NATIONAL ASSESSMENT BOARD

FOR RESEARCH AND THE STUDIES INTO THE MANAGEMENT OF RADIOACTIVE WASTE AND MATERIALS

Instituted by the law n°2006-739 of June 28, 2006

ASSESSMENT REPORT N° 9

JUNE 2015

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TABLE OF CONTENTS

HUBERT DOUBRE (1941 – 2014)

Hubert Doubre was appointed to the Commission nationale d'évaluation des recherches et études relatives à la gestion des matières et des déchets radioactifs (CNE2 - French national board for the assessment of research and studies into the management of radioactive waste and materials) in April 2007. He passed away on 2 July 2014 following the rapid deterioration of a malignant lesion for which he had been receiving treatment over the past few years. Hubert Doubre, a former student of the prestigious Ecole normale supérieure in Saint Cloud, the holder of an agrégation (high-level teaching qualification) in Physical Science, a State-certified Doctor of Nuclear Physics and a university professor, had a brilliant career and in teaching and research. He held several successive scientific management posts, firstly at GANIL in Caen and then at CSNSM in Orsay, while actively participating in numerous advisory and consulting bodies. He was acknowledged by all of his colleagues as a high-level researcher and an excellent educator. His specialism was experimental and theoretical sub-atomic physics, ranging from the synthesis of nuclei to their organisational structure. Hubert Doubre devoted significant efforts to the multidisciplinary problems of the downstream part of the electronuclear energy cycle during his leadership of the PACEN programme at the CNRS (National Centre for Scientific Research) and through his participation on this organisation's Ethics Committee. He was mindful of the scientist's role in society and had a passion for the history of science. As a member of our Board, Hubert Doubre cast a frank and enlightened eye on the research that we were responsible for evaluating and on the opinions that we submitted. All of our Board members will remember him as a colleague of firm convictions and of great ability, driven by the desire to analyse problems and the proposed solutions in the greatest of detail in order to ensure the safest possible future for nuclear waste.

SUMMARY AND CONCLUSIONS

According to the provisions of the 2006 act, the long-term management of high and intermediate-level waste involves two related components: the partitioning-transmutation of actinides found in the spent fuel of future nuclear reactors and the geological disposal of long-lived high and intermediate-level waste (LLHLW & LLILW) in accordance with the principle of reversibility. In addition, facilities in the front-end and back-end of the nuclear fuel cycle and the dismantling of decommissioned facilities produce long-lived low-level waste (LLLLW), very low-level waste (VLLW) and waste with augmented natural radioactivity. The LLLLW pose different management problems due to the very large quantities produced. Short-lived low and intermediate-level waste is stored at the Aube storage centre (centre de stockage de l'Aube – CSA).

CIGÉO GEOLOGICAL DISPOSAL REPOSITORY

The aim of the Cigéo project is to design and build a geological disposal repository for the LLHLW and LLILW included in the French industrial programme for waste management (Programme industriel de gestion des déchets – PIGD), which covers all of the waste from the current nuclear fleet. This repository should be created at a depth of 500 m in the 130 m thick Callovo-Oxfordian (COx) argilite formation at the location of the Meuse-Haute Marne (Zira) site.

The Board takes note that the construction application (Demande d'autorisation de création – DAC) for Cigéo will not be filed in 2015, despite this being a requirement of the act of 28 June 2006. The new deadline for the filing of the DAC has been set for 2017. The Board hopes that this new schedule, which is still very tight, will be respected so as to clarify the management of LLHLW and LLILW.

During the optimisation process for the deposition of exothermic packages in LLHLW disposal cells, the studies of the thermo-hydro-mechanical (THM) behaviour of the COx prompted Andra (French national agency for the management of nuclear waste) to make significant modifications to the configuration of the LLHLW zone and its situation within the "Zira" (meaning "Area of Interest", from the French acronym for "zones of interest for further investigation"). Further studies and research are required to refine the THM behaviour of the repository, improve the definition of areas in which there is a risk of the violation of the criterion of zero thermal fracturing of the COx, and assess the safety implications of such a violation. In the DAC, Andra must adopt an architectural design for the LLHLW zone enough conservative to allow for the storage of all LLHLW from the PIGD in compliance with the safety regulations. A better use of the underground space could possibly be proposed if the knowledge of these issues should improve in the future.

Studies of the resistance of disposal concrete containers filled with bituminous packages exposed to thermal conditions representative of a major fire demonstrate the strength of the packages and the chemical inertia of the bituminous matrixes. These new data ease the fears concerning fires of external origin to the packages in Cigéo facilities. Andra must continue the studies relating to the chemical stability of bituminous matrixes throughout the service life of the repository.

Andra is maintaining a dialogue with waste producers with a view to defining an initial version of specifications for the primary packages that must be disposed of. The Board recommends that Andra should be able to intervene at the earliest possible stage of the production process.

There are a number of differences in the estimation of the cost of Cigéo formulated by Andra, on the one hand, and by the producers, on the other, which are seeking to take immediate advantage of opportunities to reduce the disposal costs. The Board reasserts its concern for expenditure to be thoroughly evaluated in a conservative way and repeats its warning that safety-related costs should never to be spared for the sake of budgetary savings.

LLLLW OR WASTE WITH AUGMENTED NATURAL RADIOACTIVITY

Andra has begun exploratory research on characterising a possible LLLLW storage site within the Soulaines community of municipalities in the département of Aube, where clay series formations could allow for the creation of a repository under a reworked cover (stockage sous couverture remaniée – SCR). The Board recommends continuing the geological explorations and studying the possible environmental impacts of an LLLLW SCR repository. In particular, studies of repository evolution scenarios under different redox conditions concerning the environment must be conducted to take account of any transfer of elements into the geohydrologic system. The Board reiterates that the SCR design remains fragile for the disposal of long-lived radionuclides, as at first glance, it does not provide the guarantees offered by disposal under an untouched cover (stockage sous couverture intacte – SCI).

Areva is studying the possibility of the in situ disposal of the wastes produced by the Comurhex plant at Malvési, which are mainly waste with augmented natural radioactivity. The Board recommends continuing the work on characterising the site while specifying its containment properties and monitoring the evolution of the mineralogy of the historical waste and its behaviour with regard to leaching, as is the case for the uranium ore processing residues stored on mining sites.

REACTOR DISMANTLING

Reactor dismantling is governed by a regulatory framework in which each operator is responsible for all of the operations required for the eventual decommissioning of the site. The decommissioning and dismantling programmes currently in progress concern nine reactors shut down between 1973 (Chinon A1) and 1998 (Creys-Malville) and one uranium enrichment plant (Georges Besse 1).

The Board takes note of EDF's strategy which, beyond the dismantling of first-generation reactors, seeks to minimise the volume of waste from the dismantling of PWRs in the current nuclear fleet. The studies to identify and optimise the management of VLLW (Very Low-Level Waste) from the PWRs must be consolidated and intensified. In view of the large volumes of waste to come, the first key issue is to compile predictive disposal inventories that are as realistic as possible. The recycling and recovery of this waste must be envisaged. Considering the currently low VLLW management capacity in relation to the quantities to come, the Board requests the setting up of an industrial scheme that is capable of meeting the needs. It recommends continuing the research into possible optimisation opportunities and the R&D required to evaluate the environmental harm that could arise from such activities, especially that inherent to the transportation of waste.

PARTITIONING AND TRANSMUTATION

The act of 2006 states that research on the partitioning and transmutation of long-lived radioactive elements must be combined with the research on new-generation nuclear reactors. The CEA (Commissariat à l'énergie atomique et aux énergies alternatives – French atomic energy agency) was appointed as the developer and project owner of the "Astrid" technological demonstrator for a fourth-generation, sodium-cooled fast-neutron reactor (FNR), which uses plutonium and depleted uranium as its fuel.

Astrid must demonstrate that a certain number of innovations will ensure a higher safety level than that of third-generation reactors and integrate the lessons learned from Fukushima. The sodium-gas energy conversion system coupled with a gas turbine would constitute a major technological breakthrough in relation to previous systems. The Board recommends stepping up R&D on sodium-gas exchangers and coupling Astrid to gas turbines.

The CEA is gradually implementing a major scheme to meet the goals set for the Astrid programme. Constant progress is being made with the detailed preliminary design (DPD) of the reactor. R&D is continuing in the CEA facilities in France and in partnerships with industry at home and abroad. The Board recommends strengthening the links between the partners with a view to obtaining authorisation for construction of the reactor.

The implementation of Astrid must also provide an opportunity to study the transmutation of americium and the consumption of the Pu that remains when the decision is made to shut down an FNR fleet. The high americium-content fuel to be placed in blankets will consist of mixed UAmO2 oxide ceramics, prepared according to the powder metallurgy process like UPuO2. Studies are focusing on the optimisation of this fuel. The Board considers that the CEA should continue to conduct the fundamental analyses of the properties of U-Am oxides and their formation with a view to the subsequent production of the fuel; indeed, these mixed oxides behave differently to U-Pu oxides. The possibility of transmuting Am depends on this process.

INTERNATIONAL PANORAMA

France, Finland and Sweden are the only countries in which significant progress has been made in the process to obtain permission to create deep geological LLHLW repositories.

In Finland, the safety authority – Stuk – has issued a favourable opinion of the project presented by the waste management agency – Posiva – and the government's decision is expected in the next few months. In Sweden, the authorisation process is underway and it will take several years to obtain the opinions of the authorities concerned.

In the United Kingdom and Germany, searches have resumed for suitable sites for the location of geological repositories. Following the inhibition of the Yucca Mountain project in the United States, the dry storage of spent fuel has been adopted as an interim measure.

In 2012, EDRAM (International Association for Environmentally Safe Disposal of Radioactive Materials), which encompasses the majority of bodies responsible for radioactive waste storage, published an evaluation methodology for the cost of a geological repository. It takes account of the magnitude of the nuclear programme of the country in question, the storage depth, the characteristics and time required for the cooling of the waste, the type of packaging used and whether or not retrievability needs to be envisaged in the design of the repository. Despite the disparities among national situations, the majority of the estimations conclude that the cost of a geological repository only amounts to a few percent of the electricity production cost.

The methods used to finance geological repositories vary from one country to another. Three approaches are primarily used: provisions set aside on the producers' balance sheets, payments into a dedicated internal fund and payments into a dedicated external fund. A variety of methods are used to constitute provisions and supply funds. In Sweden, for example, the charges are paid into an external fund which places them in government obligations. The total amount of the charges is updated every three years with the authorities' consent after the costs have been updated. In Germany, the producers pay into dedicated internal funds that they can use for their investments.

The activity sector covering the dismantling of nuclear facilities and the remediation of sites is becoming increasingly important. These activities are governed by criteria and conditions that define whether sites are considered to have been rehabilitated and are thus available for non-nuclear activities. They are also governed by rules that state whether decontaminated equipment can be released or decommissioned and considered to be non-radioactive. There are significant variations in the regulations and practices from one country to another due to the lack of international harmonisation.

Finally, in March 2015, the Savannah River national laboratory in the United States released the report on the incident at WIPP (Waste Isolation Pilot Plant, Chihuahua Desert in New Mexico). Its conclusion can be found in chapter 6. This incident emphasises the critical role of quality assurance throughout the entire waste package production chain.

INTRODUCTION

The period from September 2014 to May 2015 was the Board's 8th full year in operation. This report (Report no. 9) consists of an evaluation of the research that has been presented to the Board during this period. As in previous years, a large proportion of this report is devoted to examining nuclear waste issues, monitoring the Cigéo disposal project which concerns waste registered on the French industrial programme for waste management (plan industriel de gestion des déchets - PIGD) and monitoring the Astrid programme, which focuses on the building of a prototype 4thGeneration fast-neutron reactor (FNR) and research on the transmutation of americium.

Chapter 1 is devoted to Cigéo. The Board takes note that the construction application (DAC) for Cigéo will not be filed in 2015, despite this being a requirement of the act of 28 June 2006. The delay is due to Andra's decision to take account of the conclusions of the public debate and the need to consolidate the Construction application file. Cigéo is a project conducted on a hitherto unequalled scale, which relies on numerous innovative technologies. The new deadline for the filing of the BA has been set for 2017. The Board sincerely hopes that this schedule will be respected in order to clarify the management of LLHLW and LLILW. In this chapter, the Board evaluates the changes made to the project since last year and the research on LLILW packages. On this subject, the Board is pleased to note that several of its previous observations and recommendations, especially on the fire endurance of bituminous waste packages, have led to further research that will be useful for analysing the safety of Cigéo when in service.

However, the underground storage of LLHW and LLILW is only one of the problems posed by the management of wastes in the downstream part of the nuclear fuel cycle. That is why, in chapters 2 and 3, the Board investigates the management of LLLLW and waste with Augmented Natural Radioactivity, which is less active but exists in large quantities. This management requires the creation of new storage infrastructures whose locations must be analysed with great care.

The dismantling of decommissioned facilities is also a major problem for which the studies and deliberations are at a very early stage. The Board devotes chapter 4 to this issue. Certain members of the French parliamentary office for the evaluation of scientific and technological options (Office parlementaire d'évaluation des choix scientifiques et technologiques – OPECST) have requested the immediate launch of a search for resources to implement a coherent and effective global policy in this field. The Board will be closely monitoring the essential studies and research to be carried out by the responsible bodies.

Before commenting on the key events in the international overview (chapter 6), the Board devotes chapter 5 to the "Astrid" industrial demonstrator project for a Generation IV reactor. Astrid differs in numerous respects from the FNRs built to date in France and abroad. The innovations must be successfully implemented as they should guarantee greater safety in relation to Generation III reactors. The Board has taken note of the public authorities' announcement of their willingness to continue the research on FNRs capable of the multi-recycling of plutonium and making better use of uranium resources. These are currently the only reactors with the potential to transmute certain minor actinides contained in long-lived waste.

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Since the publication of its previous report in June 2014, the Board presented its Report no. 8 to OPECST and to the relevant ministerial departments. A delegation from the Board visited Tréveray on 5 February to present its report to members of the CLIS (local information and monitoring committee) at the Bure laboratory.

The Board (cf. Appendix I) followed the same working methodology as in previous years. It conducted eight day-long hearings (cf. Appendix II), and six other shorter half-day hearings, all held in Paris, in addition to a certain number of additional meetings with legal specialists. The Board members – all volunteers – heard 83 people from Andra and the CEA, as well as from French and foreign academic institutions and industrial organisations (cf. Appendix III). These hearings brought together around sixty people on average and were also attended by representatives of the French Nuclear Safety Authority, Areva, EDF, the French Radioprotection and Nuclear Safety Institute, the central administration and OPECST.

The Board devoted a whole day to visiting the Lagunes and COMURHEX II facilities at the Areva Malvési site (cf. Appendix IV).

To prepare this report, the Board held a 2-day pre-seminar during its visit to the Areva sites at Le Creusot and Saint-Marcel near Chalon-sur-Saône. It also held numerous internal meetings, including a five-day residential seminar. A list of the Board's hearings and visits can be found in appendix II of this report. A list of documents received from the organisations that participated in the hearings is provided in Appendix V.

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Chapter 1
CIGEO

The purpose of the Cigéo project, by application of the Act of July 2006, is to design and build a geological repository for LLHLW and LLILW waste from the French industrial waste management programme (Programme industriel de gestion des déchets - PIGD). This repository should be created at a depth of 500 m, in the approximately 130 m thick Callovo-Oxfordian (COx) argillite formation found in the Area of interest for in-depth exploration (Zira) identified by Andra in the Meuse-Haute Marne area. This project has emerged from studies and research carried out over a period of more than twenty years, which demonstrated the excellent ability of the COx formation to contain the radionuclides in the waste.

Assisted by its systems project manager Gaya, Andra – acting in the capacity of project owner – is currently carrying out preliminary design studies prior to filing the construction application (DAC), which must be submitted to the government in 2017.

1.1 CIGÉO PROJECT SCHEDULE

The initial schedule determined by the act of 2006 provided for the filing of the DAC in 2015, prior to an act establishing the reversibility conditions in 2016. An updated DAC taking account of the provisions of this new act was then supposed to follow in 2017. The permit to start construction of the facility was thus expected to be announced by decree in 2018.

This schedule was revised following the public debate held between May and December 2013. To take account of the opinions and expectations expressed in the debate and consolidate the scientific case, Andra's Board of Directors decided in May 2014 to continue with the Cigéo project according to the following general approach:

- integration of an industrial pilot phase at the start of operations at the facility
- establishment of a master operating plan for disposal that can be regularly revised according to the feedback received
- involvement of civil society in the project.

Andra's priority concerns of guaranteeing safety, controlling costs and protecting and developing the host area remain unchanged.

This process will be carried out in stages according to a revised schedule that now envisages preparing for the DAC in the two following phases:

- First phase:
  - After the outline phase that had led to the establishment of a waste repository layout in consultation with manufacturers and producers in 2012 and 2013, the basic preliminary design (BPD) began in November 2013 and is scheduled to continue until mid-2015;
  - A project review by mid-2015, sponsored by the Directorate General for Energy and Climate (Direction générale de l'énergie et du climat – DGEC), and carried out by experts, will verify that the technical requirements for the transition to the detailed preliminary draft (DPD) phase have been met;

- A technical report comprising a safety orientation file (dossier d'orientation de sûreté) and a technical retrievability options file (dossier d'options techniques de récupérabilité), along with an operating master plan (plan directeur d'exploitation) and a preliminary impact assessment (étude d'impact préliminaire), in support of the declaration of public utility (déclaration d'utilité publique) procedure, shall be submitted to the French Nuclear Safety Authority (Autorité de sûreté nucléaire – ASN) at the end of 2015.

- Second phase:
  - Launch of the DPD in the second half of 2015, which will continue until the end of the first quarter of 2017. It will incorporate the results of the BPD studies and feedback from the evaluators.
  - Filing of the DAC at the end of 2017 with a view to obtaining the decree of authorisation to create the facility (décret d'autorisation de création) by 2020.

An act establishing the reversibility conditions must be passed before the authorisation to create the facility can be obtained.

This schedule is entended to be compatible with the launch of the industrial pilot phase in 2025.

For the filing of the DAC, the Board considers that this new schedule must allow for the consolidation of the technical options that have been selected by Andra. They will incorporate the levels of optimisation that emerged from the discussions with waste producers. However, the Board notes that this schedule, which no longer conforms to the act of 2006, remains very tight and all stakeholders in the project will need to adopt a very rigorous approach if the launch of the industrial pilot phase is to be ensured in 2025.

1.2 THE DESIGN OF CIGÉO, A REFERENCE SOLUTION

The optimisation studies developed by Andra since the outline phase must lead to a reference solution for Cigéo that will be submitted in support of the construction application. For this configuration, Andra intends to select only those options for which it has sufficiently robust scientific and technical justifications at the moment of filing the DAC. This means that there will still be opportunities for improvements and that they can be progressively incorporated into the project development plan, which provides for testing in the underground laboratory and at Cigéo during the industrial pilot phase. Andra has made sure that the architecture of the reference configuration is compatible with the implementation of the opportunities that it is currently planning to study.

As in the outline phase, the underground facility will consist of three main repository areas (cf. Diagram):

- An area for the disposal of HLW0 (high-level, weakly exothermic vitrified waste generating between 30 and 193 W per package) that producers will be in a position to deliver before 2040. This area will act as a pilot zone for all HLW. The length of the blind disposal cells in this area is set at 80 m, in accordance with the expertise acquired in the underground laboratory; 75 cells with centre-to-centre distances of between 8.5 m and 51 m will be required.

- An area for the disposal of LLILW, which will consist of 50 cells with an effective length of 500 m and a diameter of 9 m (excavated section of 65 m^2). After optimisation, these disposal cells will receive 7 models of storage containers, most of which can be stored in three columns and on two levels, with the remainder in one column and on one level. The LLILW cells will always have an open-ended design to allow for their ventilation and the filtration of the return air. Andra envisages developing the LLILW area in two phases to reduce the initial investment and take account of feedback from the initial phase; therefore, only four disposal cells and their access tunnels are planned for the industrial pilot phase with the rest being constructed over time.

- An area for the disposal of highly exothermic (275 to 448 W per package) HL (high-level) waste (HL1 and HL2) which may not begin until after 2075. 1,473 blind cells with centre-to-centre distances of 33 m to 51 m will be required. To continue optimising the disposal of the containers and thus reducing the costs, Andra envisages a cell length of 100 m in anticipation of the results of the testing programme scheduled to take place in the underground laboratory between now and the filing of the DAC.

The surface-underground links will arrive at the disposal level in a consolidated area intended for logistical support, which is designed in two parts to separate the operational activities from the works activities. The service facilities will be equipped with five shafts – two for the operational area and three for the works area. These shafts will be dedicated to ventilation and the movements of equipment and staff. An access ramp consisting of two 8 m diameter tubes will ensure the independent transfer of the waste packages and service (maintenance and emergency) functions by funicular railway.

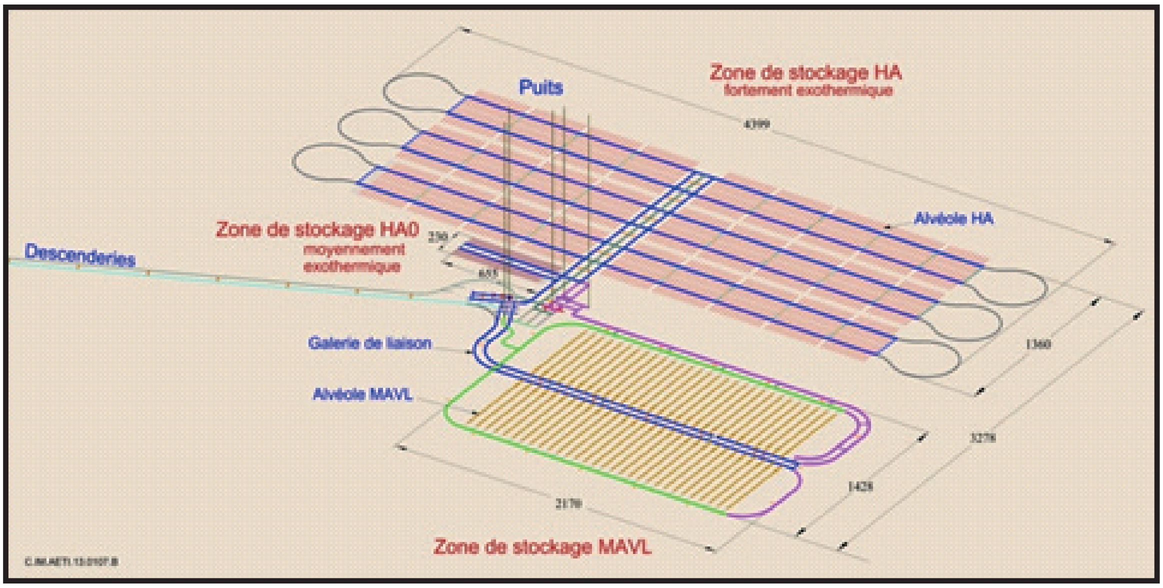
The surface facilities currently incorporate the new delivery chronology for the packages originating from version D of the PIGD. The anticipated maximum annual flow of primary packages has been reduced to 3,000 packages per year at the mid-term period, between 2049 and 2075. The initial phase of the facilities which will run until 2052 – referred to as EP1 – will receive deliveries of HLW0 and LLILW packages in their transport packaging, all to be unloaded vertically. After this date, a second phase – EP2 – will be built to accommodate horizontally unloaded packages, and will be dedicated to HL1 and HL2 packages in particular.

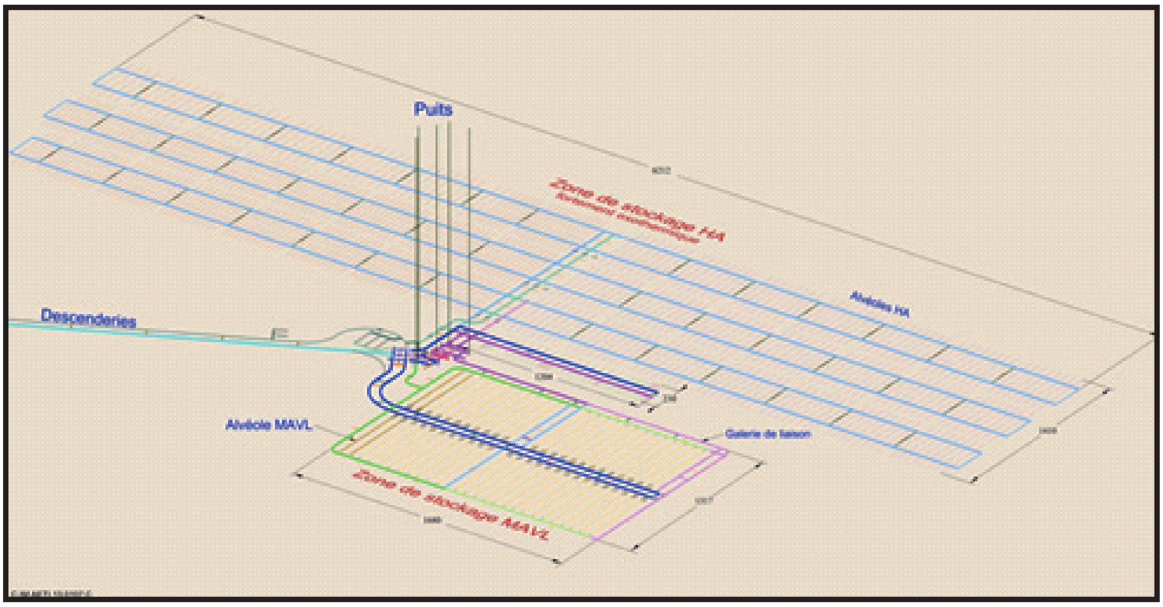
With regard to underground facilities, the main differences between the configuration proposed at the end of the outline phase and the BPD are listed below (Cf. Diagram):

- the gradual creation of the LLILW zone has led to the postponement of the direct exploration of a large part of the Zira until after the industrial pilot phase;

- the location of the HLW0 area between the HLW zone and the LLILW zone now allows for a better use of the space;

- a significant change to the architecture of the HLW zone characterised by the extension of the cells from 80 to 100 m and an increase in the centre-to-centre distance (33 to 51 m compared to 25 m in the outline). This has intensified the development of the area occupied by the HL zone which now extends almost to the south-western and north-western boundaries of the Zira. According to Andra, this adaptation was required to satisfy the hydro-thermo-mechanical criteria after refining the parameters of the bedrock by taking account of the new disposal cell lengths and a change to their loading plan. The hydro-thermo-mechanical aspects now prove to be a key aspect of the dimensioning of the HL areas; they are analysed in greater detail in the next sub-chapter and in Appendix VI.





Configuration of underground facilities: comparison of the outline (above) and BPD (below) phases, according to Andra documents

The Board considers that Andra has specified the options required for the management of the preliminary design which should lead to the filing of the DAC in 2017. With regard to the LLILW zone, it regrets that Andra has not adopted the recommendation, made in the Board's Report no. 8, to undertake an in-depth exploration of the future zone by extending the central tunnel in the industrial pilot phase. The Board is also notes the significant increase in the area occupied by the HL zone within the Zira; it is concerned that the current configuration extends to the limits of the Zira.

1.3 DIMENSIONING OF HL AREAS AND THE THERMO-HYDRO-MECHANICAL MODEL

Since the 2009 File, significant modifications have been made to the dimensioning of the HL disposal cells, during the outline phase (2012-2013) and again in the basic preliminary design phase (2014-2015).

With a view to significantly reducing the number and length of access tunnels and the number of disposal cells – and thus the costs – Andra, in consultation with industrialists and waste producers, has proposed a longer surface storage period for HL packages and an increase in the unit length of the cells.

Due to this new design, and also to the adjustment of the parameters in light of new experimental data, a reassessment of the hydro-thermo-mechanical behaviour of the HL area was required. Two types of parameters are primarily concerned by this revision. The estimation of the permeability of the COx has been adjusted downwards on the basis of a greater number of experiments permitting a more robust interpretation. This evolution is favourable from the perspective of the radionuclide containment of the bedrock. On the other hand, it is unfavourable in terms of the mechanical behaviour insofar as it facilitates the occurrence of pressure surges in the rock pores in the event of a rise in the bedrock temperature. The estimation of Young's modulus has been revised upwards; the rock thus becomes stiffer and is less likely to accommodate pressure surges of fluid. The impact of excessive water pressure is to cause effective local tensile stresses in the rock, capable of causing damage due to micro-cracking. The criterion of zero thermo-hydro-mechanical micro-cracking of the bedrock (THM criterion), has thus taken precedence over the criterion of rock temperature being below 90°C in contact with the packages, which appeared to be a critical design criterion in previous studies. Ensuring the verification of this criterion is not a simple task, as it requires more parameters, some of which are subject to greater uncertainty. According to its calculations, Andra estimates that the risk of micro-cracking reaches a critical point in the disposal area at the level of the centre-to-centre distance between the disposal cells. To ensure compliance with the THM criterion, the spacing between the cells must therefore be significantly increased. By combining compliance with the criterion and the concern for minimising the length of the access tunnels, the total surface area of the HL area is increased in relation to the previous designs and the HL area becomes oblong. At its longest point, it reaches the boundary of the Zira. The creation of the loops required for the change of direction of a tunnel boring machine then becomes problematic within the area occupied by the Zira.

Uncertainties over the values of the parameters still remain, whose repercussions on thermo-hydro-mechanical behaviour have not been fully quantified. Moreover, the consequences of a violation of the THM criterion and the possible appearance of micro-cracking have not yet been accurately measured. If this situation occurs, it must be analysed with a view to performing a more comprehensive safety assessment.

The Board observes that the studies of thermo-hydro-mechanical behaviour continued during the implementation of the optimisation process for the HL area. It notes that Andra was led to make significant modifications to the configuration of the HL area and its location within the Zira. It observes that uncertainties remain over the definition of certain parameters and that research and analyses are still required to further improve knowledge of the thermo-hydro-mechanical behaviour of the repository, improve the definition of the extension of areas in which the criterion of zero fracturing of the COx is likely to be violated and assess the consequences of such a violation for safety, taking account of the sensitivity to the parameters.

The Board considers that in the DAC for 2017, Andra must opt for a layout for the HL area zone that is conservative enough to allow for the disposal of all HL waste from the PIGD in compliance with the safety options. In a subsequent phase, as knowledge improves, optimisations could be proposed with a view to ensuring the more rational use of the underground space.

The Board once again - strongly recommends that for such complex topics, Andra should submit the studies for publication in scientific papers and journals with independent review committees. This could stimulate a debate within the scientific community concerned by these issues. At the end of this process, the credibility of the dimensioning could be strengthened.

1.4 OPPORTUNITIES FOR THE LAYOUT TO EVOLVE

Andra has identified possible ways in which the reference layout may evolve. They are referred to as opportunities insofar as they cannot be adopted on the basis of the results of robust studies for the filing of the DAC. Most of them cannot be evaluated prior to the industrial pilot phase.

Two LLILW disposal opportunities are envisaged:

- the construction of disposal cells with a large cross-section (diameter of 12 m compared to 9 m). This configuration would allow for the arrangement of packages in 3 columns and on 3 levels, thus reducing the number of cells from 50 to 36. Its feasibility relies on the package handling capacity and on the consequences for the extension of the damaged area that might result from it. It will be tested at Cigéo in the industrial pilot phase;

- the direct storage of certain primary packages without a storage container. This can only be envisaged for certain packages whose chemical, thermal and radiolytic characteristics and the risk of criticality must be specified. The reduction in terms of the number of disposal cells has not yet been evaluated.

There are several potential solutions for the disposal of HL waste:

- a better use of the underground space that could arise from a reassessment of the thermo-hydro-mechanical dimensioning based on an improvement in the modelling and a better understanding of the risks of damage to the COx;

- the extension of the disposal cells up to a length of 150 m; the resulting reduction in the number of cells would lead to a 13 km reduction in the length of the access tunnels. The feasibility of the excavation could be examined in the industrial pilot phase;

- an extension of the HL waste storage period by 20 years; this configuration would postpone the package disposal period from 2079 to 2099, thus reducing the thermal load, which would allow for a 10% reduction in the number of disposal cells required;

- a reduction in the cost of the overpack for the HL primary packages by reducing its thickness; this configuration will depend on studies of radiolytic corrosion which are currently in progress;

- the use of vitrified LLILW packages in the same format as the HL waste packages inserted as spacers among the HL waste packages.

The Board considers that the search for opportunities to improve the operation and cost of a facility that is intended for long-term service is a pertinent approach. It repeats its recommendation that the implementation of these opportunities should not, under any circumstances, be associated with a reduction in the safety of the repository, both while in service and in the long term, and that all of their impacts on Cigéo's operations should be fully evaluated. This comment is particularly pertinent to the 20-year extension of the storage period prior to disposal.

1.5 THE CIGÉO SCIENTIFIC SUPPORT PROGRAMME

The Cigéo development plan provides for a scientific programme that is specific to Andra with the aim of:

- specifying the design options,

- evaluating the project managers' technical proposals,

- developing implementation techniques,

- acquiring additional validation and demonstration items.

Andra has divided these analysis and development activities into two phases: one phase that ends in 2029 with the aim of qualifying the final system and a second phase that ends in 2034 with the aim of verifying the performance of the repository under actual conditions. This final phase will be the main focus of the industrial pilot phase which should last for around ten years after the commissioning of the first Cigéo facilities, scheduled to take place in about 2025.

The programming of the work must allow for the execution of the different components of the structure:

- LLILW zone: LLILW cell, LLILW tunnels, LLILW areas;
- HL zone: HL container, HL cells, HL areas;
- surface-underground links: shafts, access ramps, tunnels;
- surface facilities;
- closing structures.

The Board became aware of aspects of this scientific programme during its hearings.

Andra places the emphasis of its approach on research of a technological nature (overpacking and handling of LLILW packages, construction of LLILW tunnels, creation of HL disposal cells including the process for applying the filling grout to the extrados of the lining, etc.). It also considers studies of a scientific nature such as those concerning the corrosion of HL overpacks and the thermo-hydro-mechanical studies mentioned previously.

It appeared to the Board that these activities mainly concerned questions that needed to be answered for the drafting of the DAC. This is perfectly natural as the DAC must present the strongest possible solutions for all of Cigéo's functional requirements, while not excluding subsequent developments if their relevance should be supported by scientific studies.

The Board recommends that Andra should specify the scientific and technical critical design elements for the project at a sufficiently early stage of its development plan, prior to the filing of the DAC, while distinguishing between the elements that relate to the reference solution and those that concern opportunities.

Cigéo is expected to operate for a period of more than one hundred years. Unforeseen events or decisions relating to the reversibility of the disposal could lead to the extension of this period. This means that the shafts, access ramps and access tunnels to the disposal cells will need to be kept in good condition for a hundred years or so. It must be possible to retrieve the waste disposal packages (or any primary waste packages that may be stored asthey are) and they must remain in good condition for as long as possible. Indeed, they constitute the primary radioactivity confinement barrier. The repository seals must maintain the low level of permeability upon which the safety analysis is based for a period of several millennia The qualities of the tunnel coating/ support materials (concrete), packages (concrete and steel) and seals (concrete and swelling clay) must be adapted to these different time frames, given that the structural elements cannot be repaired, especially after the closure of the repository.

Andra is continuing with R&D on the different materials with a view to defining their behavioural laws and their mechanical strength, and on their degradation/corrosion under the disposal conditions: significant anisotropic stresses, contacts between materials and with the COx argillite. This R&D also seeks to evaluate the mechanical strength of the materials under accidental situations such as a fire. This R&D focuses in particular on different types of concrete and cementitious materials, the clay used for sealing cores and steel. A more detailed description can be found in Appendix VII.

The properties of the materials used to build Cigéo and ensure the safety of the repository while in service and after its closure have been studied since the origins of the concept of geological disposal. Knowledge of these materials is essential both at a fundamental level and in the context of their use for civil engineering. The R&D on materials, to be conducted by Andra prior to the filing of the DAC, must consolidate the parameters governing their evolution, in order to ensure the credibility of the models and simulations for Cigéo. The Board considers that this research is, at present, being correctly undertaken in order to validate the performance of the repository components under operational conditions and reach level 6 of the International Technology Readiness Level (TRL) scale.

Andra has demonstrated the excellent containment capacity of the COx. However, the recent experience acquired during the BPD regarding the thermo-hydro-mechanical behaviour of the rock has shown that the estimations of the values of important parameters can still be further refined and may change in ways that will have either a positive or negative impact on the performance of the repository.

The Board recommends that the development programme should define the nature of the mechanisms governing the behaviour of the COx and the associated parameters that have not yet been specified. Andra must rank the required studies and research in order of priority, while differentiating between the pre-DAC and post-DAC periods. It repeats its recommendation that Andra should propose a programme of exploration of the geological horizons through which the underground structures will pass.

1.6 LLILW

Long-lived intermediate-level waste (LLILW) occurs in highly numerous and varied forms. For Cigéo's operational needs, the LLILW packages from the French industrial programme for waste management (PIGD) are classified according to 79 families (see Appendix VII). At the Board's request, Andra compiles an operating report containing the characteristics of all package families that have been previously produced, are in process or awaiting production. The families are divided into five categories: LLILW1 to LLILW5. For categories LLILW1 to LLILW3, only waste belonging to the same category can be placed in the same cell. In principle, packages in the LLILW4 and LLILW5 categories can be placed in the same cell.

The characteristics of the packages in storage which are ready to be sent to Cigéo during the first operating period are already known and conform to the guidelines for the operating specifications. These are primarily packages stored at La Hague.

R&D is, however, being conducted for other packages in storage with a view to consolidating or supplementing the knowledge of their behaviour, either while in service or in the long term after the deterioration of the packages. These are packages for which there are doubts concerning their robustness in response to extreme events (bitumen) or their long-term behaviour (saline and organic).

A proportion of the LLILW waste has not yet been packaged and studies are in progress to define the best packaging process.

1.6.1 Bitumen encapsulated waste

a) Self-ignition and fire endurance

In its report no. 6 (November 2012), the Board had asked waste producers and Andra to study the behaviour of packages of bituminous sludge under full-scale conditions during a fire that was representative of the operating conditions at Cigéo, and to submit the results to the Board at the end of 2014. The programme of experiments on resistance to a rise in temperature was viewed in a positive light by the Board in 2013. An analysis of the results is presented in Appendix IX.

The CEA, Areva, EDF and Andra (in collaboration with universities) submitted the documents describing the experiments, the results and their interpretation to the Board within the allotted time, along with further information about the swelling of packages.

> *The Board considers that the experiments covered the entire range of composition of the bituminous matrixes that might be disposed of at Cigéo.*

Additional studies (T < 300°C) on the behaviour of bituminous matrixes at the gramme scale and at the 2 kg scale confirm the role of nitrates in exothermic reactions but do not reveal the occurrence of any runaway reactions. Under these conditions, there is no self-ignition. The bitumen used for packaging the matrixes has an ignition temperature of around 250°C. It is only self-igniting at temperatures of significantly above 300°C.

Full-scale experiments were conducted by placing disposal packages containing four primary primary packages of bituminous matrixes in a furnace at a temperature of 950°C for one hour. The disposal packages were only altered on the surface and then only locally; the primary packages remained intact. There were no traces of reactions on the surface of the bituminous matrixes. The disposal packages could be handled after the tests.

Finally, two tests, under the conditions equating to an actual fire burning for several hours at full strength until completely extinguished, were carried out with packages equipped with measuring instruments in an enclosure simulating a disposal cell. The temperature of the concrete of the storage packages reached a maximum of 600-650°C on the side exposed to the fire and the temperature of the primary packages did not exceed 150°C. After the tests, the primary packages remained intact and the storage packages could still be handled. Once again, there was little spalling of the BTH concrete containing polypropylene fibres.

The results of the varied tests are consistent and can be repeated by modelling. The test conditions broadly cover the characteristics of a fire in a LLILW disposal cell.

b) Resistance to radiolysis and swelling

The volume of hydrogen produced by the radiolysis of bitumen cannot generate swelling that is likely to damage the disposal packages, which are also permeable to hydrogen diffusion. The CEA also demonstrated that the swelling of primary packages following the uptake of water by the matrixes had no effect on the disposal package.

c) Conclusion

> *The Board considers that the studies of the resistance of packages of bituminous sludge during a major fire demonstrate the strength of the disposal packages and the chemical inertia of the bituminous matrixes during an increase in temperature of up to 300°C.*
>
> *The Board suggests making optimal use of the experimental data obtained during testing by performing inter-comparisons of digital simulations.*
>
> *The new data on the full-scale tests diminishes the fears relating to fires of external origin to the packages in Cigéo facilities and confirm their retrievability following such fires.*
>
> *The Board considers that the CEA, Areva, EDF and Andra have acquired the information required for the drafting of the safety analysis that must be submitted to the Board pursuant to its request in report no. 6.*

Furthermore, the Board requests the continuation of the studies relating to the chemical stability of the bituminous matrixes throughout the operating period.

1.6.2 Pyrophoric, saline and organic waste

a) Reactive or pyrophoric waste

"Reactive" metals mainly concern FNR control rods consisting of needles of boron carbide (B4C) containing 40 to 100 g of residual sodium per needle after treatment with water vapour. With a view to their disposal under clearly defined conditions, the CEA is carrying out studies in order to reduce the amounts of sodium substantially by using "in situ" sodium carbonation. It envisages a primary package consisting of these needles embedded in sand, with the entire assembly placed in a 1.5 m^3 stainless steel barrel. All of the needles constitute 8 packages of sodium-containing waste, which corresponds to 2 disposal packages and approximately 35 kg of sodium.

"Pyrophoric" waste refers to materials (metals or alloys) in relatively separated form that react almost spontaneously with the oxygen in the air. An inert, non-flammable matrix is used to insulate them and thus prevent the occurrence of such an oxidation. Alternatively, they may be compacted by the producers. The R&D developed by Andra aims to formulate an inert binder adapted to each metal, thus minimising the H_2O-Na reaction kinetics and hydrogen production in porous environments during the in-service phase.

LLILW magnesium waste consists of the metal fuel cladding of UNGG reactors in a Mg-Zr or Mg-Mn alloy containing traces of uranium metal. They are stored at Marcoule. The CEA is conducting studies on their recovery and the more comprehensive characterisations currently being carried out should allow for the validation of this type of package with very good guarantees of resistance in storage and disposal. 7,500 220 L packages are expected.

Aluminium or aluminium alloy LLILW are much less common structural waste items which are less pyrophoric than magnesium waste. They are stored at Marcoule. Research is being conducted into a new phospho-magnesium, boric acid and lithium nitrate-based matrix. A binder with such a composition inhibits the production of hydrogen through the passivation of aluminium. 1,300 380 L packages are expected.

The zircaloy fines originating from the cropping of needles in the assemblies are compacted and placed in CSD-C packages. This rules out any risk of reaction.

b) Saline waste packages

The current saline waste packages contain sludge and residues from the CEA's STEL evaporators (concentrates). This waste is coated in a hydraulic binder and placed in 220 L steel barrels for a total of 8,500 packages. To this can be added the packages of co-precipitation sludge from STE2 stored in silos at La Hague. Areva has designed and characterised a new package, referred to as "C5" which, after storage, will be sent to Cigéo (14,500 packages). The qualification file is ready for submission to the French Nuclear Safety Authority (ASN).

c) Organic waste packages

This waste originates from Areva (La Hague and Melox) and the CEA (civil and DAM). It amounts to approximately 40,000 packages (11,400 of which are highly alpha-contaminated). They contain a total of 3,600 tonnes of cellulose and standard polymer-based organic materials.

Two important topics are still being studied:

- gases produced by radiolysis,
- production of chelating organic molecules resulting from radiolysis.

d) Conclusion

The majority of the LLILW packages produced in line or pending delivery to Cigéo conform to the preliminary specifications which must be converted into obligations or requirements in the definitive disposal specifications.

The recovery of unpackaged waste requires the development of new packaging that is adapted to each waste family. It must be designed to meet Andra's requirements, especially:

- *increased resistance to predictable physical and chemical stresses,*
- *minimal hydrogen emissions,*
- *minimal leaching of radionuclides when the waste is in contact with water.*

For the past ten years or so, Andra and the waste producers, in collaboration with their partners at the CNRS and universities, have been conducting research into the production of new packages. The results currently obtained for waste packages containing reactive metals, soluble salts and organic matter show that hydrogen production can be significantly reduced through the use of appropriate packaging and, with regard to the release of radionuclides, it is possible to reduce the source term substantially.

The Board recommends that R&D into saline and organic waste packages should be continued to obtain even more robust evidence that their disposal cannot have an impact on safety in the long term.

1.6.3 Co-disposal of LLILW

The problem of the possible co-disposal of families of different packages in the same disposal cell was covered in Report no. 8. It is associated with the optimised filling of cells, which seeks to minimise the number of LLILW cells required at Cigéo.

There is concern that once the packages have degraded, chelating organic molecules will come into contact with radionuclides from different packages. This interaction could modify the containment and the migration of elements and of actinides in particular. Studies of this issue are continuing and involve modelling the dispersion of molecules and radionuclides from packages, on the one hand, and studies of the chelation of the actinides by thwese molecules, based on the exploitation and acquisition of thermodynamic data, on the other.

The Board considers that the characterisation of the behaviour of species that could be released by packages co-disposed of in the same disposal cell should be consolidated.

1.7 CIGÉO ACCEPTANCE CRITERIA

The package disposal process at Cigéo is based on the waste package acceptance specifications, the monitoring of compliance with these specifications and the acceptance accreditations contracted by Andra and the waste producers. This process is described in detail in Appendix VIII.

The specifications for the acceptance of waste packages at Cigéo define the characteristics and performance levels required of waste packages for acceptance into this repository.

Andra has been engaged in a dialogue with waste producers since 2012 with a view to defining an initial version of specifications for the primary packages that must be placed in disposal packages. These specifications must be finalised by mid-2015 to coincide with the end of the BPD, and then transmitted to the ASN with the safety options file in September 2015. The definitive specifications shall be defined at the time of the commissioning application for the facility.

The long process of defining the specifications consists of taking account of the characteristics of packages defined by the producers (and by Andra for the long-term properties) and comparing them with the possibilities for a reversible disposal design in the COx, which is Andra's responsibility. There are numerous problems to be resolved: containment of radionuclides in operation, examination of dispensations for several packages that do not conform to a requirement, the situation concerning historical packages, accreditations, inspections and the quest for an optimal technical and economic balance.

In its previous reports, the Board has drawn attention to the need to possess draft specifications at the earliest possible stage, which define the essential packaging guidelines and the expected requirements for the packages.

The Board takes note of the provisions implemented to produce the acceptance specifications for primary waste packages at Cigéo, especially the LLILW waste packages, which will be the first for disposal.

The discussion process currently in progress should lead to specifications for all of the packages in the French industrial programme for waste management (PIGD) provided that we continue to further our knowledge of certain packages and that certain packages are recovered.

The Board wishes to know which of the specification criteria could possibly bring the discussions between Andra and the producers to a standstill and which waste families could be to blame.

The Board is particularly attentive to any dispensations that might be granted. It wishes to know which families of packages might be concerned.

It is essential to monitor compliance with the specifications because this guarantees that packages are compliant for disposal.

The Board notes that the chain of inspections for which Andra will be responsible in order to ensure compliance with the specifications has not yet been fully defined. It involves several stakeholders including the waste producers.

The Board considers that as they currently stand, the division of responsibilities between Andra and the producers and the procedures for their transfer are not sufficiently clear. It recommends that Andra should be able to intervene at the earliest possible stage of the production process.

In general, the Board recommends clarifying the entire acceptance process for packages at Cigéo. An analysis of the causes of the WIPP incident (cf. chapter 6) emphasises the importance of the implementation of an organisational structure that guarantees constant quality assurance throughout the entire service life of the repository.

1.8 COSTS OF CIGÉO

1.8.1 Context

There are still a certain number of remaining "residual" differences between Andra and producers concerning estimates of the cost of Cigéo. The producers must set aside provisions on the basis of this cost. These differences concern the methodology adopted in addition to the value attributed to certain opportunities and certain risks. Consequently, the figure has not yet been made public by the Ministry. Andra is legally responsible for running the project and submitting an estimate of the costs to the Ministry (article L542-12 of the French Environment Code). Thanks to the experience they have acquired, the producers have developed expertise in the costing of major facilities. There has already been a significant convergence of the parties' positions following the meetings of the "Costs" Work Group established by the Directorate General for Energy and Climate (Direction générale de l'énergie et du climat – DGEC).

The gross costs will be estimated according to the current economic conditions and exclusive of taxes (VAT, property taxes and other taxes). The updated costs will be evaluated on the basis of a discount rate of 3%, excluding inflation, which seems acceptable to all stakeholders and conforms to the recommendations made by the "Lebègue" Commission.

The costs of the Bure Laboratory are currently funded by a specific tax on basic nuclear facilities (installations nucléaires de base – INB). They are thus excluded from the calculation of the provisions to which the producers are subject. There should be a clarification of how the situation will evolve in the future, especially if there is a desire to continue the R&D studies, as the Board has already requested.

The Board recommends maintaining the R&D efforts in order to analyse the feasibility of opportunities in discussions and to come closer to a consensus.

The cost of Cigéo must be estimated in compliance with article L594-1 of the French Environment Code: *"Operators of basic nuclear facilities shall conservatively estimate the costs of dismantling their facilities, or, for their radioactive waste disposal facilities, the costs of their final shutdown, maintenance and monitoring. In the same manner, they shall estimate the costs of managing their spent fuels and radioactive waste, taking particular account of the estimation established pursuant to article L. 542-12."* The National Funding Assessment Board (Commission Nationale d'Evaluation du Financement – CNEF) for the costs of managing radioactive waste is responsible for checking that these provisions correspond to the said costs.

The Board reasserts its concern for expenditure to be thoroughly evaluated in a conservative manner and repeats its warning that safety-related costs should never to be spared for the sake of budgetary savings.

In its reports, the French Court of Auditors (Cour des Comptes) has estimated that the cost of the disposal of waste from the French industrial programme for waste management (PIGD) amounts to between 1 and 2% of the nuclear energy production cost.

1.8.2 Estimation of the cost

The Board is pleased to observe that Andra and the producers agree on overall design of the disposal project, in particular with regard to the industrial pilot phase (phase 1). This concerns the excavation of the shafts and access ramp, the development of the "pilot zone" and the initial surface facilities.

As far as the following phases are concerned, it is not surprising to note that it is hard to estimate their cost at present when we are dealing with a facility whose construction and operation will last for at least a century. Given the degree of flexibility inherent to the project, changes will emerge that will lead to new estimations.

In Sweden, for example, the waste disposal project has been subject to annual re-evaluations. Experience has shown that a triennial re-evaluation of disposal costs would be more appropriate.

The Board recommends carrying out regular updates of the cost estimations for Cigéo, on a three-yearly basis, for example.

Such updates would have the benefit of avoiding sudden shocks with regard to budgets and thus the provisions.

This procedure would maintain the dialogue between Andra and the producers while allowing for regular technical and economic optimisations.

The Board, which is responsible for evaluating the studies and research, wishes to be associated with this process, given the importance of the identified opportunities and of their possible conflicts with safety and security.

1.8.3 Open questions: opportunities

For illustrative purposes, the Board shall now present several examples of divergences between Andra and producers regarding costs.

1. With regard to excavations using the tunnel-boring machine, the producers consider the figure for the excavation rate adopted by Andra (3.1 metres per day) to be too low, as prior experience, according to EDF, points to an average rate of nearly 5.5 metres per day being more realistic. The figure chosen clearly has an impact on the cost of the project.

2. The producers propose horseshoe-shaped disposal cells with an excavated section of approximately 100 m^2, allowing for the stacking of LLILW packages on three levels instead of two and meaning that 17 fewer cells (a third of the total number) would be required. Andra has adopted a more conservative and gradual approach (starting at 65 m^2, even if this means subsequently moving onto cells with a bigger cross-section). There are also major financial consequences.

3. With regard to the disposal of HL waste, the producers favour 150 m cells for the filing of the construction application. In light of prior experience, Andra adopts a more conservative view (80 and 100 m). Here too, there are significant financial implications.

4. The producers consider that the 7 mm reduction in the thickness of the HL overpack (not validated by Andra) (decreasing from 65 to 58 mm) would reduce the unit cost of the containers. Furthermore, in view of the number of overpacks (over 50,000), the serial effect must be taken into consideration on the costs, which would be reduced. However, this does not seem to be the case for the Andra estimation.

There are significant financial implications for all of these questions raised by the producers. The choices made for these opportunities must be evaluated from the safety standpoint and the Board must be informed of them.

5. Moreover, there are divergences between the producers (especially EDF) and Andra concerning the delivery chronology for HL waste. The producers have adopted a figure of around twenty years for the postponement in the delivery chronology, i.e. delivery during the 2099-2144 period. Andra acknowledges that the postponement in the delivery of HL waste would allow for the disposal of a greater number of packages per cell (thanks to the additional thermal decay), with a 10% decrease in the number of cells. However, Andra does not consider this solution to be compatible – on the industrial level – with the chronology defined in the PIGD (rapid disposal of LLILW) as it would interrupt the delivery of waste to Cigéo for a 10-year period. The producers (especially EDF) do not share this point of view as they do not consider the suspension of deliveries of packages to be synonymous with an interruption to the operation of the centre, and even less to its activities.

The Board has strong reservations about the producers' optimistic outlook because a great deal of technical expertise can be lost over a 10-year period, as we are currently seeing with the building of new nuclear reactors.

6. The producers are disputing the inclusion of expenditure relating to the observation of the environment, whereas Andra considers this to be part of the costs of the project.

The Board considers that environmental observation is of paramount important throughout the entire service life of Cigéo and thus supports Andra's position.

7. Concerning fire issues, the producers propose the drafting of agreements with local fire brigades, while Andra defines the requirements for specialist teams responsible for fire fighting and also for rescuing casualties and security surveillance.

The Board supports Andra's cautious stance regarding the security of the site and personnel.

- **Conclusion of specific observations**

Certain choices are likely to affect the safety of the repository. The Board wishes to be kept informed of the studies and research carried out in order to compile the scientific documents required to support the safety analysis.

The Board has concerns about the financial impact of the recent modification to the HL area on the cost of the repository.

The Board shall hold regular dedicated hearings to evaluate the scientific documentation resulting from any opportunities that Andra might adopt.

1.8.4 Financing suggestions

One of the key issues of the controversies between waste producers and Andra resides in the fact that establishing the global cost of the project will have a direct and immediate impact on the total amount of provisions that the producers must set aside (Article L594 of the French Environment Code). A significant increase in the cost will automatically lead to a significant increase in the provisions, which in return, could have a big impact on the liquidity position of the companies and, indirectly, on their share price. This is a major issue for Areva in particular, in view of its current financial position.

The Board reiterates that by virtue of the "polluter pays" principle desired by the legislative authorities, the waste producers must remain responsible for their waste, including the funding of its disposal.

The system of provisions is thus an appropriate mechanism, but it does not preclude further consideration of institutional adaptations that could allow for greater flexibility in the management of these provisions. Due to the importance of such an exceptional project as Cigéo, the financing system should reflect the producers' solidarity with regard to its creation.

One suggestion might be the creation of an Economic Interest Grouping (GIE [Groupement d'Intérêt Economique]), for example, that includes all waste producers and could ensure solidarity in the management of the provisions. This GIE, which could hold all of the pre-existing provisions, could also receive additional contributions.

Different financing systems exist at the international level and are presented in Chapter 6.

1.9 THE BOARD'S ANALYSIS

The basic preliminary design phase for the Cigéo project will come to an end in mid-2015. The Board observes that this phase has seen major advances in the scientific and industrial aspects of the project, which must imperatively be consolidated during the final draft phase prior to the filing of the construction application (DAC) in 2017.

The Board now emphasises the points that it considers to be particularly important:

1. *The Board commends Andra for the continuation of studies and research since 2005, especially on the thermo-hydro-mechanical properties of the rock. The revision of these properties has led to the refinement of the parameter values on the basis of a significant amount of data. Andra has drawn conclusions from this change by clearly underlining its key implications for the dimensioning of the HL area. The studies and research – experiments and calculations – must continue throughout the entire industrial pilot phase and continue while the repository is in service.*

   However, the Board has concerns about the consequences of modifying the dimensioning so close to the filing of the DAC.

2. *In addition to questions of a thermo-hydro-mechanical nature, the Board observes that the optimisation in progress runs counter to two more general principles:*

   ❍ *The repository must be closed at the earliest reasonably possible moment given the objectives of safety and reversibility. This principle applies as much to the closure of the entire repository as it does to partial closures of cells and areas. It is based on the growing uncertainties concerning the state of knowledge and techniques and the state of society in over a hundred years time, which explains the preference for passive safety in advance: "the simulations that can be performed on changes to the packages and their environment are much more precise and reliable than those that can be performed on the social state in the distant future", (CNE report no. 8). Several years ago, this same principle led to the abandonment of the solution of multi-century surface storage periods, on grounds of the burden placed on generations in the distant future. The gradual lengthening of the storage period – and thus of the period in which the repository is open – ignores this principle: in the 2009 file, the start of the disposal of HL waste was envisaged in 2045; this date has been put back to 2075 and the possibility of 2099 is already being studied. The start of LLHLW deliveries in 2099 would need to be justified by major technical considerations and an in-depth analysis of its consequences on the operation of Cigéo and its safety. This arrangement would also create a long hiatus before the end of deliveries of LLILW packages. Its consequences would need to be carefully assessed. Such a hiatus would work against the preservation of expertise and would needlessly extend the maintenance of the open facility which would continue to evolve even if it were no longer in service.*

- *The usable surface areas for storage are a resource that must be preserved, especially as they provide means of adapting to possible changes in energy policy. The new design involves making the HL area more oblong in shape, but the total area occupied is slightly larger than that shown in the 2009 File (in the "Dimensioning " scenario). In the longest part of the area, it reaches the boundary of the Zira. This could result in less flexibility, for example if more space were to be required following the stoppage of reprocessing.*

3. *The dimensioning of Cigéo requires producers to set aside provisions. These provisions are already having an impact on their finances. The optimisation of dimensioning is guided by the concern to reduce the cost of disposal. This is not an unreasonable concern.*

   However, the Board observes that the works in several areas will continue for over a century and will not start for another fifty years or so in the HL zone. The estimation of the costs of these works is associated with great uncertainties concerning the evolution of knowledge and technologies. This uncertainty makes it difficult to establish the amount of provisions required.

   The producers' desire to minimise the provisions to be set aside as of now could lead to ways of operating that are not compatible with objective choices on the scientific and technical levels. The Board notes that this sometimes leads to unproductive debates.

   The Board recommends the implementation of a mechanism that favours objective discussions between Andra and the producers in complete compliance with the principle of setting aside provisions as required by the law.

Chapter 2
LLLLW

Andra has just launched exploratory research on characterising a possible LLLLW disposal site within the Soulaines community of municipalities in the département of Aube, where sub-outcropping soft clay series formations can be found in the low-lying parts of the topography. These conditions featuring shallow clay deposits would thus require the disposal of waste under a reworked cover (sous couverture remaniée – SCR). To date, no new investigations have yet been carried out on identifying an alternative site that could allow for the storage of this waste under an untouched cover (sous couverture naturelle intacte – SCI).

2.1 THE LLLLW SITE BEING INVESTIGATED

The municipalities of Morvilliers and La Chaine, situated in the département of Aube, form part of the Soulaines community of municipalities and have hosted the Industrial consolidation, storage and disposal centre (Centre Industriel de regroupement, entreposage et stockage – CIRES), dedicated to very low activity waste since 2003. This part of the Aube département is characterised by the presence, on or near the surface, of Aptian Plicatules clay (western sector), surmounted further to the west by Albian green sand, which is, in turn, covered by Albian black clay of the Gault formation, referred to a tile-making clay. There is very little deformation throughout all of these clay-bearing levels which possess apparently attractive radionuclide containment properties, while remaining thick enough to allow for the excavation of a possible LLLLW disposal facility.

The research conducted in 2014 and 2015 consisted of creating seven cored boreholes and acquiring 120 km of 2D seismic reflection profiles.

These initial explorations allowed for the inspection of the lateral extension of the Quaternary deposits that locally cover the Aptian and Albian sedimentary series formations on the surface, and for the mapping of the thickness of the clay target layers. Their mineralogical and chemical composition remains relatively homogeneous at the regional level, with carbonate content of less than 30% and mean porosity of 30%.

Except for the area around the graben of Soulaines, situated in the southern part of the municipal area, no major faults have been observed in the rest of the area of investigation (northern half).

Eleven hydrogeological boreholes have shown vertical hydraulic transfers, moving downward in the east (recharging zone), and upward in the west. The Albian green sand, situated between the two clay layers under investigation, form part of a major aquifer on the scale of the Paris Basin, referred to as the Albian aquifer. There is very little exploitation of this aquifer in this sector, but its status as a strategic drinking water reserve for the Paris region requires the performance of the necessary impact assessments.

The Board recommends studying the possible environmental impacts of an LLLLW SCR repository. In particular, Andra must characterise the thicknesses of the indispensable vertical height of clay required, in addition to the properties and transfer rates for radionuclides and any other chemical substances, either towards the surface through the reworked cover, or towards the Albian and Aptian aquifers. Safety analyses must be carried out for varied evolution scenarios concerning groundwater and redox – oxidising or reducing – conditions affecting the environment. This safety analysis must take account of any impact on the aquifer system, especially the Albian, at the regional level.

2.2 GEOLOGICAL EVOLUTION OF THE LLLLW SITE

The option of a repository with a reworked cover (SCR) requires real trust in its durability throughout the coming millennia. Andra has thus carried out detailed studies of the morphological evolution of the relief during the last climatic periods of the Quaternary era. At the same time, it has performed simulations to anticipate the locations of areas that may be the most sensitive to erosion or, to the contrary, propitious to the deposition of new sediments, taking account of different climate change scenarios.

The Board shall be attentive to the progress made in this research. The Board reiterates that an alternative site for a repository with an untouched cover (SCI) could become essential if the scientific analysis should reveal the existence of transfers of radionuclides and chemical substances towards the surface or towards the Albian aquifer at levels above those permitted by the standards. An SCI facility would also be essential in the event of a risk of the excessively rapid erosion of the reworked cover.

2.3 CHARACTERISATION OF LLLLW

The LLLLW inventory (quantities and characteristics) presented in Appendix X of the CNE Report n° 8 and the subject of the studies on their management, remain relevant today, with a few minor variations.

There has been progress with the R&D on improving the characterisation of this waste and attempting to decontaminate it or even recover it (cf. Appendix IX).

The studies carried out by EDF with Studsvik and the CEA (completed in 2014) on decontaminating or burning the graphite from reactor stacks have shown that 14C decontamination was only partial. The process of reducing the graphite to ash has been abandoned and the storage of this ash at Cigéo is no longer relevant.

There is little chance of an incineration/vitrification process for bituminous sludge classified as LLLLW by the CEA being economically viable. The sole remaining management prospect for this waste is for it to be integrated into the LLLLW waste management process, i.e. currently an SCR repository if it satisfies the requirements of the safety analysis.

The implementation of the processing of 8,400 tonnes of standardised solid residue (SSR), in bulk storage at the La Rochelle site, for the extraction of the thorium and rare earths, or for its dehydration, is likely to reduce its volume. The decision has been postponed until mid-2016. Solvay considers that this waste corresponds more to mining-type waste.

The additional characterisation studies for the definition of storage circuits for graphite waste, ion-exchange resins and bitumen LLLLW are continuing. The new methodology for measuring 36Cl activity in graphite stacks now allows for the definition of the 36Cl inventory of the waste requiring disposal. The consolidation of the alpha activity of the bitumen waste is also in progress.

2.4 CONCLUSION

The management of the LLLW that may not be transferred to Cigéo is mainly based on the design of an SCR facility on the site currently under investigation near Soulaines. Andra must submit a report on this issue to the government and to the ASN in mid-2015, but its studies are planned to continue until 2018. It will thus be in a position to conduct a safety analysis, taking account of the consolidated characteristics of the LLLLW waste packages and of the retention/containment capacities of the clay on the site with regard to radionuclides and chemical substances.

The Board recommends that the studies in progress concerning LLLLW should be followed through to completion as quickly as possible so that the producers and Andra can make final decisions in 2018 regarding the future of this waste. It draws attention to the need to characterise the waste with regard to the risk of chemical toxicity.

The Board repeats its recommendations made in report no. 8 and reiterates that the SCR concept remains fragile for the disposal of long-lived radionuclides as, at first glance, it does not provide the guarantees offered by storage under the natural cover of an SCI facility.

Chapter 3

WASTE FROM THE FRONT-END OF THE NUCLEAR FUEL CYCLE: THE CASE OF MALVESI

In its Report no. 8, the Board reviewed the waste produced by the Malvesi Comurhex plant (Areva Malvési, Aude) and its management, mainly on the basis of the information contained in the national waste inventory and the 2012 PNGMDR (National plan for the management of radioactive materials and waste). In September 2014, it visited the liquid effluent treatment facilities, the facilities used for storing solid waste and the new Comurhex-II plant currently in the start-up phase. It has obtained additional and updated information, both about the characteristics of the waste and its current management, in addition to the strategy envisaged for the longer term. This plant produces very pure uranium tetrafluoride, UF4, and has a nominal capacity of 14,000 tonnes/per year (cf. Appendix IV).

3.1 GEOLOGY OF THE MALVÉSI SITE

With a view to the possibility of in situ geological disposal, Areva has asked BRGM to carry out a geological study of the Malvési sector. This study shall take account of several seismic profiles and consolidate borehole data providing important additional information relating to the geological surface data and the knowledge acquired by the operation of quarries situated near the site.

The sector is characterised by a set of normal Oligo-Miocene faults, most of which take root in the evaporites of the Trias, and by a set of reverse faults with a smaller pitch, which are older than the normal faults, coinciding with the formation of the Pyrenees, which are still visible in the Corbières area. The interactions between these different faults have dictated the geometry of the Oligocene basins, which consist of red strata of continental origin, whose colour originates from the deposition conditions in an oxidising environment, and dark-coloured strata deposited under more reducing conditions (presence of native sulphur that used to be mined). The Cenozoic organic matter is still at a quite shallow level and is probably immature. The bitumen encountered in the fissures of the underlying Mesozoic carbonates could thus have deeper origins. The Permian (Upper Paleozoic) is currently recognised as the main mature source rock of the sector and the origin of the only oilfield ever exploited in the Languedoc region, at Gabian (where the oils expelled by the Permian were stored in Triassic gravel-rich reservoirs situated underneath the salt).

Geochemical analyses indicate that the artesian upwelling of hyper-saline water in the outcropping Mesozoic limestone near the Malvési site are of meteoric origin, infiltrating in the Corbières hills, then leaching the deep Triassic salt layers before resurfacing at the low points, with the porous carbonates and faults acting as vertical drains located at the edges of the Oligocene sub-basins.

The Board recommends that Areva should continue its characterisation activities on the site, while specifying the containment properties of the Oligocene series in particular, as well as the origin, methods and transfer speeds of the natural fluids in the sector.

3.2 MANAGEMENT OF WASTE FROM MALVÉSI

3.2.1 Waste and effluent from the UF_4 manufacturing process

The UF_4 production process leads to the formation of several types of solid waste: insoluble oxides from the dissolution of "yellow cake" (containing 15 to 30% of miscellaneous non-radioactive chemical elements and 1% thorium including ^{230}Th) and other solids resulting from the lime treatment of the mixture of two liquid effluents. The first originates from the uranium purification phase – raffinates from the extraction of uranium by the tributyl phosphate (TBP), supernatant from the precipitation of the di-uranate – and the second originates from the fluorination stage in which the di-uranate is converted into UF_4 (solutions from the scrubbing of miscellaneous gases and dilute hydrofluoric acid).

The neutralisation solids – referred to as "sludge" – are mixtures of hydroxides of heavy elements of variable definition, calcium carbonate $CaCO_3$ and fluorite CaF_2. This sludge contains traces of uranium and natural radioelements. The supernatant from the neutralisation operation is a liquid containing large amounts of nitrates and other soluble compounds such as sodium carbonate Na_2CO_3. The separation of solids and liquids and the evaporation/concentration of the liquid part take place in a series of basins.

Since 1960, the plant has converted over 400,000 tonnes of uranium into UF4. Areva estimates the global yield of uranium recovery to be 99.83%, which leaves approximately 700 tonnes of uranium remaining in all of the on-site waste, primarily in approximately 300,000 m^3 of sludge. The processing of one tonne of uranium produces approximately 4 to 5 m^3 of effluent. Between 1960 and 1983, batches of UF_4 were produced from reprocessing uranium containing impurities such as artificial radionuclides (^{99}Tc, $^{238/241}Pu$, etc.) which are found in some of the solid waste and evaporating solutions.

3.2.2 Management of effluents by "lagooning" and solid waste by storage on the site

A series of basins are situated on 30 hectares of the 100 hectare site. The leakproof basins B3, B5 and B6 are intended for the settling of the solid fraction and basins B7 to B12 are for evaporation. All of these basins have ICPE (Installation classées pour l'environnement – Classified facilities for environmental protection) status. Since the start of the production of UF^4, a very significant proportion of the sludge (approximately 280,000 m^3) has been stored in the old B1 and B2 basins. Miscellaneous materials have also been deposited in these basins since 2004 following redevelopment work due to a landslide that affected the dykes. These basins were then covered with a layer of natural materials.

The site of B2 has a reserve facility that will contain the solid residues from basins B3, B5 and B6. Basins B1 to B6 were created over the residues and waste from a former sulphur mine, some of which have been contaminated by radionuclides from basins B1 and B2 which are not leaktight.

All of the solid waste deposits – current and future – that concern basins B1 and B2 and the site to be created, shall be included in the INB ECRIN (Basic Nuclear Facility for the contained storage of conversion residues). Since 2012, the INB and basins B3, B5 and B6 have been surrounded by a hydraulic barrier that protects the groundwater.

The mass balance and radiological evaluation of the waste (contained within and underneath the basins) are given in Appendix IV.

3.2.3 Classification of solid waste, historical and projected inventories

The solid waste that has already been produced or that will be produced prior to the start-up of the new Comurhex-II plant will be sent to INB ECRIN to join the waste qualified as historical by Areva. The physical (bulk waste), chemical (poorly defined compositions) and radiological (natural and artificial radionuclides) characteristics of the waste and especially the volumes in question, prevent its acceptance in one of the currently operational or planned waste disposal centres.

The specific activity of such wastes is generally above that of radiferous LLLLW and is within the same range as mining waste with regard to natural radionuclides, but a significant proportion of it is contaminated by artificial radionuclides. Their quantity is comparable to that of small mining waste repositories. The public authorities have also asked Areva to envisage a specific short and long-term in situ management procedure for this waste (and for the future waste) (PNGMDR Orders 2010-2012). That is why the waste from Malvési (Areva) is not counted as part of the LLLLW.

3.2.4 Short-term management – INB ECRIN

In the immediate future, Areva must continue to create the ECRIN facility in order to ensure the long-term containment of the radioactivity from its historical solid waste. It will be necessary to group together the solid waste from basins B3 to B6 inside the perimeter of the INB (basic nuclear facility) (cell to be built on the site of B2), consolidate and remodel the entire structure, and then place a multi-layer bituminous cover over the stored waste (scheduled for completion in 2017). The site of the emptied basins B3 to B6 shall be used for storing the waste from Comurhex-II.

3.2.5 Medium-term management

Areva has carried out research to implement three innovations leading to savings on reagents, a reduction in the amount of waste and better waste management. This involves the treatment of nitrate-containing effluents and the separation of the purification effluent flow from the fluorination flow. Comurhex-II has a nominal capacity of 21,000 tonnes of UF_4 per year.

The separate neutralisation of the process (purification and fluorination) flows will lead to packages of radioactive dehydrated solids containing the radionuclides from the sludge, on the one hand, and fluorites (CaF_2), on the other. The former shall be stored in hot cells, to be constructed on the sites released on the locations of basins B3 to B6. The fluorites can join the VLLW circuit.

Areva is working on the TDN project (processing of nitrates between 2018 and 2050), which aims to recover the 450,000 m^3 of liquid effluents from basins B7 to B12 and treat the new effluents in line. These nitrates will eventually be converted into solid VLLW (silico-aluminates containing the radionuclides) placed in big-bags, and into gaseous effluents (CO_2, N_2, H_2O) that can be discharged into the environment.

3.2.6 Long-term management

The modifications to the process and the commissioning of Comurhex-II must reduce the quantities of waste in the future. Nevertheless, the expected volumes remain substantial. By 2050, Areva estimates, globally and based on the current process, that the volumes will amount to 200-300,000 m^3 of dehydrated sludge (250-500 Bq/g, U and ^{230}Th) and 110-140,000 m^3 of solid nitrates (approximately 10 Bq/g, U, ^{230}Th and ^{99}Tc). These quantities, which are probably overestimated, could be revised downwards when the processes and waste are characterised in greater detail.

By around 2050, therefore, a total of nearly one million cubic metres of varied types of radioactive waste will be stored or will be likely to be disposed of at the Malvési centre, generating activity that is currently estimated at 180 TBq, to which must be added 1,300,000 m^3 of sulphur mining waste and tailings, which are partially contaminated.

Areva has thus examined the feasibility of repository for all of the waste with the obligation to submit a progress report at the end of 2014 and a feasibility studies file at the end of 2017. Several options are envisaged: a tumulus under a multi-layer, man-made cover covering the area occupied by the current site or a shallow repository (between 20 and 40 m), either in the former sulphur mine adjacent to the site (under a multi-layer cover), or in the Oligocene marl forming the substratum of the site (under a reworked cover). These studies are conducted according to the recommendations of a group of international experts.

3.2.7 The Board's comments

The Board considers that the R&D and activities undertaken by Areva should contribute to the better management of waste from the UF_4 production process. Eventually, only solid waste should remain. It will be possible to establish the stability of the future waste and its capacity to contain radionuclides without too much uncertainty. On the other hand, it will be harder to establish the final characteristics of the historical waste with a view to its disposal.

The Board recommends continuing to monitor the evolution of the mineralogy of the historical waste and its leaching behaviour, as is carried out for the uranium ore processing residues stored on mining sites.

Due to the quantities of waste deployed, the disposal site must be situated in the immediate proximity of Comurhex. The impact calculation in the safety analysis must take account of the major geological and hydrological stresses (thickness of the Oligocene series and geometry of their substratum, distribution of major faults and paleokarst, origin of the saline water upwelling around the high points of Mesozoic carbonates, and the origin of the bitumen and native sulphur found locally in the Oligocene series).

In order to adapt the concept to the local constraints, the Board emphasises the importance of having in-depth knowledge of the following:

- *radiological and chemical source terms*
- *geological and geodynamic context*
- *possible impacts of climatic origin in the medium and long term*
- *behaviour of radionuclides and chemotoxic elements within the waste and in the environment*

In the interests of consistency in the management of waste at the national level, the Board wishes to know the selection criteria for the sites and for the acceptance of packages for disposal that will be adopted. It would like to know how Andra, the National Agency, will be involved in the creation of a possible on-site repository.

Chapter 4
DISMANTLING

4.1 OVERVIEW OF THE FACILITIES CONCERNED

The facilities concerned by dismantling operations in progress are:

- the shut down reactors managed by EDF: 6 natural uranium graphite-gas (UNGG) reactors, 1 heavy water reactor, 1 PWR and 1 FNR;
- fuel cycle facilities operated by Areva;
- research facilities under the CEA's authority: pilot and reprocessing facilities (Marcoule, Fontenay-aux-Roses), military facilities (naval propulsion) and workshops.

At the same time as these operations, EDF has launched deliberations seeking to develop a global PWR dismantling strategy based on feedback from the operations in progress.

4.1.1 Reactor dismantling: the first generation

The dismantling of reactors is governed by a regulatory and financial framework in which each operator is responsible for the operations required until the decommissioning of the site and its removal from the list of basic nuclear facilities (INB). Deconstruction consists of removing the fuel and then dismantling and removing the large equipment items, eliminating the radioactivity from all premises, inspecting and then demolishing the buildings and finally decommissioning the facility. These operations are subject to administrative authorisations. They are coordinated by EDF within CIDEN (French engineering, decommissioning and dismantling and environment centre) and are carried out under the control of the ASN (French nuclear safety authority).

The current decommissioning and dismantling programmes in progress concern nine reactors shut down between 1973 (Chinon A1 – UNGG) and 1998 (Creys-Malville – FNR). The rate of progress in all of the operations exceeds 43%, and the Brennilis (heavy water), Creys Malville (SPX) and Chooz A (PWR whose authorisation decree for total dismantling was passed in 2007) reactors are currently more than 50% dismantled. For this latter plant, the project has entered its final phase with the imminent dismantling of the reactor vessel.

The key issue concerning UNGG reactors resides in identifying a disposal circuit for their graphite that will be operational by 2025. The establishment of such a circuit depends on the acceptability of receiving the graphite on LLLLW disposal sites. The ^{36}Cl and ^{14}C graphite inventory is currently specified by EDF and the CEA. EDF has developed a statistical inventory method (cf. report no. 8) which is currently being examined by the French Radioprotection and Nuclear Safety Institute. Furthermore, EDF and the CEA have analysed all of the measurements conducted on graphite waste. This inter-comparison has led to a downward adjustment of the total radiological inventory of ^{36}Cl.

> *The Board takes note of the satisfactory progress of the works allowing for the precise definition of the graphite inventory from the dismantling of the UNGG reactors. It observes that the procedures for the disposal of these LLLLW-classified materials are still to be defined.*

4.1.2 The second generation: PWRs currently in operation

EDF-CIDEN are currently analysing how to decommission a PWR while taking account of the experience acquired from the dismantling of the first generation of reactors. This involves the consolidation of a sequence of operations and the estimation of the cost of future dismantling operations.

The studies are aiming to:

- eliminate as many risks to the environment and staff involved as possible, while controlling the cost of the project;
- plan for a gradual decommissioning to release zones that are ready for deconstruction as soon as possible;
- optimise dismantling and waste management techniques, in particular by planning to reduce the volume of VLLW through the improved recovery of electric cables, steel and rubble.

EDF is seeking to maximise the standardisation of dismantling and future waste management in order to benefit from the series effect in the PWR fleet. These studies already indicate that at least 5 years will be required for the preparation of a dismantling file. To benefit from the expertise available in operation, the fuel could be removed while the facility is operating and the process could last for around five years.

Finally, attention must be paid to changes in the standards given the significant quantity of waste to be produced.

The Board takes note of EDF's strategy which, beyond the dismantling of first-generation reactors, seeks to minimise the waste from the dismantling of PWRs in the current nuclear fleet. The Board would like the studies that are being developed with a view to identifying the VLLW derived from the PWRs and optimising its management, to be consolidated and stepped up.

4.1.3 Georges Besse Eurodif plant

This plant was operated by Areva for 33 years. It carried out uranium enrichment for the production of fuel for civil nuclear reactors. Occupying a 19 ha site, it is notable for its exceptional size with 1,400 diffusion stages and over 1,300 km of pipework, amounting to approximately 200,000 tonnes of metal equipment to be dismantled.

In the operating phase, the uranium content of the facility could amount to up to 3,000 tonnes of UF6. Approximately 300 tonnes remain; the first dismantling phase consists of rinsing the facilities intended for uranium recovery - the Prism programme. The aim of this project is to extract the uranium that may subsequently be recovered and to achieve a significant reduction in the radiological inventory of equipment with a view to optimising future waste management. This operation should take three years.

The second phase in the dismantling of the facilities will consist of taking the equipment apart and then separating the metal parts for compaction. This operation will generate approximately 150,000 of steel of a relatively homogeneous grade. It should take around fifteen years and lead to a fourfold or fivefold reduction in the volume of the equipment.

Considering the relative homogeneity of the metal waste, its volume – even after compaction – and problems relating to the storage capacity of VLLW and LLLLW sites, Areva is looking into the possibility of smelting/recycling metal waste on the Tricastin site. The principle would involve melting down the contaminated steel, which will separate the steel from the contaminants, with the latter becoming LLLW (cf. the Swedish practice described in chapter 6 and in Appendix IX). The steel thus obtained could then be reused in the waste disposal circuit, e.g. for manufacturing waste disposal packages.

For an initial volume of around 900,000 m^3, the steel processing and then recycling operations could lead to a very much lower final volume of VLLW (currently estimated at less than 90,000 m^3).

4.2 WASTE DISPOSAL

Apart from the graphite originating from UNGG reactors, the resulting waste will mainly be VLLW and short-lived low and intermediate-level waste (SL-LILW). The future dismantling operations will generate quantities of waste far in excess of those produced to date (cf. Appendix IX). That is why Andra is compiling an inventory of the volumetric and radiological capacities of surface disposal centres while considering possible ways to increase their capacities.

4.2.1 Short-lived low and intermediate-level waste (SL-LILW) circuit

The outlook for the production of SL-LILW, according to successive editions of the national inventory, does not point to the saturation of the Aube disposal centre (Centre de stockage de l'Aube – CSA), created in 1992 and designed to operate for 30 years. At the end of 2013, 280,000 m^3 of such waste was stored out of a total capacity of 1,000,000 m^3. The remaining capacity is higher than the initially envisaged needs, due to a very significant reduction in the volume of waste (particularly due to the creation of the VLLW circuit). In radiological terms, the critical design element is 36Cl, which requires the careful identification of each item of waste and its activity.

According to Andra, the CSA should be able to accommodate the waste produced by the operation and dismantling of nuclear facilities that are currently operating or whose construction has been approved. According to the 2015 national inventory, the final total amount of SL-LILW is estimated at 1,900,000 m^3 which includes the contents of the Manche disposal centre (Centre de stockage de la Manche – CSM).

4.2.2 Very Low-Level Waste (VLLW) circuit

The CIRES VLLW disposal centre has a capacity of 650,000 m^3. This centre, brought into service in 2003 for a period of 30 years, housed 250,000 m^3 of waste packages at the end of 2013.

Since it began operating, the disposal forecasts have constantly been adjusted upwards, due to the increasingly strict objective for the remediation of the facilities being dismantled. According to the 2015 national inventory, the final total amount of VLLW is estimated at 2,200,000 m^3, which is nearly four times the envisaged capacity of CIRES (Industrial consolidation, storage and disposal centre). According to this same 2015 national inventory, the limits of the CIRES disposal capacity will be reached in 2020.

Even if it is feasible to envisage increasing the capacity of CIRES to 900,000 m^3, these forecasts firstly impose the obligation to create a new storage centre by the 2020-2030 horizon, and secondly show the need to improve the control of VLLW volumes.

4.3 CONTROLLING THE VOLUME OF DISMANTLING WASTE – KEY ISSUES AND RISKS

In view of the large volumes of waste to come, it is very important to formulate predictive inventories that are as realistic as possible with regard to the regulations in force.

Work on this inventory is currently in progress. By the end of 2030, waste derived from dismantling should amount to over a third of the VLLW and LLLLW produced (i.e. in comparison, as much as the VLLW and a third of the LLLLW produced up to 2010) and more than 10% of the SL-LILW. A large proportion of the inventory consists of metals and rubble.

There is a genuine risk of a significant increase in the volumes to be stored in view of the final remediation objectives that could be established in future. Therefore, the principle of a single, dedicated burial site may no longer be pertinent. The transportation of this waste (mainly rubble) by road would constitute an environmental nuisance and impose significant safety and traceability constraints. This raises the question of the disposal of the least active waste on the dismantling sites themselves. However, the storage of waste on a dismantling site requires the creation of a specially developed facility with similar characteristics to those found on current VLLW disposal sites. Such a configuration can only be considered on a case-by-case basis; the subsoil must have the required physical properties, and the agreement of the stakeholders and the safety authorities must also be obtained.

Considering the capacities for VLLW management in view of the quantities to come, the Board requests the formulation of an industrial scheme that is capable of meeting these needs. It recommends continuing the research into possible optimisation solutions in compliance with the current regulatory constraints and evaluating the environmental nuisances and risks that could arise from the chosen options, taking particular account of those inherent to the transportation of waste.

The recycling of recoverable materials extracted from dismantling waste is one of the preferred possibilities, pursuant to the basic principles of the French Environment Code and in the context of a sustainable development strategy. This is a standard practice for conventional waste. With regard to the waste derived from the dismantling of nuclear facilities, the recycling envisaged in France is strictly limited to recovery within basic nuclear facilities (installations nucléaires de base - INB).

A study focusing on the benefits and technical and economic feasibility of this type of recovery was requested in the framework of the 2012 PNGMDR. This mainly concerns the recycling of low-level metal waste and crushed materials. Uses for this waste could be found for the creation of disposal sites, e.g. Cigéo, and for the filling of cells or the creation of packages.

- The potential offered by the filling requirements for the CIRES site could allow for the acceptance of over 10,000 m^3 of finely crushed material per year. Operations that allow for crushing require special facilities whose feasibility and safety still need to be evaluated.

- Preliminary studies are currently being carried out into the recycling of metal VLLW by smelting, with an application envisaged for the Georges Besse site, which has the advantage that the materials to be processed are reasonably homogeneous. It would be based on the creation of dedicated industrial smelting and rolling facilities due to the need to ensure the future traceability of the waste. The economic viability of the recycling of metal waste by a specific circuit has not yet been established.

The French approach to the management of waste originating from nuclear facilities could lead to the storage of potentially recoverable elements on VLLW burial sites. According to Andra, over 30% of the waste described as VLLW may contain no artificial radioactivity whatsoever. This data, if confirmed, would support the argument in favour of releasing certain batches of waste with a view to their recovery in conventional circuits.

Waste from nuclear facilities and released into the conventional sector is currently recycled in Europe. It is regulated by a directive (96/29 Euratom of 13 May 1996) and by the associated technical recommendations. This recycling concerns the field of metal materials within dedicated smelting circuits. It has been implemented in Germany (Siempelkamp) and Sweden (Studsvik), where approximately 3,000 tonnes of metal waste are processed each year (cf. chapter 6). The recycling of these metals is dictated by specific rules. In Sweden, metals destined for conventional use cannot contain more than 10% of metals originating from the nuclear industry. Certain European countries thus release materials that would not be allowed by the French regulations, but products containing metals recycled from the nuclear industry are commonly imported into France.

The monitoring of the radioactivity of materials prior to release requires sufficient guarantees without which the very principle of the existence of a release threshold could not be envisaged.

> *The Board wishes to know what studies and research have already been undertaken by waste producers on measuring the very low levels of radioactivity from radionuclides contained in significant quantities and different types of materials. These studies and this research could be a useful aid to the analyses of the working group established by the ASN in order to examine the principle of a release threshold in view of European practices.*

The optimisation of waste generated by dismantling is heavily reliant on R&D activities.

The call for projects issued by Andra at the end of 2014 concerns the optimisation of the entire waste management chain (characterisation, sorting, recycling, recovery, logistics and the societal dimension). The innovations that could result from it may facilitate the emergence of new techniques and, above all, provide an opportunity to extend Andra's partnerships (waste producers, dismantling operators – intermediate-sized enterprises, SMEs – academic sector).

> *The Board wishes to be informed of the research projects adopted following this call for projects in order to evaluate whether the studies and research conducted in the framework of this call, coordinated with the work carried out by the producers and Andra, cover all aspects of optimisation. It would like to be informed of the progress made in these projects.*

Chapter 5

PARTITIONING AND TRANSMUTATION

5.1 ASTRID: REFERENCE DESIGN AND GENERAL DEVELOPMENT CONTEXT

5.1.1 Institutional framework

The act of 28 June 2006 defines a research programme on the management of all radioactive waste and materials (articles 3 and 4). Its states that studies of the partitioning and transmutation of long-lived radioactive elements must be combined with research on the new-generation nuclear reactors.

The studies of reactors are mainly conducted by the CEA in association with the nuclear energy operators (EDF, Areva, etc.) and public research institutions (CNRS [National Centre for Scientific Research], universities, etc.). Their aim is to develop the sustainable management of radioactive waste and materials. This management envisages the use of plutonium associated with uranium as a fuel in new fast neutron reactors (FNR). It involves the eventual closure of the fuel cycle of the PWR fleet currently in service and of the entire later fleet of the same power with the recycling of all of the plutonium. Finally, the studies concerning the partitioning of minor actinides with a view to their transmutation aim to reduce the radiotoxicity of the waste to a significant extent.

Research bodies (CNRS, French universities and SCK-CEN in Belgium) are investigating the opportunities that could be offered by a thorium cycle and by coupling accelerators to reactors (ADS) in order to produce energy and transmute the long-lived radioactive elements (cf. Appendix X).

At the end of 2012, the CEA submitted a report on its studies and research into the development of FNRs and the partitioning/transmutation of minor actinides to the French government and since then it has consolidated the main thrusts of this report.

The CEA was appointed as the developer and project owner of the "Astrid" technological demonstrator for a fourth-generation, sodium-cooled fast-neutron reactor (Na-FNR), which uses plutonium and depleted uranium as its fuel. The R&D carried out by the CEA also concerns the cycle of materials – uranium and plutonium – as in FNR designs (iso-generator reactor, breeder reactor or even sub-generator reactor modes), the recycling of these materials is an essential requirement. There is substantial feedback from experience acquired in France and in the other countries participating in the Generation IV International Forum on Na-FNRs.

5.1.2 The Astrid reactor: innovations for the use of plutonium

Assuming that FNRs gradually replace the current PWRs, Astrid – the prototype for these reactors – must demonstrate that a certain number of innovations will ensure a higher level of safety than for Generation III reactors and integrate the lessons learned from Fukushima. These innovations include:

- the development (patent) of a core with a low run-off coefficient,
- a significantly reduced risk of a sodium-water or sodium-air reaction thanks to the use of modular steam generators or a gas energy conversion system (collaboration with ALSTOM),
- an internal core catcher inside the main vessel,
- additional residual heat removal systems,
- major advances in instrumentation for the continuous monitoring of the core and the circuits of components containing sodium,
- detection of sodium leaks and sodium-oxygen reactions.

Astrid's design, based on these innovations, must be submitted to the ASN for approval. Some safety measures have already been analysed by the ASN.

These innovations constitute a major breakthrough concerning the future, advanced safety Generation IV FNRs. The gas energy conversion system (ECS), which avoids any contact between sodium and water, would constitute a major technological breakthrough in relation to the traditional water steam ECS.

The Board recommends stepping up R&D on sodium-gas exchangers and on the coupling Astrid to gas turbines.

As stated at the hearing on 12 March 2015, attended by its partners EDF and Areva, the CEA proposes that Astrid be dimensioned at 600 MWe for the following reasons:

- Astrid will have a dual function: acting as an irradiator and a demonstrator. As an irradiator, Astrid must validate the structural materials and the fuels that will be used for the production of its successive cores. As a demonstrator, it must allow for the extrapolation of safety data for industrial reactors of at least 1,000 MWe.

- It must be possible to determine the form of the neutron flow for the new core with a low runoff coefficient at a minimum size that will allow for subsequent extrapolations to powers of 1,000 to 1,500 MWe.

- Iso-generation (producing as much Pu as is consumed) is only possible with a heterogeneous core above a certain power level in order to maintain the Pu balance.

- The transmutation of minor actinides in heterogeneous mode requires a large neutron flow at the external core boundary, which can only be obtained at a significant power level.

- The coupling of a reactor with 600 MWe of power to the grid would allow for the partial funding of its operation.

5.1.3 Industrial partnerships and collaborations

In recent years, the CEA has intensified its industrial partnerships and R&D collaborations.

- Astrid has been awarded the "Investments for the Future" ("Investissements d'avenir") label, which requires its partners to finance the operation, at least partially, with their equity capital. 13 partners are participating in the engineering aspects of the programme (development of major components, core and structural materials, TRL qualification, safety R&D, etc.). This consortium allows for the integration of industrial concerns at the design stage. It also ensures the consortium's long-term commitment to the project.

- Numerous international collaborations have been firmly established with several countries, and in particular with the members of the Generation IV international forum: Russia (use of facilities for irradiation), United States (benchmarking of calculations for the core with a low runoff coefficient), Japan (framework agreement for R&D support for Astrid and serious accidents), Germany, Italy, Spain, Switzerland, European Commission, United Kingdom and Sweden (R&D on safety, instrumentation, accident sequence modelling, corrosion, etc.).

5.1.4 The Astrid programme

This programme includes the reactor and all ancillary studies – prior or parallel – for the developments that will follow its commissioning in order to reach the objectives established by the Act.

a) Partitioning and transmutation

The CEA and Areva manage the industrial reprocessing of the plutonium from the used UOx fuel, i.e. the separation of the uranium, plutonium, minor actinides and fission products. The plutonium thus separated can only be recycled once in a PWR due to the modification of its isotopic composition. However, FNRs can operate with plutonium regardless of its isotopic composition, which allows for the multi-recycling of Pu. R&D is thus seeking to develop processes that are essential for carrying out the industrial multi-recycling of plutonium and uranium in the FNR fuel cycle. It is also focusing on the multi-recycling of the americium required for its transmutation.

Astrid, which uses depleted uranium produced by enrichment (450,000 tonnes available in 2040) and plutonium derived from the processing of used UOx, MOx-PWR and MOx-FNR fuels, will be the first beneficiary of this R&D.

The partitioning of these elements from the used fuel, regardless of its type, occurs at key stages to allow for the deployment of Astrid:

- firstly, for the processing of MOx-PWR fuels in order to extract the plutonium from them and manufacture Astrid's first core and its subsequent refills;
- and then for demonstrating that Astrid is capable of recycling Pu from MOx-PWR, followed by its own Pu (derived from MOx-FNR), and thus allowing for the development of the specific industrial processing of the spent FNR fuel.

Significant industrial experience of the partitioning and production of FNR has already been acquired. This relates to the production of the Rapsodie, Phénix (PX) and Superphénix (SPX) fuels, in addition to the reprocessing of MOx-PWR (73 tonnes processed in the UP2 plants at La Hague) and the processing of MOx-FNR fuels originating from Rapsodie and Phénix and processed in UP2.

In support of R&D for Astrid, the CEA and Areva, working in close collaboration, envisage using the existing facilities at La Hague and Marcoule. This approach will allow for the consolidation of processes, the definition of the capacity and the optimal location of the future fuel processing and production workshops for Astrid and for a possible future fleet consisting of FNRs.

b) Scenarios

Areva, the CEA and EDF, which share an interest in closed fuel cycles and the sustainable management of materials via the use of FNRs, are continuing to study a varied range of FNR deployment scenarios.

The assumptions are based on the following data: a constant installed nuclear capacity (approximately 63 GWe), operated for 60 years with PWR of 1,530 MWe and FNR of 1,000 or 1,450 MWe. Several phases are envisaged for the deployment. Firstly, Astrid will be commissioned in around 2025, followed by the deployment of several FNRs, which will help stabilise the stock of MOx-PWR. The following phase, after 2050, provides for a fleet consisting of 40% FNRs, this configuration allowing for the multi-recycling of plutonium and, above all, stabilising the plutonium incorporated into the spent fuel. The final phase features a fleet consisting of at least 70% FNRs as long as the EPRs remain operational. At the end of their service life, the EPRs will be replaced by FNRs. The operation of a fleet of FNRs requires the use of more natural uranium.

The studies focus above all on the requirements for the transition from one phase to the next, which means constantly adapting the fuel cycle to secure the supply for the FNRs.

The results of the simulations show that the management of Pu multi-recycling dictates the dimensioning of the fuel cycle facilities (emissions of neutrons and heat by the 238Pu and 241Am in all phases) and that the recycling of the americium will require specially adapted workshops. The results are important in guiding the research to be conducted at present and defining a possible schedule for the creation of the future facilities for FNR fuel reprocessing and the americium transmutation fuel. A report is expected in July 2015.

c) Waste from a fleet of fast neutron reactors and closure of the cycle

Studies are in progress that will allow for a comparative evaluation of the waste originating from a fleet consisting exclusively of PWRs or FNRs. The first estimates reveal that the FNR fleet produces significantly less VLLW and SL-LILW; however, the volume of LLILW – mainly consisting of assembly structures and metal waste – increases by approximately 70%. In the absence of transmutation, the quantities of LLHLW (vitrified waste packages) associated with the operation of electrogenic FNRs are comparable to those produced by PWRs. In the case of transmutation with dedicated electricity-generating FNRs, these quantities depend on the management of americium.

d) Transmutation

The CEA is continuing the R&D concerning the partitioning and the shaping of compounds of minor actinides, in order to study the conditions for their possible transmutation. The Atalante installation at Marcoule will allow for the production of needles containing several tens of grammes of americium for irradiation in Astrid by 2040. After examination in hot laboratories and qualification, the production and then irradiation of needles containing increasing amounts of americium (up to several kilos) will allow for the gradual testing of the technical feasibility of transmutation with a view to moving towards industrial feasibility.

e) End of the FNR cycle

The implementation of Astrid must also provide an opportunity to study the consumption of the remaining Pu when the decision is made to shut down an FNR fleet. This is being investigated by studies currently in progress, especially at Cadarache. They are seeking to define the operating conditions for the core with a low runoff coefficient used for iso-generation, for the burning of a Capra (Consommation Accrue de Plutonium dans des Rapides – increased plutonium consumption in fast reactors) fuel with very high plutonium content. This technique, if deployed in a sufficient number of sub-generator reactors, would allow for the consumption of 50% of the final plutonium stock every 50 years with a FNR fleet that would continue to generate electricity. The other possibility investigated at the European level in the framework of the Myrrha project would be to use ADS reactors (cf. Appendix X).

The Board takes note of:

- ❍ *the reaffirmation by the CEA, Areva and EDF of the will to develop the Astrid reactor as a Generation IV electro-nuclear demonstrator reactor. This will is asserted by the signing of tripartite or bilateral agreements and in the context of the CEA-French government framework contract;*
- ❍ *the consolidation of the Astrid concept by finalising the key options and the initial safety guidelines;*
- ❍ *the participation of numerous industrialists in the project;*
- ❍ *the consolidation of national, European and international collaborations.*

5.2 RESEARCH AND DEVELOPMENT

5.2.1 Collaborations and constraints

The creation of Astrid entails a major conceptual and technological breakthrough from the Phénix (PX) and Superphénix (SPX) programmes and in relation to the other Na-FNRs in operation or being built worldwide. For this challenge, the CEA can rely on the French feedback from the building and operation of these two reactors and global feedback from Na-FNRs based on 400 equivalent years of operation of this type of reactor.

Fundamental research prior to R&D is required for the Astrid programme and for this, the CEA is closely associated with the CNRS and universities. In this way, highly fundamental physico-chemical studies are being conducted in collaboration with the academic sector (Needs, ICSM, and European programmes).

To ensure its readiness for the DPD2 and for the subsequent stages, the CEA has significantly stepped up the R&D on the major reactor components, which is being carried out in partnership with industrialists (cf. Appendix XI), especially for the critical elements such as the sodium-gas exchangers and the core catcher.

Studies on the fuel are being developed in close collaboration with Areva and in the framework of international collaborations. The collaboration with Russia involves the qualification of a prototype assembly (25 kg of MOx containing 20% Pu) that will be created in a Melox workshop and then irradiated in the Russian BN-600 reactor.

The processing of the spent fuel from Astrid will require the strengthening of the La Hague dissolution workshop, an operation which is underway in the TCP-LH workshop. In Atalante at Marcoule, the CEA is currently studying the complete dissolution of MOx-RNR samples, which is the first essential stage prior to the deployment of industrial Pu recycling. The production of the Astrid fuel is planned to take place in the AFC, which could be installed at Marcoule in conjunction with the Melox plant.

With regard to americium transmutation, the CEA is continuing to investigate the EXAm process and the studies on the production of MOx-UO_2/AmO_2. The irradiations are scheduled to be carried out abroad in the framework of collaborations. The integral EXAm test on the actual spent fuel, which was supposed to have taken place in 2014, has been postponed until 2015.

These developments are mainly reliant on the CEA's resources, although other stakeholders (industrialists, French and foreign research bodies) are also making an appreciable contribution. Budgetary constraints have caused the CEA to redefine its priorities and revise the schedule of milestones for the building of Astrid. Phase 2 of the preliminary design (PD2) has thus been scheduled for the end of 2015 when a safety orientation file (SOF) will be submitted to the ASN. If the Government decides to go ahead with the project, the DPD (detailed preliminary design) will be drawn up between 2016 and 2019, with the submission of the DAC (construction application) in 2019.

In conclusion, the Board observes that constant progress is being made with the R&D required for conducting the Astrid programme. R&D is continuing in the CEA facilities in France, in partnerships with industry at home and in collaborations abroad.

The Board repeats its recommendation for the allocation of sufficient resources to the CEA to ensure compliance with the schedule envisaging the filing of the DAC in 2019.

5.3 MATERIALS, FUEL FOR ASTRID AND TRANSMUTATION FUEL

Appendix XII provides detailed information about R&D on the materials and fuels that will supply Astrid.

5.3.1 Materials for the reactor

The core materials are the constituents of the components that allow for the development and control of the chain reaction, the internal and external core fuel assemblies and the other assemblies, control and cutoff rods (B4C). Most of the studies focus on the fuel assemblies and needles. The structural materials constitute the boiler: vessel, head and internal elements, none of which can be replaced during Astrid's service life. The studies focus in particular on the resistance of the metal materials under diverse stresses (temperature, interaction with sodium and irradiation).

R&D on the materials is also guided by feedback from the operation/maintenance of Phénix and Superphénix and, more recently, in the framework of the dismantling of these reactors and the examination of the irradiated materials under operating conditions.

In France, the CEA, EDF and Areva possess the facilities required for these studies.

The properties of the stainless steel used to manufacture the cladding and hexagonal assembly tubes are well known. The studies aim to consolidate the laws of their behaviour above 700°C and for burning rates of up to 50 GWj/t.

Ceramics of UO_2 and $UPuO_2$ that composed the fuels FNRs are knowed and their manufacturing is mastered. The CEA has adopted the industrial production process already in use at Cadarache for PX and SPX; the R&D focuses on the impact of increasing the Pu content on the process and on its simplification.

The vessel and certain internal reactor components are essential to ensuring the safe operation of the reactor for the some decades to come. They are subjected to strong permanent or cyclical stresses. The studies focus on the ageing of various types of steel under stress and irradiation, on their creep properties and their corrosion by sodium. The aim is to model the condition of the steel after 40 to 60 years for the dimensioning of elements. The current knowledge in these fields is based on data from mechanical experiments and the feedback from PX which do not exceed three decades. Similarly, the studies of steel-sodium interactions aim to specify the mechanism in operation at the interface between sodium and steel in order to dimension the components with regard to transients in the presence of oxygen.

The Board observes that the R&D methodology and approaches that will be used to qualify the materials to be used in Astrid's construction and operation are clearly defined. The R&D benefits from significant feedback accumulated over decades of activity in the Na-FNR field and in the elaboration of nuclear fuel in general. It also benefits from the expertise of the community of metallurgists and physical-chemists.

The Board reiterates that R&D must be carried out on the manufacturing processes for the vessel and internal structures to ensure the matching with the material specifications.

The Board recommends ensuring the best possible use of the feedback from Phénix acquired during its dismantling.

5.3.2 Fuel for transmutation

The fuel foreseen for americium-loaded blankets will consist of a mixed $UAmO_2$ oxide ceramics prepared according to the powder metallurgy process, like for $UPuO_2$ oxide. However, the quality of the ceramic is heavily dependent on the preparation conditions, and the chemical properties of americium are different from those of Pu. The studies focus on the search for maximum density in order to increase the Am transmutation yield, and optimum porosity to facilitate the diffusion of the large quantities of helium that will be produced in the fuel.

The Board considers that the fundamental studies of the properties of the U-Am oxides and on their shaping for the subsequent production of the americium-loaded blankets, are of paramount importance. They must be continued and developed in greater detail as these oxides behave differently from U-Pu oxides. The possibility of implementing the transmutation of Am depends on this process.

5.4 CONCLUSION

The Board advises the CEA to strengthen R&D in fields that are required to supply essential data for the provision of firm support for the submission of the construction application for Astrid, especially with regard to sodium-gas exchangers, to the coupling of the reactor to the gas turbines and the monitoring while in operation.

The Board considers that the CEA is gradually establishing a major scheme to attain the goals set for the Astrid programme. It recommends continuing to strengthen the links between industrial partners so that this project can be seen through to a successful conclusion.

The Board reiterates that the partitioning-transmutation of long-lived radioactive elements is one of the objectives laid down by the Act of 2006 and that it can only be achieved after the creation of the Astrid reactor. In the current context, it appears to the Board that the project need this objective to be clearly reconfirmed by the stakeholders.

Chapter 6

INTERNATIONAL PANORAMA

This chapter supplements the Board's previous reports and starts with an overview of the management of radioactive waste in several foreign countries.

The first part places the emphasis on the most important developments.

The second part describes the financing methods for waste disposal chosen by different countries.

The third part summarises the results of the European Arcas project and compares the cost of the fuel cycle in different scenarios.

The fourth part gives an overview of the approach implemented in different countries with regard to the decommissioning and release of materials, facilities and nuclear sites.

The final part presents the recent conclusions of the enquiry following the WIPP incident.

6.1 RECENT DEVELOPMENTS

France, Finland and Sweden are the three countries in which significant progress has been made in the process to obtain permission to create geological repositories. These are countries in which sites have been identified and studied and in which construction licensing applications have been, or are about to be, submitted to the authorities.

In Finland, the safety authority – Stuk – has informed the government that the encapsulation plant and the disposal facility for spent fuels proposed by Posiva could be built under conditions of guaranteed safety. The government's formal decision to grant the building permit is expected before the summer.

In Sweden, the authorisation process for the disposal site is in progress, according to the provisions of the Environment Code and the decree on nuclear activities. The authorities (the Environmental Court, the safety authority, the municipalities concerned and the government) are expected to make their final decision in several years' time).

In two European countries – the United Kingdom and Germany – searches have resumed to identify suitable sites for the location of geological repositories. In both cases, the governments have defined a step-by-step approach with the increased participation of stakeholders, especially local ones.

In the United Kingdom, a body with responsibility for this task – Radioactive Waste Management (RWM Ltd), has been created by the Nuclear Decommissioning Authority, NDA, as an independent subsidiary. The process to be used for choosing a site is currently being defined. RWM has prepared a series of geological criteria in order to limit the choices before involving candidate municipalities.

In Germany, a Site Selection Commission has been established. Its members have very different backgrounds. The works began on the basis of a law adopted in 2013, which defines the site selection methodology. The working methods of the commission which is responsible for proposing a site in 2031 are not clearly defined.

In the United States, no concrete measures have yet been implemented following the Blue Ribbon commission's report. Several legal proceedings are still underway following the withdrawal of the Yucca Mountain project by the current administration. The operating companies are also demanding compensation from the US-DOE, given that the spent fuel is still stored locally on their sites, even after finalisation of production by the plants. After shutdown, there is strong pressure for the fuel to be removed, so that the site can be released for other activities. In this context, the regulator (NRC) has decided that the long-term dry storage of the spent fuel constituted a safe and reliable solution for several decades. This is the interim solution that will be implemented by the USA in the future.

Canada has published the *Canadian National Report for the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste*[1], drawn up in the framework of a collaborative project involving the government, the nuclear sector and the regulatory body. The waste producers are responsible for the financing, organisation, management and operation of waste management and disposal facilities. The management is regulated by the Canadian Nuclear Safety Commission – CNSC. The report stipulates that the spent fuel and radioactive waste are currently managed in temporary storage facilities that are safe, secure and environmentally acceptable. They are constantly monitored by permit holders and the regulatory body. For the long-term management of its spent fuel and radioactive waste, Canada undertakes to implement approaches that do not impose an unwarranted burden on future generations and is endeavouring to seek solutions.

The Nuclear Waste Management Organization (NWMO) is responsible for the long-term management of spent nuclear fuel. The report describes the progress made in the selection process for the site of a storage facility in deep geological formations. The process consists of successive phases that gradually reduce the number of study areas among the eligible communities. At present, 10 communities have been selected for the second phase of studies, which will consist of geological assessments followed by the drilling of boreholes. The NWMO estimates that it will take several years to identify a preferred site in a host community that has been duly informed and is prepared to accommodate it.

An opinion on the management of low and intermediate-level waste has been issued by a federal commission appointed after the filing of a construction application by Ontario Power Generation. The commission recommends that the government authorise a storage facility in geological formations at a depth of a 680 metres in argillaceous limestone on the site of the Bruce nuclear complex, situated in the municipality of Kincardine in Ontario. The storage facility would be built and operated by Ontario Power Generation.

6.2 COSTS OF A GEOLOGICAL REPOSITORY

6.2.1 The EDRAM association

In 2012, the International association for environmentally safe disposal of radioactive materials (EDRAM), whose members include the main bodies responsible for radioactive waste disposal (Andra, BfS/DBE, NAGRA, RWM, NWMO, ONDRAF/NIRAS, POSIVA, SKB, ENRESA, NUMO and OCRWM) published a methodology for evaluating the cost of a geological repository. This was the first time such an exercise had been undertaken due to the significant time lag between expenditure, which is spread over several generations, and the funding which must be provided by the current generation that produces the waste. These revenues must be managed via mechanisms that are not likely to disadvantage future generations in relation to the current generation, or vice-versa. The estimation of future costs, and consequently the total amount of provisions to be set aside from the present day to cover these costs, requires a balance to be struck between the expected benefits of economic growth and the risk of unforeseen expenditure that only increases over time.

[1] Canadian Nuclear Safety Commission (CNSC) 2014, CC172-23/2014F-PDF

Several parameters influence the cost of a repository. The most important is the size of the nuclear programme of the country in question. Other parameters, which differ from one estimate to another, are the depth of the repository, the cooling time of the waste prior to its disposal, the type of packaging used, whether or not retrievability needs to be incorporated into the design of the repository, and the direct disposal of spent fuels or of vitrified waste. Despite all of these differences, the majority of the estimations conclude that the cost of a geological repository only amounts to a few percent of the electricity production cost.

The methods used to finance the management of waste and dismantling vary from one country to another. However, the main objective is always the same: to ensure that all of the required funds are available for use at the right time. Three main types of approaches are favoured by different countries.

- Provisions entered on the balance sheet
    - The sums required to cover the costs of the management of waste and dismantling are entered on the company's balance sheet as debt. As the work on the project progresses, the company must make sure that it has sufficient assets and liquidities to pay for it.
- Dedicated internal funds
    - Payments are made into a dedicated fund throughout the entire service life of a nuclear facility. The company manages the funds internally. The fund management rules vary, but several countries allow the funds, or a proportion of them, to be reinvested in the company, subject to compliance with certain guarantees.
- Dedicated external funds
    - Payments are made into a fund that is either managed by the government or by a group of independent trustees. The management rules vary. Certain countries demand that the use of the fund be reserved for the management of waste or dismantling, while other countries allow producers to borrow from the fund to finance their investments.

The waste management chain involves different stages that include the producer's activities, e.g. the operation or dismantling of plants, the packaging of waste, their temporary storage, transportation and packaging with a view to disposal, in addition to the disposal itself. The calculation of the cost depends on the stages that are considered to form part of the geological disposal project. Beside the cost of the technological stages, there are also the costs of R&D, administrative procedures, training, economic and social support, etc., which may vary significantly from one country to another.

The initial difficulties encountered by the EDRAM ad-hoc work group include non-standardised terminology, differences in building regulations, techniques and standards, accounting techniques and evaluation methods. The ad-hoc work group thus adopted a simple and standardised cost matrix structure.

Provisions are set aside by different means including charges when waste is accepted by the organisation responsible for their future disposal. These charges depend on the quota of waste concerned in the global cost of disposal and therefore on their inventory (physical, chemical, biological nature and respective volumes), in addition to the schedule envisaged for their disposal. The global cost also depends on the site (type and depth of the host rock, lengths and volumes to be excavated, chosen technologies, etc.), the packaging and final containment methods, and the rate of disposal. To this must be added the spending on human resources, estimated for the entire service life of the site, including the monitoring period, and thus over a period covering several generations. Finally, this total must also include the costs of R&D and of maintaining knowledge throughout the service life of the repository.

If relevant to the country or management mode in question, EDRAM shall also add the following to its cost matrix: transport costs, fitting of overpacks, support for local communities, historical costs, VAT and, of course, a contingency reserve. Finally, the methodology includes the choice of a discount rate – either fixed or variable over time – to take account of the "value of time" for the different phases in the life of the repository: preparatory works, construction, operation, closure and post-closure monitoring.

The benefit of this method is that it offers the likelihood of there being little disparity between the estimated costs and the actual costs, provided that the physical and radiological inventories are well known and the term is in the not too distant future. The disadvantages are that in the long term, the available technologies, the unit costs of the raw materials and labour, in addition to the regulatory and societal requirements, may have evolved in unforeseen ways. The method seems to be a compromise between the desire to avoid overestimating the future costs so that capital is not tied up unnecessarily, and the desire to protect future generations from the underestimation of these same costs.

The EDRAM document also proposes a methodology for comparing disposal costs in different countries. To this end, a common monetary unit with a value in the same reference year must be chosen. The costs must be nominal with no margin for contingencies. Indeed, these contingencies vary significantly from one country or scenario to another. It should also be noted that economies of scale only apply to a proportion of the costs.

6.2.2 Sweden

In Sweden, companies that are licensed to operate nuclear plants are responsible for the safety of their plants and for the disposal of the spent fuel and waste that are generated. They are also responsible for the decommissioning and dismantling of reactors at the end of their service life. These responsibilities include any measures required until the complete decommissioning and closure of the repository.

The most important measures consist of planning, building and operating the required facilities and of carrying out R&D for this purpose. These measures are financed by the waste-producing companies which must therefore pay licence fees into a dedicated fund managed by the State throughout the service lives of these plants and even longer if necessary. In addition to the licence fees, the companies must offer additional guarantees to the State in order to ensure the funding of the repository in the event of unforeseen circumstances.

The licence fees are paid into the fund which places them in public securities via the National Debt Office. The licence fees already paid are used to fund the expenditure undertaken by SKB.

The system of licence fees and securities is determined by the government. This constitutes the regulatory framework for the funding. Barsebäck Kraft AB – a company whose two reactors were shut down in 1999 and 2005 respectively, may still have to contribute to the funds.

The holder of the operating licence for the reactor pays the licence fee calculated on the basis of the electricity generated. This currently concerns three licence holders: Forsmark Kraftgrupp AB, OKG Aktiebolag and Ringhals AB. These companies are also owners of SKB, the company responsible for managing fuel and waste. On behalf of its owners, SKB is responsible for all of the measures required for the management and storage of spent fuel and waste.

The holder of the license to operate the repository, in consultation with the other owners, shall calculate the management and disposal costs for the spent fuel and waste generated, in addition to the costs of dismantling plants. The licence holders have appointed SKB to perform these calculations.

The government has decided that SKB's calculations shall be submitted to the Swedish safety authority in order to prepare proposals for licence fees and guarantees. The files submitted by SKB are monitored by a dozen or so experts (legal, scientific and financial) belonging to a department of the Swedish safety authority. Decisions concerning the total amounts are made by the government after discussion with the parties concerned (SKB, the owners of SKB and the safety authority).

The licence fees are paid and the securities are pledged, when required, both during the service lives of the reactors and after their final shutdown, dismantling and disposal of fuels and waste.

The amount of spent fuel and waste to be disposed of depends on how long the reactors have been operating. The regulatory framework stipulates that SKB's calculations shall take account of an average life span of 40 years for each reactor currently in operation.

Alongside the licence fees, the companies must provide two types of guarantees, one covering the licence fees that have not yet been paid and the other concerning contingencies. The total amount of the guarantees becomes due if the companies fail to honour their obligations to pay the licence fees or if the assets in the fund are deemed to be insufficient.

The regulatory framework stipulates that the cost calculation shall be updated and submitted to the authorities every three years.

The Swedish financing system was created in around 1980 and has been gradually improved since then. The fund contains sufficient amounts to cover the expenditure for the next 25 years. Licence fees and investment incomes should be sufficient to cover the remaining costs. The guarantee payments are placed in large, stable companies. The system's great strength can be attributed in great part to the triennial review of costs and decisions concerning licence fees and guarantees. In this manner, the estimations can be corrected over time.

The total cost to be covered is estimated at €14 billion. This sum covers the previous and future costs of the nuclear programme of 12 reactors with an overall capacity of 10 GWe, operating for 50 to 60 years. This sum applies to the transportation, temporary storage, encapsulation and geological storage of all of the spent fuel, in addition to the dismantling of the 12 reactors. It also covers the waste disposal costs, except for waste destined for non-geological disposal. The cost of the encapsulation and storage plant accounts for approximately one third of the total cost. Dismantling is estimated at €2.5 billion. The annual licence fee has varied between 1 and 4 öre/kWh since 1982 (0.1 – 0.4 centime/kWh).

6.2.3 Belgium[2]

As the cost of a repository depends on its envisaged inventory, the classification of the waste destined for disposal must be specified. At this point, it should be noted that Belgium does not consider irradiated fuel to be a waste.

As proposed by ONDRAF/NIRAS – the national body responsible for the management of radioactive waste and enriched fissile materials, Belgium has adopted a three-category classification based mainly on the level of activity and the lifetime of radionuclides:

- *Category A waste* is packaged, short-lived, low and intermediate-level waste containing very limited quantities of long-lived radionuclides. It is destined for surface disposal.

- *Category B waste* is packaged, low and intermediate-level waste, contaminated by significant quantities of long-lived radionuclides.

- *Category C waste* is packaged, high-level waste containing large quantities of long-lived radionuclides. Its thermal power when packaged is and will continue to remain high for a long time beyond the period currently considered for its storage.

Category B and C waste requires disposal in geological layers.

The cost of geological disposal has been estimated according to a bottom-up approach, in which the costs are evaluated by adding up all of the current and anticipated costs (building, equipment, management, human resources, working time and any other costs relating to the disposal activity). All of the costs are allocated according to a chronology with milestones. A discount rate is applied

[2] Main sources: Plan Déchets (Waste Plan), NIROND 2011-02 F, September 2011 ; Cost evaluation of geological disposal of category B&C waste…,NIROND TR 2009-15 E, August 2009.

[3] Contract ENER/2012/NUCL/SI2.643067, http://www.bruegel.org/fileadmin/bruegel_files/Events/Event_materials/2014/February/Synthesis_economics_nuclear_20131127-0_reduced_size.pdf

at a value of 2% excluding inflation. The grand total corresponds to the total cost of the project as a current net value on the chosen scheduled date. As the methodology was devised prior to the EDRAM recommendations, it is not totally compatible with them. Among other costs, it includes transport costs and a contingency margin of 30 to 35% of the global cost. This percentage is consistent with the estimation of the "Synthesis on the Economics of Nuclear Energy" study carried out for the European Commission in 2013[3].

The most recent estimate (2009) of the total, non-updated cost is approximately €3 billion2008 over a century, including technological and project contingencies, for a geological repository in Boom clay at an approximate depth of 220 metres, assuming the complete reprocessing of all commercial fuels (7 plants with a service life of 40 years and an installed capacity of 6 GWe).

The proportion of this total cost due to construction is 40% or less, whereas the share of the cost for human resources that can be directly attributed to geological disposal is 40% or more. The total cost of the R&D activities relating to geological storage in Belgium, including the costs of the underground laboratory, was estimated at approximately €360 M2008 for the 1974-2014 period, or approximately €9 M per year. The cost of the R&D activities will be largely determined by the exact scope of the future decision in principle on geological disposal and any additional societal demands.

This estimate is the fruit of multi-disciplinary work overseen by ONDRAF/NIRAS, in collaboration with Synatom (the company responsible for managing the entire fuel cycle for Belgian plants, from the procurement and management of the fuel through to its disposal site) and the Belgian State (responsible for certain historical waste). The engineering company Belgatom supplied much of the technical data and the unit costs to be used. Andra carried out a final reading prior to a peer review performed by the German firm DBE-TEC, drawing on its experience of building and managing facilities at Morsleben, Gorleben and Konrad, which were also designed for possible geological disposal.

The estimate of disposal and R&D costs allows ONDRAF/NIRAS to set the prices for accepting waste deliveries. Each waste producer's share of the final disposal cost is determined according to the quantities and nature of the waste envisaged in the realistic production scenarios that each producer is required to submit to the organisation. A margin is applied to cover contingencies.

The methodology and resulting prices are discussed by the ONDRAF/NIRAS technical committee whose members include the representatives of waste producers. After agreement, the prices are officially applied in contracts.

A Nuclear Provisions Commission has been created in the framework of the *Act on the provisions set aside for the dismantling of nuclear plants and for the management of irradiated fissile materials in these plants.* It consists of six representatives of the public authorities, two experts and three representatives of Synatom. The Commission has consultative and monitoring powers for the constitution and management of provisions for the dismantling of nuclear plants and the management of irradiated fissile materials. As soon as Belgium has made a decision in principle on the disposal of wastes in categories B and C and on the reprocessing of irradiated fissile materials, the Commission will play a more important role regarding provisions concerning final disposal.

6.2.4 Finland

In Finland, nuclear operators pay contributions into an external and national fund for the management of radioactive waste, managed by the Ministry for Trade and Industry. The contributions are due during the plants' first 25 years in operation. Operators can borrow up to 75% of the fund for their own investments. In 2014, the fund had accumulated €2.38 billion.

9,000 tonnes of storage capacity is provided for spent fuel, corresponding to approximately 4,500 copper containers. The current plans provide for 137 tunnels with a total length of 42 km, occupying a surface area of 2 to 3 km^2. 1.3 million m^3 of bedrock will need to be excavated.

The most recent cost estimate for the geological repository at Olkiluoto is €3.3 billion for all radioactive waste. This includes storage operations lasting until 2120 (€2.4 billion) and dismantling (€200 million). The current capacity of the installed fleet is 2.8 GWe, but this will increase due to the EPR under construction.

6.2.5 Germany

The producing companies pay contributions into dedicated internal funds that they manage and can use for their investments.

6.2.6 Spain

Contributions are paid into an external fund managed by the public corporation Enresa, on a prorata basis at the rate of €3 /MWh sold. Enresa is responsible for dismantling plants and waste management.

6.2.7 United Kingdom

The government, through the Nuclear Decommissioning Authority (NDA), is responsible for the long-term management of the country's historical liabilities and commitments. These operations are financed by the government on an annual basis (approximately €2.8 billion) and by the NDA's commercial revenues (approximately €1.6 billion). The annual budget for liability remediation activities is approximately €4 billion.

The private sector (EDF Energy) possesses a fund for nuclear liabilities to cover the dismantling costs of AGR and PWR-type reactors situated at Sizewell B. The costs of the fuel cycle and short-term liabilities are entered as provisions in the company's accounts.

6.2.8 United States

For several decades, nuclear electricity-generating companies have been paying $0.1 cent/kWh into a dedicated nuclear waste fund, managed by the Department of Energy (US-DOE). The DOE has the legal responsibility for managing spent fuel from plants. The fund has now accumulated approximately €40 billion. As the DOE has been unable to fulfil its obligations for the storage and disposal of spent fuel, it has been obliged to stop collecting funds and had already to reimburse some producers. By the end of 2013, 33 lawsuits and 26 decisions had led to refunds amounting to €3.2 billion. It is estimated that the total reimbursements will be limited to 22 billion if the DOE can honour its commitments from 2021.

For the historical waste on the 107 national laboratory sites, the DOE has launched a major environmental remediation programme in which 90 sites have been cleaned up to date. This entails a global financial effort of around €6 billion per year.

6.3 COMPARISON OF FUEL CYCLE COSTS – ARCAS PROJECT

A comparison of the fuel cycle with transmutation in an ADS and/or a FNR has been carried out in the framework of the Arcas project[4].

A comprehensive report on the findings of this project is outside of the framework of this report[5]. Nevertheless, it is worthwhile mentioning some of its aspects and findings.

The reference scenario considered in the framework of the project is derived from the Pateros project (Partitioning and transmutation European roadmap for sustainable nuclear energy; 2006-2008, R&D FP6, 11 countries, 17 partners including the CEA, CNRS and Areva). The following scenario was analysed: the UOx and MOx fuels from a fleet of PWRs are reprocessed in order

[4] ADS and fast Reactor Comparison Study in support of Strategic Research Agenda of SNETP, R&D FP7, 1/10/2010 – 31/03/2013, 8 countries and 14 partners, including the CNRS, CIEMAT, KIT, JRC-ITU, NRG, NRI, UPM, TNB, NNL, UNIMAN and SCK•CEN)

[5] http://cordis.europa.eu/docs/results/249704/final1-arcas-249704-project-final-report-final-description-of-main-s-and-t-results-foregrounds-.pdf

to separate the U, Pu and the minor actinides from the fission products. The fission products are stored in a geological repository. The Pu and the minor actinides are recycled in several cycles in the ADS or FNRs. Without reference to the legislation in force in the different countries, the aim is to recycle the spent fuel from countries that are abandoning their nuclear programmes in order to remove all of the minor actinides from it, manage the plutonium and stabilise the minor actinide inventory of countries that are continuing to exploit nuclear energy.

The code used for modelling the cycles (COSI6 – ver. 6.0.1) was developed by the CEA.

Two methods were used for calculating the costs. The first method, based on a GIF tool (NRG-G4Econs), calculates the cost per kWh and per tonne of actinides destroyed. The other method (developed by the CNRS) considers different scenarios: RNRs alone; RNRs and ADS; ADS alone. The costs, estimated in an independent manner by the CNRS and NRG according to the two different methods, include the investments, operation and the fuel cycle.

The following conclusions emerged:

- as soon a fleet of PWRs is in operation, if transmutation (Pu and minor actinides) is not imposed, the operators will have no economic reason to choose it. The direct storage of the spent fuels (spent UOx and MOx) will remain the preferred technology;

- if the Pu is managed in a fleet consisting of PWRs and FNRs, the minor actinide inventory can be stabilised by their transmutation in dedicated FNRs;

- despite the high construction and operating costs of the ADS, their transmutation efficiency is such that only a small number of them will be required in relation to the number of FNRs dedicated to transmutation. For a fleet with the same electrical capacity, the cost of the ADS reactors would thus be offset by a larger proportion of PWRs continuing to operate in a single stratum.

The general conclusion of the project is that the costs of the different scenarios are of the same order of magnitude. The double-strata, closed cycle scenarios would cost 15 to 30% more, but there would be a significant reduction in long-lived waste. The single-stratum scenario using FNRs exclusively would be more expensive. These estimates are based on future technologies that are still being developed and thus remain indicative.

6.4 DISMANTLING AND DECOMMISSIONING OR RELEASE

The dismantling of nuclear facilities and the remediation of sites on which nuclear activity has taken place are increasingly important sectors of the nuclear industry. This concerns old research reactors, the first and second generations of electricity-generating reactors and certain nuclear weapons-related facilities. In the United States, the United Kingdom and Russia, these dismantling and remediation activities are particularly important for specific and complex reasons.

These dismantling and remediation activities are governed by criteria and conditions that define whether sites can be considered to be rehabilitated and available for non-nuclear activities. They are also governed by rules that state whether decontaminated equipment can be released or decommissioned and managed in the same way as non-radioactive equipment.

In terms of definitions, a distinction is made between exclusion, exemption and release. Due to the lack of international harmonisation, there are significant variations in the regulations and practices from one country to another.

Exclusion is the decision to remove certain materials, facilities or practices from regulated control. For example, this could concern natural radioactivity, such as that found in the human body, certain types of mineral water and the Earth's crust.

Exemption is a generic decision that, in principle, is made in order to exempt materials, facilities or practices from regulated control, as their characteristics are such that they pose no risk to health and that any control would consequently be superfluous. An example is the exemption of foods in which the concentration of artificial radionuclides is below a certain predefined threshold.

Release (or decommissioning) is a retroactive decision, made by the inspection body, to release materials or facilities originating from a human activity that is itself monitored, provided that their radiological characteristics are below a certain threshold. The release criteria are based on the "trivial dose" concept, which implies that there is no risk of contamination and that the dose is below the standards of the regulations in force. Internationally, an individual dose level of below 10 μSv/y is considered to be "trivial". An example is the release of decontaminated materials originating from the dismantling of nuclear facilities. In general, the release threshold is higher than the exemption threshold.

National approaches to the notion of a release threshold are strongly dependent on the classification of radioactive waste. In France, there are thus four major categories of waste, i.e. high, intermediate, low and very low-level, but there is no release threshold for materials. Belgium employs three categories of waste – high, intermediate and low-level – and applies release thresholds. In Italy, there are also three categories of waste and several release regulations. Germany also has three categories of waste and release criteria.

The Swedish approach is described below, as an example, in greater detail:

- The radiation safety authority – SSM – has published the regulations for release (SSMFS 2011:2). They have been in force since 2012. They cover materials, facilities, buildings and sites that may have been contaminated during authorised nuclear activities. The aim of the regulations is to facilitate the rational and satisfactory use (from the perspective of radioprotection) of these materials, facilities, buildings and sites. The regulations specify the requirements regarding measurements, reports and safety assessments. They also include tables – per radionuclide – of the levels of surface contamination or concentration that satisfy the release criteria. The release itself can only be decided upon by the SSM, after the submission of a detailed report by the operator of the facility.

- On the Studsvik Nuclear site at Nyköping, the potentially contaminated metal – mainly iron and a little copper – is melted down. This produces contaminated residues, on the one hand, which are returned to the producers (national or international), and releasable metal, on the other. The latter accounts for 80 to 90% of the processed metal. It can be recycled in the conventional sector subject to compliance with a certain number of rules.

6.5 UPDATE CONCERNING THE WIPP INCIDENT

In March 2015, the Savannah River US national laboratory publicly released the "Waste Isolation Pilot Plant Technical Assessment Team Report"[6], prepared on behalf of the Department of Energy (DOE).

The main observations are:

1. Drum no. 68660 contained chemically incompatible substances.

2. The lid on this drum was unable to withstand the increase in temperature and pressure.

3. This drum was the sole cause of contamination of the site.

4. The cause of the uncontrolled reaction occurred exclusively inside the drum and was not due to any external factors.

[6] SRNL-RP-2014-01198, http://www.wipp.energy.gov/Special/TECHNICAL_ASSESSMENT_TEAM_REPORT.pdf

5. The thermal and pressure effects caused the damage observed in the area.

6. There was no explosion.

The main conclusion of the report was that the barrel (no. 68660) contained chemically incompatible substances. The configuration of the materials inside the container, combined with this incompatibility, caused an uncontrolled exothermic reaction. The rise in pressure due to the build-up of gas inside the drum caused the lid to move. This allowed the gases and radioactive materials to react with the air and other materials outside the drum, causing the damage observed in the WIPP P7R7 zone.

It should be noted that no members of staff were situated inside the storage area at the time of the incident and that no external contamination of the staff was observed. However, laboratory analyses showed that 21 people showed signs of a slight internal contamination. Traces of radioactive substances were observed outside the site. It should also be noted that, at least for the drum in question, the quality assurance procedure for the continuity of procurement was flawed: the absorbent material used originally was replaced by an unqualified organic absorbent material.

The enquiry revealed that there had been failings at the Los Alamos National Laboratory in the understanding and application of the rules that had been established for the manufacture of these waste drums. This incident emphasises the critical role of quality assurance throughout the entire waste package creation chain.

Appendix I

COMPOSITION OF THE ASSESSMENT BOARD

JUNE 2015

Jean-Claude DUPLESSY – Chairman of the National Assessment Board, Member of the French Academy of Sciences, CNRS Project leader emeritus.

Pierre BEREST – Expert invited by the National Assessment Board, Project leader at Ecole Polytechnique.

Adolf BIRKHOFER – Expert invited by the National Assessment Board, Professor emeritus at the Munich Technical University.

Frank DECONINCK – Honorary Chairman of the Belgian Nuclear Research Centre, SCK CEN – Professor emeritus at Vrije Universiteit Brussel.

Pierre DEMEULENAERE – Professor of Sociology, University of Paris-Sorbonne (Paris IV).

Robert GUILLAUMONT – Member of the French Academy of Sciences (chemistry section), Member of the French Academy of Technologies – Honorary Professor at the University of Paris XI Orsay.

Maurice LAURENT – Honorary director of the Parliamentary office for evaluating scientific and technological choices.

Emmanuel LEDOUX – Vice-chairman of the National Assessment Board – Honorary project leader at the Paris Ecole des mines, Mines-Paristech.

Maurice LEROY – Vice-chairman of the National Assessment Board, Associate member of the French national academy of pharmacy, Professor emeritus at the University of Strasbourg.

Jacques PERCEBOIS – Director of the CREDEN (Centre for economics and energy law research Professor emeritus at the University of Montpellier.

Gilles PIJAUDIER-CABOT – Director of the research laboratory for complex fluids and their vessels – Senior member of the Institut Universitaire de France – Professor at the University of Pau and the Pays de l'Adour.

François ROURE – Professor and scientific expert at IFP-Energie Nouvelles, Adjunct professor, University of Utrecht.

Claes THEGERSTRÖM – Former President of SKB (Swedish company in charge of managing nuclear fuel and waste), Member of the Royal Swedish academy of engineering science.

Appendix II

ORGANISATIONS HEARD BY THE BOARD

22 October 2014:	CEA – Materials for Astrid and its fuel
23 October 2014:	Andra – Clarification of the schedule until the Construction application (DAC) - Final outline for Cigéo: phases and optimisations
26 November 2014:	CEA & CNRS – Which programmes and materials for transmutation? What upstream research needs to be implemented and coordinated?
27 November 2014:	Andra – Cigéo scientific support programme 1) up to the DAC; 2) phase 1 of Cigéo
21 January 2015:	Andra & Producers – LLLLW: review of sites and research in progress
22 January 2015:	CEA & Producers & Andra – Results of the Bitumen research programme
11 March 2015:	Andra – Specifications for LLILW packages
12 March 2015:	CEA – Astrid and FNR deployment scenarios

RESTRICTED HEARINGS

17 September 2014:	CEA - morning
17 September 2014:	EDF - afternoon
18 September 2014:	Andra - morning
18 September 2014:	Areva - afternoon
13 November 2014:	ASN - morning
18 March 2015:	CNE/ANDRA technical meeting

BOARD HEARINGS

5 February 2015:	Hearing of the Board by the CLIS

CNE2 VISITS

04 September 2014:	Visit to the Lagunes and COMURHEX II plants on the Areva Malvési site
09 April 2015:	Visit to the Areva Creusot forge and Areva Saint-Marcel sites – Areva Chalon-sur-Saône

Appendix III

LIST OF PERSONS HEARD BY THE BOARD

ANDRA

ABADIE Pierre-Marie
BAUER Corinne
BUMBIELER Frédéric
BOISSIER Fabrice
BONNEVILLE Alain
BOSGIRAUD Jean-Michel
BRULHET Jacques
BUMBIELER Frédéric
CAIRON Philippe
CALSYN Laurent
CLAUDEL David
CRUSSET Didier
DUBIEZ-LEGOFF Sophie
DUPUIS Didier
DUPUIS Marie-Claude
DUTZER Michel
GERARD Fanny
GETREY Christophe
GIFFAUT Eric
HARMAN Alain
HURET Emilia
KRIEGUER Jean-Marie
LABALETTE Thibaud
LAGRANGE Marie-Hélène
LEVERD Pascal
LIEBARD Florence
ROMERO Marie-Ange
ROUX-NEDELEC Pascale
SCHUMACHER Stephan
SENENTZ Gérald
SEYEDI Darius
TALANDIER Jean
TALLEC Michèle
VOINIS Sylvie
YVEN Béatrice

AREVA

BOSSE Emilie
BOSGIRAUD Jean-Michel
DREVON Caroline
DUBIEZ-LEGOFF Sophie
FORBES Pierre
GAGNER Laurent
LEBRUN Marc
SENENTZ Gérald
VALOT Vincent

CEA

ADVOCAT Thierry
BEHAR Christophe
BOUFFARD Serge
BOULLIS Bernard
CARON Nadège
CHABERT Christine
COURCELLE Arnaud
FRIZON Fabien
GARNIER Jean-Claude
GAUCHE François
JOMARD Gérald
JOURDA Paul
LATGE Christian
LE FLEM Marion
PARET Laurent
PEYCELON Hugues
PELLETIER Michel
PETESCH Cécile
PHELIP Mayeul
PILLON Sylvie
PIKETTI Laurence
ROBIN Raphael
ROUAULT Jacques
SATURNIN Anne
SAUSAY Maxime
TOURON Emmanuel

CNRS

BACRI Charles-Olivier
BIARROTTE Jean-Luc
BILLEBAUD Annick
BOUSSON Sébastien
DAVID Sylvain
MARTINO Jacques
RENAULT Anne

EDF

BLAT Martine
BUTTIN Jérémy
LAUGIER Frédéric
LASSABATERE Thierry
PACQUENTIN Didier
PAYS Michel

Appendix IV

THE BOARD'S VISIT TO MALVÉSI SITE

In its Report no. 8, the Board reviewed the waste produced by the Malvési Comurhex plant and its management, mainly on the basis of the information contained in the national waste inventory and the 2012 PNGMDR (National plan for the management of radioactive materials and waste). This plant produces very pure uranium tetrafluoride, UF_4, which is then converted to hexafluoride, UF_6, at Pierrelatte. This type of fluoride is the compound that allows for the industrial enrichment of natural uranium into ^{235}U which gives it its "nuclear" quality. The enriched uranium is used to produce UOx fuel for thermal neutron reactors. The nominal capacity of the Malvési plant is 14,000 tonnes of UF_4 per year. In September 2014, the Board visited the liquid effluent treatment facilities, the facilities used for storing solid waste and Areva Malvési's new Comurhex-II plant. During this visit, it obtained additional updated information, concerning the characteristics of the waste and its current or proposed future management.

WASTE AND EFFLUENT FROM THE UF_4 MANUFACTURING PROCESS

The plant has treated and continues to treat several commercial uranium products. The main product is Yellow Cake (YC) containing di-uranates/uranates ($U_2O_7(NH_4)_2$, $U_2O_7Mg_2$, $MgUO_4$), depending on the provenance, and other uranium compounds (U_3O_8, UO_4, hydroxides, etc.). The YC is delivered in 220 L drums. It originates from the conventional processing of uranium ore, which eliminates virtually all of the radioactive daughter products of natural uranium isotopes above 234U (however, there are remaining traces of 226Ra and 231Pa); YC also contains 15 to 30% of non-radioactive miscellaneous chemical elements (Na, Si, Fe, S and Ca, etc.) and thorium (up to 1%) containing the 230Th isotope.

The UF_4 production process begins with dissolving YC in nitric acid and clarifying the solution to separate the insoluble oxides (Si, Ti, etc.) which are removed as waste. This dissolution produces nitrogen oxides which are recycled. The next stage is the purification of the uranium by solvent extraction with the TBP diluted in kerosene. This leads to nitric solutions with varying levels of acidity (extraction raffinates) containing all of the elements and radioelements other than uranium. The uranium is recovered from the organic extraction solution after being returned to a weak nitric acid solution from where it is precipitated out to the $U_2O_7(NH_4)_2$ state. The supernatant rejoins the initial solutions. All of the solutions constitute the purification effluent flow: aqueous nitric acid solutions containing different elements in addition to traces of uranium due to in-line process losses.

The ammonium di-uranate is then converted into UO_3 via hot air when placed in a furnace. The UO_3 is then continuously converted into UO_2 and UF_4 in a second furnace. This is firstly supplied with NH_3 (UO_3 reduction zone at 800°C) and then with anhydrous HF (UO_2 fluorination zone at 350/450°C). A surplus of HF is required. The gaseous effluents from the last two conversion reactions (N_2, H_2O, NH_3) and surplus HF are recovered and processed. This generates what amounts to a second liquid effluent flow of fluorides and hydrofluoric acid. The nitrogen oxides produced during the preparation of UO_3 are recycled.

The two process flows are combined and neutralised with lime, which leads to the precipitation of the heavy elements in the form of reasonably well defined hydroxides, calcium carbonate, CO_3Ca, and fluorine, F_2Ca (solid fraction or sludge), producing a liquid supernatant containing a high proportion of nitrates and soluble salts such as sodium carbonate, CO_3Na_2. The mixture that forms the final process effluent is sent for lagooning.

Since 1960, the plant has converted over 400,000 tonnes of uranium into UF_4. Areva estimates the total uranium losses at 1.7%, which leaves approximately 700 kg of uranium remaining in all of the waste on the site. The processing of one tonne of uranium produces approximately 4 to 5 m^3 of effluent. Between 1960 and 1983, batches of UF4 were produced from reprocessing uranium containing impurities such as artificial radionuclides (^{99}Tc, $^{238/241}Pu$, etc.) which are found in some of the solid waste and evaporating solutions.

MANAGEMENT OF EFFLUENTS BY "LAGOONING" AND SOLID WASTE BY STORAGE ON THE SITE

A series of basins cover 30 hectares of this 100-hectare site, which allow for the settling of the solid fraction in the final effluent from the plant (basins B3, B5 and B6) and the evaporation/concentration of the liquid fraction (basins B7 to B12). All of these basins have ICPE (Installation classées pour l'environnement – Classified facilities for environmental protection) status. The liquids are transferred from the settling basins to the evaporation basins by gravity. Since the start of UF_4 production, a very large proportion of the sludge and miscellaneous materials has been placed in storage, starting in 2004, in two former basins (BI and B2) and then covered with a blanket of natural materials. On this site, there is a reserve facility that will contain the solid residues from basins B3, B5 and B6 (see below). B3 is currently used for the management of water on the site and is built on dried sludge mixed with earth. Basins BI to B6 have been created on top of residues and tailings from a former sulphur mine, which have been partially contaminated by radionuclides from basins B1 and B2, which are not leaktight, unlike basins B3 to B6 which are. In the area occupied by the evaporation basins, there is also an isolated rainwater regulation basin containing sedimented sludge. All of the solid waste deposits – current and future – in basins B1 and B2 are and shall be included in the INB ECRIN (Basic Nuclear Facility for the contained storage of conversion residues). Since 2012, the INB and basins B3, B5 and B6 have been surrounded by a leaktight, multi-component, 10 m-deep underground wall, anchored in the underlying clay to a depth of 2 m, which is designed to retain the percolating water and thus protect the groundwater. The runoff water is recovered separately. All of this water is treated. The mass balance and radiological evaluation of the waste (contained within and underneath the basins) are given in table 1.

CLASSIFICATION OF SOLID WASTE, HISTORICAL AND PROJECTED INVENTORIES

The solid waste that will be produced prior to the commissioning of the new Comurhex-II plant (scheduled for 2015) is considered by Andra to be historical waste. The physical (bulk waste), chemical (poorly defined compositions) and radiological (natural and artificial radionuclides) characteristics of the waste, and above all the volumes in question, prevent its acceptance at one of the currently operational or planned waste disposal centres. Its specific activity is generally above that of radiferous LLLLW and its natural radionuclides are within the same range as mining waste; however, a significant proportion of this waste is contaminated by artificial radionuclides. Within the waste, uranium and thorium are associated with different mineralogical phases currently being identified and on which Andra is carrying out leaching experiments. The quantity of waste is comparable to that found in small mining waste repositories. The public authorities have also asked Areva to envisage a specific short and long-term *in situ* management procedure for this waste (and for the future waste) (PNGMDR Decrees 2013-2015, articles 7-II and 16-II). That is why the waste from Malvési (Areva) is not included in the current LLLLW inventory.

The modification of the current UF_4 production process, combined with the commissioning of Comurhex-II, must lead to a change in the current production and management of effluents, and lead to the production of VLLW and LLLLW packages. Furthermore, Areva envisages a major project for the processing of evaporating solutions that will also lead to the production of VLLW packages (see below and table 1). This should lead to a reduction in the quantity of process waste to come. Nevertheless, the accumulated volume of "new waste" remains substantial. Areva estimates the overall total at 250-300,000 m^3 by 2050. This quantity could be adjusted downward over time.

In total, therefore, by around 2050, the Malvési centre will house nearly 500,000 m^3 of VLLW and LLLLW-type waste which is likely to be disposed of there. To this must be added 1,300,000 m^3 of sulphur mining waste and tailings, a proportion of which is contaminated. For example, a typical mining waste storage site such as Bellezane contains 1,550,000 tonnes of mining residues generating activity of ^{226}Ra of 50 TBq. The total activity is increased by a factor of 5 if the activity of the uranium, ^{230}Th and mining tailings also stored on the site are included.

SHORT-TERM MANAGEMENT – INB ECRIN

Major operations have taken place since 2000: recovery of waste liabilities, retention wall around bassins B1 to B6 and construction of the new Comurhex-II plant. At the same time, major R&D programmes have also been launched: modification of the UF_4 production process, transformation of liquid effluents into solids and storage of waste packages. Finally, the INB ECRIN has also been established.

In the immediate future, Areva must continue to create the ECRIN facility in order to ensure the long-term containment of the radioactivity from its historical solid waste. After 50% dehydration, the solid waste from basins B3 to B6 (35,000 m^3, 410 Bq/g, 4 to 6 kgU/tonne) will be consolidated in this INB (basic nuclear facility). It will be kept in the storage area next to basins B1 and B2. After the consolidation and remodelling of all of the waste, a multi-layer, bituminous blanket will be laid over the top of it (scheduled for completion in 2017). The site of the emptied basins B3 to B6 shall be used for storing the non-historical waste. The management of waste from Malvési also entails the rinsing and decontamination of YC drums, their compaction and transportation to the CSA (several thousand barrels per year). The INB ECRIN's accommodation and radiological capacities are 400,000 m^3 and 120 TBq respectively.

MEDIUM-TERM MANAGEMENT, COMHUREX-II

The R&D conducted by Areva has led to three innovations culminating in lower reagent consumption, less waste and improved waste management. These innovations are the separation between the purification and fluorination flows, the gradual commissioning of Comurhex-II and the processing of nitrate-containing effluents. Comurhex-II has a nominal capacity of 21,000 tonnes of UF_4 per year.

The separate neutralisation of the process flows will firstly lead to packages of solid, dehydrated radioactive LLLLW packages containing the radionuclides from the purification effluents, which will be stored in hot cells to be constructed on the newly available sites of basins B3 to B6, and secondly, fluorines originating from fluorinated effluents, which can be added to the VLLW circuit. The Comurhex-II process differs from that used in the previous plant primarily by directly converting the uranyl nitrate into UO_3 while separating the effluent flows. Denitration by the "isoflach" process in the presence of natural gas will produce nitrogen oxides that will be recycled in nitric acid. Finally, the major TDN project (processing of nitrates between 2018 and 2050), aims to recover the 450,000 m^3 of liquid effluents from basins B7 to B12 and treat the new effluents in line. These nitrates will eventually be converted into solid VLLW placed in big-bags (silico-aluminates containing the radionuclides), and gaseous effluents (CO_2, N_2, H_2O) that can be discharged into the environment. Without going into detail, the chosen process (Studvik THOR pyrolytic/vapour treatment) has a treatment capacity of 18,000 m^3 of lagooned effluents per year, leading to the production 5,700 m^3 of solid waste.

LONG-TERM MANAGEMENT

Areva is examining the feasibility of a repository for all of the waste and is required to submit a progress report to the authorities at the end of 2014 and a feasibility studies file at the end of 2017. Several options are envisaged: a tumulus under a multi-layer, man-made cover over the area occupied by the current site, a shallow repository (between 20 and 40 m), either in the reducing clay of the adjacent former sulphur mine (under a multi-layer cover), or in the Oligocene marl forming the substratum of the site (under a reworked cover). These studies are conducted according to the recommendations of a group of five international experts.

THE BOARD'S COMMENTS

The Board considers that the research and projects undertaken by Areva are consistent with improving the management of waste from the UF_4 production process. Eventually, only solid waste should remain. The characteristics (chemical and radiological composition) of the waste to be produced should be controlled due to the start of the separate neutralisation of effluents and the TDN programme. Consequently, it will be possible to ascertain their stability and radionuclide containment capacity without too much uncertainty, which will facilitate the impact calculations in the safety analysis for their disposal. On the other hand, the final characteristics of the historical waste will be harder to establish with a view to its disposal. The Board emphasises the need to monitor the evolution of its mineralogy and behaviour with regard to leaching, as is the case for the uranium ore processing residues stored on mining sites. The compositions of these types of waste are similar.

The research into the possibility of on-site waste disposal is consistent with the searches for an LLLLW and VLLW site currently being conducted by Andra. However, the Malvési project is an imposed storage site presenting a restricted choice of locations for facilities. The Board emphasises the importance of knowing the source terms, geological, geodynamic and hydrogeological contexts and the behaviour of radionuclides within the wastes and the environment with a view to adapting the concept to the local constraints.

TABLE 1

<table>
<tr><td colspan="12">Estimated values of the quantities and activities of waste stored on the Areva Malvési site
(Areva and Andra data – National waste inventory)</td></tr>
<tr><td>B1</td><td>B2</td><td>B3</td><td>B5</td><td>B6</td><td>B7</td><td>B8</td><td>B9</td><td>B10</td><td>B11</td><td>B12</td><td>Bx</td></tr>
<tr><td colspan="2">280,000 m3 of dry sludge and miscellaneous materials, 90 TBq to 490 Bq/ (410,000 tonnes) (1)</td><td colspan="3">56,700m^3 of settled sludge, 9 TBq to 230 Bq/g (70,000 tonnes) (2)</td><td></td><td></td><td></td><td></td><td></td><td></td><td>(3)</td></tr>
<tr><td colspan="5">1,300,000 m^3 of sulphur mining tailings of which 200-300,000 m^3 are contaminated at a level of 1 Bq/g.</td><td colspan="7">450,000 m^3 of liquids containing 130,000 tonnes of nitrates (U at less than 0.5 mg/L in U, ^{226}Ra and ^{99}Tc)</td></tr>
<tr><td colspan="12">Details
U at 51 MBq/kg (^{238}U,^{234}U,^{234}Th^{234}Pa), at equal activity, U is 5 10^4 times more abundant than ^{230}Th</td></tr>
<tr><td colspan="12">1- 490 Bq/g of which 380 Bq/g are due to alpha emitters. 50%U and 50% ^{230}Th and 1% other (^{99}Tc-3,1 Bq/g, $^{238\ \mathrm{to}\ 242}$Pu-22 Bq/g, ^{241}Am-1,8 Bq/g, ^{237}Np- 1 Bq/g)
77,000 m^3 of sludge (U hydroxides, ^{230}Th and impurities of YC - Si, Fe, Na, Ca, V, Mo - and CaF_2 contaminated
by artificial radionuclides, 162,000 m^3 of mixed sludge and earth, 43,000 m^3 of covering materials.</td></tr>
<tr><td colspan="12">2- 230 Bq/g of which 160 Bq/g are due to alpha emitters. 23,000 m^3 of miscellaneous waste mixed with inert earth under B3, U for 50 GBq and Ra for 25 GBq, 23,000 m^3 (28,600 tonnes) in B5, U, ^{230}Th, Ra for 6.3 TBq and 14,300 m^3 (17,700 tonnes) in B6 for 2.8 TBq</td></tr>
<tr><td colspan="12">3- Bx: regulation basin, 80,000 m^3 of sedimented mud for 0.4 TBq containing 8.9 tonnes of U, 119 tonnes of Cd, 126 tonnes of Cu, 2 tonnes of Hg and 4 tonnes of Se</td></tr>
<tr><td colspan="12">Estimated values of the quantities and activities of future process waste
(Areva and Andra data, PNGMDR)</td></tr>
<tr><td colspan="5">Purification flow (sulphates)</td><td colspan="3">Fluorination flow (fluorines)</td><td colspan="4">TDN waste (aluminosilicates)</td></tr>
<tr><td colspan="5">450 m^3/year, 1,700 Bq/g, 19.3 kgU/tonne</td><td colspan="3">900 m^3/year, 3 Bq/g, 60 gU/tonne</td><td colspan="4">5,700 m^3/year, 17 Bq/g, 0.5 gU/ tonne</td></tr>
<tr><td colspan="12">All of the solids will contain approximately 50% water and a fraction of soluble salts equal to not more than 10%.</td></tr>
</table>

Appendix V

LIST OF DOCUMENTS SUBMITTED TO THE BOARD IN 2014-2015

ANDRA

- Follow-ups by Andra to the Cigéo project following the public debate – Andra – 2014.
- Update of the French industrial programme for waste management (Programme industriel de gestion des déchets - PIGD) – 3 June 2014.
- Annual progress report on work carried out in the underground research laboratory in 2013 – 24 June 2014.
- Operating report on knowledge of LLLLW packages envisaged for Cigéo – summary of 30 June 2014.
- Progress and sustainable development report – Andra – 2013.
- Andra Journal – National Edition – No. 18 – Summer 2014.
- Marine and petroleum geology document – Vol. 53 - Maurice PAGEL - May 2014.
- Development plan for components of the Cigéo project – Variation according to the TRL scale (ISO 16290:2013) – 5 September 2014.
- Les essentiels 2015 – National inventory of radioactive waste and materials (Inventaire national des matières et déchets radioactifs).
- Report on the equations and values of the parameters adopted in the full-scale simulations of the Thermo-Hydro-Mechanical behaviour of the Collovo-Oxfordian formation – 27 March 2015.
- Factual comparison of the temperatures and actual maximal stresses attained in the Callovo-Oxfordian formation for HL area dewsigns studied since the 2005 File.
- Operating report on knowledge of LLLLW packages envisaged for Cigéo – summary of 30 March 2015.
- Radioactive waste: monitoring report – International monitoring of geological disposal projects for high-level and/or long-lived waste and on the management of radioactive waste April 2015.

CEA

- 2013 Progress report – CEA/DEN/DIR – June 2014.
- 2013 Annual Report.
- Sodium-cooled nuclear reactors – monograph of the Directorate for nuclear energy (Direction de l'énergie nucléaire) – November 2014.
- Memorandum on energy – 2014 Edition.
- Elecnuc – Nuclear plants worldwide – 2014 edition.
- Results of the R&D programme on the behaviour of bitumen-coated packages – 23 December 2014.

Appendix VI

CIGEO: DIMENSIONING OF THE HL ZONE

SPECIFICITY OF HL PACKAGES

In contrast to the majority of intermediate level (IL) waste, high level (HL) packages emit significant amounts of heat. However, the power they produce diminishes over time. For this reason, IL waste will be disposed of first, whereas the majority of HL waste is or will be stored on the surface for several decades, especially in the Areva facilities at La Hague, in order to reduce the power it will be releasing when it enters the disposal facility. The disposal of HL waste is currently being envisaged from 2075 rather than 2045 as mentioned in the 2009 File, and the possibility of 2099 is also being considered.

However, packages of "HLW0" (high-level, weakly exothermic vitrified waste), which will occupy 75 cells, will be an exception to this practice. This type of waste already exists in small quantities and its thermal power is generally moderate. Therefore, it can be disposed of sooner than the other HL waste packages – referred to as HL1 and HL2 – which are more numerous (and have been estimated to require 1,473 cells). More of this waste will be produced in the future in the plants belonging to the electro-nuclear fleet.

Even after the storage period, when these packages are placed in the disposal facility – from 2075 for HL1-HL2 and in around 2040 for the HLW0 – their thermal power will remain high. This will lead to a gradual increase in the temperature of the surrounding rock, of up to several tens of degrees in proximity to the waste packages. The heat thus generated will spread, by thermal conduction, throughout an increasingly large volume of rock before it is eventually released at the surface after a period that is calculated in millennia. However, as the power produced decreases over time, the phenomenon will firstly reach a peak: the temperature in proximity to the packages will reach its highest level (after less than a century in the walls of the HL2 cells but after several centuries at the mid-point between two parallel cells) and will then decrease due to the reduction in the source power and the transfer of the heat throughout the entire body of rock.

DIMENSIONING OF THE HL AREAS

An HL area is generally rectangular in shape. It consists of a series of access tunnels in a parallel configuration, allowing access to the disposal cells laid out at right angles to them. The cells are generally arranged in parallel rows interrupted either by the access tunnels or by around twenty metres of rock between the ends of two consecutive cells. The area is thus criss-crossed by tunnels and rows of cells. The number of cells per row is a key issue in the current optimisations.

The maximum temperature reached in proximity to the packages, exactly when it is reached and other values such as the total energy produced, the spatial and temporal distribution of temperatures and the stresses imparted on the rock, depend on how the repository is dimensioned. This takes account of the duration of the initial storage period for the packages, the length of the cells, the number of packages in a cell, the spaces left between consecutive cells and the spacing between parallel cells.

The dimensioning must take account of another constraint. The disposal permit, which could be granted in around 2020 after the construction application has been examined, will mention a clearly defined underground zone within which the repository must be created. The boundaries of this zone are currently defined by the "Zira" (from the French acronym for zones of interest for further investigation) (CNE Report No. 6). The Zira must be capable of accommodating all of the waste produced by the current fleet. The quantity of waste to be accommodated is defined by agreement on the basis of the current plants having a service life of 50 years and assuming the reprocessing of the materials concerned. These assumptions could become obsolete if the service life of the plants is extended, for example, or if reprocessing is stopped. In the event of the stoppage of reprocessing, the non-reprocessed spent fuels will require direct disposal and will emit more thermal power than the HL waste whose disposal is envisaged at present. Dimensioning margins must therefore be incorporated so that more numerous or "hotter" packages can be accommodated if necessary.

THE DIMENSIONING DEFINED FOR THE CONSTRUCTION APPLICATION (DAC) MUST BE ROBUST

Apart from HLW0, which requires rapid disposal, the disposal of HL waste could thus begin in 2075. It is likely that the dimensioning devised today will not be exactly the same as that adopted in the future: techniques will have changed, knowledge will have progressed and there will be greater hindsight with regard to the behaviour of the rock. Nevertheless, it is essential to demonstrate today – or more precisely in 2017 when the DAC is filed – that we can at least submit a robust design that satisfies all of the acceptance criteria, both for the operating period in which the repository is open, and for the subsequent period during which safety must be maintained in a passive manner. Future generations may modify the design that is adopted today, but proof must be provided right now that such a design does at least exist.

It is thus essential to offer a solution which, although it may certainly be revised, is robust and sufficient.

THERMAL DIMENSIONING CRITERIA FOR AN HL AREA

The heat generated by the waste is generally transferred throughout the rock by conduction. The ventilation of tunnels plays a minor role. Conduction is a well-documented physical process that can generally be quite accurately modelled; indeed, the values of the parameters that describe it in the subsoil are much more consistent than those describing the run-off of water, for example. Spatial and temporal distributions of temperature can thus be calculated reliably. A degree of uncertainty is inevitable but it remains slight.

We need access to criteria that allow us to judge whether the temperatures calculated according to the chosen design are acceptable. The criteria concern the working conditions (temperature of tunnel walls and electrical equipment below 50°C, and temperature of ventilated air below 26°C in the presence of personnel), the resistance of materials (temperature of bituminous packages below 30°C, temperature below 90°C in any part of the rock, below 450°C in the middle of the glass in HL packages and 70°C on the outer edges when contact with water occurs after several millennia, below 65°C in concrete even when there is no ventilation and 80°C in the event of an incident) and, in the much longer term, the absence of mineralogical transformations (that could be caused by a temperature of 70°C maintained for 10,000 years). In the older designs, the criterion of temperature below 90°C had a special significance because meeting this criterion tended to mean automatic compliance with all of the other criteria.

MECHANICAL DIMENSIONING CRITERIA FOR AN HL AREA

The heating up process also has mechanical consequences: the expansion of the rock and water modifies the stresses and the pore pressure in the rock. The first criterion –the absence of shear failures – seems easy to achieve. The second relates to a lack of effective tensile stresses which are liable to cause thermo-hydro-mechanical micro-cracking. Due to recent dimensioning changes, this second mechanical criterion is currently the most critical and means that the criterion of a maximum temperature of 90°C is of secondary importance.

It has the following origin: the rise in temperature around a cell leads to an expansion of the rock and thus to additional stresses. The analysis of these stresses is complicated by the fact that their distribution is the result of two asymmetrical effects. A single cell generates cylindrical isotherms, additional radial compressive stresses and orthoradial stresses which are less compressive and may even be tensile stresses. On the other hand, there is an asymmetry between the horizontal direction, along which there is a large number of parallel cells whose thermal effects are cumulative and limit the movements, and the vertical direction, in which there is a single layer of cells, which means that movements are not prevented.

Furthermore, the rock contains water which, as it heats up, expands much more than the solid fraction. The pressure of the water contained in the pores thus increases. As the rock is not very permeable, the water finds it difficult to flow out and the pressure only decreases very gradually. It is known that micro-cracking of the rock can occur if the water pressure is greater than one of the stresses exerted in the rock (the difference is referred to as the "Terzaghi effective stress", which therefore must not be positive, i.e. a tensile stress). In reality there is a small margin, known as the tensile resistance, but it is commonly not taken into account as a precaution. From the perspective of the Terzaghi criterion, there is competition between two forces: the increase in water pressure and the modification of the stresses, which are both caused by the heating up of the rock. The compressive stress, which increases to a lesser degree, is the most critical as this stress is likely to be the first to violate the Terzaghi criterion. The combination of symmetries means that the most critical point is situated at the mid-point of the cells, where the orthoradial direction coincides with the vertical direction. The pore pressure in the rock increases whereas the vertical stress does not change, leading to significantly increased compression.

The resulting risk is the appearance of micro-cracking that will tend to occur in the horizontal direction, thus leading to an increase in the initially very low permeability and possibly allowing for greater local movements of water.

This problem had already been envisaged by Andra in the 2005 and 2009 Files, but it was brought into sharper focus from 2012 onwards.

NEW PARAMETER VALUES, FEWER CELLS AND TUNNELS

Since the submission of the 2005 File, Andra has been continuing with its mechanical investigations. In addition to the testing of specimens in surface laboratories, new data have been revealed by numerous tests, including thermomechanical testing carried out in the underground laboratory. 3D seismic analysis, carried out in 2010, has also provided information about the speed of waves in the different layers penetrated and thus about the stiffness of the material on a larger scale. These data have led to the parameter values being more clearly specified. This revision is necessarily quite a long process: the rock is heterogeneous and anisotropic, the techniques used are based on very different principles and each of them have possible biases which must be evaluated; a large volume of data must therefore be gathered before deciding to modify the value of a parameter in the calculations. Moreover, the poro-thermo-hydro-mechanical behaviour of the argillite is undeniably complex. This process has led to several changes in the knowledge of parameters.

The mean permeability of the rock appears to be lower than was initially presumed. There are numerous measurements based on transitory methods, but their interpretation requires an initial model of sufficient complexity. More recently, they have been supplemented by steady-state measurements, which are less dispersed and thus easier to interpret. Andra thus considers these values to be reference values as they have become more numerous. There does not seem to be any correlation between the variability of permeability and the vertical or horizontal position in the layer. The median value adopted by Andra has decreased by an order of magnitude in relation to that adopted in 2005. This change confirms that diffusion will be the primary mechanism in the radionuclide transfer, which will consequently be extremely slow in some cases (actinides). On the other hand, it also implies that the dissipation of the excess water pressure caused by the heating process occurs even more slowly than was initially envisaged.

It is also observed that Andra's calculations do not take account of the anomalies in the natural interstitial pressure that were thought to have been revealed during the borehole tests; the reasons for this should be explained.

On the contrary, the stiffness of the rock – and therefore the stiffness of the "box" containing the water – appears to be greater than envisaged. In the 2005 and 2009 Files, the Young's modulus was 4,000 MPa. Andra now has the results of approximately 200 laboratory tests; the Young's modulus values are dispersed with a mean of 5,900 MPa and a median of 5,300 MPa. The values inferred from the speed of sound in the layer are more than double (16,700 MPa throughout the whole of the COx) – and even higher for the upper part of the COx, referred to as the "silto-carbonated unit" (Unité Silto-Carbonatée – USC). In principle, the difference should be smaller. Andra explains this anomaly by the non-linear behaviour of the rock, which has been clearly identified in the laboratory:

the stiffness is a decreasing function of the applied stress. However, sound waves only provide small variations in stress; the thermal variations that will be affecting the Callovo-Oxfordian formation will create higher thermal constraints and to describe them, we must therefore choose a lower stiffness value than the "dynamic" stiffness. Andra has thus chosen values of 6,000 MPa (and 12,500 MPa in the USC), which are higher than in 2005 and 2009, and also higher than the mean of the results of the recent laboratory testing, but considerably lower than the "dynamic" values. This is not an unreasonable choice, but some uncertainty still remains; in addition, Andra is examining the value of 9,000 MPa as a variant.

Furthermore, there is a smaller increase in water pressure following a rise in temperature, if all other factors remain equal.

It is noted that the maximal temperature criterion used two main parameters relating to the rock: its thermal conductivity and diffusivity (K^{th}, k^{th}). In addition, the Terzaghi criterion uses the elastic constants, the Biot modulus and coefficient, permeability, the differential expansion coefficients and porosity (E, v, M, b, K_h, α_m, α_O, ϕ_0 ..), which significantly increases the uncertainties, especially given our conviction that the behaviour is in fact non-linear, which means that these coefficients, at best, give no more than an average value.

In addition, the dimensioning of the repository has changed. In consultation with the producers, Andra has sought to optimise its design. One of these optimisations involves extending the storage period prior to disposal by around twenty years (85 years instead of 60-70 years in 2005 and 2009). This results in a lower level of thermal energy released (1.6 10^{10} J/m^2 over 8.4 km^2 instead of 2 10^{10} J/m^2 over 8 km^2 in the 2009 basic scenario). Another optimisation is to envisage longer cells (100 m as the reference for HL1 and HL2 cells and 80 m for HLW0, instead of 40 m in 2005 and in 2009; 150 m is envisaged for the future). Along a complete row of cells, perpendicular to the access tunnels, we thus find packages with slightly reduced thermal power, arranged in cells with seemingly fewer dividers. Between two rows of consecutive cells, there is either an access tunnel to the cells – but fewer of them – or around twenty metres of rock between the ends of two cells. One row of cells intersected with around fifteen access tunnels in the 2009 File; there are now six tunnels in the current design. One row of cells consisted of thirty 40 m cells in the 2009 File, whereas there are now twelve 100 m cells in the current design. Therefore, while the thermal power per unit of length of a cell has hardly changed (129 W/m instead of 125 W/m: the power has diminished but some dividers have been removed from the cells), the mean thermal power along one entire row of cells has a increased considerably. The length of one row of cells has been significantly reduced (1.4 km instead of 2 km).

CONSEQUENCES OF THE NEW DESIGN: MORE SPACE BETWEEN THE ROWS OF CELLS

As might be expected, the calculations show that this design does not make it harder to conform to the criterion of keeping temperature below 90°C in the rock (the highest temperatures are reached adjacent to the cell and thus depend primarily on the thermal power per unit of length of one cell, which has hardly changed), but that by increasing the thermal power along a row, we also increase the temperatures at a greater distance, which has an impact on pore pressures.

Andra has carried out numerical calculations with these new mechanical parameter values and this change of geometry.

The two modifications – improved knowledge of physical parameter values and the shortening of cell rows – make an approximately equal contribution to the increase in the pore water pressure, at a given constant spacing.

With the spacing provided for in the 2009 File, high water pressure values are thus attained. The Terzaghi criterion is no longer satisfied when the initial spacing is maintained; it is breached at the mid-point between two cells in particular. To restore compliance with this criterion while maintaining a high mean linear density, the spacing between individual cells must be significantly increased. In the example of the HL-C2 packages, the spacing between two cells, which was formerly 14.4 m in the 2009 File, was extended to 36 m at the end of the Basic Preliminary Design (BPD) (and up to

51 m for other packages). The cells are longer, but there are fewer of them in the same row, and the HL area has become oblong: it occupied a 4 km x 2 km rectangle in the 2009 File, which became 6 km x 1.4 km at the end of the BPD. There are considerable economic benefits to the changes: the number of cells drops from 5,000 to 1,500 and the linear length of the tunnels is halved from 100 km to 50 km, as there are fewer but longer tunnels. One disadvantage is that in its widest dimension, the HL area reaches the boundary of the Zira, and could even extend beyond it if we take account of the loops that must be created at either end to accommodate the turning of the tunnel-boring machine, if this excavation method is adopted.

CONCLUSION

The dimensioning of the HL area must include checks on compliance with several thermal and mechanical criteria. It can be observed that within one family of similar designs, a single criterion is often a "critical design" criterion: if it is satisfied, then the others are automatically satisfied as well. In the 2009 designs, keeping the temperature of the rock below 90°C proved to be a critical design criterion. In the recent designs, the absence of micro-cracking plays the same role. Temperature projections, which are based on the laws of thermal conduction, are more robust than projections of effective stresses, which are based on more complex physical laws and on the choice of more numerous parameters that are harder to obtain. The Board thus considers that the designs currently proposed are likely to require greater margins.

The Board judges that Andra has conducted a good preliminary analysis of the problem posed by the appearance of effective tensile stresses and thus of possible thermal micro-cracking. Nevertheless, much work still needs to be done. At the very least, new calculations taking account of the non-linearity of thermo-poro-elastic behaviour should be performed. The extension of areas likely to be subjected to effective tensile stresses should be analysed, in addition to their sensitivity to the parameters of the model. From this latter perspective, the uncertainties over these parameters need to be further specified and their impacts on the calculation results should be better evaluated. More fundamentally, the consequences for EDZ, the status of the associated water, any consequences of the deferred impacts, and the role that might be played by phenomena that have not yet been taken in to consideration, such as the production of gas, must continue to be evaluated. This requires quite a long-term effort and for the filing of the DAC in 2017, a sufficiently conservative design will need to be adopted to allow for the fact that there are still gaps in some of the required knowledge.

Certain in-situ thermal tests have already been carried out by Andra, but they have been on a small scale and although the changes in temperature have been correctly measured and explained, the same cannot be said for the water pressure in rock pores and even less so for the stresses and displacements. It may be beneficial to develop larger-scale tests, perhaps requiring long periods of time, with the emphasis on designing them to allow for high-quality measurements of the largest magnitudes.

The connection must be made with safety concerns and thus with possible changes in permeability after closure. Thermal micro-cracking may lead to an increase in natural permeability. We are not sure of how to assess the severity of this problem if it should appear. Two strategies could be envisaged. The first might involve adopting a very conservative design in order to retain large margins with regard to the appearance of micro-cracking. Another strategy could involve extending the analysis up to the consequences of micro-cracking and then evaluating its impacts in terms of safety, which might demonstrate that conservative dimensioning remains a reasonable approach. From this latter standpoint, it seems that the increase in permeability due to micro-cracking occurs mainly in the horizontal direction, and that a very small amount of micro-cracking could reduce the pore volumes sufficiently to cause a substantial drop in the water pressure and stop the extension of the area subject to micro-cracking. Although the permeabilities envisaged in 2005 were greater than at present, they nevertheless made diffusion the dominant mode for radionuclide diffusion. To a certain extent, these considerations relativize the fears. Conversely, however, the criteria have been examined by Andra on a local scale and the Board has not yet been able to form a clear idea of the extent of the zones that are likely to be affected by micro-cracking, or of how sensitive this extension might be to uncertainties. In summary, we do not yet know how to assess the distance separating the appearance of an initial micro-crack, which would be benign in isolation, from the creation of a large, continuous damaged zone. This study must be carried out. It could benefit from comparisons with analogous geological situations. For example, cases of the natural thermal hydro-cracking of clay materials have been described.

Appendix VII

CIGEO: R&D ON MATERIALS

SPECIFICITIES OF CIGÉO EXCLUDING SURFACE FACILITIES AND OBJECTIVES FOR R&D INTO MATERIALS

Cigéo is expected to operate for a period of 100 to 120 years. Unforeseen events or decisions relating to the reversibility of the disposal could lead to the extension of this period. This means that the shafts, access ramps and access tunnels to the disposal cells will need to be kept in good condition for a hundred years or so. It must be possible to retrieve the waste disposal packages (or any primary waste packages that may be stored as is) and they must remain in good condition for as long as possible. Indeed, they constitute the primary radioactivity confinement barrier. The repository's seals must be "leaktight" for periods of several millennia, which means that they must retain the low level of permeability on which the safety analysis is based (10^{-16} m^2). The qualities of the tunnel coating/support materials (concrete), packages (concrete and steel) and seals (concrete and swelling clay) must be adapted to these different time frames, given that these elements cannot be repaired, at least after the closure of the repository.

R&D concerning these materials thus seeks to establish the laws of behaviour concerning their mechanical strength and their degradation/corrosion under the disposal conditions: strong anisotropic stresses, contact between the materials and the CO_x argillite, or quite simply to evaluate their strength in accidental situations such as a fire.

FEEDBACK, TOOLS AND SCHEDULING OF R&D STUDIES ON MATERIALS

There is a significant amount of feedback on concrete, steel and clay of the bentonite type that have previously been used to build industrial structures of different sizes on surface or sub-surface levels. Andra's knowledge files reflect the knowledge on these materials that was updated in 2012, especially in fields associated with geological disposal. As far as possible, they take account of the fact that the time periods to be considered for Cigéo are unusual and longer than those for which the resistance of materials is generally certified.

Andra is carrying out research on materials in parallel or concomitantly with the technical tests in which they are implemented, either in its own facilities, in the facilities of its consortium of laboratories, in the Meuse Haute-Marne (LSMHM) underground laboratory, or through its participation in major international collaborations dedicated to geological waste disposal. The Board has covered these points in its previous reports.

Andra's Development Plan (PDD, submitted to the Board in October 2014) establishes the rate of research for Cigéo's components: surface-underground links, LLILW cells, HL cells, closing structures and for their sub-components. For each component, Andra has drafted logigrams covering the period until the industrial commissioning of Cigéo in around 2035, with the industrial pilot phase allowing for their qualification for the desired requirements (level 9 of the TRL scale). The PDD lists the key issues, studies and research to be conducted up to the sub-component level.

The Construction application (DAC) filing date, at the end of 2017, is the first R&D milestone, particularly for the materials. By this date, Andra must have verified that they are fitted to fulfil their functions assigned in representative environments (TRL 6).

By comparing 2012 "Dossiers de connaissances" (Knowledge File) and the 2014 "PDD" documents, we can monitor the changes in R&D, especially regarding materials and their implementation. The PDD for Cigéo presents a long-term view that will need to be updated if necessary.

CONCRETE AND CEMENTITIOUS MATERIALS

Concrete will be used for the surface-underground links, tunnel linings and LLILW cells, the manufacture of containers for disposal packages for many types of LLILW waste, the filling of annular space in the HL cells and the sealing support blocks. The properties of the different types of concrete depend on their formulations which may vary greatly. Concrete items such as containers may be reinforced or contain fibres. Concrete is used in a perfectly controlled manner in civil engineering and this material can satisfy numerous storage requirements. For all that, there are certain situations in which Andra is required to conduct studies. The R&D thus focuses both on the formulations of certain types of concrete and on the implementation of concrete or cementitious material (pumping, self-compacting, etc.) required for the construction of large items or structures.

a) Surfacing materials

The coatings of the different tunnels and linings of LLILW cells, in variable thicknesses of around one metre depending on the applications, shall consist of high-performance concrete (HPC). HPC formulas based on Portland cement and more complicated types of cement have been perfected. Andra's R&D concerns implementation testing on different scales for the dimensioning and preparation of full-scale demonstrators in the industrial pilot phase.

b) LLILW containers

LLILW containers protect the primary waste packages and must therefore conform to the operating requirements – particularly robustness if dropped during stacking – and resistance to thermal stress in the event of an accident. They must allow for the venting of radiolysis gases, withstand gamma radiation and limit the dissemination of radioactive substances under all circumstances. Andra has selected 5 types of concrete containers – C1 to C5 – with several variations in lids and internal layouts (empty weight of 8 to 12.5 tonnes, weight of reinforcements and or metal fibres of 0.4 to 0.8 tonnes, loaded weight of up to 18 tonnes). Andra envisages the construction of 72,000 containers. An HPC type of concrete is used whose formula was developed in 2005 in response to the requirements for disposal in LLILW cells. Prototypes have been created and tested. The remaining work mainly consists of optimising the implementation of the HPC on an industrial scale. Andra is studying the possibility of the co-disposal of several packages containing different types of waste. Amongst other factors, their compatibility depends on the migration (diffusion and sorption) of water-soluble chemical species from the breakdown of organic material through the concrete of the containers. There is a risk of them complexing the radionuclides and making them more mobile in the COx. Andra is running a research programme on co-disposal which includes the topic of the migration of organic products in concrete.

c) Annular space

To minimise the mechanical stresses exerted on the liners and the effects of the argillite oxidation products on corrosion (see above) Andra is analysing the formula of a cement-based material (concrete/bentonite slurry) to be used for filling the annular void between the liner and the argillite altered by the excavation of the HL micro-tunnels. These studies are being accompanied by application tests.

d) Support blocks

The seals are based on the principle of using a swelling clay core of around 30 metres in length trapped between two low pH concrete support blocks. The gradual hydration of the core ensures contact, due to expansion, with the argillite that has global permeability of 10^{-16} m^2 although the permeability of the swelling clay is much lower (10^{-18} m^2). Each seal (5 shafts and 2 access ramps, tunnels and cells) has a specific configuration. It requires the total or partial removal of the coating. The seals are large structures in which numerous mechanical and physico-chemical phenomena come into play. The low-pH types of concrete, studied by Andra over 10 years or so, have special formulas. They are binary or ternary mixtures of Portland cement, silica fume (SF) and fly ash (FA), with a very high SF and FA content. They are mainly designed to minimise damage to the argillite or clay that comes into contact with them by reducing the pH of the alkaline mixture (3 pH units lower than a traditional type of concrete). To this end, Andra is analysing the properties of several formulas (THM parameters) in addition to the rheology of the different concrete in relation to their implementation and the physico-chemical modifications of the interfaces between this concrete and the argillite or clay. These modifications could be limited to a thickness of just ten or so centimetres in 100,000 years for the concrete (and an insignificant amount for the swelling clay). Andra is also using mock-ups to study several geometrical forms of support blocks capable of resisting the thrusts of the swelling clay.

SWELLING CLAY

The swelling clay used in the seal cores is a mixture of MX80-type bentonite in powdered, pelleted or brick form with cement. Andra possesses a very large amount of feedback on the hydration of bentonite (swelling pressure according to dry density, permeability to water and gases, etc.) and on its behaviour in representative mock-ups of seals (interaction with the immediate environment, concrete, argillite, etc.). Experiments are underway in the LSMHM that should produce results before 2017. In particular, Andra is studying the hydro-mechanical-gaseous behaviour of the area of damaged argillite around the seals under the thrust due to the swelling (self-sealing) responsible for the reduction in the overall permeability of the seals.

STEEL

Steel will be used for the lining of HL cells, the containers of HL waste packages and certain LLHLW packages. Andra is thus analysing the corrosion of the steel under the constantly changing environmental conditions of HL cells filled with waste packages: increase in temperature and pressure, variation of radiation intensity and gradual transition into a reducing environment saturated with water.

a) Lining

The French industrial programme for waste management (PIGD) provides for the storage of HLW0 waste packages in cells (the storage of HL1 and 2 packages is scheduled for 2070 and afterwards). The lining of an HLW0 cell will consist of interconnected tubes (diameter of 76.2 cm and thickness of 2.54 cm) over a distance of 80 m, at slope of 2%. It must ensure the mechanical protection of the HL container for the longest possible time, but it is not watertight. Andra has currently adopted a grade of steel already used in the oil industry: API 5L X65.

In the absence of oxygen and at a temperature of below 90°C, the extrados of the lining corrodes in direct contact with the argillite (generalised corrosion), like all steel, and loses about 10 micrometres of its thickness per year but at increasingly slower rates due to the formation of protective corrosion products through the passivation of the steel. In oxidising environments and during transitory acidic conditions (oxidation of pyrites in sulphuric acid and ancillary reactions), the speeds are higher, as under radiation above a dose rate of 80 Gy/h. For several years and on numerous samples, Andra has been studying the effects of the main parameters characterising the local environment of the lining (O_2 content, pH, T, dose rate, etc.) in addition to the micro-structural elements that come into play (formation of oxide/hydroxide/iron carbonate and iron hydroxy-chloride). In situ experiments have been set up in the LSMHM. There have been concerns about the phenomenon of corrosion under stress leading to micro-cracking. The presence of a cementitious material in the annular cell space, which leads to a high pH level in contact with the extrados of the lining, promotes the passivation of the steel and prevents any risk of intra or inter-granular cracking.

Multi-scale studies of the mechanical resistance of lining are being conducted on mock-ups (with and without an annular void) in order to discover/specify conditions of radial bending (out-of-roundness, resistance of joints, etc.) and buckling. They began several years ago and Andra is capable of performing simulations which show that the chosen characteristics of the lining can guarantee the existence of a gap between the packages and lining for a time far exceeding the operating life of the repository and can protect the packages for a long time afterwards (hundreds of years). Experiments on buckling are continuing in the laboratory, taking account of feedback from the oil industry on buried tubes. The characteristic time for the appearance of buckling could be 500 years.

All of these experiments obviously require the most in-depth knowledge of the physico-chemical and thermo-hydro-mechanical behaviour of the argillite, which is Andra's constant concern.

Between now and 2017, Andra will be able to simulate the corrosion and thermo-mechanical deformation of the lining and make choices regarding its dimensioning and construction. In 2017 Andra must devise a credible design for the disposal of LLHLW in which the role of the hydrogen produced by the corrosion of the steel will be an important point. The hydrogen generated by the corrosion of the steel in the lining will be the first substance produced in large quantities in the repository, which makes the type of steel an important choice.

b) LLILW containers

Two types of steel containers – C6 and C7 (empty weight of 1 to 1.5 tonnes, loaded weight of up to 15 tonnes) will be used for storing 240 large packages of LLILW containing no (or very little) organic matter. The grade of steel used is likely to undergo generalised corrosion.

c) HL containers

All of the HL waste packages (55,000 of which 4,150 HLW0 packages are only slightly exothermic) must be containerised to limit the dose rate to 10 Gy/h on the external wall, prohibit the arrival of water in contact with the primary package during the thermal phase (<350 years) and then while the activity is decaying. The length of the period during which the containers remain intact plays a key role in analysing the safety of the repository after its closure. Andra has selected six models of containers, five of which will have a single or double-package containers for HLW0 primary packages. Andra is studying the possibility of using P285NH forged steel for their construction. Prototypes have been created according to processes qualified by Andra as reliable and reproducible. The first HLW0 packages with power of under 200 W/package shall be stored in double-package containers (2 m long, 63 cm in diameter and 6.5 cm thick).

The HL container will eventually be in the presence of water and then argillite and/or the material used for filling the annular space. During the degradation of the lining, the HL package will also be subjected to occasional mechanical stresses. The simulations (to be continued) show that under the harshest conditions, the steel will remain within its elastic deformation range for several hundred years. This could thus help to minimise the risk of corrosion under stress. Nevertheless, studies of the mechanical resistance of structures featuring microcracking are envisaged.

Andra is conducting the same type of study on the steel of the HL container (and the weld on the lid) as on the steel of the lining: corrosion and mechanical resistance under degraded conditions in its environment.

In 2017, Andra will be in a position to choose the thickness of the container and draft a container definition, production and inspection file.

DIRECT DISPOSAL OF PRIMARY LLILW PACKAGES

Andra is studying the possibility of the direct disposal of 70,000 primary packages (out of the 176,000 to be disposed of). In the absence of a storage container, it must be demonstrated that the different materials that constitute these primary packages are capable of ensuring the mechanical robustness required for their entry into the repository, followed by their resistance to stresses in incidental or accidental situations (e.g. fire) and to physico-chemical alterations. Whatever the circumstances, the packages must not lose their containment properties over the required periods. This research is the producers' responsibility.

CONCLUSION

The properties of the materials used to build Cigéo and ensure the safety of the repository while in service and after its closure have been studied for as long as the concept of geological disposal has been studied. The materials (concrete, steel, clay) and their implementation in Cigéo's structures must, in general, ensure the containment of radioactive substances prior to the closure of the repository and for as long as possible thereafter, even though the safety analyses reveal that this containment will ultimately rely on the properties of the COx. Considerable feedback has been acquired both at a fundamental level and in the context of their use for civil engineering. The R&D on materials, to be conducted by Andra prior to the filing of the DAC, must specify/consolidate the parameters governing their evolution, in order to ensure the credibility of the models and simulations for Cigéo.

Appendix VIII
ACCEPTANCE SPECIFICATIONS FOR PRIMARY LLILW PACKAGES AT CIGEO

The definition of the process for depositing waste packages in cells at Cigéo is a prerequisite for the commissioning of the repository. This process is based on the waste package acceptance specifications, the monitoring of compliance with these specifications and the acceptance accreditations contracted by Andra and the waste producers.

The specifications for the acceptance of waste packages at Cigéo define the characteristics and performance levels required for waste packages for acceptance into this repository. They guarantee, subject to checking that the packages conform to the specifications, that any package will comply with the safety requirements both in operation and after the closure of the repository. The process for defining the specifications involves acquiring the most detailed knowledge possible of the families of primary waste packages and disposal packages, and making the packages compatible with the concept of reversible disposal. This is based on the safety analysis that will ultimately be reflected in the General Operating Rules (GOR) for Cigéo. This process is organised according to an iterative dialogue between Andra and the producers. This dialogue is essential to Andra, both for obtaining information about the packages and for the design of the repository. It is also indispensable to the producers with regard to the management of their waste and to ensure that their own packages will be accepted at Cigéo. Economic optimisation is a key concern of both Andra and the producers with regard to the disposal of all waste packages, be they in storage, currently in production or not yet produced. The specifications will be approved by the French nuclear safety authority (ASN) prior to the commissioning of Cigéo.

Depending on the types of packages, the monitoring of compliance with the specifications shall include several inspections carried out both on the producers' sites and in the Cigéo surface facilities, and by the operators, Andra and ASN.

Finally, approvals between Andra and the producers shall provide official authorisation for their waste to be stored at Cigéo.

Feedback from the operation of the CSA is beneficial to this complex process.

FAMILIES OF PACKAGES, KNOWLEDGE AND UNCERTAINTIES

The PIGD may be described in several ways according to the characteristics to be emphasised. For the disposal of packages at Cigéo, Andra has defined families of waste sharing similar characteristics that allow for the same mode of management in operation (e.g. the production process and attached documents, radionuclide and chemical content, thermal power, dose rate, container, packaging matrix, etc.). For the Cigéo PIGD, there are currently 79 families of LLILW packages and 19 families of LLHLW packages which are defined on the basis of over 100 knowledge files for 232,000 packages. These 98 families are all attached to the 31 "families" in the national inventory of radioactive waste or – for LLILW packages – to "categories" LLILW1 to LLILW6 that differentiate among packages by grouping them according to whether they can be disposed of in separate cells (LLILW1 to 4) or are suitable for co-disposal in the same cell (as is currently the case for LLILW4 and 5). As the R&D studies on co-disposability are still in progress, the current categorisations may not be final.

It is clear that knowledge of the characteristics of packages plays an important role, firstly for the definition of families and ultimately for the acceptance specifications. The levels of description in the "knowledge files" (dossiers de connaissances) for the packages differ according to whether the packages have already been produced – especially the oldest, are currently being produced or will be produced in the future. To take account of these uncertainties, Andra assigns a level of knowledge to each family (0 to 3), which allows it to integrate margin factors into advances in the design of the repository (e.g. from 1.5 to 10 for the radiological inventory comprising 144 radionuclides) and to engage in discussions with the producers on how to improve or rework the packaging of waste, or even to continue characterising packages or repackage them. Andra may also continue to study the long-term performance of the packages.

It is clear that higher levels of knowledge of the packages mean that smaller margins can be allowed for in the design of Cigéo. This provides for a more convincing demonstration of safety. At the Board's request, Andra compiles an extremely accurate and up-to-date tracking report on changes in the status of knowledge about the packages, which acts as the basis for drafting the acceptance specifications.

SPECIFICATIONS

a) Regulations and methodology

In general, any radioactive waste repository in operation will possess acceptance specifications for waste packages. The order of 7 February 2012 compels producers to comply with them and to package their waste accordingly. For a future repository such as Cigéo, the production of packages is subject to approval from the ASN, which makes its decision according to Andra's guidelines on the design of the repository in order to ensure the operational safety (in normal and downgraded situations in addition to incidental and accidental situations) and safety after closure.

The methodology for drafting the specifications is based on an iterative comparison of the characteristics of the packages and the design of their repository. On the basis of the knowledge of packages provided by the producers, Andra investigates several possibilities at each stage of the design studies: acceptance of the package without alterations according to the draft specifications of the time, possible improvements/adaptations to the package, revision of the repository design, etc. The specifications are thus gradually drafted as the dialogue between Andra and the producers progresses. If, at the end of the process, the packages fail to meet the final specifications that will then correspond to the General Operating Regulations (GOR) for the repository, they must be taken back. The regulations also provide for the fact that the GOR for a nuclear facility may change over time.

b) Preliminary and final specifications

Since 2012, Andra has been maintaining a dialogue with waste producers with a view to drafting an initial version of the preliminary acceptance criteria for primary waste packages at Cigéo. This must be finalised in mid-2015 to coincide with the end of the basic preliminary design (BPD), and then sent to the ASN with the safety orientation file (SOF) in September 2015. The specifications currently being drafted exclusively concern the primary packages that will be placed in disposal packages (other specifications may cover the direct disposal of primary packages). The version of the specifications that will be enclosed with the construction application (DAC) in 2017 shall, on the basis of the detailed preliminary design (DPD) studies, incorporate the latest adjustments in compliance with the GOR for Cigéo that will also be included in the DAC. Finally, the definitive specifications shall be defined when the commissioning application for the facility is submitted.

Andra and the producers are establishing an organisational structure to manage this long process which aims ensuring that the packages shall fulfil their assigned roles in the repository. Numerous problems must be solved to ensure compliance – under operational conditions – with the containment requirements for radionuclides (possible use of a containing disposal package), dispensations for several packages that do not conform to a requirement (this situation potentially concerns historical packages), accreditations, inspections and the quest for an optimal technical and economic balance.

c) Characteristics of the specifications

The specifications currently being drafted (and those to follow) are not drawn up per family of packages but cover all of the families, unless this is necessary for certain families due to specific requirements. The accreditations, on the other hand, concern each family.

The specifications cover the requirements for the container, the materials used for the packaging of waste, its immobilisation and the filling of the container. They differentiate between the packages produced and those to be produced. They include quantitative values for certain requirements, qualitative values and declarative or limit values for other requirements, and finally identify operational requirements such as drop resistance. The quantitative requirements are upper limits (weight, geometry, void fraction, surface contamination, hydrogen emissions and prohibited substances). The qualitative requirements concern organic substances and the declarative values concern radiological and chemical inventories. They also cover the administrative conditions and inspections to which the packages must conform and incorporate a certain degree of flexibility for special cases.

The declaration of radionuclide inventories (and of chemical toxins), which are what makes radioactive waste so unique, relates to long-term safety, pursuant to the ASN's 2008 Guide.

MONITORING OF COMPLIANCE WITH SPECIFICATIONS

In the package acceptance process, it is essential to monitor compliance with the specifications because this guarantees the compliance of packages for disposal.

The specifications partly relate to the manufacture of the packages, which are manufactured on producers' premises according to production standards authorised by the ASN (production specifications). Thes production standards involve one or more inspections of the quality of all packages and allow for the recording of any nonconformities. Since 1990, Andra has been monitoring inspection activities for the production of LLILW and LLHLW packages (examinations of documents, audits and inspections of processes and tracking of conformity recorded in the quality assurance programmes [QAP] and quality control programmes [QCP]). The methods of this monitoring are defined in agreements adapted to each situation. This activity on the production sites is an initial verification of compliance with the specifications. For packages manufactured before 1990, the only package conformity documents are those drafted by the producers and the ASN. Andra's monitoring is in this case document-based. Before being sent to Cigéo, the stored packages will be inspected by the producers, to check, in particular, on compliance with the transport regulations.

For the acceptance of packages into Cigéo, Andra, in agreement with the producers, will perform inspections to check that the packages correspond to those that were actually sent (in-line inspections for all packages) and to check on compliance with the producers' specified and declarative values recorded on the specifications (inspection by the sampling of packages within a family on the producer's premises prior to shipping to Cigéo). These inspections shall be performed by Andra in the Cigéo surface facilities (or by Andra on the producer's premises).

The sampling-based inspections shall concern a percentage of the packages from each family. This percentage shall mainly depend on the extent of the knowledge about the packages, the quality assurance system implemented during the manufacturing of the packages (especially traceability) and the possible existence of operators other than the producers in the package production chain. The main aim of this process is obviously to ensure the most representative sampling of the packages. Several parameters will be measured by non-destructive methods in the Cigéo surface facilities: radiological inventory (gamma spectroscopy), absence of prohibited waste, homogeneity, voids (imaging), alpha emitters (neutron/photon interrogation), thermal power, degassing and drum corrosion. The possibility of micro-sampling is also envisaged. In the event of the nonconformity of packages, a disposal solution will be sought: additional packaging, a new disposal solution, special exemption or return to the producer's premises.

These inspections require the installation of one (or two) high-performance, multi-purpose measurement stations at Cigéo. Certain measurement techniques are already available but others must be developed to attain Level 8 on the TRL scale and this will require significant R&D. Techniques such as the measurement of degassing take a long time to implement. The aim is to strike the optimum technico-economic balance.

CONCLUSION

The drafting of the acceptance specifications for primary waste packages at Cigéo is the result of an iterative process between Andra and the waste producers. It involves taking account of the characteristics of packages defined by the producers (and by Andra for the long-term properties) and comparing them to the possibilities for a reversible repository design in the COx, which is Andra's responsibility. The technico-economic optimisation of their disposal involves seeking a convergence between the possibilities of improving the robustness of the primary packages and the avoidance of failures, and takes account of the safety requirements. The specifications thus confirm the requirements that must be met by the packages for acceptance into Cigéo.

These specifications are bound to evolve as the project progresses.

Andra and the producers are currently working on the preliminary acceptance criteria that will accompany the DAC in 2017. The final specifications will be ratified by the French nuclear safety authority (ASN) prior to the commissioning of Cigéo. They shall incorporate the progress made by all partners at this time, and shall conform to the GOR for Cigéo.

Andra and the producers are also working on a way to inspect compliance with the specifications. In this respect, the Board takes note of the acceptance process for waste packages at Cigéo, which seeks to foster a collective understanding of the problems of managing waste packages, but it draws attention to the fact that this process needs to be transparent.

Appendix IX

R&D ON THE DISPOSAL OF VLLW, LLLLW AND LLILW

GENERAL INFORMATION

The optimisation of the distribution of radioactive waste among the existing or planned management circuits is being analysed in the framework of the PNGMR (especially for the 2010-2013 and 2013-2015 periods). The destination of the waste is defined according to three major criteria: safety, technical possibilities and economic possibilities. This optimisation requires knowledge of the characteristics of the waste and of the technical possibilities for its industrial processing and packaging. It also requires the prediction of the behaviour of the packages in storage and disposal. R&D is being carried out by the producers following the expression of needs issued by Andra, either for the operation of repositories or for the evaluation of their long-term safety. The R&D carried out by Andra concerns the long-term behaviour.

For the majority of the waste and according to their resources, Andra and the producers possess the knowledge and tools required to carry out a technico-economic optimisation; nevertheless, R&D are still in progress for certain types of waste. This concerns waste for which there are doubts as to the robustness of the packaging in relation to an extreme event, such as an outbreak of fire while the repository is in operation (bituminous waste) or a behaviour that is hard to predict in the long term in a disposal situation after the degradation of the packages (pyrophoric, saline or organic waste). The waste to be packaged and the packages already created are in storage on the producers' premises. The waste to be packaged is, in particular, LLILW derived from the activities of UP1 and UP2-400 plants. The data and requirements concerning waste packages are constantly updated to minimise the uncertainties.

The accommodation capacity of the repositories is an important aspect of the optimisation. In this respect, the dismantling of reactors and fuel cycle facilities will generate significant amounts of miscellaneous waste:

- SL-LILW can be disposed of at the CSA with LLILW and HL waste being sent to Cigéo, along with the other technological and operating waste to come;
- the dismantling of first-generation reactors (6 UNGGs, Brennilis, Chooz and SPX) will produce 300 tonnes of LLILW and 50,000 tonnes of SL-LLILW.

The projected VLLW flow, including approximately 500,000 m^3 from dismantling (including 115,000 tonnes from first-generation reactors), poses several potential management problems:

- the reduction of quantities to be placed in the repository,
- increasing the capacity of CIRES (Industrial consolidation, disposal and storage centre),
- and the search for a new VLLW disposal site,
- the LLLLW from dismantling and the graphite from the UNGGs (17,000 tonnes), are earmarked for the future SCR (disposal under a reworked cover) repository.

Here, we shall be examining the results of the latest studies concerning these types of waste (excluding an economic assessment).

VLLW

Before the end of its service life in 2033, CIRES, in its current mode, will not be able to accommodate more than 650,000 m^3 of VLLW. It is already almost half-full. The projected volume of VLLW to be disposed of by around 2030 is 1,300,000 m^3. The capacity of CIRES could be raised to 950,000 m^3 by increasing the volume of the trenches without changing the surface area occupied by the centre. At this time, this would still be insufficient for the disposal of all VLLW. That is why Andra is seeking a site for the creation of both an LLLLW repository and a VLLW repository, and is investigating possible ways of using certain VLLW as fill materials (crushed rubble) or as materials for manufacturing waste containers (steel). This would allow for the removal of around ten thousand cubic metres of waste from the VLLW inventory per year.

In this context, the recycling of metal materials from Eurodif is being studied by Areva and could be extended to the steel from reactors. The process could involve smelting steel from the temperature exchangers, pipes, stages and metal components from the Georges Besse (GB1) enrichment plant, after prior surface decontamination ("Prisme" operation). It could be implemented under conditions allowing for an additional decontamination, by concentrating the natural residual surface contamination in slag. A pilot plant is envisaged.

Approximately 150,000 tonnes of steel are concerned and this process could reduce the volume of waste by 100,000 m^3, as only 40,000 m^3 would be sent to CIRES in the form of slag. The decontaminated steel could be reused in the nuclear industry (e.g. for disposal containers). However, many different aspects – technical and economic, but also safety-related, such as the introduction of metal containers into Cigéo in addition to those already envisaged – need to be considered before the recycling of such steel can be approved.

If the smelting process is not adopted, the decontaminated metal waste (except for certain parts) will be cut up and packaged in 5 m^3 concrete-encased packages into which a cement slurry is injected, and then sent to CIRES. The ceramic diffusion barriers from GB1 will be ground up and delivered to CIRES in big-bags. In total, this amounts to approximately 200,000 m^3 of waste in 10 to 15 years.

Another option for reducing the VLLW inventory is to envisage a waste release threshold that would allow for the recycling and use of materials outside the nuclear sector. Indeed, the French regulations lead to waste without any impact on health being placed in VLLW repositories. This option is being examined by the ASN and in the framework of the PNGMDR. An important aspect of this question is the implementation of an inspection of the radioactivity of materials prior to their release. This inspection is no simple matter due to the very low level of activity and the large quantities of materials.

The waste producers and Andra are acutely aware of the problems of managing VLLW waste. The quantities of VLLW that will originate from the nuclear fleet and associated facilities at the end of their service lives and after their dismantling, have been underestimated and are still uncertain. They should amount to around 2 million cubic metres.

LLLLW

The LLLLW inventory (quantities and characteristics), presented in the CNE Report No. 8 (in Appendix X on page 75) and the studies of its management, remain valid, with few exceptions (specific activities of certain types of waste). Here, we shall be presenting the results of research on improving their characterisation and the processing envisaged for their acceptance or refusal at an SCR repository.

The studies conducted by EDF with Studsvik (completed in 2014) and the CEA on the decontamination of graphite from reactor stacks and its possible burning, have shown that almost all of the ^{3}H and ^{36}Cl can be eliminated from the mass of crushed graphite, but that the ^{14}C decontamination is only partial. The process developed consists of gradually heating the graphite in a controlled atmosphere (carboxygasification), capturing the emitted gases and then treating the captured effluents.

However, this decontamination treatment is not efficient enough for the ^{14}C, with regard to the emission standards for this radioactive element, to envisage the combustion of the graphite. The building of a decontamination prototype has been suspended.

The CEA was supposed to submit a report on the incineration/vitrification possibilities for packages of bituminous LLLLW packages in 2013 but this was postponed until 2015 as the priority was given to studies on the fire endurance of packages of this waste. Considering the outcomes of previous experiments (2003-2005), there is little chance of an incineration/vitrification process being viable. Therefore, at present, the sole prospect for the management of bituminous packages classified as VLLW is disposal in an SCR repository.

Solvay envisages two possibilities for the treatment of SSR (standardised solid residue originating from the processing of monazite for the extraction of rare earths), stored on the La Rochelle site (8,400 tonnes). The first involves the extraction of thorium (35 tonnes) and rare earths (1,000 tonnes) from it, which would reduce the volume from 7,200 to 500 m^3 of VLLW (part of the Valor project on processing all of Solvay's thoriferous materials). The second treatment would involve the dehydration of the SSR, thus halving the volume to be disposed of in the SCR repository. The decision on implementing these treatments has been postponed until mid-2016. Solvay considers that SSR is more of a mining-type waste and that surface disposal would be consistent with the provisions being studied for Malvési waste. As of now, an increase in the radiological capacity for thorium at CIRES needs to be examined.

The report on LLLLW management to be submitted by Andra to the government and the ASN in mid-2015 will allow for a design study of the SCR repository, taking account of the results of the geological investigations in progress on a potential site near Soulaines. This study will be conducted from 2016 to 2018. At the same time, the studies in progress on the additional characterisation of graphites and waste bituminous matrixes shall be continued along with the studies required for the possible acceptance of graphite liners, UNGG waste, ion exchange resins and LLLLW bitumen in Cigéo. Finally, other large quantities of LLLLW not yet recorded might appear in the future (dehydrated sludge and gypsum from Malvési).

LLILW

a) Bituminous sludge packages

To evaluate the operating safety of the disposal of primary packages of bituminous matrix sludge in LLILW cells (bituminous packages, families F2-3-04 and 05, F2-4-03 and 04 in the 2012 national inventory), their fire resistance and radiolysis-related hydrogen emissions must be known. Over the long term, the effect of the pressure from their swelling on the disposal package and on the argillite must be known, in addition to the source terms of salts and radionuclides. In its report no. 6 (November 2012), the Board asked waste producers and Andra to study the behaviour of bituminous packages under full-scale conditions during a fire that was representative of the operating conditions at Cigéo, and to submit the results to the Board at the end of 2014.

The CEA, Areva, EDF and Andra (in collaboration with universities) launched a high-priority programme of experiments on the fire endurance of bituminous packages and submitted the documents describing the experiments, the results and their interpretation to the Board within the allotted time, along with further information about the swelling of packages. This data was accompanied by additional documents in response to the Board's questions.

The composition of the sludge produced by the STEL at Marcoule (1966-1996), by the STE2 and STE3 workshops at La Hague (1989-2012) and their packaging have varied over time, producing six characteristic families of bitumen packages. It is possible to obtain highly representative compositions of the matrixes for each family of packages (min. and max. compositions) and as a consequence, to prepare matrixes (non-radioactive) of the same composition for the experiments. The matrixes result from the heat treatment of the sludge during its incorporation into the bitumen. They consist of numerous solid compounds used for the co-precipitation of the radionuclides contained in the effluents from the spent fuel treatment workshops.

The programme of experiments was approved by the Board in 2013. It consisted of three sections:

- study of the thermal characteristics of the matrixes at the gramme scale (T< 300°C),
- behaviour of the matrixes at the 2 kg scale (T< 300°C),
- behaviour of disposal packages containing four primary packages of matrixes in a fire.

In the last two types of experiments, the containers and the matrixes were equipped with thermocouples and the experiment sequences were filmed. The experiments in section 1 required the preparation of 105 and 48 samples to cover the variabilities of the compositions from the STEL and STE2/3. The experiments in section 2 concerned 17 samples covering the 6 families of matrixes and those in section 3 concerned 6 disposal packages, each containing 4 primary packages.

◆ **Resistance of bituminous matrixes to a rise in temperature**

The characteristics of the components of the bitumen matrixes could lead to exothermic reactions, or even to runaway reactions, due to a rise in temperature. This explains the need to measure the parameters governing thermal emissions in samples whose compounds vary in composition (typically around ten). A microcalorimetric study was carried out. There are numerous cross-interactions between compounds but a method was established per experimental design to identify the importance of each one and ultimately establish the quantitative modelling of their contribution to the measured parameters: heat emitted, power, maximum temperature of the sample, etc. The modelling was carried out for temperatures of up to 300°C for all families of bituminous asphalt. These numerical results have no chemical significance in the sense of the identification of chemical reactions among compounds. However, they do reveal and confirm the predominant role of nitrates in the heat emissions and show that there are no runaway reactions, under any circumstances, in the temperature range investigated.

The experiments on the 2 kg samples set out to model the heat transfer and consecutive physical modifications in the bituminous matrixes heated incrementally up to 300°C on their periphery (constant T for 1 hr 30 min). At this temperature, the centre of the bitumen rises to 250°C and there is no ignition.

The fronts of physical change in the contents of the bitumen packages spread from the drum wall inwards due to the effect of a steep temperature gradient. When the exterior temperature is constant, heat spreads by conduction with the appearance of a vitreous transition at 50°C and then a change of density at 100°C, followed by the appearance of convection at 150°C. The bituminous matrix thus moves to a given location and, for a short moment, reaches a temperature at which exothermic reactions occur. However, not enough energy, calculated on the basis of the modelling of the microcalorimetric experiments, is released to affect the physical mechanisms in action. There is no instantaneous release of heat capable of producing localised self-heating. The modelling shows that this would still apply to a 200 kg mass of bitumen due to its low thermal conductivity.

The spatial and temporal temperature measurements allow for the modelling of the observed phenomena and their extrapolation to the scale of a primary waste package, both for the increase in temperature and the convection movements. For a package subjected to maximum temperatures of 150°C at the surface of the container, the modelling shows that there is no accumulation of matrixes. The temperature of 150°C corresponds to that reached in the full-scale fire tests (see below).

◆ **Resistance of packages during a fire**

The experiments were conducted in two phases. In the first phase, three disposal packages – one containing barrels of pure bitumen and the two others containing barrels of bituminous matrixes from STEL and STE3 – were consecutively placed in a furnace and subjected to a temperature of 950°C for 1 hour. The B2-type disposal packages (6 tonnes, with metal reinforcement, HPC-type concrete containing metal fibres both with and without polypropylene fibres at 1kg/m^3, and

12 cm-thick walls), in addition to the primary packages (drums and contents), were equipped with numerous thermocouples (around one hundred for each disposal container and the same number for the four packages). The recording of data from each test lasted for over 40 hours.

The following results were obtained:

- the temperature never exceeded 150°C on the walls of the primary packages, 90°C three cm from the walls and 50°C in the middle of the package;
- the disposal packages not containing polypropylene fibres suffered from spalling in isolated patches to a depth of 1 cm and 0.3 cm for the fibrous packages;
- no traces of chemical reactions were apparent on the surface of the bituminous matrixes and the walls of the primary packages were unaltered;
- The disposal packages could still be handled after the tests.

In the second phase of the tests, two disposal packages equipped with instruments and identical to those used in the previous tests were exposed to a wood-fuelled fire in a chamber simulating an LLILW cell (80 m^3, consisting of cellular concrete and fire brick walls, with ventilation of 0.8 m^3/s). Two fire tests, with an in situ disposal container, were conducted in order to adjust the experimental conditions. The fire generated power of 1.5 MW for 1 to 2 hours and then burned at lower power until it was naturally extinguished.

The results obtained for the two consecutive tests under actual conditions with instrumented packages reveal that:

- the temperature on the concrete of the disposal packages reached a maximum temperature of 600-650°C on the face exposed to the fire, which spalled, but to a lesser extent on the HPC concrete with polypropylene fibres;
- the temperature never exceeded 150°C on the walls of the primary packages and 110°C three cm from the walls;
- the primary packages remained intact and the disposal packages could still be handled.

The results of all tests were consistent and can be repeated by modelling. The test conditions broadly covered the characteristics of a fire in a LLILW cell, such as might be caused by the ignition of a forklift.

Other tests are envisaged in the experimental facilities that have been set up. They will allow for the testing of the resistance of other disposal containers for bitumen packages that are seemingly more robust than those used in the first test as their concrete walls are 20 cm thick.

◆ **Resistance to radiolysis and swelling**

The bitumen radiolysis that takes place inside bituminous matrixes mainly produces hydrogen. This is solubilised (5%), diffuses, and coalesces, causing the bitumen to swell, and then escapes from the bitumen to occupy the free space left in the drum during filling. The H_2 production yields, according to the energy absorbed by the bitumen, are known and codes can be used to estimate that in the most radioactive bitumen packages, the production cannot exceed 10 L per year. And this drops to 3 L per year a few years after their production.

Therefore, the volume of hydrogen produced by radiolysis cannot generate swelling likely to damage the disposal packages, which are also permeable to hydrogen diffusion. At most, the hydrogen occupies the empty space in the primary packages and in the disposal packages of between 100 and 200 L.

The influence of the presence of H_2 in the industrial packages of stabilised bitumen (several years after their production) or in the expansion space of the packages was not tested during the heat and fire tests. It remains slight and consequently has no predictable impact on their behaviour. For example, there is negligible heat gain from hydrogen during combustion.

The CEA also demonstrated that the swelling of primary packages following the uptake of water by the matrixes had no effect on the disposal package.

b) Packages of reactive and/or pyrophoric metal waste

The expression of R&D requirements for this waste dates back to 2007; programmes were launched at this time by producers (collaborations with universities and the CNRS), and a review will be carried out in 2015. The majority of this waste still awaits packaging.

The reactive metals are primarily control rods from FNRs being dismantled. They consist of needles of boron carbide (B_4C) containing 40 to 100 g of residual sodium per needle after treatment with water vapour. For their disposal, the CEA envisages using a primary package consisting of these needles embedded in sand and placed in a 1.5 m^3 stainless steel drum (F2-4-15 family). For all of the needles, this will amount to 8 packages of sodium-containing waste, which corresponds to 2 disposal packages.

Under operating conditions, the presence of sodium in the pore volume of B_4C leads to fears of significant heat and hydrogen emissions in the event of contact between sodium and water. Studies (by Andra, CEA and EDF) are in progress with a view to substantially reducing the quantities of sodium (35 kg for the 2 disposal packages), using the *in situ* carbonation of the Na. This project requires in-depth knowledge of the kinetics of H_2O-Na reactions in porous environments.

"Pyrophoric" waste is metallic material (metals or alloys), either in solid form or in pieces of varying sizes, that react almost spontaneously with the air. To prevent their oxidisation, they need to be embedded in a non-flammable matrix. The compaction of fines has the same aim as it allows for a significant reduction in the reactive surface area of the waste and thus of their pyrophoric nature. However, even when insulated in this manner, the degradation of a package could allow for the corrosion of these metals by water. This could occur quite quickly for aluminium in cement: 300 micron per year. Unless a corrosion inhibitor to be mixed with the matrix is found, the R&D developed by Andra will focus on two topics in order to evaluate the possible release of hydrogen while Cigéo is in operation: optimising the formula of an inert industrial binder adapted to each metal, and measuring the speed of corrosion under different conditions.

LLILW magnesium waste (F2-4-09 family) consists of the metal fuel cladding of UNGG reactors in a Mg-Zr or Mg-Mn alloy containing traces of uranium metal. It is stored in drums (in bulk, discs or pieces) in 17 ditches at Marcoule and amounts to 1,100 tonnes (plus 500 tonnes of SL-LILW). The CEA is studying their recovery.

According to the chosen process, the existing drums would be compacted and the flattened discs obtained would be placed in threes in a stainless steel drum (220 L) and then embedded in a silica-sodium aluminate binder containing sodium fluoride (corrosion inhibitor). Each new drum could contain up to 150 kg/package of pyrophoric metal. The chosen calcium-free binder, referred to as a geo-polymer (MGeo), contains less water than Portland cement and its physical (fluidity and porosity) or even chemical properties (control of pH, possible addition of salts) can be controlled. A series of measurements have been carried out on samples and prototype packages, either inactive or containing radioactive cladding.

They show:

- very low hydrogen emissions (0.2 L/package/year),
- excellent resistance to heating (up to 500°C) and to radiolysis (microstructure maintained up to 10 MGy),
- good mechanical strength,
- no leaching of magnesium by cementitious water (only the Na is leached),
- protection of the uranium metal from any reaction with water by the binder.

Additional studies are continuing on performing a more complete characterisation of such a package that offers very good guarantees of resistance under operating and disposal conditions. 7,500 packages are expected.

Bulk or compacted LLILW metal from the structure of "non-UNGG" fuel assemblies processed in UP1, are generally embedded in concrete casings or metal drums in a hydraulic or cement/bitumen binder.

Structural aluminium or aluminium alloy LLILW originating from UNGG fuel stored at Marcoule is much less common and is less pyrophoric than magnesium waste. It cannot be embedded in cement. Research is being conducted into a new phospho-magnesium, boric acid and lithium nitrate-based matrix. It inhibits the production of hydrogen through the passivation of aluminium. More than 1,300 380 L packages are expected (F2-4-07 family).

The zircaloy fines originating from the cropping of needles in the assemblies are compacted and placed in CSD-C packages. This rules out any risk of pyrophoric reaction.

c) Saline waste packages

The current saline waste packages consist of the sludge and residues from the CEA's STEL evaporators (concentrates) coated in a hydraulic binder, with the resulting product contained within 200 L to 700 L metal (or concrete) drums or containers (8,500 packages, 3 300 m^3, mainly belonging to the F2-5-02 and 03 families). This waste contains large amounts of miscellaneous radioactive precipitates, some of which are highly soluble in water. In addition to emitting hydrogen from radiolysis, in the long term, they may be sources of solutions containing very high levels of salts and mineral chelating agents. These solutions may modify the migration of radionuclides (near-field and far-field).

The future saline waste packages will result from the recovery and packaging of co-precipitation sludge from STE2, which is still stored in silos at La Hague and whose chemical composition and activities have been defined. A proportion of this sludge has been coated in bitumen but the ASN prohibited any further packaging of this type in 2008 for reasons of safety at the UP3 plant. Areva has therefore designed and characterised a new package, referred to as "C5" which, after storage, will be sent to Cigéo (14,500 packages, 3,900 m^3).

The dried and pelleted sludge with magnesium stearate as an additive will be placed in "breathing" 316 L stainless steel containers equipped with a Poral plug (210 L, 555 kg with 326 kg of sludge) and embedded in sand. Areva has shown that:

- the chosen type of steel, as envisaged, resists generalised or stress-induced corrosion in the presence of water vapour (storage at La Hague or deposited in ventilated LLILW cells) or liquid water (long-term). Corrosion by pitting in the presence of chloride ions originating from salt spray-rich atmospheres (storage at La Hague) or from products of the radiolysis of the sludge (HCl) can even be ruled out;
- this steel resists even the most extreme chemical aggressions;

- in storage or when deposited in the cells at Cigéo, the sole source of hydrogen (4 L/year, or exceptionally 15 L/year) is radiolysis in the pellets, as the presence of free liquid water in and around the packages is excluded under the operating conditions.

Furthermore, Areva has measured the production yields of gases from the radiolysis of the inorganic (hydrolysed and hydrated) and organic (TBP, stearate) chemical compounds contained in the irradiated synthesis sludge (1 MGy) and the sludge from STE2. Areva has also measured the radiological and chemical source terms of the leaching of the pellets by cementitious water (pH 11 to 13). These source terms are very low in radionuclides, nitrate ions and sulphate ions and in sodium, due to the unexpected formation of an insoluble compound ($Na_3NO_3SO_4,H_2O$). The chemical species released, including traces of organic acids derived from the radiolysis of the magnesium stearate, have been identified and the migration of the radionuclides has been modelled.

It has taken several years to complete the three fields of study required for the characterisation of the C5 package. The qualification file is ready for submission to the French Nuclear Safety Authority (ASN).

d) Organic waste packages

These packages are numerous and varied in terms of their origins (Areva La Hague and Melox, CEA civil and CEA DAM), the organic materials contained therein (chlorinated or non-chlorinated polymers, cellulose, resin, etc.), and their activity, especially alpha if they contain traces of plutonium and americium. Approximately 40,000 packages contain a total of 3,600 tonnes of organic materials (families: F2-3-10 and 13, F2-4-11, F2-5-04 and 05). The alpha radiolysis of organic molecules generates hydrogen and other gases such as CH_4, HCl, HF, etc. and degrades them. The water-soluble degradation products (WSDP) are mixtures of multi-functional organic molecules with chelating properties. In the long term – as is the case for saline waste, but for other reasons – they may modify the migration of radionuclides in the near and far fields of the repository. It has been estimated that there will be 11,400 packages (6 600 m^3) with very high levels of alpha-emitter contamination, and 17,900 packages (15,000 m^3) containing less alpha-emitter contamination.

The prior and current studies conducted by producers on the radiolysis of industrial packages containing organic waste, associated with numerous generic studies carried out by the CEA, CNRS and university partners on the radiolysis of "organics", have allowed for the creation of a substantial database (over 3,500 items) of the radiolysis yields of gaseous and non-gaseous organic products ("Prelog" [from the French acronym for "Polymers under radiolysis for the study of organic leachates"] database). Through the use of operational codes, Prelog allows for the modelling of situations encountered under operating conditions, in addition to long-term forecasts.

Similarly, considerable progress has been made in the identification of WSDPs, but the mixtures of degradation products are so complex that much more work will be required to identify all of them. Further studies are currently being conducted in an effort to identify WSDPs in detail and ascertain their ability to chelate actinides. The compounds studied originate from the radiolysis of polyurethane, polypropylene and polyvinyl chloride and all of them possess at least the carboxylic function. These studies are consistent with those carried out by Andra on the production of gases, WSDPs and the migration of radionuclides under all disposal conditions.

e) Other waste

Many types of non-packaged technological or operational LLILW are currently kept in storage and more will be generated by the dismantling operations currently in progress. The typologies of waste already produced and to be produced over the next 20 years are very varied. They could be packaged in new or previously used packaging. The choice requires the continuation of intensive R&D, not just on the characterisation of waste but on the packaging itself.

The majority of industrial packages for LLILW manufactured in line or pending delivery to Cigéo satisfy the guidelines for disposal specifications jointly defined by Andra and the producers. These specifications must be transformed into obligations and requirements in the final specifications.

The recovery of unpackaged waste requires the development of new packaging that is adapted to each waste family. This packaging must in practice:

- increase the resistance of the packages to predictable physical and chemical stresses,
- minimise hydrogen emissions,
- minimise the leaching of radionuclides when the waste comes into contact with water.

For the past ten years or so, Andra and the waste producers, in collaboration with their CNRS and academic partners, have been conducting research into the production of new packages. The results currently obtained for waste packages containing reactive metals, soluble salts and/or organic matter show that hydrogen production can be significantly reduced through the use of appropriate packaging and, with regard to the leaching of radionuclides and other elements, it is possible to reduce the source terms of these packages substantially.

Appendix X

R&D ON THE ADS (ACCELERATOR-DRIVEN SYSTEM)

An ADS facility is the product of coupling a high-power proton accelerator with a sub-critical fast-neutron reactor via a spallation target. An industrial ADS could be used as an intense fast-neutron source for the transmutation of minor actinides (MA) and the fission of plutonium, or indeed, in this case, to generate electricity above what is required for its operation. From this perspective, the ADS and FNR will either be complementary if the FNRs are exclusively electricity-generating, or competing if the FNRs can transmute MAs. From the French perspective, the time for making a choice between the ASD and FNR for MA transmutation is in the distant future.

Several countries are researching the design of a prototype ADS. The project at the most advanced stage – Myrrha (Mol, Belgium) – benefits from several years of European development. It involves coupling a 600 MeV, 4mA linear accelerator, whose reliability far exceeds that attained at present (factor of 10), to a sub-critical 65 to100 MWth reactor powered by a MOx fuel (30 to 35% Pu) cooled by a lead-bismuth (Pb-Bi) liquid eutectic and a Pb-Bi spallation source with a window. Myrrha can also act as a fast neutron source for the irradiation of materials for FNRs (CNE Report No. 7, page 38 and http://myrrha/sckcen.be).

The Board is monitoring the R&D that is being conducted into the feasibility of this facility and the components of its three parts.

Since 2007, the IN2P3, SCK-CEN and CEA have been involved in the project to couple a 14 MeV fast-neutron source (Genepi) – continuous, pulsed or continuous with beam interruptions – to an experimental, zero-power reactor (Venus) consisting of uranium metal and lead rods (Guinevere project, Euratom FP6). This project has given birth to Freya (Fast Reactor Experiment for hYbrid Applications, FP7), dedicated to studying the criticality of Venus under different configurations and different operating conditions for Genepi. This coupling of the two parts of the experimental assembly aims to develop sub-criticality monitoring and control procedures that are essential for demonstrating the safety of an ADS. Freya should continue for another two years and explore the reactivity of cores more closely resembling those in Myrrha.

Such developments of instrumentation for the accurate measurement of the criticality and sub-criticality of an ADS (currently better than 200 pcm) are clearly essential. The Guinevere facility has demonstrated that these measures are reproducible and there are plans for it to act as a mock-up for Myrrha.

The linear proton accelerators in service suffer from instabilities during their normal operation, including beam interruptions, which make them unsuitable, in their current state, for use in an ADS. Moreover, the beam must be continuous and the power of this type of accelerator does not exceed 1.3 MW, whereas Myrrha will require power of up to 2.4 MW. An ADS accelerator must be perfectly stable (1 to 2 stoppages per month lasting less than 3 seconds). The project in progress – MAX (Myrrha Accelerator eXperiment, FP7, 2011-2014) – coordinated by the CNRS with a view to improving the technology of the future Linac supraconductor for Myrrha – involves more than 10 partners (http://ipnweb.in2p3.fr). The desired qualities of Linac's components that have already been selected (injector, low and high-energy acceleration cavities, amplifiers, etc.), or already successfully tested, allow us to envisage the creation of a reliable prototype. However, it is essential to continue the research into the integration of components so that their performance can be definitively characterised. Certain components have never been completely manufactured. This is the key issue of the accelerator component of the vast Myrte (Myrrha Research and Transmutation Endeavour, 7 partners, 2015-2018) programme whose overall aim is to consolidate the design of Myrrha's injector. Myrte also includes components on transmutation by ADS. The European Spallation Source (ESS) under construction in Sweden will use a 2 GeV (5 MW) linear proton accelerator that shares a lot of similarities with the model envisaged for Myrrha. Joint research is also being carried out by the CNRS and the CEA as part of a collaborative programme involving 17 countries. In general, for all of the accelerator applications, the increased reliability leads to improvements in the time of use and higher quality results for research and industrial applications.

The Chinese ADS (100 MWth), which is similar in design to Myrrha, is scheduled for commissioning in 2023.

The development of the Myrrha project takes account of the lessons learned from the Megapie project that demonstrated the feasibility of a liquid Pb-Bi spallation source under helium: continuous 575 MeV proton beam of 1.74 mA (for 1 MW of power), depositing 650 kW into a few litres of the 80 litres of the source liquid circuit. The design/construction of the source began in 1999 and it was installed at PSI in 2006. After irradiation experiments lasting four months, the results were analysed over a three-year period. This source was then transported to Zwilag for dismantling and the destructive analyses of materials have just finished. This amounts to 15 years of R&D. This is a typical example of an international nuclear physics experiment (8 countries involved, managed by France) prior to the launch of a major project such as Myrrha, an experiment which itself requires ancillary R&D in numerous fields (the CEA is currently preparing to publish feedback from Megapie).

The lessons to be learned from Megapie concern neutron production and flows, the neutron activation of materials and the physics/chemistry of Pb and Bi spallation, the thermo-hydraulics of the Pb-Bi eutectic for heat removal, the resistance/corrosion of the target materials and of its window in particular, and the inventory of radionuclides produced in the eutectic, some of which are volatile ($^{208-210}Po$, ^{129}I, etc.). The results have allowed for the validation of different computation codes. The operation of Megapie has shown the key points to be examined in order to overcome failures that might be anticipated on the safety level.

Megapie has attained all of its scientific and technological objectives and the results obtained will make an essential contribution to the use of a Pb-Bi as a spallation fluid and also as a coolant. They show that additional studies must be carried out for Myrrha regarding the choice of the configuration of the spallation target (Myrte project and internal SCK-CEN programme).

The Myrrha project is being consolidated. This consolidation must continue until 2018. At this time, a pre-construction design should be available prior to the decision to be made by SCK-CEN and its international partners regarding the start of the construction of Myrrha.

Appendix XI

ASTRID: ENGINEERING

Astrid's design is based on research into several *Major Components* and their assembly. It is progressing through each consecutive preliminary design (PD) phase (basic PDs: BPD1, BPD2 and the detailed PD: DPD3), and reviews of the choices of options that define the R&D priorities that are required to move from one phase to the next.

The CEA is the project owner and the strategic coordinator of the project involving around 600 people.

At the end of 2015, the CEA must submit:

- a final report on AVP2 (BPD2) summarising the 2,350 technical files drawn up during the BPDs and justifying the choices for the DPD (3 years from 2016-2019);
- the Safety Options File (SOF) for Astrid, demonstrating how the design counters the anticipated risks encountered in different operating situations.

MAJOR COMPONENTS

Astrid has the following major components:

- the reactor core and the primary circuit (1500 MWthh, 2,700 tonnes of sodium at between 400 and 550°C);
- fuel assemblies (300 hexagonal tubes, 65,000 needles, 4.7 million annular pellets amounting to 26 tonnes of $UPuO_2$ and 16 tonnes of UO_2;
- the internal storage system for spent fuel (SF) assemblies and the system for their removal under a gas hood via a rotor in the vessel and replacement with new fuel assemblies;
- the advanced safety control system (traditional system and an additional system that is separate from the traditional system);
- the electromagnetic system for releasing cutoff rods at the Curie point;
- the core catcher for the entire core with means of guiding corium flow through the diagrid-grating;
- 4 secondary circuits for thermal energy conversion (375 MWth), with 3-circuit temperature exchangers (primary Na-Na, secondary Na-Na and tertiary Na-N_2 or Na-H_2O). In the first case, the water vapour generators (VG) would operate at under 150 bar and 500°C with 1 vapour line for energy conversion. In the second case, gas generators (GG using N_2) would operate at under 180 bar and 515°C with 2 gas lines;
- the residual heat removal systems (EPuR): two with Na coolant and one cold air source (5 exchangers immersed in the vessel) and one with oil coolant and one cold water source (2 exchangers on the internal security vessel);
- the anti-seismic isolation system for the vessel, 3 primary pumps, 4 VGs or 4 GGs and 5 EPuRs;
- the operating inspection system (100% inspection) and repair system (ISIR) with new measurement technologies.

All of these components are innovative even if they are developments of the traditional components that were previously used. The nitrogen-based energy conversion system (ECS) operating on a Rankine thermodynamic cycle is a major innovation. The other major technological innovation is the core catcher.

TOOLS

To develop the Astrid concept, the CEA needs tools adapted to the size of the experiments and modelling to be carried out. The former are/will be included in major platforms and the latter will be carried out by multi-scale simulation software programs.

a) CEA testing platforms

Platforms for neutron physics, such as Masurca (undergoing renovation) or for underwater studies (thermohydraulic loop), such as Giseh, exist and are operational as they play an essential role in the studies for PWRs. Platforms for the development of "sodium" technologies for instrumentation (neutron), detection (sodium leaks), diagnosis and repair (ISIR) must be upgraded or built. The R&D associated with sodium technologies requires the development of numerous ultrasound-based sensors and the reconstitution of acoustic signals in image form. For the moment, the only operational platform is Papirus, offering a variety of low-volume (from one litre to 3 m^3) facilities for studying Na-N_2 exchangers (up to 40 kWth). It is a development of the Diademo facility.

Large sodium test loops dedicated to full-scale R&D, the qualification of components (by similarity or full-scale tests) and for advanced cleaning processes will be brought together in the CHEOPS (French acronym for Testing hall and circuits for large sodium components) platform. The design and construction of this facility on the Cadarache site began in mid-2014 and the first tests are scheduled to be held in 2019. It will cover varied needs, with the main ones being:

- the thermomechanical qualification of different types of assemblies covering the different operating modes (steady state and transient states),
- the qualification of a sodium-gas heat exchanger module (2/3 scale), a sodium gas heat exchanger (1/10 scale), and the apron sealing on intermediate exchangers,
- the qualification of handling chain elements under sodium: slide gate valves, hood brake, hood grabs, demonstration of the efficiency of cleaning processes, etc.
- the qualification of instrumentation and ISIR (In Service Inspection and Repair) techniques: long-distance telemetry, thermo-hydraulic instrumentation, qualification of intervention equipment under sodium (holders, inspection and repair tools),
- the validation of transfer (material and heat) models in the reactor cover gas, mechanisms for limiting thermal transfer under the cover slab, and cover-slab feed-through technologies, etc.
- the validation of cold trap operation processes, etc.

The Plinius facility, designed to allow for the study of corium-Na interactions at the hundreds of kilogrammes scale, is planned for the future. In the framework of its collaboration with Russia, the CEA also has access to technical platforms dedicated to sodium.

b) Scientific computation tools (SCTs)

The CEA is developing a new generation of additional codes for the numerous simulations required for the assembly of major components at fuel level (pellets, needles, housing, assemblies), for the core (neutron and thermo-hydraulic simulations) and the reactor system.

PARTNERSHIPS AND COLLABORATIONS

The CEA has established numerous national and international collaborations and partnerships.

a) R&D collaborations

Russia: The CEA-Rosatom agreements concern irradiations in BOR-60 (2012-2014) and the establishment of a work group in preparation for adding an Astrid assembly to BN-600. The CEA also benefits from facilities for qualifying materials and fuel samples in addition to Russian experimental platforms.

India: Agreements with IGCAR and BARC on R&D concerning safety/serious accidents in FNRs

China: Discussions on defining the scope of collaborative operations are in progress

USA: The CEA has carried out checks on the calculations of Astrid's core neutronics, which have been confirmed by the DOE. The DOE is interested in Pu-burning FNRs.

Japan: The CEA has signed agreements with JAEA and, in 2014, important agreements were entered into with METI and MEXT for the joint development of engineering and R&D studies in support of the DPD for Astrid. It is a preferred partner.

b) Engineering, R&D and qualification partnerships

In addition to the R&D conducted in France with NEEDS and in the framework of the three-party agreement signed by the CEA, Areva and EDF, the CEA benefits from the work of the Generation IV Forum and, within ARDECO (Astrid R&D European COoperation), has brought together several laboratories with expertise in materials, fuels, measurement and modelling in the nuclear and safety fields, from: Switzerland, Italy, Germany, United Kingdom and Sweden.

The most important partnerships for the development of the Astrid project are those established by the CEA with 13 industrialists concerning the engineering of the major components and also on the R&D associated with technical innovations, organisational assistance and safety. The industrialists are partly funding their works with their equity capital. Six of them are directly involved in designing Astrid: Areva in association with a Japanese consortium (boiler, auxiliaries, command and control), SEIV (hot examination cells), Alstom (ECS), Bouygues (civil engineering) and Jacobs (equipment and infrastructures). The core design is reserved for CEA in association with Areva.

GOVERNANCE

The CEA has deployed an infrastructure designed to manage the collaborations and organisation of the project. This involves aspects such as reviewing the choices of options, expressions of R&D needs and the qualification of Astrid according to the TRL scale.

QUALIFICATION OF OPTIONS

The definition of the design goes hand-in-hand with the qualification of the options, the major components, equipment, civil engineering and functions ensuring either the operation (sodium thermo-hydraulics, instrumentation under sodium and repairs under sodium) or safety (incidents and serious accidents).

Appendix XII
ASTRID: R&D ON MATERIALS

ASTRID

a) FNRs, specificities of Astrid and aims of R&D on materials

The design of Astrid's reactor marks a major breakthrough in relation to the design of the previous Na-FNRs built in France (Rapsodie, Phénix-PX and Superphénix-SPX trilogy) and of the other FNRs built abroad. It is the forerunner of the Generation IV Na-FNRs. It needs to demonstrate the industrial feasibility of an Na-FNR with a service life exceeding 40 years and a target life of 60 years, and mastery of the total fusion of the core with a radionuclide containment capacity eliminating any need for the long-term evacuation of populations. This leads to "specificities" that raise questions regarding the choice of structural and core materials, under nominal as well as incidental or accidental operating conditions. The materials must have very high levels of mechanical strength and be corrosion-resistant to fulfil their functions without failing. The answers to the questions can be found in feedback from previous experience and in new research to consolidate/define the semi-empirical laws of behaviour of the materials, most of which are currently based on time frames of much less than 60 years.

b) Feedback and resources for studies, R&D

The feedback is derived from 35 years of R&D on French Na-FNRs carried out by the CEA, EDF and Areva, incidents and difficulties encountered during the operation/maintenance of these reactors and examinations—during dismantling—of irradiated metallic materials under operating conditions. In this respect, the PX—an electricity generating reactor that is also experimental—should be the biggest source of information. A programme to optimise the exploitation of what the CEA refers to as the "PX treasure" has already been established. All of the MOx fuel assemblies for the FNRs have been manufactured in the Cadarache Pu workshop. Finally, the latest developments in the production of PWR MOx in Melox are available for the transition to FNR MOx.

The R&D on the materials of the FNRs has been underway for a long time and several samples of metals previously irradiated in PX are being examined,along with the results of recent experiments (referred to as "PX end-of-life", 2009-2010).

The CEA, EDF and Areva possess facilities and tools adapted to R&D on Na-FNRs, even if only for activities specific to studies involving metallic sodium. They also possess powerful design software and simulation codes for designing and dimensioning components. There is a significant lack of fast neutron irradiation equipment in France, although this is overcome by access to foreign FNRs in the framework of international collaborations.

The R&D is being carried out in the perspective of the construction of Astrid, but with a longer-term aim of archiving all of the basic data. The CEA is developing the RCC-MRX code (on the basis of the 2012 version), incorporating the feedback from all French nuclear operators on metal reactor materials and the technical rules for the design and mechanical construction of nuclear reactors. This code helps answer the questions concerning any potential radiation damage and the thermo-mechanical stresses on steel, according to the current state of the art.

c) Schedule of materials studies for Astrid

The CEA's R&D should continue throughout the end of the AVP2 (BPD phase 2) until the end of 2015, and then during the DPD from 2016 to the end of 2019. It will focus on the core and structural materials and will have the following objectives:

- The construction of a first core (start-up core) with a burn-up rate of 60 GWj/t, which is close to the current burn-up rate for UOx and MOx fuels in PWRs. This rate will be gradually raised to 120 GWj/t by following a qualification plan. This corresponds to the TRL scale that evaluates the maturity level of a technology, which needs to be increased from TRL 6 to

TRL 7-8. The core design currently corresponds to TRL 4 qualification (validated design). Between now and the end of 2019, the R&D for the design must allow for a two-level increase on the scale (from 4 to 6).

- Improvement of the RCC-MRX code to take it through to maturity by incorporating the results of R&D on structural materials.

All of the data and technical developments will thus be available to facilitate preparations for the construction of Astrid.

The R&D on materials benefits from the required resources and substantial feedback accumulated over decades of activity in the Na-FNR field. The methodology and approaches that will be employed to qualify the materials to be used in Astrid's construction are clearly defined. As for all of the studies relating to Astrid, they require significant human and financial resources that must be earmarked right now. They also require analyses of the irradiated materials (feedback to come) to be integrated into the dismantling schedule for the core and for the removable components of PX which should take place between 2015 and 2029.

d) Core materials

These are all of the components that allow for the development and control of the chain reaction, the internal and external core fuel assemblies and the other assemblies, control and cutoff rods (B4C). Most of the studies focus on the fuel assemblies and needles.

◆ Steel

In the nominal state, the stainless steel (Fe/Ni /Cr/minor elements, C< 0.1%) of the fuel needle cladding and the hexagonal assembly tubes are subjected to intense radiation from fast neutrons within a temperature range of 450 to 650°C (over a height of approximately 2 m from the top to the bottom of the needles). The $UPuO_2$ ceramic fuel pellets are irradiated by the neutrons, the fission fragments and the radiation from numerous radionuclides produced in the fuel, under a radiant temperature gradient of approximately 500°C/mm (2,000°C in the middle and 500°C at the edges; the pellets have a diameter of approximately 6 mm with a central hole of 1 mm). 300 to 500 W/cm of power is released, depending on the position of the needles in the core. These temperatures may be even higher during thermal transients. The metal materials must therefore be especially strong, particularly the cladding which forms the first radionuclide containment barrier, not just during their time in the reactor but also during cooling and while the assemblies are stored underwater for decades. The physico-chemical phenomena that are produced in the steel and fuel are well known. However, more studies on assessing the impacts on the resistance of materials are required at both the fundamental and development/innovation levels, in order to ensure the safety of the future FNRs and of Astrid in particular. Indeed, the Generation IV Na-FNR reactors must demonstrate that they are safer than Astrid, which will be at least as safe as the EPR-type Generation III reactors. The resistance of the materials is one of the keys to safety.

The latest specification for the steel grades that were used for the cladding of Rapsodie, PX and SPX is AIM 1 austenitic steel (15% Cr, 15% Ni, stabilised with Ti – 4% – and cold-drawn, with a face-centred cubic structure). The greatest fear concerning the needle cladding is its deformation, which is partly caused by the swelling and creep under irradiation of the steel due to the formation of cavities. This is a function of the number of movements that an atom undergoes under the effect of neutrons (dpa), the temperature T and time t. The dpa is associated with the average burn-up rate (ABR), the number of split atoms and the neutron fluence (1 dpa closely corresponds to an ABR of 1 GWj/t, to 1% fission or to a fluence of 2 10^{21} n/cm^2 for E_n > 0.1 MeV). The variations in deformations (dpa, T, t) are well known for the latest AIM1 precursors over extended ranges of parameters (up to at least 110 dpa). They still need to be confirmed for AIM1 and to be extended to 130 dpa (or even 150 dpa) and above 700°C. To this end, needles irradiated in PX and samples irradiated in BOR60 are currently being examined (Zèbre, Plavix, Oliphant 1 bis and Tiramisu programmes/experiments). Alpha and heavy ion irradiation experiments are also being conducted at the same time. The creep of AIM1 under irradiation at T > 650°C is also being analysed.

As for the needles, the grade of steel used for the hexagonal tubes of the assemblies is based on feedback from French Na-FNRs. It is EM10 martensitic steel (Cr 8%, Mo 1% with a quadratic-centred structure) which does not deform under neutron irradiation of up to 155 dpa and T < 600°C (Boitix-9 programme/experiment) but it can become fragile at significantly higher temperatures and lose its properties through phase transformation at around 850°C. Further research is required on its behaviour under non-nominal conditions during temperature transients (incidental or accidental conditions). Studies of its creep and phase change properties at high temperature (T of between 700 and 1000°C) are in progress.

In conclusion, the behavioural laws for AIM1 and EM10 steel should be clearly defined between now and the date of the decision to build Astrid. They will broadly cover the fields of dpa and T in the specifications. The Astrid programme will itself contribute to the continuation of studies in the run-up to the industrial mastery of the production of the cladding and assemblies of the future Na-FNRs and to the qualification of new types of steel (e.g. martensitic ODS steel).

◆ **Fuel**

The fuel needles for Astrid will contain pellets of $UPuO_2$ and UO_2 ceramic (with U depleted to ^{235}U). The mixed oxide is actually the fuel for FNRs and UO_2 constitutes the fertile part (blanket) that allows for the production of the Pu. The physico-chemical phenomena produced both in the pellets (solid or annular) when the reactor is operating, and in contact with the needle cladding are documented and, for the most part, clearly understood. Indeed, they have been studied for a long time. Databases are available corresponding to burn-up rates of up to 150 GWj/t. The behaviour of needles has also been modelled. The latest Germinal V2.2 computation code used by the CEA allows for the correct prediction of the behaviour of the needles, such as their deformation (partly due to the expansion of the pellets), the formation/progression of the central hole, the internal corrosion of the cladding, the progression of the fusion front in the event of a power transient or a variation in the sodium flow rate, etc. Nevertheless, its validation is continuing on the basis of the precise modelling of material reorganisation phenomena in the oxides and examinations of needles irradiated in PX. This study benefits from international collaborations.

The quality of the fuel ceramic, measured by its capacity to avoid damaging the cladding (and the overall operation of the needle), depends on its manufacturing procedure. For Astrid, the CEA has adopted the Coca process that proved its worth in Rapsodie PX and SPX (106 tonnes of $UPuO_2$ oxides were pelleted at Cadarache using 21 tonnes of Pu between 1963 and 1999). It consists of mixing UO_2 and PuO_2 powders. Research is underway in an effort to simplify the process and develop it to include better implementation conditions for the powders (grinding, flowability, pelleting and lubrication) and increase the Pu content (from 20 to 30%). The delivery of the first annular fuel pellets corresponding to the dimensions of Astrid is expected in late 2015.

◆ **Production and qualification of the first fuel assemblies**

Extensive feedback is available to inform the production of needles and assemblies. Technological improvements are being examined with a view to automating the production of assemblies for Astrid.

Preliminary studies for the design of the Fuel Production Workshop (FPW) are in progress (CEA and Areva) with a particular focus on taking account of the variation in the isotropic composition of the Pu (increase in ^{238}Pu). At this point, it should be noted that the following requirements apply firstly to the testing and then to the operating phases of Astrid, three years before its start-up: the production of 100 assemblies/year (first core) and then 90 per year for 9 years, followed by 63 per year for 11 years (aggregate total of 1,800 assemblies). One assembly will contain approximately 210 needles.

If there are no problems with the production of the first assemblies, their "in-pile" qualification prior to the start-up of Astrid will have to be carried out collaboratively because France does not possess its own fast neutron irradiation facilities. Such collaborations are envisaged with Russia in BOR60 and BOR600 (or with Japan in Joyo, if it is recommissioned). The "off-pile" qualification (mechanical and thermal testing in water and sodium) may be carried out in facilities belonging to the CEA (Giseh and Cheops) and Areva (Creusot Loire). Astrid's assemblies will then be qualified to increase from 60 dpa to 120 dpa or more.

In conclusion, the "fuel" and the assemblies will be manufactured using a well-established and understood process and technology. The latest lessons that the CEA may learn from the examinations of assemblies from PX and the experiments that took place prior to its shutdown will consolidate the knowledge. Only the in-pile qualification of the separate or assembled components will be dependent on the effective availability of fast-neutron irradiation facilities.

◆ **Structural materials**

These are the boiler materials: the vessel and its head, in addition to other components that cannot be replaced (internal) during Astrid's service life (60 years). The problems thus relate to the resistance of the materials and consequently their ability to ensure safety during decades of service. The feedback in this regard comes mainly from the lessons learned from PX which operated for an equivalent period of 15 years at full power, and also from the "structural materials" R&D for SPX and the methodology used to conduct it. An Expert Work Group on PX Materials (Groupe de Travail Expertise Matériaux – GTEMP), a three-party group involving CEA, EDF and Areva, now exists which analyses the lessons to be learned from PX and provides guidance for the examination of components for the ongoing R&D requirements for Astrid.

316L(N) austenitic stainless steel (Cr 17%, Ni 12%, Mo 2.3%, nitrogen-stabilised) has been chosen for the internal components and the tank because it offers the best compromise between mechanical properties when irradiated and resistance to corrosion by sodium –characteristics that have always been sought for the internal components of Na-FNRs. The internal components (vessel, pump, pipework, etc.) are constantly in the presence of sodium at between 400 and 550°C. They – especially the vessel – are subjected to low levels of irradiation (compared to the core materials) but this continues for decades. Above all, they undergo continuous and cyclical mechanical operating stresses. Under these conditions, and over time, the main phenomena that must be controlled for the steel are ageing under stress and irradiation, creep and corrosion by sodium. The R&D consists of defining reliable laws of behaviour over the longest possible period based on existing data from feedback and new, long-term laboratory measurements on test pieces (fatigue-recovery and creep under load) in temperature and tension ranges (500-700°C) and (70-1000 MPa) respectively, that broadly cover the envisaged conditions for Astrid. For creep, two time-dependent mechanisms come into play: over short periods (< 1 year), concerning an elastic deformation (striction), followed over the longer term by the formation of cavities along the grain boundaries. The time taken for the transitions between the two states to appear is linearly inversely proportional to T. The modelling of "life span", e.g. before the failure of test pieces or the appearance of cavitation cracks in the steel and the comparison of the models with the substantial amount of experimental data, give durations of up to 25 years but with significant margins of error/dispersion of values (factor of 2 to 3). The experiments in progress will allow for the validation of periods of up to 35 years or more, in ten years' time. The general aim is to model the condition of the steel after 40 to 60 years of exposure to stresses in operation in order to dimension the components, while applying safety coefficients. The observation of abnormally high creep rates over long experimental times calls for the studies to be continued. There are also fears of other types of damage that may affect stainless steel, especially to welds (cracking under recovery and under stress). The parts of the components in question have been identified and the studies in progress must lead to the definition of the conditions required to prevent sudden failures.

R&D on the long-term resistance of 316L(N) steel has an important role to play in ensuring the reliable dimensioning of Astrid's internal components. The fundamental studies and R&D on the creep states for steel over periods exceeding several decades must therefore be continued in order to consolidate the extrapolations of damage to the metal and thus reduce the uncertainties.

The sodium coolant, although very pure, contains traces of oxygen (5-12 ppm) and sources of oxygen pollution appear during the operation of the reactor. Under the operating conditions for an Na-FNR reactor and depending on the amount of oxygen, oxides such as Na_2O and $NaCrO_2$ may form.

For steel in the presence of a "sodium/oxygen" flow, the great fear is for corrosion to occur at 500°C, thus altering the surface of the steel. Models of the corrosion kinetics have been produced (thicknesses of corrosion layers and of the depletion of elements) which depend on the oxygen content, the T and the sodium circulation speed, but the mechanism is not fully understood despite numerous microscopic observations of the steel/sodium interfaces on test pieces (Corrona facilities) and of the PX components for example. It is basically formed by sodium chromite ($NaCrO_2$) with chromium depletion of the steel, penetration of Na into the austenite and rearrangements of the major and minor elements leading to the formation of iron carbide and modification of the austenitic structure. The oxygen transients are obviously the most damaging because, in normal operation, the oxygen initially contained in the sodium is quickly consumed. The transients lead to the formation of Na_2O which is itself corrosive. The 890 tonnes of sodium in the PX vessel contained tens of ppm of Fe, Cr, Ni and C towards the end of its operating life. The R&D seeks to obtain a corrosion model that is consolidated by the integration of mechanisms and a source term model for the transfer of material.

The experience from PX shows that corrosion by sodium had a low impact on the internal reactor components. Nevertheless, it is important to acquire more detailed knowledge of the mechanisms that come into play, especially from the perspective of a 60-year service life for Astrid (and commercial Na-RNR). Over time, sodium corrosion and the specific thermo-mechanical modifications to 316L(N) steel cause relatively superficial damage to the components, especially the vessel, and consequently impose a time limit beyond which the continued operating safety of the reactor will need to be demonstrated. The aim of the R&D activities in progress is to consolidate the predictions and postpone this time limit for as long as possible. They are on the right track to do so.

f) Other R&D carried out by the CNRS and the CEA

The CNRS is conducting research into nuclear energy through two of its institutes (INC and IN2P3) and through its participation in the ANCRE Alliance. Within this broad framework, it is coordinating the NEEDS project that includes the "Materials" collaborative project supported by the CEA, Areva and EDF. Of all the projects chosen and financed following the 2013 invitation to tender on metal and ceramic materials, those of interest to Astrid concern the behaviour of boron carbide under irradiation and the segregation of minor elements of steel under irradiation. The studies have only just begun and the Board has only been informed of a few preliminary results, such as the amorphisation of B_4C after 4 dpa of damage.

TRANSMUTATION

a) Materials for americium transmutation

In the transmutation design for americium-loaded blankets (couvertures chargées en américium – CCAm), the transmutation fuel assemblies could be positioned on the outer edge of an Na-FNR core, e.g. in the form of a ring of 80 assemblies. The transmution fuel, which is currently the subject of R&D, will be a mixed U-Am oxide from the series of solid solutions:

U_1-$xAm_xO_{2\pm\delta}$ (non stoichiometric, with $0.075 < x < 0.5$ and δ variable according to the proportions of U^{4+}, U^{5+}, Am^{3+}, Am^{4+}). In its last report, the Board reviewed the status of research into the synthesis of these solid solutions, their properties and their behaviour under irradiation (Appendix VIII of Report no. 8). The development of a fuel for the industrial transmutation of americium will require decades of R&D during which Astrid will play a key role as it will allow for the qualification of the needles and then of the assemblies. Before reaching this point, the studies are initially focusing on samples of oxides and mechanisms prefiguring the needles.

The density of the transmutation fuel assemblies must exceed 95% of the theoretical density (TD), and the morphology of the mixed oxide grains must allow for the proper diffusion of helium which is produced in large quantities while the fuel remains in the reactor, and all within the temperature range of 800-1000°C. These properties mainly depend on the method of preparing oxides in ceramic form, on both thermodynamic and kinetic levels. It must be possible to prepare the fuel for the americium-loaded blankets (CCAm) on an industrial scale.

The CEA and the industrialists have very extensive experience of "powder metallurgy" for the preparation of oxides and especially U-Pu oxides for MOx fuels. The latest results in the field of U-Am oxides show that the simple traditional reactive sintering of UO_{2+x} and AmO_{2-x} (450 MPa, 1300°C, 4 hr) leads to densities below 90% of the TD. However, if the powder mixture is firstly left to react for several hours to obtain the solid solution before sintering (at around 1,800°C, under Ar/H_2 at 4%), densities of above 95% can be obtained if x does not exceed 0.15 (in the "UMACC" – Uranium Minor Actinide Conventional Sintering process). The density diminishes beyond this proportion of Am. Finally, if a "co-converted" powder is sintered (x = 0.15) i.e. obtained from an aqueous solution (Exam process and oxalic co-conversion studied elsewhere) or by roasting a resin exchanged with U(VI) and Am(III) (CRMP – Calcinated Resin Microsphere Pelletisation process – studied elsewhere), the same result is obtained but at 1,300°C. However, this takes two to three times longer, although a 600°C increase in temperature is also obtained. The sintering rates relating to the diffusion of ions correspond to several states. The activation energy for the sintering of the mixed U-Am oxide is higher than for the UO_2 (580 compared to 400 kJ/mole). Even if we start with a solid U-Am solution, it is hard to obtain high densities if x > 0.15.

In addition to this research, results have also been obtained in the framework of the NEEDS programme run by the CNRS and in numerous collaborative projects including the CEA.

This involves controlling the morphology of neodymium oxide Nd_2O_3 by roasting/sintering (700-1200°C in the atmosphere) oxalates of this element, $Nd_2(C_2O_4)_3$ $(H_2O)_x$ yH_2O (x and y variable). These oxalates are precipitated at ambient temperature in the presence of additives that are not related to the reagents. These additives and their implementation influence the morphology of the oxalates and their hydration, and these characteristics then have an impact on the hydration of the oxides. Oxide densities of 98% of the TD may be obtained at above 700°C. As neodymium and americium are homologous and the Nd and Am oxalates are isostructural, these studies point to possible methods of optimising the morphology of mixed U-Am oxides.

The next step is the XANES (X Absorption Near Edge Structure, INE-AKA line at Karlshrue) characterisation of U-Am oxide obtained by co-conversion using resin containing U(VI) and Am(III). The XANES spectra at the LIII threshold of U or Am are used to determine the oxidation levels of the U and Am ions, e.g. according to the temperature. These data help with the interpretation of complicated roasting/sintering mechanisms for mixed oxide in the air or under Ar/H_2. Thus, for $U_{0.85}Am_{0.5}O_{2\pm\delta}$ in air above 300-350°C, U^{4+} and Am^{3+} predominate, and under a reducing atmosphere above 700°C, U^{4+} and Am^{4+} predominate. For all that, these mechanisms have not yet been completely defined. The XANES measurements are broadly applicable for investigating oxidation levels in synchotron light lines dedicated to radioactive materials (as found at Karlsruhe) and equipped with appropriate mechanisms. In this respect, a facility capable of performing in situ XANES measurements on samples undergoing examination is being developed, with the support of the NEEDS programme. The essential part of this experiment is a furnace capable of attaining 2,200°C in a controlled atmosphere (O_2, H_2, N_2, Ar).

The behaviour of helium at between 800 and 1,000°C in U-Am oxides is a problem approached by studying samples of mixed oxides irradiated with neutrons and by studying the implantation/ diffusion of helium ions in these oxides under diverse conditions. The Marios and Diamino experiments (Appendix VIII of Report No. 8) will allow for the measurement of the release of helium and its location in oxide grains on samples irradiated by thermal neutrons at 20-30 GW/, which are significant burn-up rates for a CCAm fuel. For the moment, the implantation of $^3He^+$ ions (5 MeV, profile centred on 1 micron, 0.3 atom %) with several accelerators (CEA and CNRS) have focused on UO_2. Their location in the grains can be viewed and their diffusion by annealing can be measured. All of the measurements reveal slow intra-granular diffusion (blocked by the defects) and fast diffusion at the grain boundaries. This allows for an initial modelling of the He behaviour according to the temperature. The effect of the presence of fission gas is simulated by the concomitant implantation of krypton and iodine ions. This presence inhibits the diffusion of helium below 1,000°C.

The studies to come – both experimental and theoretical – will be aiming at modelling the behaviours of He and Xe in the CCAm fuel: diffusion, solubility and swelling, etc. No research concerning nuclear energy can be carried out without neutron and/or accelerated ion irradiation.

The EMIR Federation consisting of six French accelerators (http://emir.in2p3.dfr) offers experimenters a wide range of light to heavy ions in a vast energy range. It coordinates access to the beams based on the choices of projects.

Numerous publications concerning these above-mentioned studies have been produced in recent years.

b) Conclusion

All of these results show that is not/will not be easy to obtain a U-Am transmutation fuel and that it will certainly be a much more complicated process than for a UPu MOx. The Board considers that the fundamental studies of U-Am oxides and their formation must be continued, or even stepped up, according to the programmes that have been launched, as we must rise to the challenge of demonstrating that the possibility of industrial americium transmutation will not fail due to fuel-related problems. Any results obtained prior to using Astrid as a fast-neutron source will contribute to the success of the experiments that will continue with this reactor for the qualification of fuel for americium-loaded blankets (CCAm).

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244 boulevard Saint-Germain • 75007 Paris • Phone: +33 1 44 49 80 93 & +33 1 44 49 80 94

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