construction of the rigid design test drift (GCR) in the direction of the major horizontal stress allowed examination of the conditions for production of a concrete liner poured in-situ about 6 months after excavation. Long-duration monitoring of this drift will provide information on (i) its loading and its behaviour over time, (ii) the development of the initial argilite fracturing after installation of the liner, and in particular the permeability;
- the construction of the GRD2 drift with a thick sprayed concrete liner (45 cm), installed as soon as possible after excavation.

The comparison of the behaviour of these drifts and the flexible design drift (GCS) will make it possible to understand the relationship between deformation of the terrain and the loading of a rigid liner.

In phase 2 of the test programme on the HLW vaults, a 40m vault was excavated, with an instrumented sleeve. The aim is two-fold; (i) to complete the data on the evolution of the argilite/sleeve interface, (ii) to test sleeve examination methods, in particular using optical fibres. The sleeve was made completely watertight, including on the drift wall, in order to see whether the outer ring fills with water and if so at what rate.

Characteristics of the Callovo-Oxfordian argilites

Load measurements on the instrumented tubing in small diameter boreholes continued. After more than a year of testing, the loading tends to become isotropic. A traction test was attempted on one of the tubes to measure the steel/argilite friction coefficient. The friction forces exceeded the strength of the thread, which failed at just over 60t. The heating tube installed for the tube elongation check (TEC) was subjected to a first heating cycle up to 55°C from 7th June to 9th September 2011. After total cooling, a new cycle was launched in January 2012 and reached a temperature of 90°C. With regard to the TED experiment (tube elongation test), the heating phase with three probes is currently being completed. It will be followed by a controlled heat reduction phase. A preliminary feasibility check on the long-duration diffusion test began in the autumn of 2011. The detection capability of the sensors is verified using sealed sources of ^{22}Na. This test will continue into 2012.

Tests on drift sealing components

A complete circumferential groove 30 cm wide and 2.5 m deep was made in an experimental drift parallel to the major horizontal stress (GET drift) in mid-2011. It was the subject of deformation measurements during and after it was excavated. Its walls are still stable and show no signs of change.

Planning of the experimental programme until 2014

Technological tests

The next steps in the technological tests on drift construction will on the one hand concern the excavation of a large diameter structure, the tunnelling machine assembly chamber, for which the excavated diameter is 7.8 m, and on the other, the excavation with the tunnelling machine of 80 m of drifts to test the installation of prefabricated segments. The same structures will be built in the direction of the minor stress between 2014 and late 2015. Experimental phase 3 for the HLW vaults will begin with building of a longer vault, with a new and more powerful machine. In late 2012, full-scale thermo-hydromechanical behaviour (THM) testing will be implemented in and around a HLW vault in its reference configuration.

Characteristics of the Callovo-Oxfordian argilites

The CDZ experiment (mechanical compression of the EDZ) will continue with a new loading cycle after soaking of the fractured zone. This will end with sudden removal of the wall, to observe the reaction of the EDZ. A new experimental arrangement was put into place to acquire date on the chemistry of the pore water of the argilite at 80°C. A sampling system for monitoring the development of the liquid and gaseous phases extracted from a significant volume of argilite raised to a temperature of 85±5°C thanks to four heating boreholes. New diffusion experiments are envisaged as of early 2014, if the feasibility tests prove conclusive. The aim is to have transfer distances of several decimetres, significantly higher than the centimetre scale damage zone around the injection chamber and to assess the anisotropy of the transfer properties due to the stratification of the formation. The tracers used will be ^{36}Cl, ^{22}Na and actinides.

Tests on drift sealing components

With regard to the closure of the underground structures, technological or performance tests on the individual components are scheduled. The performance test on a portion of the swelling argilite seal core (NSC experiment) is scheduled to last several years, in order to monitor resaturation and ultimately estimate the equivalent permeability of the system (core, interfaces, EDZ). In order to finalise the method for interrupting the damaged zone by means of a groove filled with swelling clay, tests will be carried out in 2012 on a mock-up to define the filling method. If these tests prove positive, a complete groove will then be filled with swelling clay, followed by a forced hydration phase. The full-scale implementation test on the complete seals (swelling clay core and low pH concrete support blocks) will be carried out in an installation outside the underground laboratory. It will be part of the European DOPAS project, starting in late 2012.

3.3 Contributions to phenomenological understanding of the disposal centre

Here we look at four topics on which, by means of experimentation and modelling, significant progress has been made in understanding the properties and behaviour of various components of the disposal centre, more specifically the geological medium.

3.3.1 The damaged zone in the argilites around the disposal structures

The initial damaged zone (defined as being that following excavation) is today characterised starting from the structure walls by:

- a zone, referred to as the EDZ, corresponding to a dense 3D network of interconnected fractures;
- a zone, referred to as fractured, characterised by fractures which are only slightly connected, if at all.

The EDZ and the fractured zone are interleaved and have an overall elliptical shape; the dimensions of the small axes of the EDZ and the fractured zone are virtually identical, whereas those of the large axes differ significantly (several radii).

The experimental data as a whole underlines the self-sealing capability by swelling of smectite minerals and mechanical closure of the damaged argilites. These mechanisms intervene very rapidly and lead to the restoration of very low water permeability.

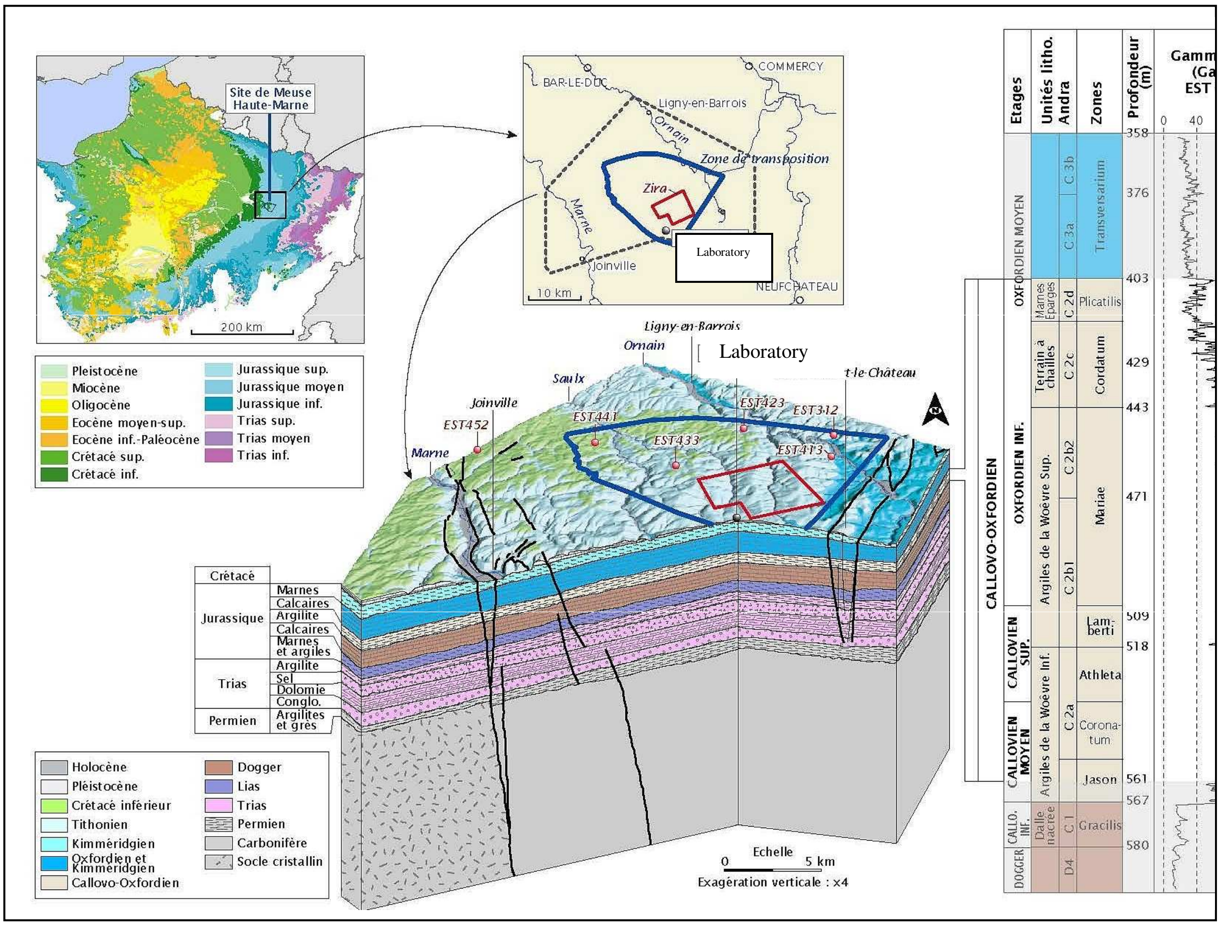
3.3.2 Transfer of solutes to the Callovo-Oxfordian layer

The analysis of the solute transfer conditions in the Callovo-Oxfordian layer of the Transposition Zone, based on the detailed characterisation of the hydro-dispersive parameters of the Callovo-Oxfordian (permeability, anion and cation diffusion coefficients) underlines the predominance of diffusion throughout the Zone.

3.3.3 Flows in the surrounding formations and their evolution over the next million years

From the Zone of Interest for In-Depth studies (ZIRA), and at the scale of the Transposition Zone (see diagram below), the hydraulic trajectories in the Oxfordian and the Dogger are on the whole homogeneous both in terms of direction (North for the Oxfordian and South-East for the Dogger) and rate (about 1 km per 100,000 years for the Oxfordian and 1 km per 20,000 years for the Dogger). The vertical hydraulic load gradients in the

Callovo-Oxfordian on the ZIRA are low, less than 0.1 m/m in absolute terms, and are primarily descending on the ZIRA. The simulations based on the surface terrain erosion rates show very little evolution of the hydraulic load gradients and associated trajectories in the surrounding formations above and underlying the Callovo-Oxfordian over the next million years.



Siting of the Cigéo disposal centre project

The hydraulic-gas transient

The data acquired from the experiments (on samples and in the underground laboratory) carried out by the gas transfer laboratory grouping, as well as via the European Forge programme and numerical simulations based on a more faithful representation of the source terms and gas transfer parameters, led to the identification of new elements. The capacity for easy transfer of gas in the argilite fractures and in the assemblies of clay pellets used to create the seal cores, leads to the consideration of gaseous hydrogen transfer in the central zone and its passage through the access structures to the Oxfordian, as being as probable as its confinement in the waste zones. The hydrogen only migrates in dissolved form and by diffusion through the Callovo-Oxfordian layer to the Carbonaceous Oxfordian and the Dogger. The hydrogen gas pressures in the repository are far lower than those of the argilite fracturing domain.

3.4 Scientific outlook

Here we identify several essential points on which research is being carried out to improve the overall understanding of the behaviour of the repository elements.

Corrosion kinetics of low alloy steel metal components and the coupling with the hydraulic behaviour of the vaults in the production and migration of gases. The degree of uncertainty surrounding the corrosion kinetics of low alloy steel components of HLW vaults (sleeve and over-pack) is still high; more specifically, there is as yet no complete explanation for the evolution of corrosion kinetics which is are first high and then fall or even remain constant over time.

3.4.1 Mechanical behaviour of structures and repository

The representation of the damaged zone as a fractured environment and the long-term deferred argilite deformation rates are two avenues for work involving both experimentation and numerical simulation. The saturation of the seals is today evaluated at full scale, notably using simplified representation models of the hydromechanical behaviour of the pellet assemblies. The specific local and transient effects during saturation must more specifically be evaluated to consolidate the design domain adopted and ultimately the control of the phenomenological evolution of the seals.

3.4.2 Radiological inventories and source terms of certain waste

The ^{129}I inventory is the result of a factory summary and is a maximum of 1% of the total inventory of initial spent fuels. New analytical techniques should be able to show that the ^{129}I levels are lower than those currently being considered in the inventory. The same applies to ^{36}Cl, with a more precise evaluation of the contents in the glasses and the structural elements potentially leading to savings, the scale of which has yet to be determined.

Observation-oversight of the repository
Significant progress has been made in the field of R&D on sensors (optical fibres, spectrometers, miniaturisation, wireless transmission). Efforts must be continued, more specifically to harden the sensors, ensuring their durability and independence, but also to develop means of merging the data that will be acquired during operation of the repository and decision-making systems.

Micro-nano approach to processes
Certain Thermo Hydro Mechanical and Chemical (THMC) processes, notably those taking place at the interfaces, require a small scale approach in order to improve our understanding of them. Work will therefore be needed for modelling and quantifying these processes. This research will in particular be carried out in a project supported by the NEEDS cross-cutting programme.

3.5 Storage research

3.5.1 Studies and research on innovative storage concepts

In order to explore the potential innovations highlighted in the interim report it published in 2009, Andra has initiated a technical and optimisation study to examine in greater detail three concepts for HLW waste and one concept for ILW-LL waste, all assumed to take place on the surface. These concepts were drawn up in particular to ensure greater versatility for the packages accepted. Furthermore, a more detailed study of the oversight systems has been initiated.

HLW in ventilated shafts concept

In the ventilated shafts concept, racks containing six primary packages or three disposal packages make the storage system more versatile. As a counterpart, the heating power of the primary packages is limited to 1,000 W, as opposed to 2,000 W in the glasses storage extension – south-east (E-EV-SE) at La Hague. The racks are stacked vertically in closed stainless steel shafts. Cooling is by means of natural ventilation. This concept is suitable for storage, after an initial thermal decrease period, of a large number of heavily exothermic vitrified waste packages for significant cooling up to 85 years and beyond. Its versatility means that it is able to take HLW packages of various dimensions removed from disposal.

The more detailed study aims notably to increase the unit thermal power of the HLW packages that can be accepted in this storage concept and to improve the oversight possibilities, while looking to reduce the mass of mobile handling equipment which would make it necessary to reinforce the storage vaults.

The HLW on slabs concept

The storage of compacted hulls at La Hague was designed to take standard compacted hulls and end-pieces (CSD-C) waste packages. The average thermal power of these packages is about 15 to 20 W. A storage concept for HLW packages in their primary or disposal form, with a thermal power of less than 500 W was in 2009 derived from that for storage of compacted hulls (ECC). Ventilation is mechanical and horizontal, with partial recycling of the outgoing hot air, in order to regulate the humidity. The storage building responsible for radiation protection and handling is remote-operated. The primary or disposal packages are placed vertically on a base positioned on the building slab(s). This concept is versatile and ensures the accessibility of each package for monitoring. It would be suitable for creating buffer storage capacity or to take packages retrieved from disposal. Additional studies concern improvements to heat removal.

The HLW in concrete modules concept

A storage concept for HLW packages in their primary or disposal format, with a thermal power of less than 1,500 W, was derived in 2009 from that of the existing NUHOMS® concept for spent fuels. The HLW packages are placed in storage containers in groups of 24 to 30 primary packages or 15 to 16 disposal packages. These containers are closed and ensure package confinement and protection. They are placed horizontally in concrete bunkers cooled by circulating natural ventilation in contact with the container. The bunkers provide radiation protection. This storage concept will be optimised for improved interfacing with the transport system.

The ILW-LL storage concept

The primary packages grouped in racks, or the disposal packages, are pre-stacked and then placed in a long row on mobile beams, for adjustment to the various dimensions of the racks and disposal packages.

4 CONTINUED RESEARCH INTO SEPARATION-TRANSMUTATION

4.1 Purpose and implications of the research:

The purpose of separation-transmutation is to remove the minor actinides from the ultimate waste as they are the main contributors to its long-term radiotoxicity and to the residual thermal load after the decay period.

By the end of 2012, in accordance with the requirements of the Act of 28th June 2006, CEA must issue a file on the studies and research carried out into separation and transmutation, jointly with that carried out on the new generations of nuclear reactors and on accelerator driven reactors dedicated to the transmutation of waste, in order to be able to make an assessment of the industrial prospects for these technologies. The following aspects of the studies carried out during the period 2010-2012 should be underlined:

- the transmutation of americium and curium would reduce the long-term radiotoxicity of ultimate waste by a factor of up to 100 at 1000-10,000 years, but without contributing any gains with regard to the radiological impact of the repository. Andra has identified the fact that the significant retention in the Callovo-Oxfordian argilites confined them in the near field and that the activity flux associated with the minor actinides leaving the host formation was negligible);
- transmutation of americium alone would have limited impact on long-term toxicity but would help reduce the repository footprint by a factor of 2 to 5 (a factor 5 reduction would require prior storage for 120 years) for HLW waste alone;
- overall performance is limited by the waste produced beforehand ("initial stock"), as well as by the ability to absorb the inventory at the end of life of the fleet;
- the transmutation of minor actinides only has any sense if multi-recycling of plutonium is utilised;
- transmutation is only effective if it leads to the fission of the actinides: in this respect, fast neutron systems are the most appropriate.

Transmutation is a complex operation, which requires the recovery of elements of interest (separation of the minor actinides), and then their fission recycling in the reactor, with several options being possible: homogeneous or heterogeneous solutions, dedicated stratum.

The research carried out at CEA has validated minor actinide separation processes and certain transmutation systems on real fuels on a laboratory scale; these concepts still need to be consolidated and, prior to industrialisation, would require experimentation on a larger scale.

Implementation would only be possible with the deployment of fast neutron systems in the fleet. The impact of such a strategy, in terms of both reactors and the cycle, was the subject of an initial analysis based on all the criteria to be considered. Industrial implementation cannot be envisaged for the waste generated or engaged by the current NPP fleet.

In the light of current knowledge, retrieval of the minor actinides from vitrified waste for the purposes of transmutation would not seem to be conceivable. The implementation of the transmutation option does not obviate the need for geological disposal of the ultimate waste.

The aim of the research for the period 2013-2015 will mainly be (depending on decisions which could be taken after the 2012 deadline stipulated by the 28th June 2006 Act):

- to consolidate the separation concepts developed for retrieval of the minor actinides;

- to continue with development of fabrication processes of fuels charged with minor actinides;
- to continue with experimental irradiation concerning the various concepts envisaged for the transmutation of the minor actinides, and to clarify the possibilities for demonstrations in the ASTRID prototype and the MYRRHA facility;
- to fine-tune the technico-economic assessments according to the various deployment scenarios;
- to continue with upstream, exploratory or fundamental research in this field.

4.2 The separation of the minor actinides

The research has led to the development of specific extractants and separation processes, successfully tested in the laboratory, for each of the solutions envisaged: Am extraction (EXAm), extraction of Americium and Curium (SANEX) and grouped extraction of all the actinides (GANEX).

For the period 2013-2015 this will mainly involve:

- for the EXAm concept, continuing with "complete experimentation", envisaged for a few kg of spent fuels, from reprocessing up to the production of AmO2 pellets. This experimentation will allow laboratory scale testing of the sequence of the various individual operations (separation, conversion into oxide, fabrication of pellets), but also of the various related operations (in particular the management of effluents and by-products);
- optimising the processes, primarily the EXAm concept for the retrieval of americium. This will essentially involve seeking to reduce the size of unit operations and the flows generated, by looking to work with more concentrated material flows;
- conducting exploratory research on alternatives to the molecules and processes being studied;
- continuing studies to consolidate the processes in order to better examine the conditions for possible industrial deployment (technologies, monitoring and control systems in particular).

4.3 Fabrication of fuels containing minor actinides

The fabrication of compounds containing minor actinides must take account of the specific nature of these actinides, more specifically their radioactivity (alpha, gamma, neutrons), especially for recycling concepts referred to as "heterogeneous", with fuels containing about 10% minor actinides on a UO_2 support.

For the elements of interest, primarily americium, the research concerns:

- the production of oxide powders obtained by means of U-minor actinides co-conversion processes (in particular co-precipitation followed by co-calcination);
- the fabrication of compounds rich in minor actinides, paying particular attention to the production of very fine particles, which are potential sources of irradiation (eradication of crushing steps from the powder mixing processes, exploratory studies concerning alternatives such as gelification processes);
- studies to develop remote-operated technological components (conventional maintenance in gloveboxes impossible), and the development of advanced robotics concepts (which could also benefit the plutonium fuels fabrication studies);

- studies concerning compound cladding materials, taking particular account of the large quantities of helium generated.

The period 2013-2015 will also involve continued study of pilot facilities to prepare for transmutation experiments on fuel cladding pins.

4.4 Experimental burnup

The studies are continuing in order to finalise the transmutation devices and validate the correct behaviour in-reactor of the transmutation fuels with respect to the various concepts envisaged. We will first of all focus on the transmutation of americium:

- ***for homogeneous recycling***, further to the previous experiments in PHENIX which validated the concept up to levels of a few % of minor actinides, this will involve looking at the high burnup fractions (about 100 GWd/t) envisaged for fast reactors;
- ***for the concept of recycling in blankets***, this involves conducting mini-disk scale experiments (less than one gram) while attempting to reproduce the operating characteristics of fast reactor blankets during burnup in experimental reactors (temperature conditions in particular) and the production of gas corresponding to the targeted transmutation ratios. It is on this mode of transmutation, for which the studies are least advanced, that most of the efforts over the period 2013-2015 will focus.
- ***for the concept of recycling in a dedicated layer (Accelerator Driven System - ADS in particular),*** post-burnup analyses of FUTURIX fuel samples irradiated in PHENIX under an international CEA-DOE-ITU programme will be carried out (as of 2013).

The possibility of demonstrating an ADS prototype, such as the European MYRRHA project in Belgium to demonstrate burnup of fuels dedicated to the transmutation of minor actinides, will also be examined. The aim will also be to specify the nature and scale of the demonstrations that could be carried out in the ASTRID prototype or in the MYRRHA facility. During the 2013-2015 phase, efforts will more specifically focus on precisely identifying the domain chosen for the ability to transmute the minor actinides in ASTRID, by in particular identifying any "threshold effects" on the dimensioning of the reactor (domains for which the aim of the demonstration significantly affects the dimensioning of the reactor).

4.5 The scenarios studied

The studies carried out over the period 2006-2012, which will be reported in the file presented by CEA in late 2012, allowed an assessment – on the basis of various criteria (gains in waste management, but also costs and detrimental effects throughout the fuel cycle) – of the impact of the implementation of minor actinide separation-transmutation options.

During the period 2013-2015, CEA will focus on:

- consolidating these assessments: estimation of uncertainties, more detailed quantification of certain criteria, updating of certain technological parameters on the basis of advances in research, whether for critical reactor concepts or for ADS;
- together with the industrial partners and coherently with the decisions to be taken by the public authorities, drafting of possible deployment scenarios for such options in the French NPP fleet. The options will be envisaged for the ASTRID prototype sodium-cooled fast neutron reactor and, at a European level, for the MYRRHA facility.

4.6 Upstream research

Upstream research to support separation and transmutation studies will be carried out:

- for aspects related to radiochemistry and to studies of separation concepts in Atalante at CEA Marcoule, and within the Marcoule Separation Chemistry Institute (ICSM). A contribution to CNRS's NEEDS programme is also being envisaged. Research is looking at identifying and controlling the fundamental phenomena which govern the selective extraction of the elements of interest, the actinides, and the exploration of original concepts;
- for aspects linked to transmutation, through the use of analytical irradiation (PROFIL irradiations) to validate nuclear data at CEA and validate elementary nuclear data within the framework of the CNRS' NEEDS programme and European programmes, the acquisition of nuclear data concerning the elements of interest and the sensitivity studies will be continued.

5 CONDUCTING RESEARCH TO SUPPORT THE SAFETY ANALYSIS FOR DISPOSAL PROJECTS

IRSN has taken the necessary steps to produce the appraisals of the safety files which Andra will be presenting for the creation of new radioactive waste disposal facilities. The areas which warrant major efforts include that of the safety of a deep geological disposal facility. The focus of the research activities at IRSN on this subject is different from those for which Andra is responsible. They can call on far more limited resources and concentrate on a smaller number of subjects, aiming to provide the independent support necessary for the future appraisals. In this respect, the deadline of the 2013-2015 plan coincides with a key date for IRSN, which will be required, on behalf of ASN, to examine the file prepared by Andra to support the application for creation of a deep geological disposal facility. With this in mind, IRSN will utilise all the new knowledge acquired by itself or by the scientific community, more specifically since the examination of the 2005 File on the feasibility of a geological disposal facility in a layer of clay. For the period 2013-2015, the Institute more specifically intends to:

- **increase its research efforts on the behaviour of the repository confinement barriers, more specifically during the transition period,** beginning during the operations phase and continuing after its closure**.** In this respect, the phenomena to be considered are notably**:**
  - **thermal, hydric and mechanical (THM) phenomena** liable to affect the performance of the components of the disposal facility. Over the coming three years, IRSN will focus its efforts on the one hand on continuing the seal experiments in the Tournemire experimental station, in order to assess the key parameters which govern the overall performance of the seals (SEALEX tests), and on their modelling and, on the other, on understanding and modelling the effects of gases inside a repository. On this point, IRSN is a contributor to the European Forge project, more specifically concerning the evaluation of the gas formation mechanisms and the numerical simulations of the expected effects. With regard to the mechanical behaviour of the disposal vaults (appearance of the EDZ, role of supports, etc.), IRSN will supplement the simulations of this behaviour by taking account of the observations made in the various laboratories, notably that in Bure;
  - **the main factors in the physico-chemical evolution of the components of the disposal facility.** The studies initiated aim to clarify the influence on safety of the chemical processes during the various phases in the life of a disposal facility. On this point, the Institute intends to continue the study of the possible effects of bacterial growth on steel corrosion and the study of the phenomena of radiolysis and the

degradation of waste packages, as well as to step up its effort to understand the complex cement/iron/clay chemical interactions. The experiments carried out for this purpose in the Tournemire experimental station will concern the impact and duration of the oxidising transient in the vaults simulating the HLW type vaults (OXITRAN tests) and the durability of the concrete and near field disturbance of the clays under the effect of temperature (CEMTEX tests) or of artificial ventilation. IRSN will also take part in the experimental studies carried out in the Mont Terri underground laboratory (Switzerland) concerning the evolution of clayey materials under the effect of an alkaline pH;

- **complete the study of the characteristics of importance for the confinement capacity of the geological barrier,** over and above the analysis of the field data concerning the site studied by Andra and the assessment of the limits of the survey methods employed, the Institute will continue its efforts to study the differential fracturing of the clays, which notably aim to provide an explanation of the presence or absence of fracturing in various clay formations. At the same time, by means of the FRACTEX programme to be implemented in the Tournemire experimental station, IRSN plans to study the transport properties associated with areas of the medium subject to little disturbance. IRSN will also complete its knowledge of the transfer properties associated with the indurated clay formations, by taking part in the TAPSS2000 national programme associated with deep drilling to 2000 m by Andra on the Meuse /Haute-Marne site and the research programmes associated with the project to drill through the entire clay layer of Mont Terri;
- **to consolidate its ability to perform global modelling of the disposal facility.** In this respect, it will continue its assessment of the influence of the hydraulic schemes of the Meuse/Haute-Marne site by means of specific hydrogeological models, integrating all the new data acquired. IRSN also intends to make a particular effort on simulation of radionuclide transfers in the geological medium, by means of the MELODIE computer code. Finally, IRSN intends to delve more deeply into defining possible very long-term evolution scenarios for the disposal facility and its environment, on the basis of the scientific knowledge available on the subject.

IRSN does not carry out its research in isolation. At the national level, it is involved in numerous cooperative ventures with a network of renowned scientific partners, organisations, schools and universities. Together with CNRS, the Institute will also take part in the NEEDS programme. IRSN has also renewed the MoU established with Andra and allowing joint research work to be carried out in accordance with the provisions allowing compliance with the necessary ethical rules. At an international level, IRSN aims to work on European projects. In addition to the Forge programme in the 7th Framework Programme (FP) in which it is a participant (see above), the Institute is coordinating the SITEX project, a grouping of 15 appraisal organisations and safety regulators, which aims notably to create the conditions, Europe-wide, for interconnecting the research on technological developments, supported by the IGDTP platform, and that concerning the safety of disposal facilities, in which the independent organisations of the nuclear licensees must necessarily take part. International partnerships, notably with Japanese (JAEA, JNES), Canadian (CCSN) and Russian (SEC/NRS, IBRAE) organisations have also been created around particular IRSN research projects.

Finally, the Institute recalls its desired to continue to make its experimental resources available to the French and foreign scientific community, in particular the clay medium experimental station at Tournemire (Aveyron), which since 2007 has been integrated into the IAEA centres of excellence.

6 GAINING A CLEARER UNDERSTANDING OF THE SOCIAL DIMENSION OF WASTE MANAGEMENT

The involvement of the Human and Social Sciences (HSS) in the field of radioactive waste and materials management is justified upstream by the desire to make the various recommended solutions more robust. Their acceptability, which in the end is political in nature, is made easier when all the phenomena involved are dealt with in an appropriate framework, without ignoring their socio-economic, environmental, political, cultural, etc. aspects, and the various scientific and technical issues involved are interconnected. One-dimensional, inward-looking R&D has little chance of helping technical projects succeed, as is shown by the history of nuclear waste management in France prior to 1991. The aim of HSS research is thus to integrate the social aspects into the various on-going projects and ensure that they all work together in a cross-disciplinary system. Collaboration with researchers from these diverse backgrounds must from the outset aim to create specialised communities on subjects of common interest with the operators.

The topic of reversibility was the first to be given this level of priority by Andra. Several scientific events, in particular the organisation of two symposia, resulted from this work, as did the publication of the collective work "Making radioactive waste governable. Deep disposal undergoes the reversibility test". An economic sciences PhD dissertation was also defended on this topic.

Andra also launched the "memory" project in 2010, which on the one hand comprises work designed to continue to create and improve the memory of and records about the facilities and, on the other, scientific studies concerning materials ageing and issues specific to human and social sciences (HSS). Scientific studies into the ageing of materials consisted in testing the permanent ink/paper combination by means of standardised tests. Durability studies on other media for the longer term are currently being defined. They will concern non-paper media for writing and engraving, in particular studies of surface markers to be installed on the cover over the centres and the production of sapphire disks as demonstrators for a memory medium, the longevity of which could be up to a million years. As for HSS studies, an initial bibliographical approach is planned, in order to define a framework for any research to be included in the Agency's scientific programme. The envisaged work is based around the following topics:

- the longevity of languages and symbols, in order to determine for what reasonable time current or dead languages can be known and what the communication solutions could be once these languages cease to be known;
- institutional conservation of written works, sounds, images, objects, etc. by specialised French and international organisations, to analyse the preventive measures taken to limit deterioration over time and encourage assimilation and transmission by future generations;
- long-term digital archival, more specifically by organising an intelligence watch in this field, which is beginning to become organised and which, within the next few decades, could open up new prospects for the long term;
- the archaeology of techniques and landscapes, incorporating man-made changes and geodynamic changes, as well as the possibilities of memory with human creations (use of the in-fill of surface-underground links as a memorisation tool);
- the memory of "legacy" repositories not managed by Andra, which exist in various places in France (uranium mines, nuclear tests, etc.);
- the foreseeable changes in society;
- the inclusion of preserving the memory of repositories in teaching programmes on nuclear energy, heritage and memory;

- transmission of memory between generations via social networks on the internet.

Andra is also taking part in international work on memory within the "Preservation of record knowledge and memory" working group set up by the Nuclear Energy Agency (NEA/RWMC/RK&M).

More recently, in 2011, Andra set up a multidisciplinary steering committee to create a grouping of cross-cutting human and social sciences laboratories (GL-SHS). It comprises researchers from CNRS, SciencesPo Paris, Ecole des Hautes Etudes en Sciences Sociales, the Institut Francilien Recherche Innovation Société, Mines ParisTech and other university institutions. The general central research topic for the grouping is "transmission between generations and understanding of long time scales". This is because the time-frames involved in the Agency's activities, in particular in the management of the most highly radioactive waste, is indeed unique when compared with other industrial areas. It raises particularly complex questions which notably concern the ability to anticipate events over long periods of time and to ensure that they are managed. The approach adopted was to look at the practices and tangible arrangements made to produce the Cigéo geological disposal project. The question arises of the transmission to future generations of the means and resources for intervention on the fate of this project. In order to encourage and promote cross-pollination between perspectives and exchanges, the GL-SHS research programme is jointly put together by the members of the steering committee and the pilot Andra. It is built around the following three core subjects: governance, knowledge and memory; socio-economic evaluation.

The first part deals primarily with integrating social aspects and scientific and technical elements into the decision-making processes. Profound changes have been made in the field of radioactive waste governance in recent decades. The question now is to ensure that the long time entailed by research (and by radioactive waste) is in phase with industrial time and political time. These time-related approaches are the responsibility of numerous players with interests that may often conflict, and lead to organisational changes.

The second point above all concerns the robustness of knowledge over the long-term and the transmission between generations of the information, practices and knowledge necessary for the Cigéo project. The specific knowledge production system put into place at Andra to demonstrate the feasibility and safety of reversible geological disposal of radioactive waste, in particular the use of modelling and numerical simulation tools, is of particular interest to science historians. It is the characteristic traits of modern techno-sciences and the complexity of the relationship between science and society that can also be examined from a fresh perspective. The techno-scientific understanding of long time-scales, or more precisely of the future, is an object of research which is currently arousing great hopes on the part of the social sciences. The problem of understanding long time-frames is not independent of that of memory, for which the socio-anthropological perspective preferred by the GL-SHS could prove to be highly instructive. Despite the efforts made since the 1980s by those in charge of managing radioactive waste around the world, in particular the Scandinavian countries with institutional archival and the United States with markers, little academic research has however been carried out on the question of multi-millennia memory in this field.

The third point is based on the study of socio-economic evaluation methodologies and practices applicable to the Cigéo project, as well as directories of the players involved, or liable to be so, in this evaluation. One aspect is to analyse the respective functions and roles of the different types of evaluation (ex ante/ex post/ex nunc, internal/external, etc.), how different forms of knowledge are integrated into them, and the place given to forward planning and scenario writing in them. The comparability with similar projects through the study of "major project" case studies was also given particular attention.

Finally, CNRS's NEEDS places HSS at the heart of nuclear questions and envisages looking at the question of time in a more general manner, from the risk management and assessment viewpoint. This perspective requires adopting an approach that is both retrospective concerning methods of memorising sites entailing a risk (polluted sites, mining residues, etc.) and forward-looking concerning the pooling of data and their transformation into operational knowledge in appropriate information systems (for example, for management of contamination according to differentiated time-scales). This positioning will more broadly enable concrete answers to be given, based on past experience in order to envisage scenarios for assuming public responsibility for the transfer of knowledge. With regard to the management of radioactive waste and materials over long time-frames, the issues are not only the memorisation of data but more generally the actual process of recording a trace by the players involved and the decision-making styles that are most appropriate to such time-scales. Particular attention will be given to the following questions:

- What are the social implications of the various possible long-term management solutions (storage, transmutation, geological disposal, etc.)?
- What decision-making modes are associated with the various long-term management options?
- What are the technical and political consequences of the absolute need for reversibility?
- How can the knowledge necessary for the operation or decommissioning of today's equipment be passed onto future generations?
- How can one ensure justice and fairness across the territory and between the various generations? What are the underlying ethical issues?
- What are the impacts on man and the environment of the various radioactive waste and materials management choices?
- What is the role of the experts and the citizens with regard to the public decision concerning a situation of long-term uncertainty?

In answering these questions, the aim is not social engineering in order to make the envisaged solutions more acceptable, but on the contrary to fuel the debate and the public choice and thus strengthen the ties between science and society. The aim of the NEEDS (nuclear, risk, society) programme is to ensure that knowledge progresses and that programmes, networks and diverse skills are established, while addressing the need for transparency and sustainability which today characterises the public debate. This programme also intends to build on the HSS knowledge acquired on the topic of nuclear waste, more specifically through the considerable work done on this question at CNRS.

Appendix 5: Concepts and plans for the post-closure period

1 CONCEPTS AND PLANS FOR THE PERIOD FOLLOWING THE CLOSURE OF INSTALLATIONS CLASSIFIED ON ENVIRONMENTAL PROTECTION GROUNDS

1.1 Very low level waste disposal centre

The operation of the VLL waste disposal centre is regulated by order 2012040-0002 authorising the operation of this first installation classified on environmental protection grounds dedicated to the disposal of radioactive waste. This order is derived from the regulations applicable to the disposal of hazardous waste (ministerial order of 30th December 2002, amended). Andra also wanted to follow the same methodology for assessment of the long-term impact of the VLL waste disposal centre as that already used for the low and intermediate level waste disposal centres, the Manche disposal centre and the LLW/ILW waste disposal centre in the Aube.

The order thus presents the requirements of resources imposed on the hazardous waste disposal facilities by the regulations as well as the additional requirements specified as a result of the safety assessments carried out for all the phases in the lifetime of the facility, from the construction phase to the post-oversight phase.

In accordance with the authorisation order, Andra will propose a project to the Prefect defining the institutional controls to be applied to all or part of the facility, no later than one year after the end of the operating period. These institutional controls could prohibit the building of constructions and structures liable to impair the conservation and oversight of the site covering. They should also ensure that the means of collecting leachates before sealing of the shafts at the end of the oversight phase are protected and that confinement of the waste emplaced is durably maintained. Moreover, the purpose of the oversight phase will be to monitor the evolution of the disposal facility for a period of at least thirty years after the last waste is emplaced, and its conformity with the forecasts and the order of the Prefect. For this purpose, controls will be maintained, more specifically:

- regular upkeep of the site (ditches, cover, ponds, fencing, etc.);
- geotechnical observations of the site, with regular and at least annual updating of the topographical survey;
- periodic measurement of the quality of the water collected from the centre and discharged into the environment and checks on compartments of the ecosystem in the near environment of the VLL waste disposal centre.

All of these measurements will be to verify the absence of radioactive or chemical pollution in the environment of the centre. They will be able to ensure early detection of any behavioural anomalies and anticipate any remediation measures.

Following this oversight phase, the record of the centre will at the very least consist of institutional controls entered into the land registry.

2 CONCEPTS AND PLANS FOR THE PERIOD FOLLOWING THE CLOSURE OF BASIC NUCLEAR INSTALLATIONS

The legislative framework applicable to Basic Nuclear Installations for the period after closure of the facilities, is more specifically based on:

- the Act on transparency and security in the nuclear field (TSN Act 2006-686 of 13th June 2006 codified) which specifies that the transition of a BNI to the oversight phase is subject to authorisation (Article L.593-25 of the Environment Code) and that the administrative authority can apply institutional controls around this BNI (Article L.593-5 of the Environment Code);
- decree 2007-1557 of 2nd November 2007 which specifies the content of the authorisation application file for transition to the oversight phase. This file in particular contains: the impact assessment, a safety report, a risk management study, the general oversight rules and, as applicable, the institutional controls (see Art. 43)
- the order of 7th February 2012 setting out the general rules for BNIs. Chapter V of this order concerning radioactive waste disposal facilities stipulates that: *"In compliance with the objectives set forth in Article L. 542-1 of the environment code, the choice of the geological environment, the design and the construction of a radioactive waste repository, its operation and its entry into the oversight phase are defined such that protection of the interests mentioned in Article L. 593-1 of the Environment Code is ensured passively against the risks presented by the radioactive or toxic substances contained in the radioactive waste after entry into the oversight phase. This protection must not require intervention beyond a limited oversight period, determined according to the radioactive waste disposed of and the type of disposal repository. The licensee justifies that the chosen design meets these objectives and justifies its technical feasibility."*

2.1 The Manche disposal facility

From the regulatory viewpoint, the Manche disposal facility (CSM) is a basic nuclear installation (BNI n°66) dedicated to the surface disposal of low and intermediate level, short-lived waste. The creation authorisation decree dates from June 1969. The transition of the facility to the oversight phase was authorised by decree 2003-30 of 10th January 2003. This oversight phase is conventionally scheduled to last a period of three hundred years and includes a discharge license dating from 10th January 2003. In 1996, on the basis of the conclusions of the Commission assessing the situation of the Manche disposal facility (known as the "Turpin Commission"), it was decided that the *"site may not be relieved of all controls"* after this oversight period. Andra thus confirmed the need to maintain and eventually transmit the memory of the site and take all necessary steps to limit the nature of the constructions or equipment which could be installed on it.

The concepts and plans for the post-closure period comprise: the design of the facility, oversight and maintaining a recorded memory:

- <u>measures concerning the design were taken by the licensee during the operating phase</u>. Thus, after closure, the disposal facility corresponds to a mound in which the waste packages disposed of in the structures are protected from climatic hazards by a low-permeability cover; an effluents management system recovers water that has infiltrated through the cover and/or into the disposal facility. The water recovered is transferred to the AREVA-La Hague treatment installation, in accordance with the discharges authorisation order;

- decree 2003-30 authorising transition to the oversight phase mentioned that the licensee must ensure oversight appropriate to the facility and its environment. This is defined in the regulatory oversight plan, which includes monitoring of the cover, the confinement of the disposal structures and the discharges from the centre. This plan specifies that the results are regularly sent to ASN (annual report) and to the public (summary of the annual report presented to the CLI). The decree also defines that protection of the facility against the risks of intrusion and malicious acts is guaranteed for the duration of the oversight phase. Furthermore, the decrees stipulates that every ten years, the licensee shall study whether or not the oversight and protection measures applicable to its facility need to be updated;
- in terms of maintaining a memory and record of the facility, three avenues have been identified:
  (i) long-term archival of the information: decree 2003-30 defines the requirements concerning the long-term archival of information:
  - *Detailed memory*: the documents are duplicated on permanent paper and archived in two separate places, in the Manche disposal facility and in the French National Archives. The archive is updated every 5 to 10 years depending on developments in the Centre;
  - *Summary memory:* an initial version of this document of about a hundred pages was submitted to ASN and the CLI in 2008. This document should be revised as and when the safety reviews are carried out, in order to incorporate all the operating experience feedback gained from the oversight phase. When it is considered as having been stabilised, it will be printed on permanent paper and widely distributed, as stipulated by the technical prescriptions;
  (ii) information of the public, in particular during the oversight phase, more specifically via exchanges with the local information committee (CLI) and through communication measures;
  (iii) the draft application for the creation of institutional controls to minimise the risk of intrusion into the disposal facility for as long as possible after the oversight phase. Such institutional controls were suggested by the Turpin Commission and envisaged by Andra, in the 2009 safety report, pursuant to Article 31 of Act 2006-686 of 13th June 2006.

2.2 The Aube waste disposal facility

From a regulatory viewpoint, the Aube LLW/ILW waste disposal facility, which took over from the Manche disposal facility, is also a basic nuclear installation (BNI n°149). The creation authorisation decree of 4th September 1989, was modified by decree 2006-1006 of 10th August 2006 plus its discharge license of 21st August 2006.

With regard to the period following operation, the creation authorisation decree for the LLW/ILW waste disposal centre notably stipulates that: (i) during the oversight phase, "*the structures shall be protected by a cover of very low permeability" and "the facility shall continue to be monitored for a time allowing radioactive decay of the radionuclides with short or intermediate half-lives, to a level presenting no further significant radiological risk.";* (ii) following the oversight phase, *"it shall be possible for the land occupied by the facility to be used normally without any radiological restriction […] no later than 300 years following the end of the operations phase".*
In addition to the regulatory aspects, Andra also follows the recommendations of RFS I.2 which defines the fundamental safety objectives, the design bases for a repository and the monitoring of the facility during the operations and oversight phases.

In the same way as the CSM, the concepts and plans for the LLW/ILW waste disposal centre's post-closure period comprise: the design of the facility, oversight and maintaining a recorded memory:

- the measures concerning the design were taken by the licensee during the operating phase in accordance with the requirements of RFS I.2:
  (i) the limitation of initial activity: the radioactive waste accepted by the LLW/ILW waste disposal centre is waste with a short or intermediate half-life, with limited quantities of long-lived radionuclides or those with low or intermediate specific activity. The aim is for the activity of the radionuclides disposed of to have significantly decreased during the 300 years of facility oversight;
  (ii) confinement of the waste is ensured by the package and the structure (including the cover and the infiltrated water collection networks) during the operation and oversight phases and by the geological formation on which the disposal facility is sited, notably during the post-oversight phase;
- the provisions concerning oversight of the facility and its environment. At closure of the centre, in accordance with decree 2007-1557, Andra will apply for authorisation to make the transition to the oversight phase and shall propose general oversight rules. A decree will authorise transition to the oversight phase. The monitoring approach currently employed during the operations phase will in principle continue during the oversight phase. This oversight relies on a certain number of measurements (notably radiological, chemical, water table heights, hydrological, climatological) the monitoring of which over a period of time should make it possible to: (1) verify the correct working of the disposal facility, ensuring the absence of any unacceptable dissemination of the radionuclides initially contained in it; (2) detect any abnormal situation or development in order to identify and locate the causes and initiate the necessary remedial measures; (3) gain sufficient understanding of the disposal facility evolution mechanisms; (4) assess the radiological and chemical impact of the disposal facility on the population and the environment and monitor its evolution, in order to verify compliance with the regulatory requirements; (5) ensure protection of the facility against the risks of intrusion and malicious acts;
- provisions concerning maintaining the recorded memory: Andra relies on the reference solution developed for the CSM, for which preparations are made as of the operations phase. The CLI should also continue during the oversight phase and thus allow public information and consultation.

2.3 The Cigéo disposal facility project

The Safety Guide for the final disposal of radioactive waste in a deep geological formation was issued by ASN in 2008. This guide defines:
- the fundamental safety objective: protection of the health of individuals and of the environment is the fundamental safety objective of the disposal facility. After closure of the disposal facility, protection of the health of individuals and the environment should not depend on oversight and institutional controls, which cannot be maintained with any degree of certainty beyond a limited period;
- the design bases and safety principles;
- oversight and maintaining the recorded memory: a facility oversight programme must be put into place during the construction of the disposal structures and until closure of the facility. Certain monitoring provisions could also be maintained after closure of the facility. The need to implement this oversight should be taken into account as of the design of the disposal system. The recorded memory must be maintained after closure of the site.

The Cigéo disposal project will be designed in a deep geological layer, the Callovo-Oxfordian, to allow long-term confinement of the substances contained in the High Level and Intermediate Level long-lived waste. According to Article L542-10-1 of the Environment Code, *"a disposal* *facility in a deep geological formation for radioactive waste is a basic nuclear installation".* The Cigéo project thus falls under the regulations applicable to BNIs, as defined in §1.2.

In accordance with the regulatory framework, notably the order of 7th February 2012, and the above-mentioned ASN Safety Guide, the Cigéo disposal facility project is designed to evolve from active safety to entirely passive safety, where no human intervention will be required. After operation, the facility will be closed and enter the oversight phase.

As with surface facilities, the concepts and plans for the period following closure of the planned Cigéo disposal facility comprise the design of the disposal installations, oversight and maintaining the recorded memory:

- provisions concerning the design: to meet the post-closure safety objectives, the deep geological formation disposal facility is designed to be able to guarantee and demonstrate safety during operations and for a long time following its closure, with regard to both man and the environment, while being reversible for a period of at least 100 years. In accordance with the regulations and the ASN Guide, the underground disposal facility shall, once closed, meet the post-closure safety objectives passively. The safety of the facility is thus based on a range of components for confining the radioactivity and isolating the waste from any possible external hazards;
- the provisions concerning oversight of the facility and its environment. Steps shall be taken to maintain the memory and ensure monitoring for as long as possible. Monitoring of the environment is envisaged prior to construction (initial baseline state), during construction and throughout the operating period. This could be continued after closure of the underground facility and decommissioning and dismantling of its operating installations on the surface. This oversight will meet the regulatory requirements concerning the monitoring of the impacts of the facility. All of these measurements will be to verify the absence of radioactive or chemical pollution in the environment of the centre and ensure that it is functioning correctly. The long-term environment observatory (OPE) offers a framework for monitoring the environment before and during construction and operation. An oversight programme is also designed with respect to post-closure safety to monitor a certain number of parameters in the underground facility during its operating phase. The means implemented for post-closure oversight of the Cigéo project will notably be based on experience feedback from the surface facilities;
- provisions concerning maintaining the recorded memory focus primarily on: transmission to future generations, to inform them of the existence and contents of the facility and provide them with knowledge enabling them to understand their observations, to facilitate any actions or to transform the site. At present the reference solution adopted by Andra to guarantee a memory of its disposal centres is based on five measures: (i) two "active" memory systems to ensure that a recorded memory is preserved in the short and medium terms, and (ii) three "passive" systems for the longer term. This reference arrangement must be implemented for the Cigéo disposal project, with the need for the memory to be maintained following closure of the facility for as long as possible, and at least for five centuries.
At this stage of the project, the reference solution implemented in the Manche disposal facility constitutes the basis for the memory system to be implemented for the Cigéo disposal project.

2.4 Low level, long-lived waste disposal project

Andra is drawing on the "General safety orientations report concerning the search for a site for disposal of long-lived waste of low specific activity" published by ASN in May 2008. It thus stipulates that:

(i) after closure of the disposal facility, protection of the health of individuals and the environment should not depend on oversight and institutional controls, which cannot be maintained with any degree of certainty beyond a limited period;

(ii) with regard to the oversight phase, the designer must consider the means of ensuring this oversight as of the design of the disposal facility.

The concepts and plans for the period following closure of the LLW-LL project are closely linked to the concepts developed, to the site(s) chosen for the disposal facilities and to the nature of the waste disposed of. Steps shall be taken with respect to oversight following closure of the disposal facility. They shall be studied and clarified as the design studies progress. They shall be based on all the operating experience feedback from the licensee, Andra, concerning the other centres.

Appendix 6: Inter-governmental agreements concluded in France for management of spent fuel or radioactive waste (agreements in force, given in chronological order)

1 AGREEMENTS IN FORCE, LISTED IN CHRONOLOGICAL ORDER

1 – Switzerland:
Exchange of letters constituting the agreement between France and Switzerland with regard to the COGEMA reprocessing contract, signed on 11th July 1978.

2 – Netherlands:
a) Agreement in the form of an exchange of letters between the Government of the French Republic and the Government of the Kingdom of the Netherlands, concerning the reprocessing in France of spent fuel elements, signed in Paris on 29th May 1979.
b) Modifying agreement dated 9th February 2009, published by decree 2010-1167 of 30th September 2010

3 – Sweden:
Exchange of letters constituting the agreement between France and Sweden regarding the reprocessing contracts, signed on 10th July 1979. Additional exchange of letters constituting the agreement between France and Sweden, signed on 10th July 1979.

4 – Spain:
Exchange of memoranda constituting the agreement between France and Spain on radioactive waste from spent fuels produced by the Vandellos I NPP, signed on 27th January 1989.

5 – Japan:
Cooperation agreement between the Government of the French Republic and the Government of Japan for the use of nuclear energy for peaceful purposes, signed in Tokyo on 26th February 1972. Protocol modifying this cooperation agreement (set of three annexes, a report and an exchange of letters), signed in Paris on 9th April 1990.

6 – Australia:
Arrangement between the Government of the French Republic and the Government of Australia concerning the implementation of a reprocessing contract concluded between COGEMA and the Australian Nuclear Science and Technology Organisation (ANSTO), in the form of an exchange of letters, signed in Paris on 27th August 1999.

7 – Italy:
Agreement between the Government of the French Republic and the Government of the Italian Republic concerning the reprocessing of 235 tons of Italian spent fuel, signed in Lucca on 24th November 2006, published by decree 2007-742 of 7th May 2007

8 – Germany:
Agreement in the form of an exchange of letters between the Government of the French Republic and the Government of the Federal Republic of Germany concerning the transport from the French Republic to the Federal Republic of Germany of packages of radioactive waste from reprocessing of spent fuel, signed in Paris on 20th and 28th October 2008, published by decree 2008-1369 of 19th December 2008

9 – Monaco:
Agreement between the Government of the French Republic and the Principality of Monaco concerning the entry into French territory of radioactive waste from Monaco, signed in Paris on 9th November 2010.

2 AGREEMENT CURRENTLY ENTERING INTO FORCE

1 – Netherlands:

Agreement between the Government of the French Republic and the Government of the Kingdom of the Netherlands, concerning the reprocessing in France of Dutch spent fuel elements, signed in The Hague on 20th April 2012.

Ministry for Ecology, Sustainable Development and Energy
General Directorate for Energy and Climate
Arche Nord – 92 055 La Défense Cedex
Tél. : 01 40 81 21 22
www.developpement-durable.gouv.fr

French nuclear safety authority (ASN)
15-21 rue Louis Lejeune – 92 120 Montrouge
Tél. : 01 46 16 40 00
www.asn.fr

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