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^{O} 11

MAY 2017

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FOREWORD

In 1991, in a first law in France on the management of radioactive waste, Parliament, conscious of the specific character and novelty of the problems posed, entrusted a 15-year assessment of the state of advancement of research in this field to a National Assessment Board (CNE) made up of twelve independent and voluntary individuals. Under this law, CNE assessments give rise to an annual report to Parliament, submitted to the Parliamentary Office for Evaluation of Scientific and Technological Options (OPECST). This first Board (CNE 1) made a total of 13 reports.

In June 2006, a second law on the management of radioactive materials and waste confirmed the existence and role of the Board, which became the current CNE 2, and this document constitutes its 11th report to Parliament.

The Board evaluates current research and makes recommendations to aid decision-making by public authorities.

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This year, the Board held 11 one-day hearings, generally bringing together around sixty people representing all sector stakeholders. It also held 5 closed hearings and made several visits (see appendices I to V). For this 11th report, it has taken account of documents submitted up to 1 May 2017.

As in previous years, the Board has devoted a large part of its work to the analysis and evaluation of research and studies on the Cigéo project being conducted by the National Radioactive Waste Management Agency (Andra). In applying the provisions of the 2006 law, which expressly states that long-lived high- and medium-level radioactive waste should be stored in "deep geological strata", Andra is currently preparing to develop an underground repository in a clay formation situated at the boundaries of the Meuse and Haute-Marne departments. The Cigéo project is already very much under way, since the construction authorisation request could be submitted in 2018.

The 1991 and 2006 laws also recommend that research be conducted on the separation and then transmutation of long-lived radioactive elements present in this waste in order to reduce long-term radiotoxicity. The Board regularly monitors this research. This report reviews recent developments in this research and, in particular, the progress of the Astrid project, of which the Alternative Energies and Atomic Energy Commission (CEA) is the prime contractor. Astrid is intended to study the industrial feasibility of a new breed of fast-neutron power reactor, one of the means currently envisaged for the industrial transmutation of the elements in question.

The Board is also evaluating the problems related to the clean-up of contaminated sites, the recovery of stored waste and the dismantling of many nuclear installations. In particular, these activities will produce extremely large quantities of very low-level radioactive waste, which now warrants an optimal management solution.

In all countries confronted with management of waste downstream of the nuclear power cycle, the deep geological repository is considered the reference solution, as pointed out in an OPECST report in 2014. The Board, benefiting from input from its foreign members, provides an annual progress report on research carried out in the main countries with a nuclear industry.

After the writing of this report and prior to its presentation to the Parliamentary Office for Evaluation of Scientific and Technological Options (OPECST), the bodies responsible for assessing the safety of nuclear installations expressed strong reservations about the disposal of bituminous waste in Cigéo; they would advocate searching for a neutralization of the chemical reactivity of the waste packages, rather than their disposal as foreseen today by Andra. The choice depends on a scientific debate on the flammability of bituminous waste. This point requires further studies which will be evaluated by the Commission.

COMPOSITION OF THE NATIONAL ASSESSMENT BOARD

Jean-Claude DUPLESSY – Chairman of the National Assessment Board – Member of the French Academy of Sciences – Emeritus Research Director at the CNRS.

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Gilles PIJAUDIER-CABOT – Vice-chairman of the National Assessment Board – Professor of Civil Engineering, Director of ISA-BTP, LFC-R – Senior member of the Institut Universitaire de France.

Claes THEGERSTRÖM – Emeritus Chairman of SKB (Swedish company in charge of managing nuclear fuel and waste) – Member of the Royal Swedish Academy of Engineering Sciences.

SUMMARY - CONCLUSION

According to the provisions of the 2006 law, the long-term management of long-lived high- and intermediate-level waste (LLHLW & LLILW) has three components: its industrial storage, its disposal in geological repositories and the separation-transmutation of long-lived radioactive elements. In addition, the nuclear industry and the dismantling of decommissioned facilities produce waste of lower activity which requires specific management, in particular because of the large quantities produced. This report evaluates the progress of studies and research on these themes.

CIGÉO GEOLOGICAL REPOSITORY

The purpose of the Cigéo project is to build and operate a geological repository for LLHLW and LLILW radioactive waste. This repository should be created at a depth of 500 m in the 130 m-thick Callovo-Oxfordian (COx) argillite formation in Meuse - Haute-Marne. It has benefited from more than twenty years of studies and research carried out by Andra and the scientific community, notably in the underground laboratory at Bure.

The models developed to calculate the sizing of underground structures are convincing because of the care taken with the qualification of the thermo-hydro-mechanical (THM) behaviour of the argillite. Continuous improvement of the physico-chemical modelling of the repository has made it possible to refine quantification of water and gas flows both during the operation of Cigéo and after its closure.

Andra must now apply these models to finalise plans for Cigéo. Andra must clarify all the criteria used to validate the configuration and sizing of the structures envisaged.

Andra plans to submit the construction authorisation request (DAC) in 2018. The Board considers that the DAC should describe a design that is entirely feasible using existing technologies. Given the centuries-long duration of the project, the construction authorisation decision should also allow the implementation of improvements or technological developments without degrading safety. The Board draws attention to the specific nature of this construction authorisation decision.

In view of the complexity of the installation and its operation, the Board recommends the creation of a three-dimensional interactive digital mock-up of Cigéo, to train operators and finalise the procedures to be implemented. The project will begin with an industrial pilot phase, an essential step that will end with a complete demonstration of the control of Cigéo's operations. The industrial pilot phase should also be used to characterise, *in situ* and at full scale, the different constituent elements of the seals.

The Board does not consider it desirable to deliberately leave open each filled cell until the termination of a section of the repository. It recommends putting in place, progressively during the operation of Cigéo, a sealed isolation structure enabling each filled cell, thus isolated, to evolve in passive mode in relation to the geological environment. These cells would be monitored continuously.

Following submission of the DAC, a new estimate of the total cost of the repository should be established. The costs of the industrial pilot phase will be precisely determined. Given the need for liquidity throughout the project, the cost estimate for Cigéo must include the costs associated with its funding.

In order to implement the provisions of the 2016 law (reversibility review and citizen participation), the Board suggests the creation of a specific body to manage the evolution of Cigéo in a transparent manner. This body would ensure the annual monitoring of the operational master plan presented by Andra. Where developments in the repository appear desirable, this body would evaluate and assess them from the point of view of their consequences, based on the opinion of the competent authorities and commissions. These development projects would also be the subject of consultation between all stakeholders, including Andra, waste producers and citizens. Once these exchanges and deeper analyses have been carried out, this body would be responsible for proposing a draft decision to the State.

SEPARATION AND TRANSMUTATION

Astrid is a technological demonstrator of a fast-neutron reactor (FNR). It will supply electricity while sustainably recycling plutonium from the reprocessing of spent fuel. In accordance with the 2006 law, for which transmutation aimed at reducing the radiotoxicity of long-lived nuclear waste is a fundamental component, Astrid will evaluate the industrial feasibility of the transmutation of minor actinides. Astrid should also establish the conditions for increased plutonium consumption in order to stop production of nuclear electricity without wasting a large stock of plutonium.

It is in this context that the Board is following the developments of the Astrid project.

The Board notes that the R&D conducted under the detailed preliminary design for the gas energy conversion system (gas-ECS) has progressed well and should lead to bringing the Astrid nitrogen ECS version to the same level as the Astrid steam-water ECS version.

The implementation of Astrid involves the development of specific fuel assemblies that will have to be qualified; this qualification is scheduled for inclusion in the international collaboration framework.

The Board draws attention to the need for a long-term vision regarding changes in the country's nuclear power installations so that all stakeholders can implement adequate and optimised R&D. Indeed, the development and qualification of the FNR fuel cycle undertaken by the CEA, EDF and Areva will take several decades. In addition, large-scale robotised automation of new fuel fabrication and reprocessing plants will be necessary for the radiation protection of workers.

Today, considering the technical and radiation protection constraints, only the transmutation of americium is envisaged. Scientific and technical feasibility experiments are under way and require access to irradiation tools that are sparse in the world. The Board recommends that these studies and developments should also be part of a long-term policy.

CLEAN-UP AND WASTE MANAGEMENT

All basic nuclear facilities must be cleaned-up and dismantled after they are shut down. This procedure sometimes implies recovery of waste that is stored there. R&D to develop the equipment has been conducted for more than a decade and the actual waste recovery operations will continue for at least two decades. These operations continue under normal conditions.

The study carried out by the CEA and the producers provides important information concerning the storage of bitumens. Still to be verified, however, is the possibility of a waste package catching fire and the fire spreading to the entire sector. The Board recommends further experiments on this subject. Finally, it is necessary to re-evaluate alternatives to incineration of these bitumens based on updated data.

Very low-level waste (VLLW) represents a considerable volume and Andra expects that its Cires repository, even when extended to 900 000 m^3, will be full by 2030; a second repository must then be opened. Andra estimates that about half of VLLW has such low activity that it could be placed in simplified repositories. The Board has already recommended ensuring a consistent policy for the management of low-level waste, whether or not it originates from a nuclear industry. It considers that waste management policy should be based solely on toxicity studies . Isolation and containment times with regard to the biosphere must also be defined, taking account of society's expectations. The issues of a release threshold and low doses clearly underlie these questions.

All countries using nuclear energy consider that geological disposal of LLHLW-LLILW is the reference solution.

The most advanced European countries are Finland, which has begun construction of its granite repository at a depth of 430 m, and Sweden, where the procedures for authorisation to build a granite repository are expected to come to fruition in 2018.

In the United States, the procedure for granting authorisation for Yucca Mountain is restarting. Canada is looking for repository sites in geologically well-adapted areas that also benefit from a societal agreement.

Accelerator-Driven Systems (ADS) have been proposed as alternatives to fast reactors for the transmutation of actinides. Research continues internationally, principally in the framework of the European Myrrha project, piloted by the Belgian SCK●CEN and included in the ESFRI road map.

Most countries with a nuclear industry have already undertaken dismantling operations (reactors, fuel plants, reprocessing plants (etc.). They show that the technologies are available to perform dismantling while observing all the conventional and nuclear safety standards. The methodologies for estimating the costs have been validated.

The Board has analysed the conditions for the release of VLLW in different countries. The experience of countries with a release threshold shows that regulations, combined with strict procedures and controls, can ensure the protection of populations. The materials thus released can be reused without restriction, even in consumer goods. European and international harmonisation of approaches to the methods for release of VLLW would therefore seem desirable. The Board reiterates its recommendation for an in-depth reflection by France on this issue.

CHAPTER 1: CIGÉO

1.1 INTRODUCTION

The purpose of the Cigéo project, by application of the law of June 2006, is to design and build a reversible geological repository for LLHLW and LLILW radioactive waste as part of the French industrial waste management programme (PIGD). This repository should be created at a depth of 500 m in the 130 m-thick Callovo-Oxfordian (COx) argillite formation in Meuse - Haute-Marne. This project has emerged from studies and research carried out over a period of more than twenty years, especially in the underground laboratory at Bure, which demonstrated the excellent ability of the COx formation to isolate the waste and then sustainably confine the radionuclides it contains.

Assisted by the system project manager the Gaiya group (Technip-Ingérop), Andra, as prime contractor, carried out the initial preliminary design up until June 2015. After a project review commissioned by the Directorate-General for Energy and Climate (DGEC), the project entered the detailed preliminary design phase, which is expected to close with the submission of the construction authorisation request (DAC) scheduled for mid-2018.

To prepare the DAC submission, Andra has developed a safety options dossier (DOS) in two sections, one for Cigéo in operation and the other for Cigéo after closure. These documents are accompanied by a dossier of technical options for recoverability (DORec) and a proposed operational master plan.

The Board has analysed these documents (see Appendix VI) in the light of the R&D results. Although the documents prepared by Andra had been finalised before the enactment of the law of 25 July 2016 defining the reversibility of a deep geological repository, the Board has examined the consequences of reversibility for the next operational master plan for Cigéo that Andra will have to prepare.

Finally, the decree and order for the National Plan for Management of Radioactive Materials and Waste (PNGMDR) for the period 2016-2018 were published in the Official Journal in February 2017. The PNGMDR recommends studies on the evaluation of storage requirements for waste destined for Cigéo. It prescribes a working programme concerning the conditions of acceptability for the waste that will go into the repository. It also envisages expanding the inventory of waste to be stored, by including waste that would result from any possible change to the French nuclear power policy.

This chapter presents the Board's reflections on the progress made with the Cigéo project between June 2016 and April 2017.

1.2 THE WASTE INTENDED FOR CIGÉO

Acceptance of waste packages in storage must be carried out through an approval process. This process must be defined for the commissioning of Cigéo.

> *The Board reiterates its request to have more details on the package approval process and, in particular, the studies and research undertaken to obtain the full picture.*

The inventory of waste subject to reversible disposal in deep geological strata was specified in the decree issued on 23 February 2017 establishing the PNGMDR requirements. It includes a reference inventory and a reserve inventory (Appendix VII). Cigéo will be funded and built to receive the reference inventory. This inventory corresponds to the scenario of reprocessing all spent fuel from the existing fleet of nuclear plants. It takes into account the uncertainties over the lifetime of the fleet and the volumes that will be produced in the performance of waste recovery and conditioning projects.

The PNGMDR 2016-2018 devotes a large forward-looking section to the question of waste in the reserve inventory, which includes spent fuel in particular. It asks Andra to carry out work on the feasibility of the storage of nuclear material which would be reclassified as waste under the reserve inventory. Andra interprets these requests as Cigéo adaptability studies, and relies on generic knowledge.

Studies on the storage of spent fuel in Cigéo are being requested as a precautionary measure.

The Board considers that studies carried out to date on possible storage of spent fuel in clay are highly inadequate. Under the 2006 law, however, this deficiency has no consequence for the DAC, since this law only provides for the storage of ultimate waste in Cigéo; the spent fuel, which is not ultimate waste (article L542-1-1 of the Environment Code), is therefore excluded.

1.3 THE BOARD'S ANALYSIS OF THE SAFETY OPTIONS DOSSIER

Andra submitted the different documents making up the Cigéo safety options dossier (DOS) in June 2016 and the investigating procedure by the Nuclear Safety Authority (ASN) is under way. The objectives of the DOS and how it will be investigated by ASN are specified in Appendix VIII.

The 2006 law provides that the National Assessment Board for Research and Studies into the Management of Radioactive Materials and Waste (CNE) gives an opinion on the construction authorisation request (DAC) for the Cigéo deep geological reversible repository and sends it to the Parliamentary Office for Evaluation of Scientific and Technological Options (OPECST). To prepare the opinion that it must submit on the DAC, the Board has analysed, with regard to the R&D, Andra's proposals reported in the DOS. In its analysis, it has verified that the design of the installation reconciles, in accordance with the 2006 and 2016 laws, the particular characteristics of a deep reversible repository in which waste packages are introduced without intention, *a priori*, to remove them.

The Board notes that Andra's documents were drawn up before the law of 25 July 2016 was passed. Subsequent versions of these documents will need to be adjusted to comply with it.

The Board reiterates below the main conclusions of its analysis of the DOS, DORec and the operational master plan. The detailed development of this analysis is described in the complete documents published by the Board in November 2016 (See www.cne2.fr & Appendix VI). Details concerning more particularly the research that will lead to the DAC are included in section 1.5 of this report.

1.3.1 The methods of closure of the repository with regard to reversibility

The Board recalls that, according to the principles of the law, Cigéo must be designed as a robust, reversible repository intended ultimately to be closed to ensure long-term passive safety; closure is gradual, while guaranteeing the exercise of reversibility through the implementation of recoverability. Since Cigéo is intended to receive ultimate waste, the recovery of one or more packages can only be envisaged, on principle, in the case of a malfunction in the storage process. Indeed, in the operation of Cigéo, a deep geological repository, the large-scale recovery of waste cannot be considered a normal operation.

Reversibility is based on the ability to make timely decisions in the context of operation of the repository.

The Board does not consider it desirable to deliberately leave open each filled cell until the termination of a section of the repository. It recommends putting in place, progressively during the operation of Cigéo, a sealed isolation structure enabling each filled cell to evolve in passive mode in relation to the geological environment; these cells would be subject to a continuous monitoring programme contributing to the feedback required to exercise reversibility.

1.3.2 The role of the operational master plan

The reversibility reviews, planned at five-year intervals, will be an opportunity to decide whether or not to isolate the cells. Certainly, every stage in the progressive closure of Cigéo complicates the retrieval of a waste package but it increases passive safety. The operational master plan must therefore analyse in depth the methods used in this isolation strategy. Andra must demonstrate that the sealed isolation structures of the cells, which the Board recommends, fulfil all protection functions in incidental or accidental situations, in particular in the event of fire.

The operational master plan is a document that is intended to evolve with the operation of the repository. The Board requests that the methods for the progressive closure of the repository should be specified, with simultaneous consideration of the operational and long-term safety objectives. It recommends that each successive version of the operational master plan defines in an instructive way the objectives and guidelines of the project and takes into account the contingencies likely to affect its development.

In accordance with the 2016 law, Andra must define the methods that will allow it to take advantage of the feedback from the conclusions of a societal consultation.

The operational master plan thus becomes a structuring element in the governance of Cigéo.

1.3.3 Cigéo in operation

Cigéo will be a complex installation because of its size, its dual location at the surface and at depth, its centuries-long operating life, and the simultaneous work on construction, operation and closure.

In operation, the traceability of packages must be ensured, implying that the documentation should be accessible to operators at all times.

Control of the interface between the “works area” and the “operational area” will be essential to ensure safety in operation. The DAC must clarify the measures taken to ensure the safety of personnel in these two areas simultaneously and to analyse accidental situations. It should also address the integration of Cigéo's maintenance process, its impact on operational planning and especially on safety in the event of failure or shut-down for scheduled maintenance.

In view of the complexity of the installation and its operation, the Board recommends the creation of a three-dimensional interactive digital mock-up of Cigéo, to train operators and finalise the procedures to be implemented.

1.3.4 The industrial pilot phase

The industrial pilot phase provided for by the 2016 law is intended to demonstrate complete control of the industrial management of the repository. The smooth running of the industrial pilot phase will be the key element in taking the decision on final commissioning of the repository. Andra expects it to take around ten years.

The Board considers that the industrial pilot phase should last for as long as is necessary to validate the technical options and facilitate normal operation.

1.3.5 Demonstration of safety after closure

From the perspective of the safety assessment that will accompany the DAC, it seems important to size the installation in a robust manner. The demonstration of long-term safety relies mainly on modelling of the release and migration of radioactive chemical species through the components of the repository and the geological environment to the outlets and the biosphere.

The Board recommends clarifying the interlinking of the various models used to represent the phenomena at the various scales involved in the demonstration of safety. As for the evolution scenarios of the structure over the long term, it requests that a sensitivity study should be presented to assess the effect of the variability of the material parameters on the simulation results. It considers that the choice of the parameters associated with the environments altered by the presence of the structure should be better underpinned.

1.4 WORK UNDERTAKEN IN THE SCOPE OF THE DETAILED PRELIMINARY DESIGN

1.4.1 What content for the DAC?

As we approach the end of the detailed preliminary design phase and the resulting DAC, the level of document detail is queried. It should be sufficient to underpin the safety analysis while guaranteeing a certain adaptability to allow the project to evolve. The documents should take into account the centuries-long duration of the project, which will require the implementation of optimisations according to pre-defined methods.

The notions of the preliminary design are defined in Decree No. 93-1268 of 29 November 1993 relating to project management missions entrusted by public prime contractors to private-sector providers. The one related to infrastructures will be considered here:

“The purpose of preliminary design studies shall be: a) To confirm, in the light of the additional studies and knowledge acquired, the feasibility of the solution selected and to determine its main characteristics; b) To propose a topographical layout of the main structures; c) To propose, where appropriate, a breakdown into performance stages and to specify the duration of this performance; d) To enable the prime contractor to take or confirm the decision to carry out the project, to finalise the programme and to determine the necessary means, financial in particular; e) To estimate the

provisional cost of the work, distinguishing between expenditure by structural part and by type of work and indicating the uncertainty attached to it, taking into account the basis used for the estimate; f) To permit the establishment of the fixed remuneration package in accordance with the conditions laid down in the project management contract.

The preliminary design studies also include the preparation of the dossiers to be submitted, if necessary, for obtaining the building permit and other necessary administrative authorisations which fall within the competence of the prime contractor, and assistance to the prime contractor during the course of their appraisal. "

The examination of the construction authorisation request for the Aube repository (CSA) typically illustrates the degree of precision chosen for obtaining construction authorisation. This file includes the diagrams defining the layout of the structure (plans and sections), records describing the criteria used for sizing, the methods of verification of these criteria that contribute to demonstrating safety and finally a detailed description of the structure, which is necessary, for example, to initiate consultations with a view to its construction.

Such a level of detail does not apply directly to Cigéo, since the construction of the structure will extend over 150 years. Differences between the work as planned tomorrow and as it will be performed in more than one hundred years are inevitable. These differences would at least pave the way for improvement now envisaged by Andra and provide a basis for subsequent decision-making by the authorities.

The Board considers that the purpose of the construction authorisation request should be to describe a repository design that is feasible with current technologies, forming a reference configuration.

The ensuing construction authorisation decision should also allow the implementation of improvements or technological developments without degrading safety. The Board draws attention to the specific nature of this construction authorisation decision.

1.4.2 The link between research and the engineering

In the absence of a "target" reference solution for submission of the DAC, the studies and research undertaken by Andra do not distinguish between work on technological obstacles that have to be overcome to arrive at a reference solution and that relating to its possible improvement.

At one year from submission of the DAC, the research programme for Cigéo should be focused and show a clear link between the necessary knowledge requirements and the engineering options envisaged.

The sizing of a structure consists in proposing its precise geometry and the technical specifications that satisfy all the criteria envisaged, and more precisely the parameters that must be kept below a permissible threshold (deformability, maximum stresses, temperature, diffusive flux, etc.).

- The thresholds result either from research or from previous knowledge of the materials and systems used.
- The values to be compared with these thresholds are the result of calculation methods detailed in a sizing sheet (which must therefore form an integral part of the DAC). These are the loads.

The thresholds, like the loads, are weighted by safety coefficients. For the thresholds, these depend on the variability of the properties of the materials and the uncertainties of measurement. For the loads, these coefficients are provided by the uncertainties in execution, modelling and operation.

The level of knowledge required to set the thresholds or to determine the loads must take the uncertainties into account. Thus the thresholds are adjusted to the level of knowledge, and are sometimes very restrictive due to a lack of thorough knowledge and sometimes more refined, knowledge permitting.

Sizing is a recurrent process which generally starts from a preconceived fixed geometry, resulting for example from the engineer's experience. Sizing is then adapted in line with the default criteria. The result is a configuration that verifies all the criteria considered. In the case of Cigéo, we have not been sent this list of criteria. Often, only the final result is presented. It is precisely this absence of traceability in the sizing process that may prevent evaluation of the research that must be carried out in order to arrive at an acceptable sizing.

The Board recommends that all the criteria used to size the reference configuration, in the light of current knowledge, should be made explicit as soon as possible.

Moreover, the safety margins calculated for each criterion and for a given sizing will lend direction to a possible optimisation process. If, after optimisation, a variant degrades a criterion which had such a safety margin that it is not affected by a small decrease of it, and if the variant is less expensive, then it should be preferred. The next step is to estimate the cost of each variant, which can be used to rank them and to make decisions.

Research useful for DAC submission can therefore be divided into two avenues:

- research that promotes understanding of the operating conditions of the structure and hence reduce the uncertainties on the loads;
- research that promotes knowledge of the properties of the materials and therefore pushes back their limits of operation while maintaining the robustness required to ensure the safety of the repository.

Such an approach has not been presented to date.

Once the sizing methods for a reference solution have been established, the Board recommends systematically comparing the research undertaken with the load criteria and calculation models used to establish that reference solution.

1.4.3 Uncertainties

The description of the future behaviour of a repository site is necessarily subject to uncertainties. Such uncertainties are natural and must not, under any circumstances, preclude a project simply by their existence. On the contrary, it is advisable to integrate them into the definition of the project in order to arrive at a robust solution.

The uncertainties (detailed in Appendix IX) are mainly divided into two groups:

1) uncertainties in the measurements of the environmental properties;
2) uncertainties linked to models that do not exactly take account of the different processes affecting these environments.

In addition to these two groups of uncertainties, random events occur, for which it is difficult to determine the probabilities of occurrence.

Uncertainties therefore exist, but methodologies are available to manage them. Depending on the objective to be achieved, a set of scenarios is defined or an uncertainty analysis is carried out. This involves defining the modelling chain and identifying the most influential parameters in the overall chain, i.e. those that have a major impact. It is possible to focus on these parameters to estimate any uncertainties regarding the final responses and to deduce safety coefficients or optimisation opportunities.

The Board wants estimation of the uncertainties in the different data to be continued, links between the models constituting the modelling chain to be clearly described and the approach adopted to channel uncertainties along this chain and thereby deduce the safety coefficients to be made explicit.

1.5 RESEARCH UNDERTAKEN BY ANDRA FOR THE DAC

This research concerns the behaviour of the repository in operation (1.5.1) and after closure (1.5.2 cont.).

1.5.1 Sizing of the underground structures

The aim of Andra's research into the sizing of the underground structures is to ensure that these structures hold throughout the operational lifetime. For this, Andrea has a very important information base. It comes from twelve years of research in the underground laboratory where experiments were conducted in conditions close to those of Cigéo. Experiments in surface laboratories supplement this information, as well as discussions and interactions with teams of researchers from agencies in other countries and from numerous academic teams in France and around the world.

The approach is based on the aggregation of knowledge of different parts of the system.

Analysis of the natural behaviour of the rock makes it possible to obtain the key parameters for the construction of models of the mechanical behaviour of the clays. The argillite layer consists of two units: the silt-carbonate unit (the upper third of the layer) and the clay unit targeted for the structure. The main characteristics are high compressive strength, delayed deformation (especially in the clay unit) and a water swelling capacity, linked to the presence of smectites.

The behaviour of the clays around the structures is studied in order to understand and quantify the mechanisms of fracture creation when digging the structure modifies the state of local stress. Studies reveal two types of fracturing in the clay unit: an area of dense, connected fractures close to the wall (the excavation damaged zone or EDZ), and beyond that a zone of fractures not connected to one another. The orientation of the fractures is anisotropic, in relation to the direction of the main stresses in the environment. The associated convergence is also anisotropic. It develops in two stages: convergence at high speed during the first few months and then at a very low residual speed for several years. For the silt-clay unit, only unconnected fracturing develops, with very low convergence.

Finally, the studies are complemented by analysing the impact of excavation methods on fracturing. A significant effort has been put into the models used, with the comparison of different modelling concepts (semi-analytical models, equivalent homogeneous medium, discrete fractures).

The Board considers that the calculations presented are convincing. They should allow for testing the impact of different technological solutions on the durability of the structure.

The design choices are based on the use of standard methods, proven for this type of site. Nevertheless, long-term safety will require adjustments, in particular for convergence management: a combination on the one hand of the early blockage of convergence to avoid the creation of the

EDZ and, on the other hand, an initial free convergence phase that limits the residual stress on the walls. An interesting concept is proposed. It consists of associating the technique of bolting while progressing (to limit the creation of the EDZ at the front face) with the application of a compressible coating (compressible lining segments, filling) to allow for initial convergence and hence limiting the residual stress.

The Board recommends moving from the research phase to the engineering phase by applying the methods developed for the sizing of galleries and their linings.

1.5.2 THM sizing of the high-level areas

The aim of this research is to demonstrate that the heat released in the high-level (HL) cells during the thermal phase does not cause damage in the host rock. The verification is based on models, validated using experiments in underground or surface laboratories. The simulations must demonstrate that, in the selected scenarios, the stress field remains below the fracturing limit.

The approach is based initially on the use of simple models. These models are then made more complex to take into account a more elaborate phenomenology. Finally, sensitivity tests make it possible to rank the phenomena and the key parameters. Important work has thus been done to test the robustness of the simulation results in all uncertainties: geological variability, chosen law of behaviour, modelling choice.

These studies have been able to demonstrate the relevance of a 2D approach compared to the 3D approach, more costly in computational time. The choice of a thermo-poro-elastic behaviour model is also sufficient to analyse the behaviour of the far (inter-cell) field; the thermo-poro-elasto-plastic model is, however, essential to represent the behaviour in the immediate vicinity of the cell, taking into account the fracturing in the EDZ.

The simulations also give operational information on the scenarios to be taken into account: low impact of the spatial variability of the thermal power, interference with other phenomena (gas and chemical transients) that are not relevant because they are limited to the vicinity of the cells. Taking into account the history of loading of the cells is also negligible.

The Board appreciates the care taken in validating the THM behaviour models and their conditions of use. It considers that Andra has equipped itself with the necessary tools to represent the operation of high-level areas. Andra must now move from the model to sizing.

Andra wishes to use this calculation approach to arrive at a *prudent* model for DAC submission. These models can be fine-tuned and the margins narrowed, with data from the first few years of operation of the HL0 high-level area allowing improvements to be suggested for the HL1 and HL2 areas within the framework of Cigéo's flexibility.

1.5.3 Transient resaturation and the release of gas

Contact with the atmosphere during the construction and operation phases of Cigéo will cause partial desaturation of the cement and rock in the immediate environment of the structure. The research undertaken aims at evaluating the characteristic time and mechanisms of resaturation. It is also a question of quantifying changes in pressures and flows during this transient phase and until the return to equilibrium.

Andra is relying for this on an understanding of the processes acquired during dedicated experiments or on analogues. Transposition in time and space is carried out using validated numerical models, using experiments and through code intercomparison exercises. The approach has been applied to the various components of the repository and then at the scale of the geological layer.

From the initial hydrogeological model, the simulations represent the effect of excavation, ventilation during the operation phase, and post-closure resaturation. The calculations can be used to determine the extension of the unsaturated zone, the local pressures and the water flows during resaturation. Different phenomena have been taken into account: flow of water and gas, production of hydrogen in the repository, its dissolution and migration. Couplings with thermal (heat release by HL cells) and osmotic (LLILW with a high saline load) processes are also evaluated.

The Board notes that the continuous improvement of the physical description has made it possible to refine the quantification of the flows and the duration of the transients, without challenging the general conclusions of the 2005 dossier.

Despite the very low permeabilities, the large surface area of contact with the host rock in the cells favours the resaturation of the residual voids of the structure by the water from the Callovo-Oxfordian formation rather than the water that can pass through the seals. In the absence of other processes, resaturation would be effective after a period of the order of a few thousand years. However, the generation of a large quantity of hydrogen, resulting from corrosion of the steels, radiolysis of the water or even bacterial activity, substantially slows down resaturation. Resaturation times are therefore estimated at a hundred thousand years for the HL areas and several hundred thousand years for the LLILW areas. Technological voids are saturated last, since the resaturation is dominated by capillary effects.

On a smaller scale, seals not yet resaturated do not constitute a gas barrier. The outer part of the linings (compressible zone and connected fracture zone) in direct contact with the structures allows convective gas circulation inside Cigéo. This transfer limits the pressures in the structure, with a maximum, depending on the scenario envisaged, between 5 and 12 MPa within a few tens of thousands of years. These calculations highlight the high pressures. They were not taken into account in the THM behaviour of the structure and in the seals of the surface-to-bottom connections.

Andra has demonstrated its ability to quantify flows of water and gas during the transient phase of resaturation, taking into account the coupling of many processes.

The Board requests that it should be presented with the consequences of high gas pressures on the evolution of the repository components.

The healing of the connected fracture zone is related to the resaturation of the argillites. Gas pressures limit water ingress and therefore delay healing; conversely, the connection of the fractures facilitates convective transfer of gas and the ingress of water. The effect of hydrogen production on the healing of the connected fracture zone has not been quantified.

The Board recommends analysing healing and its effects on the THM behaviour in the resaturation phase or, failing that, to demonstrate that these effects are integrated into the package of scenarios studied.

1.5.4 Migration of radionuclides

The safety of the repository after closure ultimately depends on the very slow migration of the radionuclides in and outside of the repository. The first point has an impact on the co-storage of different categories of package in the same LLILW cell. Co-storage is studied with a view to optimisation of Cigéo. The second point concerns the migration of radionuclides during the transient phases of resaturation of the storage installations, then finally in the stationary phase in the saturated COx, from the packages to the outlets in the biosphere. The R&D conducted by Andra on these two subjects is detailed in Appendix X.

What prevents co-storage of packages is a possible interaction between radionuclides migrating from one package and degradation products of organic materials coming from another package. Over a decade, Andra has acquired data to simulate phenomena that would lead to a mixture of diffusion plumes and to exothermic chemical reactions or to the appearance of radioactive species that would migrate faster than in the absence of co-storage .

The Board considers that in order to decide on co-storage in the same cell, it is necessary to specify the values of the parameters related to the migration of the organic molecules and of the complexes they form with the radionuclides of interest in a heterogeneous environment representative of a degraded cell.

Regarding the migration of radionuclides outside of the repository, diffusion studies yield values on the characteristic flows and times. Andra has based the post-closure safety of Cigéo on calculations made on the saturated COx. It has undertaken to refine these predictions by taking into account the transient phases of resaturation. These phenomena are extremely complex.

The Board considers that Andra has demonstrated its ability to carry out calculations on the migration of radionuclides, explicitly taking into account the transient phases. These calculations could usefully be carried out for the DAC, in addition to the stationary calculations.

1.5.5 The role of the seals

The properties of the Callovo-Oxfordian formation play a central role in passive safety after closure. However, the surface-to-bottom connections collect all the water drained from the repository. It is indispensable to demonstrate that the transfer of the radioactivity by the surface-to-bottom connections and the galleries also meets the requirements for safety after closure.

Two design principles are used to limit flows through the surface-to-bottom connections. The grouping of the surface-to-bottom connections and the blind-corner design prevent the emergence of load gradients within the structure and hence avoid horizontal convective transfers. Subsequently, the seals of the surface-to-bottom connections are located within the silt-carbonate unit in order to avoid the presence of a connected fractured zone linked to excavation which could bypass the sealing.

Andra has started the experimental demonstration of the effectiveness of the seals: installation of a clay plug, measurement of the resaturation times and permeability, seal anchorage test.

Due to the saturation times and flows that remain very low by construction, the sealing efficiency was initially demonstrated through a series of internationally conducted trials to characterise the principles of operation of a seal and to determine its properties. An experimental phase of *in situ* placement of the constituent elements of the seal at full scale will then be necessary; it will characterise the technical implementation methods. Finally, the properties of these constituent elements feed into the hydrogeological models from which the respective flows of the different repository outlets are calculated.

The Board recommends that the experimental characterisation of the various constituent elements of the seals in an underground environment should continue, at full scale, during the industrial pilot phase.

The calculations are based on a normal evolution scenario, in a realistic hydrogeological scenario and with the main control parameters assigned to the measured or estimated average values. The calculations give the quantification of the flows through all the reference surfaces and in particular through the seals. The reference calculations are supplemented by a sensitivity study: construction of adverse evolution scenarios and "what if" scenarios including a failure of the clay core of the seals.

Current Andra calculations along with sensitivity studies show that transfers through the surface-to-bottom connections are always very low. In the worst case, the maximum dose rate by the surface-to-bottom connection at 100,000 years is less than 1 µSv/year.

The Board approves the methodology for calculating radionuclide flows at the outlets.

The Board recommends that Andra should verify the effectiveness of the seals during the transient periods.

1.6 FUNDING, METHODOLOGY AND CONTRACT ENGINEERING OF THE PROJECT

One year after the order published in the Official Journal in January 2016 on the cost of Cigéo, some critical elements remain in estimating the expenses related to this exceptional industrial project. As a reminder, the order fixes a "target cost" of €25bn, under the economic conditions of 31 December 2011. This cost is estimated over a 140-year period starting in 2016, namely 10 years of design and construction of the first structures, 10 years of pilot phase, 110 years of operation and progressive development of the repository and 10 years for closure.

It should be noted that the cost decided upon constitutes a reference for the establishment of provisions for waste producers. Legislation states that the fixing of this cost constitutes neither an authorisation for the project, nor a decision on its safety level. It encourages Andra to remain active on the main optimisation avenues identified, in compliance with the safety requirements set by the ASN. This estimate is therefore intended to evolve.

The provisions of the three producers for the management of LLHLW-LLILW waste are respectively: €7bn for EDF, €765m for Areva, and €1.9bn for the CEA. The total represents an increase of approximately €2bn in the provision for Cigéo in accordance with the order evaluating these costs. The coverage rates of the stakeholders are presented as being close to 100%. EDF has a provision of €8.25bn for the long-term management of radioactive waste including, outside of Cigéo, €250m for graphite and €1bn for other low-level waste. Areva's provisions are evaluated by sector based on unit management costs and waste volumes. The cost taken into account is calculated per m^3 of storage packages. It should be noted that all these evaluations, presented in 2017, refer to the end of 2015. It is understood, therefore, that no changes have been made in the last two years.

For a project spanning 140 years, discount rates and inflation are crucial because they condition the pace of future spending. These rates are chosen by each of the operators within limits fixed by the State.

The Board recommends ensuring that the discount rates are chosen to best preserve intergenerational equity.

Several critical elements emerge in terms of changes in the predicted cost of Cigéo:

- The logic remains relatively “short-term”: the dynamic evolution of the project remains that envisaged in 2014, with some detail on the industrial pilot phase and very little on subsequent sectors, the progressive closure and the rejuvenation of the installations;
- All uncertainties, defined as opportunities and risks that can positively or negatively affect the cost and funding of the project, as well as the uncertainties surrounding estimates are totally absent; this also applies to possible changes in nuclear power policy.

Over such a long horizon, it is obvious that quantifying uncertain events and assigning them a probability is a complex exercise. Moreover, this was one of the differences between Andra and the producers before the publication of the order. Nevertheless, the current logic will have to evolve in order to arrive at a precise and shared vision of the first investments. The cost of the industrial pilot phase is currently set at €6bn.

The Board requests that the costs associated with the industrial pilot phase be reassessed with precision.

The evolving aspect of the costs, while provided for in the order, remains difficult to grasp. The only dynamic element provided to the Board is the evaluation of the marginal costs of the LLILW and LLHLW cells according to the storage scenario up to 2143, distributed along a *trajectory presented as unique* (and therefore without uncertainty). Work in progress, to be completed in March 2017, will improve the optimisation scenarios envisaged for the DAC submission horizon.

With the submission of the DAC, a new estimate of the total cost of the repository should be established taking into account the optimisation paths. This estimate must include costs related to the funding of Cigéo.

The Board notes the absence of a strategy for contract engineering in the industrial pilot phase. The contractual arrangements for the project management of the funicular railway have been agreed and commit the parties to a firm price. It seems that the rest of the work will be carried out within the framework of contracts awarded according to the classic public procurement format. As with any infrastructure project, and particularly in the case of underground works, uncertainty as to the cost of the work “as carried out” cannot be overlooked. It is important, therefore, to contain these uncertainties as far as possible by defining the exact cost of construction, linked to construction methods, the feasibility of which will be ensured to the best of current knowledge.

The Board considers that contract engineering is involved in the evaluation of the costs of the repository and asks Andra to integrate this dimension into its studies as from now.

1.7 GOVERNANCE

1.7.1 Governance of Cigéo

As with the creation of all basic nuclear installations, Andra will have to conduct the Cigéo project in compliance with the requirements of the construction authorisation decree. These requirements, listed in the decree of 2 November 2007, require the beneficiary of the construction authorisation decree to describe precisely the technical characteristics of the installation, its operating principles, the operations to be carried out there and the various phases of this implementation, any "significant change" only, in principle, being made in accordance with the original procedure.

However, it is clear that an installation as innovative as Cigéo, designed to be operated for at least a century, will undergo technical evolutions that will require continuous adaptation of the original specifications. Ensuring the reversibility of the repository, imposed by the law of 25 July 2016, is also one of the major constraints for which Andra and the State authorities will have to devise new provisions.

The law on reversibility has defined it as an essential principle of the Cigéo project, which gives an evolving aspect to the decisions making up this structure: they are reviewed regularly (every five years), so that the chosen configurations can be subject to change: the public authorities may decide to continue the project, to pause the project, or to go back (on the basis of the possibility of recoverability of the packages which must be facilitated by the design of the structure).

However, this reversibility is a more general principle: from a concrete and practical point of view, it also allows local and particular evolutions based on the fact that the initial configuration of the project must not be fixed for a structure with an implementation period of more than a century and which will pass through various phases. This is due to the fact that, from a technical point of view, improvements and optimisations may appear necessary in the light of the progress in knowledge and taking into account feedback, not to mention the general evolution of circumstances or fluctuations in energy policy. It is therefore appropriate that the project retains a dimension of flexibility and guarantees the possibility of relevant modifications to the configuration of the structure.

Can the governance of Andra, as it is currently organised, make it possible to respond to the evolving nature of the project?

In 2006, OPECST, in a report on radioactive waste management, noted that in order to meet its missions over the next few decades, Andra should benefit from: “a tidying up of its status”. This adaptation of Andra's governance principles will be all the more difficult to implement because, as required by the 2016 law, citizens' participation must be guaranteed throughout the lifetime of the repository.

To tackle these various imperatives, the Board suggests the creation of a specific body to manage Cigéo's evolution in a transparent manner.

This body would ensure the annual monitoring of the operational master plan presented by Andra.

Where developments in the repository appear desirable, this body would evaluate and assess them from the point of view of their consequences, based on the opinion of the competent authorities and commissions.

These development projects would also be the subject of a collaborative assessment by all stakeholders, including Andra, waste producers ***and citizens****. The debate would at the same time deepen the subject, inform all stakeholders, and associate them with the definition of desirable developments.*

Once these exchanges and deeper analyses have been carried out, this body would be responsible for proposing a draft decision to the State.

1.7.2 Governance of the downstream cycle

The adoption and implementation of these proposals could give the governance of Cigéo the flexibility that would appear necessary, while allowing the “societal vigilance” demanded by the National Association of Local Information Commissions (ANCLi) in a recent report.

However, radioactive waste management is only one of the problems associated with the downstream nuclear fuel cycle. Thus the dismantling of old nuclear installations, the reprocessing or storage of spent fuel, the transmutation of actinides, the development of a fast-neutron reactor sector, etc. will also have to be taken into account.

The Board considers that the issues related to the downstream governance of the nuclear fuel cycle are all issues that could be effectively addressed by a single decision-making body capable of providing a coherent, summarised vision of what will constitute one of the most important projects that our country, like all countries that have developed a nuclear industry, will have to implement.

CHAPTER 2:

SEPARATION AND TRANSMUTATION

The 2006 law provides that long-lived high-level and intermediate-level waste (LLHLW and LLILW) coming mainly from the current fleet made up of PWR-type thermal reactors, will be stored in deep geological strata. The Cigéo project meets this requirement. This law also provides that research should be undertaken to study the industrial feasibility of separation and transmutation of long-lived radioactive elements. Since the recovery of vitrified waste is impossible, only waste from a possible future fleet is concerned. The corresponding studies and research should be carried out in relation to those conducted with the new generations of nuclear reactors. It is now established that, among long-lived radioactive elements, only the minor actinides could be transmuted in fast-neutron reactors (FNRs).

> *The geological storage of LLILW and LLHLW and research on the transmutation of minor actinides are two fundamental aspects of the 2006 law. The R&D must be carried out on both fronts to preserve the balance desired by the legislator. It is in this context that the Board is following the developments of the Astrid project.*

2.1 THE ASTRID PROJECT

2.1.1 Context

The 2006 law has entrusted the CEA with the task of producing a technological demonstrator of the Generation IV fast-neutron reactor, Astrid, which should provide the necessary elements for an industrial choice to deploy an FNR fleet. It could take place in the second half of the century (see Appendix XI). It should be remembered that MOX fuel, rich in plutonium (Pu), resulting from UOX fuel, can only be used once in PWRs due to changes in the isotopic composition of the plutonium. For an FNR, the MOX can be recycled endlessly, whatever the plutonium isotopes, hence the choice of the objectives for Astrid, which must therefore:

- supply electricity with a safety level at least equivalent to that of the EPR and incorporate all the recommendations of the IAEA and the ASN resulting from the analysis of the Fukushima accident;
- validate the sustainable use of plutonium from the reprocessing of PWR MOX and later FNR MOX (300 tonnes of plutonium contained in MOX in pools in 2040) and depleted uranium from the enrichment of natural uranium (approximately 450,000 tonnes of depleted uranium by 2040);
- assess the feasibility of the transmutation of minor actinides;
- establish the conditions for increased plutonium consumption in order to stop production of nuclear electricity, as required, without wasting a large stock of plutonium.

These various requirements are the subject of much research and technological development. This report describes the latest developments as of 2017.

2.2 PARTNERSHIPS AND COLLABORATIONS

Since September 2010, the CEA has been prime contractor for the design studies of the Astrid project, and has finalised the initial preliminary design of the reactor with a water-steam energy conversion system (ECS). During the detailed preliminary design phase, which will cover the period 2016-2019, the CEA will compile a dossier with a nitrogen ECS. This would avoid any risk of contact between sodium and water, a risk that exists for the few sodium FNRs currently in use in the world.

The CEA has set up a large consortium to develop the large amount of knowledge required to execute Astrid. Six hundred people contribute to the project. Strategic and operational steering are provided by the CEA in close cooperation with EDF and Areva. In addition, Alstom, Japan Atomic Energy Agency (JAEA), Mitsubishi Heavy Industry (MHI), Rolls Royce, Toshiba and Bouygues, among others, are providing their support in design, assistance and R&D. For the latter, the CEA is also the pilot of the recent European partnership ARDECo (Astrid R&D European COoperation), which brings together a number of foreign universities, the CNRS and EDF-R&D. To this is added the Generation IV International Forum partnership, European networks and even the European Commission.

The latest bilateral agreements with Japan (JAEA, MHI, etc.) go beyond detailed preliminary design studies and make Japan a major partner. Japan plays a scientific, technical and financial role.

The absence in France of a fast-neutron research reactor, since the closure of Phenix (1 February 2010), severely penalises research into future nuclear power. Specific international co-operation to qualify components of Astrid has therefore been established with Russia, for irradiations in Bor-60 and BN-600, with India for the physics of serious accidents, with China for experiments in the Chinese Experimental Fast Reactor (CEFR), with the United States in the field of neutronics and power transients of the core or with South Korea for access to supercritical CO_2 loops, which would have a better yield than nitrogen for energy conversion.

2.2.1 Time line

The choice of exploring new technological options in a complete break with existing or under-construction sodium FNRs, consolidating them in facilities still under construction and searching for a funding plan for the Astrid reactor, have imposed a new time line on the Astrid programme. It is currently being revised, with the introduction of a new governance entity from 2020 that will include the main participants.

The time line is shown with certainty until 2019, the date of the end of the detailed preliminary design, with two milestones:

1) end of 2017: submission of a progress report to the authorities (choice of technological options, Astrid gas ECS dossier, costs, Astrid gas safety options dossier);
2) end of 2019: submission of the files required to decide on the continuation of the project (definition dossier, technical needs specifications). The continuation of the programme, including organisational methods and funding, will then be decided.

The CEA, EDF, Areva and the supervisory authorities envisage, after the detailed preliminary design, a design consolidation phase for Astrid of 4 years, a development phase and a construction phase. These phases would proceed under the responsibility of the new governance.

Thus the decision to build Astrid, which would initiate the development phase of a future FNR fleet, would not be taken until 2024. This chronology assumes that the safety appraisal leading to the construction authorisation decree would have been carried out during the consolidation phase. The construction of Astrid would take from 8 to 10 years after the construction authorisation decree, with start-up of the reactor towards 2040.

The dates for a start-up of the Astrid fuel cycle facilities will be consistent with the reactor construction time line. The end of the initial preliminary design for the fuel fabrication plant is planned for the end of 2019. The detailed preliminary design for this plant will be performed during the design consolidation phase for the reactor. Currently, three options are envisaged for the fuel fabrication plant: an establishment in La Hague or Marcoule (Melox), or construction in Japan under the partnership established for Astrid. The qualification of the assemblies will be in Russia, in BN 600. It is programmed from 2016 to 2025.

2.2.2 Detailed preliminary design 2016-2019

The year 2016 was devoted to the selection of the expected characteristics of Astrid, to define its configuration for the detailed preliminary design. An extensive amount of work has made it possible to clarify or improve this configuration in order to reinforce safety and operability. This concerns the handling of the assemblies, the design and installation of the sodium-nitrogen exchangers and the two nitrogen lines (15 MPa, 510°C) to the electric generating machine room, the hot cell for handling spent fuel, the coupling of the core and the corium collector in the event of melting of the fuel assemblies, the circuits for removal of residual power and the optimisation of the general architecture in terms of structure and cost.

Significant advances have been made in particular concerning the sodium-nitrogen heat exchanger and the ECS circuits. Its integration into the Astrid design has a strong impact on the layout of buildings attached to the reactor. The nitrogen ECS decreases the thermodynamic efficiency by a few percent compared to a water-steam ECS. Studies are under way to seek to reduce this negative impact. There are now no blocking points in the integration of the nitrogen ECS. Qualification of the ECS and all Astrid components has yet to be obtained using the dedicated technology platforms, available or under construction, which are Diademo, Cheops, Giseh, Papirus, Masurca and Plinus2 (Appendix XI).

Areva-NP is studying in particular the inclusion of the “serious accident” which would lead to meltdown of the core. This involves the development of several devices of which the more innovative are those for the transfer of corium to the recovery zone and the corium collector itself, which, in addition to its performance at extreme temperatures, must be compatible with the presence of sodium. The first device benefits from Russian and Japanese feedback on metallic protections, in place of protection with refractory oxides that was first considered. The development of a dual system for evacuation of residual power, a cooling system for the reactor in shut-down and a cooling system by radiation through the vessels is supported by the CEA, JAEA, MHI and Areva-NP. Finally, Astrid's earthquake resistance is ensured by anchoring the reactor on an independent slab. All these devices contribute to preventing the release of radioactivity to the environment, whatever the circumstances.

Another area in which Astrid is innovative is in the instrumentation (core and ex-core) providing access to continuous monitoring. The very important advances achieved with Phénix were the basis for the development of neutron, thermal, hydraulic, sealing and geometry measurement systems. These are major developments in terms of safety for the detection of leakage of tertiary fluid in the sodium for a water-steam ECS or for a nitrogen ECS or for the leakage of sodium in pipes or components.

The Board considers that R&D conducted under the detailed preliminary design, notably on the nitrogen ECS (energy conversion system), is progressing well. Its objective is to bring the nitrogen ECS version of Astrid to the same level as the water-steam ECS version.

2.3 THE CYCLE OF PLUTONIUM-RICH FUEL

2.3.1 Introduction

The fabrication of the first Astrid core (25 t of MOX and 15 t of UOX) will use 5 t of plutonium from PWR UOX. Areva has shown that PWR MOX fuel assemblies can be reprocessed by making modifications to the Purex process used in La Hague. However, these modifications cannot be applied to future industrial reprocessing of all spent MOX in a mixed EPR - FNR fleet. The high content of plutonium and minor actinides in the spent MOX fuel from Astrid or future FNRs will require not only the development of a new process, but also new facilities for fuel fabrication and reprocessing of spent fuel. To move towards an FNR fleet, it will be necessary to demonstrate that

Astrid can use plutonium from PWR MOX and then spent FNR MOX. A decision to introduce FNRs to the fleet can then be taken after validation of the data provided by Astrid. These developments will lead to the implementation of a new fuel cycle.

The following considerations are based on the hypothesis of a mixed EPR and FNR fleet, which, evolving in stages, would lead to an FNR fleet according to a scenario, given in Appendix XI, by the CEA, EDF and Areva. Two plutonium-rich spent fuels (EPR MOX and FNR MOX) would then be reprocessed (see Appendix XI). The former is in strategic industrial reserve in the pools at La Hague. It will have to be recovered, to bring it for reprocessing.

2.3.2 Stock of plutonium - Storage of PWR MOX

Assuming the gradual development of an industrial fleet of FNRs, the first FNR after Astrid could be launched around 2060-65. In order to obtain the necessary plutonium, this will require reprocessing of the spent PWR MOX assemblies that are currently under water storage at La Hague (1724 tonnes at the end of 2015 containing 103 tonnes of Pu); this is a 6% PWR MOX. The first assemblies were put in the pool around 2000 and today the strategic reserve of MOX is increasing by 120 t/year, i.e. an increase in the plutonium reserve of about 7 tonnes per year, in addition to the 103 t available in 2015.

In the pool, the fuel assemblies are maintained at 50°C and the pressure in the rods is 8 MPa, conditions under which no evolution is expected. However, on recovery from the pools and dry transport to the reprocessing plant, the MOX temperature and internal pressure increase to approximately 400°C and 18 MPa for one month. The CEA and Areva have studied the ensuing modifications to the MOX and fuel sheaths and are testing the mechanical strength of the assemblies.

The CEA studies have shown that there is no technical difficulty associated with the long-term storage of spent PWR MOX fuel under water (see Appendix XI).

2.3.3 MOX reprocessing

The spent PWR MOX fuel assemblies differ from the spent PWR UOX fuel assemblies (industrially reprocessed at La Hague) only by the content and nature of the plutonium, which is richer in ^{238}Pu, and by a higher content of minor actinides. The structure of spent FNR MOX fuel assemblies will be very different (stainless-steel sheaths, hexagonal stainless-steel tube) and the spent fuel will be even richer in plutonium. The plant at La Hague is not suitable for the industrial reprocessing of such fuels.

The multi-recycling of nuclear material from an FNR fuel will require the conversion of plutonium and uranium into solid mixed oxide for the fabrication of fresh FNR fuel assemblies. Exploratory R&D, which is expected to gain momentum, is being developed by the CEA, EDF and Areva to prepare for future reprocessing. This involves improving, or even completely changing existing technologies and processes; this concerns:

- cutting rather than shearing the MOX assemblies to recover the spent fuel;
- melting of decontaminated sheath pieces, structural members and cutting fines for conditioning;
- continuous dissolving of oxide fuel without loss of plutonium;
- a new process for extracting Pu(IV) and U(VI) without a redox step;
- new processes for cold co-conversion of uranium and plutonium and fabrication of oxide pellets;
- the vitrification of fission products and minor actinides without prior calcination of the nitrates produced when dissolving the oxide;
- on-line operation with automation/robotics.

Based on promising trials on all these points, the CEA is considering the creation of a demonstration workshop for recycling technologies coupled with the Astrid reactor.

The replacement of the Purex process with a new process is a key point in the future reprocessing of MOX (see Appendix XI). In Purex, separation of plutonium is achieved by oxidation-reduction operations involving chemical reagents inducing a “chemical risk”. Increasing the concentration of plutonium and consequently the reagents would increase this risk. Consequently, the CEA, EDF and Areva have undertaken studies, at the laboratory stage for now, to achieve this separation. They aim to replace the extracting agent in the organic phase of the Purex process with another agent which then makes it possible to simply experiment with the acidity of the aqueous solutions in order to effect the separations. Initial results show that the separation of plutonium and uranium from fission products and minor actinides is possible. The fission products remain in the dissolved solution of the spent fuel. This innovative, even revolutionary, approach will require considerable R&D effort. It will take time to arrive at an industrial process. The R&D will be long, but essential to prepare the future fuel cycle.

The Board considers that the programme now started by the CEA, EDF and Areva to ensure that the fuel cycle associated with the FNRs will be industrially ready in time is justified. Indeed, the challenge of implementing the entire Astrid programme and the operation of future FNRs, which requires access to the plutonium contained in the PWR MOX and later in the spent FNR MOX, is considerable . The development and qualification of a new cycle will take decades. This is consistent with the hypothesis of introducing an FNR to the fleet around 2060.

2.3.4 From Astrid to a fleet of FNRs

In the deployment of an FNR fleet, EPRs and FNRs will coexist for a long time, just like the reactors in the current fleet will coexist with EPRs. Furthermore, it is essential that France can continue to reprocess spent fuel. In other words, there must be continuity in the fleets and it is imperative to preserve the knowledge and the plants that will make it possible to operate a closed cycle (Appendix XI).

The Board draws attention to the need for a responsible and long-term vision of the evolution of the country's nuclear power installations so that all stakeholders can put in place optimised R&D in support of this fleet.

2.4 SEPARATION OF AMERICIUM AND TRANSMUTATION

2.4.1 Choice of americium

FNRs allow the transmutation of minor actinides with an efficiency that depends on the element under consideration. It should be remembered that long-lived fission products are not transmutable. Today, considering the technical and radiation protection constraints, only the transmutation of americium is envisaged. Transmutation of neptunium could be envisaged, but that of curium would lead to the implementation of almost insurmountable radiation protection measures (Appendix XII). The elimination of ^{241}Am from future vitrified waste packages would significantly reduce their thermal load, which would result in a reduction in the high-level waste storage capacity from 150-170 to 20 m^2/TWhe (deduced from studies for Cigéo).

It should be noted that this involves reprocessing spent FNR MOX assemblies beyond the extraction of uranium and plutonium, in order to extract the americium. This reprocessing needs to be fast (5 years) to limit continuous formation of ^{241}Am from ^{241}Pu present in the spent FNR MOX.

The transmutation can be carried out according to two concepts:

- heterogeneous with a fuel ($UAmO_2$) placed in 10-20% americium-bearing blankets (AmBB) placed at the periphery of the core;
- homogeneous with a $UPuAmO_2$ fuel with 1 to 2% americium.

As designed, Astrid's core can accept 3 assemblies of $UAmO_2$ or 1 assembly of $UPuAmO_2$ without affecting the neutronics.

2.4.2 Obtaining the americium

Over the past decade, the CEA has developed a strategy for the fabrication of these assemblies, which begins with the implementation of the EXAm process (for EXtraction of Americium). After separation of uranium and plutonium, this allows the specific isolation of americium. The fabrication of AmBBs, their reprocessing after irradiation and then their re-fabrication, pass through many stages:

- implementation of the new extraction process combining uranium and plutonium,
- implementation of the EXAm process,
- conversion of Am oxalate to AmO_2, fabrication of $UAmO_2$ pellets for the AmBBs,
- then, after irradiation, reprocessing of the irradiated AmBBs and re-fabrication of the AmBBs, since the transmutation yield of the americium will be only a few tenths of a percent per pass through the reactor.

The “full EXAm” experiment in Atalante has been in progress since 2010 to recover 2 to 3 g Am in the form of $UAmO_2$ in order to make 4 mini-rods foreshadowing the AmBBs which will be irradiated at the ATR (USA) after 2019.

The fabrication of $UAmO_2$ oxide with a high density (> 94% of the theoretical density) and a controlled porosity (15%) is controlled by the CEA.

2.4.3 Irradiation of americium

Numerous experiments have been carried out on the irradiation of various oxide candidates to become transmutation fuels of the minor actinides in homogeneous mode (FNR fuel, U, Pu, low content of minor actinides) or heterogeneous (FNR fuel, U, minor actinides). In addition, the European scientific community, led by SCK-CEN in Belgium, will study the feasibility of transmutation in an accelerator-driven sub-critical reactor (ADS) using a specific fuel – Pu, minor actinides, high content of minor actinides. This approach would also require the reprocessing of irradiated fuel to obtain the isolated elements for transmutation.

All experiments carried out on americium-bearing fuels aim to test their suitability for irradiation (amorphisation, restructuring) and helium release for fuels with a high americium content. Today only the transmutation of americium, in AmBB ($UAmO_2$) fuel with 10-20% Am, is envisaged in France.

R&D on transmutation fuels is carried out in numerous Euratom programmes (Fairfuels, Pelgrimm, etc.) and in the numerous links that the CEA has forged (ITU, DOE, etc.).

Assessment of the industrial feasibility of americium transmutation will require modification of the fuel fabrication plant and the reprocessing plant built to supply Astrid: the fuel handling chain and the hot cell will be affected, as will the configuration of Astrid.

The CEA has mastered both the fabrication and the post-irradiation examination of the rods used during recent decades for the study of the transmutation of minor actinides. Astrid's design allows assemblies to be accommodated, but no concept of irradiation assembly has been qualified to date. Moreover, the time required to go from the fabrication of the assemblies to their examination after irradiation is about

30 years. In addition, separation and transmutation studies require access to irradiation tools now based on partnerships that are rare around the world. The Board recommends that these studies receive coherent support and form part of a long-term policy.

CHAPTER 3:

CLEAN-UP AND WASTE RECOVERY

All basic nuclear facilities must be cleaned-up and dismantled (C&D) after they are shut down. These operations are complicated because they often involve heavy, nuclear-based technologies that are subject to regulatory constraints. They therefore result in large investments that are programmed over decades. They must be preceded or accompanied by studies on the radiological status of the site and the objective to be achieved in the deconstruction and recovery of the soil, the inventory of the waste that will be produced, its characteristics and conditioning and, if the facility houses waste in storage, recovery and conditioning of this waste. Waste recovery operations require significant R&D beforehand, because not all the waste stored may benefit from conditioning already used in the nuclear industry (see Appendix XIII).

The specific R&D associated with waste recovery falls within the broader framework of waste treatment and conditioning (T&C). This research is conducted and supported by waste producers. It is developed according to three principles: recycle if possible, reduce the volume of waste, and reduce its chemical reactivity. The optimisation of the T&C to produce primary waste packages is based on the criteria that characterise a good package, which the Board assessed in report No. 10.

The fabrication of a new primary package complies with waste packaging specifications, or with a conditioning standard submitted to the ASN if there are no specifications, which is the case for almost all waste recovery and conditioning. If production authorisation is given, producers consider that the package is eligible for storage. The R&D is all the more sustained because the waste presents properties that raise fears of behaviour incompatible with storage safety options. Packages are therefore always produced in compliance with a standard covering three requirements: production specifications, control and traceability including the methodology and the characteristics to be safeguarded. During the R&D, Andra is consulted by the waste producers and always gives the ASN an opinion on the suitability of the package's characteristics for storage. The ASN ultimately gives the regulatory authorisations. In fact, exchanges between producers, Andra and the ASN are continuous and start at the beginning of R&D.

Waste packages resulting from clean-up and waste recovery operations that cannot be disposed of in a very low-level waste (VLLW) or low- and intermediate-level short-lived waste (LILW-SL) repository are put in storage. This is the case for low-level long-lived waste (LLLW) and long-lived intermediate-level waste (LLILW), which are awaiting the opening of Cigéo. Dedicated storage facilities can be built if necessary.

In Appendix XIII, the Board details the clean-up and waste recovery operations undertaken by the CEA and Areva.

3.1 C&D AND WASTE RECOVERY AND CONDITIONING OPERATIONS: THE STATE OF PLAY

The CEA and Areva show the fastest possible C&D and waste recovery strategies within the constraints outlined in the introduction and the search for multicriteria optimisation. Budgetary constraints have been particularly strong in recent years. The major projects for the CEA and Areva are the workshops of the former reprocessing plants UP1 (Marcoule) and UP2 400 (La Hague) and, for Areva, also the enrichment plant for natural uranium GB1 (Pierrelatte). The reprocessing operations generate a wide range of waste, especially since UP1 and UP2 400 did not have all LLILW conditioned on-line at the time they were in operation. EDF is also showing the fastest possible strategy for natural uranium graphite gas reactors (UNGGs) and the first pressurised water reactors (PWRs), but EDF has had to review the dismantling technique for the UNGG reactors and their dismantling will be spread out over a longer period than expected. Andra is undertaking remediation of orphan sites polluted with radium.

The scale of the work in progress and still to come has led the CEA to set up a new governance of C&D and waste recovery by creating a Dismantling directorate for civil plants. This new directorate should define priority projects and strengthen their implementation. Areva and EDF have also reorganised dismantling governance.

The dismantling waste to come is mainly VLLW (concrete, scrap metal, rubble). It represents around 60% of all VLLW that will be produced (operation and dismantling). The LILW-SL waste (equipment close to the core of reactors or clean-up waste) represents 40% of all LILW-SL waste produced. Finally, LLLW waste (graphite stacks) and LLILW (metal parts near cores) account for 30% and 10% respectively of all waste in these categories.

The quantities of waste to be packaged and the number of primary packages produced from the operations that have and will take place at the different CEA and Areva centres are detailed in Appendix XIII. The diversity of waste and storage conditions means that all cases of C&D and waste recovery are special cases, both by the means to be implemented and by the duration of the operations to be programmed. Their only common characteristic is the extent of the resources to be developed.

The Board notes that the operations undertaken by the CEA and Areva will take place over several decades. R&D to develop the equipment for waste recovery has been conducted for more than a decade and the actual waste recovery operations, as programmed, will continue for at least two decades. The Board also notes that recovery and conditioning operations are continuing normally and that large items of equipment for waste recovery are going to be operational.

The Board wishes to know the progress and the new programme for EDF's operations.

3.2 R&D FOR CONDITIONING OF NEW WASTE

The R&D covers 6 classes of waste typical of waste recovery and conditioning operations and containing the following components: graphites, bitumens, polymers, reactive metals, non-reactive metals and cement-type materials. For the first four, the processing/conditioning aims to minimise the risks by making them inert. Research on graphites was assessed in report No. 10. This report describes the R&D covering the following 3 classes (see Appendix XIII). Finally, the R&D for the last two concerns only reduction in volume.

3.2.1 Bituminous waste

The behaviour of bitumen packages in warehousing and storage has been extensively studied because they are not inert waste. Indeed, the composition of the bitumen mixes they contain gives rise to fears of an unstable situation that could ultimately lead to a fire. In its reports Nos. 9 and 10, the Board examined the results obtained by the producers and Andra on the reactivity of bitumen mixes to an increase in temperature and on the fire resistance of primary packages, simulated but chemically representative and placed in their storage package. The CEA has extended its studies to mixes aged under radiolysis, which differ from fresh mixes by a higher viscosity and the presence of hydrogen bubbles.

The Board considers that the study carried out by the CEA in conjunction with the producers provides important and credible information concerning the storage of bitumens. The fire resistance is compatible with very high temperatures which do not *affect the integrity of the packages for at least one hour according to standard ISO 834.*

Since the fire load in the various parts of Cigéo is not yet known with precision, it remains, in the hypothesis of a package catching fire, to study its possible spread to the whole sector.

The Board recommends experiments on simulated bitumen mixes, in the absence of experiments on samples of real waste, making it possible to verify 1) whether the exothermic reactions are decoupled from the presence of hydrogen during a rise in temperature, 2) to what extent an uneven distribution of the salts would accelerate the exothermic reactions under the effect of a rise in temperature.

An alternative to the storage of bitumen packages would be their incineration, which would remove all questions about their stability. The CEA, which controls this technique, has conducted some incineration tests using a double plasma torch in the Shiva facility at Marcoule. From these tests, carried out essentially between 2003 and the end of 2005, the CEA concludes that it is now very difficult to envisage the industrialisation of a process for the incineration of bitumen packages and the vitrification of the residues, because all the reactions involved are incomplete and very difficult to control. The essential difficulty, in addition to the cost that the CEA considers to be prohibitive, stems from the presence of refractory salts carrying radioactivity, which can only be completely decomposed thermally at very high temperatures.

The Board considers that it is appropriate to continue exploring the incineration of these bitumens, even if the few experiments carried out do not appear conclusive. Direct incineration of the packages would in fact involve control of the gases and particles emitted and the development of appropriate conditioning for the non-decomposed radioactive residues. Previous heat treatments to remove the refractory salts could possibly overcome the obstacles to the process.

3.2.2 Polymer waste

In waste packages containing organic materials, such as polymers, radiolysis produces hydrogen and corrosive species or species that can lead to the formation of complexes with radionuclides contained in the waste. This can entail risks, in the long term for example, such as accelerating the migration of radionuclides into the COx. To this category belong the so-called “alpha contaminated” LLILWs, mixtures of metal waste and plutonium-rich organic waste from MOX fuel fabrication plants. To eliminate these risks, the CEA, in partnership with Areva and Andra, is developing the Pivic process, which consists of incinerating the waste and vitrifying the residues (*see* Appendix XIII). The package resulting from the treatment, called a Pivic package, will contain a two-phase glass-metal solid, the radioactivity being confined to the glass already used for the conditioning of fission products and minor actinides (industrial package CSD-V). The R&D began at Marcoule in 2011 and is scheduled over a period of 15 years, with the aim of arriving at an industrial facility.

The Board understands that there is currently no major scientific barrier to the development of an incineration-vitrification-fusion facility for “alpha-contaminated waste” and that the project is financed. It notes the willingness of the CEA, Areva and

Andra to bring the project to a successful conclusion. The Board will monitor its scientific and technological progress.

3.2.3 Waste containing reactive metals

Waste containing metals such as magnesium, zirconium and their alloys is reactive. Hydrogen is given off in contact with water. This waste comes mainly from the reprocessing of UNGG fuels. The diversity of these fuels and the evolution of the reprocessing technology during the period in which they were produced have led to very different types of “magnesian waste”, with packaging that cannot be identical, although based on sealing by cementing in appropriate containers. The R&D focuses on the formulation of a cement-type hydraulic binder. This formulation must lead to the waste being made inert, which is understood as the minimisation of the hydrogen production per package. It must allow for easy industrial application (low heat of hydration, fluidity) while ensuring, in the primary package, strength (homogeneity, resistance to compression), resistance to radiation and long-term behaviour compatible with its storage.

The Marcoule waste consists of UNGG fuel sheaths separated by the graphite that surrounds them. They are whole, crushed or compacted and stored dry in bulk. The CEA has selected a “geopolymer” mineral binder which is a sodium fluoro-silicoaluminate prepared from aluminium silicate and a fluorinated solution of sodium silicate. It has excellent mechanical strength, and is resistant to alpha and gamma irradiation and leaching. The magnesian waste from La Hague is mixed with other wastes which also contain traces of reactive metals and graphite. Sorting is virtually impossible, so that Areva has finally chosen a slag (aluminium, calcium and magnesium silicate) mixed with a solution of sodium hydroxide as a binder for the coating of this waste. It limits hydrogen production to a level that does not alter the mechanical performance of the final package.

The Board has not received any new information on the packaging of other wastes containing reactive metallic materials: sodium, aluminium or indeed graphite waste.

The Board takes note of the increase in R&D on the packaging of waste containing reactive metals. The CEA and Areva have optimised binders to control hydrogen production. The results obtained show that this production can be considerably reduced and controlled. However, it is essential to limit the quantity of metals per package. The CEA and Areva face a major challenge given the quantities of waste to be managed.

CHAPTER 4:
LOW-LEVEL AND VERY LOW-LEVEL WASTE

4.1 GENERAL INFORMATION

These types of waste present a wide typological variety and ill-defined mass activities within the limits of a few Bq/g to a few hundred Bq/g. They are produced in enormous quantities in a very delocalised way. They contain varied radionuclides, mostly long lived, but their common characteristic is to have low, very low or even non-existent radiological impacts, whatever the management situation envisaged. However, their storage is subject to a regulation that is explained in Appendix XIV and the producers of this waste are bound by obligations towards the ASN.

These obligations have just been reorganised in accordance with the provisions of a decree and its implementing order, taken following the publication of the PNGMDR 2016-2018. These provisions set out numerous actions to be carried out by waste producers and by Andra in the years to come. This will lead them to provide the ASN with numerous reports following on from studies already undertaken, or to be undertaken. The Board cites in Appendix XIV the structuring articles for the demands relating to very low-level (VLLW) and low-level long-lived (LLLW) waste.

There are two broad categories of VLLW:

- those containing only natural radionuclides: uranium mining waste including tailings and processing residues, radium-bearing waste, uranium conversion waste from Malvesi, non-nuclear waste, so-called historic waste (and some uranium- and thorium-bearing materials if they were reclassified as waste);
- those that potentially contain artificial radionuclides mainly produced by the implementation of nuclear fission energy: technological waste from the operation and dismantling of nuclear reactors and cycle facilities.

The Board has already addressed in its previous reports a few aspects of the management of all these VLLWs and the related R&D. It has stressed that the management of future VLLW under current regulations will encounter difficulties due to the quantities to be taken into account, which go well beyond the storage facilities of the industrial grouping centre for warehousing and storage (Cires). It has also pointed out that the site currently being explored in the Communauté de Communes de Soulaine (CCS) appears to be well suited to the storage of VLLW, but that the storage of LLLW should be the subject of a thorough safety analysis.

In this chapter, the Board examines the management of VLLW by separating mining waste and waste from the non-nuclear industry whose treatment leads to enhanced radioactivity (Tenorm: technologically enhanced naturally occurring radioactive materials), from nuclear VLLW and excluding that of Malvesi. It also makes a quick point on the management of LLLW. Indeed, it has been made aware of recent developments in the management of all these wastes. In particular, the last reflections on the management of VLLW and LLLW move towards global management, consistent and proportionate to their radiological and chemical hazards, both for humans and for the environment, with surface or sub-surface storage of most of them remaining the common outcome. To support the new management scenarios, Andra refers to the storage of low- and intermediate-level short-lived waste (LILW-SL) at the Aube repository (CSA), which has been in operation for decades and the Cires storage facility. Finally, the notion of very low-level waste (VVLLW) is put forward.

The management of waste categories discussed here often depends on several Codes and regulations that are listed in Appendix XIV.

4.2 MINING WASTE

The quantities, activities and locations where tailings and uranium processing residues are stored are given in the Mimausa inventory prepared by the Institute for Radiological Protection and Nuclear Safety (IRSN). Areva has identified the tailings that have been dispersed in the past and carried out work to restore radiological standards when the radiological impact exceeded threshold values defined with the ASN. Many of these tailings are now in storage.

The mineralogical evolution of mining residues that concentrate the radioactivity is monitored by Areva. Under the action of water, minerals form which strongly sorb uranium and radium and other radioactive elements. Areva has modelled the two-phase chemical system that they form with water on contact. The concentrations of radionuclides in the water are low. Waste water from repositories is decontaminated before being released into the environment.

Areva is collecting data on mineralogical developments in repositories. This data is essential for modelling their long-term behaviour, and in particular for predicting concentrations in waste waters.

The Board considers that physico-chemical monitoring of all mining residues and waste water from repositories should be systematically continued in order to understand the evolution of the source term and to allow the modelling of their radiological impact in the long term.

4.3 TENORM

40 Very little Tenorm waste, of the VVLLW type, is stocked in the 4 authorised ISDDs (Hazardous Waste Storage Facilities) and ISDNDs (Non-Hazardous Waste Storage Facilities). This represents around 10% of the Tenorm of about 50 million tonnes. Small amounts of Tenorm will go to Cires or will be classified as LLLW. The vast majority is deposited at the production sites.

The management of Tenorm will be amended by ordinance following a 2013 Euratom directive on protection against ionising radiation. A decree is with the Council of State. Tenorm below an exemption value will be conventional waste that can be recovered. Above this value they will become naturally occurring radioactive substances (SRON) which, depending on their activity, will go to ISDD and ISDND for levels below 20 Bq/g or to repositories for VLLW or LLLW. These latter cannot be recovered.

Many non-nuclear industries will have to check the mass activity of their waste. The application of the provisions of the decree will require the modification of all existing regulations (see Appendix XIV). The Board has taken note of the management of Tenorm in England. It comes mainly from the oil industry. It is stocked in two centres.

The Board understands that the transposition of the European directive leads to a new situation for both the Tenorm producers and the quantities of Tenorm waste to be managed. It wishes to know, in application of the provisions of the decree, the quantities of Tenorm that would go into the different types of storage and the methods of their orientation.

4.4 HISTORIC WASTE

This is VLLW with an non-natural radioactivity of a few Bq/g or Tenorm, which is not under Andra's responsibility but that of the producers. It is listed in the national inventory of 2015. The CEA's large-volume deposits (100 to 150 000 m^3) have been undergoing environmental monitoring for decades and the water analysis of the monitoring piezometers shows no radiological impact. The CEA wishes to leave them in place. The others, in smaller quantities, will be recovered during C&D operations. The same applies to EDF.

4.5 VLLW AND THE NUCLEAR INDUSTRY

Andra predicts that the storage capacity of Cires of 900 000 m^3 will be reached by 2030 and that a second VLLW repository will be required despite the efforts of producers to reduce the volumes of waste to be stored. It is envisaged that this second repository will be in Soulaines (see report No. 10).

The latest forecast, once the whole fleet is dismantled, estimates the volume of VLLW at two million m^3. For EDF and Areva, a massive influx of VLLW will occur after 2033, while for the CEA, on the contrary, a decrease is expected after 2033. The CEA will thus be the main contributor to Cires. This data (Appendix XIV) depends, however, on the levels selected for the clean-up of the facilities.

Andra estimates that 40 to 50 % of the volume of VLLW in fact corresponds to VVLLW. This could then go into *in-situ* repositories, simpler than Cires, of the ISDD-ISDND type, installed, for example, at the large C&D centres. The notions of VVLLW and its storage remain to be defined.

The Board observes that this last point has a direct bearing on the use of a release threshold, which does not exist in France but is in force in Germany and the United Kingdom (see International Panorama).

In this context, the recovery of the steels from dismantling Areva's GB1, EDF reactor GVs and other steels, by melt-decontaminating, remains valid. EDF and Areva are planning to submit a safety options dossier to the ASN in 2018 for a demonstrator including the management of secondary waste and plan the construction of the demonstrator in 2027. The recovery of other VLLW such as concrete rubble is under consideration. Even where recovery is technically feasible, the economic dimension remains a major variable in decision-making.

A clear proposal for VLLW management including real recycling possibilities is expected for 2023.

In its report No. 10, the Board recommended ensuring consistency in the management of low-level waste. The recommendations of the Directorate-General for the Prevention of Risks on Naturally Occurring Radioactive Substances now need to be harmonised with those of the ASN on VLLW. The Board considers that waste management policy should be based solely on studies characterising its toxicity. Isolation and containment times with regard to the biosphere must also be defined, taking account of society's expectations. The issues of a release threshold and low doses clearly underlie these questions.

CHAPTER 5:
FUNDAMENTAL RESEARCH

Since the enactment of the 2006 law, Andra has started the Cigéo project, which will lead to the storage of waste from the current nuclear fleet. The CEA, for its part, has started the Astrid project, which will lead to the construction of a industrial demonstrator FNR with its fuel cycle. How is fundamental research associated with these two major industrial projects?

Fundamental research is essential, because it demonstrates the scientific feasibility of the project and establishes the laws of behaviour of the components. It continues with the demonstration of technical feasibility, which makes it possible to cross the 9 levels of the Technology Readiness Level (TRL) scale. The last level of the scale is reached when all the components and their operation have been qualified in a quasi-industrial situation. Fundamental research is then used again for improvements and subsequent innovations.

The fundamental research for Cigéo and Astrid, and more generally research for waste management and new nuclear reactors, is mainly undertaken by the CEA, the CNRS and the universities, most often in collaboration with the major nuclear stakeholders. Beyond the national collaborations, there are major European and international programmes in which French teams participate.

The Board has described in its previous reports the general characteristics of fundamental research carried out at national level under the 2006 law. It has evaluated the results on several topics and made recommendations. As a continuation of these evaluations, the Board has continued its hearings with researchers on nuclear materials and waste. In this chapter, the Board looks at research developed under the Needs programme. It also refers to the research of a few national stakeholders. The results obtained are presented in detail in Appendix XV.

5.1 NEEDS

Needs is an incentive programme led by the CNRS; it involves the major stakeholders in research and the nuclear industry. Its aim is to bring together a national community of researchers in unifying projects (PFs). Research projects are in response to calls for tenders. Some calls are targeted to promote priority themes. They give rise to structuring projects. Often several laboratories of the partner organisations are involved in the selected projects.

This year, Needs presented results in three areas:

- the acquisition of missing nuclear data for the simulation of Gen IV reactors;
- the search for radioactive waste containment matrices to limit the production of hydrogen;
- the behaviour of porous environments (clay and cement) under thermal, hydraulic, mechanical, chemical and gas pressure stresses, these being environments involved in geological storage.

This choice shows the diversity of the Needs research topics.

Needs has found a working dynamic to select and develop research topics of interest to major French nuclear companies. The projects, developed and implemented rapidly, provide a solid scientific basis for applied research. This type of collaboration differs from traditional, bilateral or European collaborations. Needs generates numerous publications and gives visibility to the CNRS and the universities in the nuclear field.

The Board considers that Needs is a step in structuring a nuclear research community. Needs contributes important and original information that is indispensable and its *continuation is a key element of research related to the 2006 law. For this activity to continue in a more constructive and sustainable manner, funding arrangements should be found to enable multi-year projects. Indeed, the Board observes that Needs receives annual support, which only allows short-term actions to be launched.*

5.2 NATIONAL STAKEHOLDERS

Andra launched two calls for tenders (2014 and 2015), using the structure of the ANR, to initiate innovative solutions in the management of VLLW, which had not until recently been the subject of a research programme. The Board observes that there is complementarity between projects selected by the ANR and certain Needs projects. Andra wishes to maintain the momentum created by these two first calls for tenders and to reinforce the synergy between the research supported through different sources of funding.

The CEA has taken a new step in understanding the behaviour of nuclear glass. It has characterised the structural properties of highly irradiated glass as it will be in the long term and the layers of glass degraded by water in the COx. These structures control the different leaching regimes of the glass, responsible for the long-term release of radionuclides. The CEA has also presented the Board with the results of its research in the field of radionuclide migration. They aim to understand and predict the transport of radionuclides across the geosphere, in support of demonstrating the safety of a geological repository. Finally, the CEA has an important programme in nuclear toxicology to understand how radioactive or non-radioactive but chemo-toxic substances interact at the molecular level with cell constituents. A large multidisciplinary community from the CEA and all French organisations involved in radiation protection or toxicology participates in this programme.

In 2010, Subatech's radiochemistry group launched a major programme for modelling the interactions of ions and organic molecules with clays, COx argillites and cement-type materials. It targets the theoretical and experimental study of the sorption and the mobility of these entities.

Finally, the Environmental Risks Group of the Radiation Protection Division of the IRSN is carrying out research on the assessment of the radiological risk for humans and the environment in any exposure situation (chronic, accidental) and is also developing a programme for the management of contaminated soils. Improved knowledge of the mechanisms of radioactivity transfer in the biosphere thus underlies these activities. Experimental resources have been installed on several sites contaminated by nuclear accidents or placed under radiological supervision.

The Board recalls that innovations under development in reactors, the fuel cycle and waste management come from principles based on the results of fundamental research. These developments will remain at the top level only if the research continues unabated. Moreover, nuclear safety can only be credible if it is based on a deep knowledge of the phenomena driving the short- and long-term evolution of materials and the biological mechanisms involved in radiation protection.

The Board considers that the fundamental research presented to it on topics related to the 2006 law meets expectations. The data obtained is of high quality and supports the developments already undertaken within the framework of this law.

The Board does not detect the presence of strategic orientations for fundamental research in the mid and long term. It recommends that the CNRS, universities, the CEA and Andra present a national, multi-year and coordinated plan for the development of fundamental research in the nuclear field.

5.3 HUMAN RESOURCES IN NATIONAL RESEARCH

The Board returns in Appendix XV to the human resources allocated to fundamental research. They are decreasing, both at the CEA and in the academic world. The Board recalls that fundamental research upstream of the R&D, which has also seen its resources weakened, is essential to establishing the scientific knowledge on which the French strategy for the management of radioactive materials and waste is based. The reduction in resources means that fundamental research and the R&D on which it is based are managed only in the short term. The particular case of radiochemistry is very important because it is broadly related to radioactivity and radioactive matter and is therefore at the heart of the disciplines that support this strategy.

The Board reiterates its previous recommendations to support fundamental research and training in line with the need to manage radioactive materials and waste. This inescapable need requires, and will continue to require, high-level experts to apply their skills in the many roles that are indispensable to achieving it in complete safety.

CHAPTER 6:
INTERNATIONAL PANORAMA

6.1 INTRODUCTION

As in previous reports, the first part of this chapter describes recent developments since the publication of report No. 10.

As a reminder:

- report No. 7 contained an international overview detailing the situation up to 2013;
- report No. 8 especially focused on the organisation of the management, financing and estimated cost for a geological repository and on the international approach to reversibility/recoverability;
- report No. 9 analysed in more detail approaches concerning the cost of a geological repository and the problem of the decommissioning or release of dismantling materials;
- report No. 10 presented the conclusions of the Board's study tour to Poland, the Czech Republic and Hungary.

This chapter analyses:

- the advancement of research into geological storage;
- the availability of critical mock-ups, important for the neutron study of core configurations and for the validation of neutron calculation codes;
- some experiments in the dismantling of nuclear installations, including the methodology of cost estimation;
- and, finally, the practices of releasing very low-level waste in Germany and the United Kingdom.

The international framework and the distribution of fast neutron sources, which have not undergone major changes, are repeated in Appendix XVI.

6.2 RESEARCH LABORATORIES AND GEOLOGICAL DISPOSAL SITES

6.2.1 Recent developments

a) Finland

In November 2015, the Finnish government granted Posiva Oy a licence for the construction of a geological repository. Since then, preparations for the construction phase have advanced well and the safety authority, STUK, concluded in November 2016 that Posiva Oy was in a position to start construction. Following this decision, Posiva Oy and Yit Construction Ltd. signed a contract for the excavation of the first tunnels. The duration of this phase of the project is estimated at two-and-a-half years. The value of the contract is around €20m and the labour force, including subcontractors, represents around 100-125 people years. YIT started excavating in December 2016.

The next contracts related to the construction of the repository will be awarded as the project progresses. They will concern the excavation of the wells for the future container lift and the excavation of the station at - 430 m where the containers will be received. Exploratory excavations as well as sealing tests have been carried out.

b) Sweden

In March 2011, SKB, which manages spent nuclear fuel and waste, submitted a construction authorisation request for a deep geological repository in granite rock at Forsmark in the municipality of Östhammar. Another request concerned the encapsulation plant for spent fuel assemblies, adjacent to the Clab warehouse facility where they are located, in the municipality of Oskarshamn.

The authorisation requests are examined both by the Swedish radiation protection authority (SSM) under the law on nuclear activities and by the Land and Environmental Court (mark- och miljödomstol, MMD) in accordance with the Environmental code. Many issues were raised by stakeholders and submitted to the court or the safety authority. SKB responded by providing the additional information requested or by explaining how problems could be addressed during the construction and operation of the facilities.

The review of the request has now been completed and the Court has determined that it is prepared to go ahead with the various procedures. They include 4-5 weeks of public consultations in September 2017, both in Stockholm and at the planned sites for the repository (Forsmark, Östhammar) and the encapsulation plant (Simpevarp, Oskarshamn). The court must then give its decision in the following months. In a statement made in June 2016, SSM considered that SKB was in a position to comply with the nuclear safety requirements it had recommended. Accordingly, SSM recommended that the Court should authorise the repository.

SSM based its assessment on the fact that SKB had demonstrated:

- that the choice of Forsmark as the preferred location for the repository is well-founded;
- that the solution adopted for final disposal of the waste is preferable to the alternatives;
- that it has the capacity to develop and operate encapsulation and storage facilities in compliance with safety requirements.

SSM emphasised that it proceeds by stages in the authorisation process. This means that if the Swedish Government makes the policy decision to grant a licence, SSM will examine, as construction and operation proceed, if the operations envisaged by SKB meet the safety requirements. At each stage, SKB must further refine its safety analysis report to demonstrate the long-term radiological safety of the repository.

Recently, SSM recommended that the government should approve the latest research and development programme by operators of nuclear power plants with respect to radioactive waste management. The nuclear industry in the country has to submit such a programme every three years.

It is therefore likely that the non-political licensing process will be completed by the end of 2017. There will then follow the political decisions of the two municipalities concerned (they have a right of veto) and the government. The municipality of Östhammar, where the repository is planned, has decided that a local consultative referendum will take place on 4 March 2018. At present there is a clear majority in favour of a repository in Östhammar. Whatever the outcome of the local referendum, it is the municipal council that will have to make the legally valid decision regarding the response to the government.

The final decision to authorise the construction of the repository and encapsulation plant must be made by the government. Although it is likely that by the spring of 2018 it will receive clear and positive notifications from the safety authority, the Court and the two municipalities concerned, it is likely that the government decision will be taken only after the elections in September 2018. SKB plans to start construction around 2020 with operations starting around 2030.

c) United Kingdom

In July 2014, the British government published a White Paper entitled “Implementing Geological Disposal”. The identification of potential sites for a geological repository will prefer an approach based on collaboration with communities that would be willing to host it. Such a disposal facility could be situated in England, Wales or Northern Ireland.

The White Paper defines three main actions to be taken by the government and the company Radioactive Waste Management Ltd (RWM) before the process of choosing the site begins:

- provide information on the geology of the regions of England, Wales and Northern Ireland ("National Geological Screening" programme);
- prepare the work to define the conditions under which interested "communities" would participate in discussions and benefit from investments;
- develop land-use planning procedures to guide future applications for geological repository construction.

These actions are under way and RWM is awaiting the government's green light to launch the procedure.

Scotland has its own policy for the management of radioactive waste. RWM will work with the Scottish government to ensure that radioactive waste in Scotland is managed safely.

d) United States

President Trump has requested $120 million in the US budget for financial year 2018 to restart the licensing procedures for Yucca Mountain.

As a reminder: Yucca Mountain in Nevada was the principal site studied and partially built for the disposal of spent fuel in the United States. In 2002, the "Yucca Mountain Development Act" was approved by Congress and signed by President Bush. Yucca Mountain thus became the site officially proposed for installation of a repository. In June 2008, the Ministry of energy (US-DOE) submitted an application for authorisation for the site to the competent authorities (Nuclear Regulatory Commission, NRC), but the Obama administration stopped the project.

The sending of transuranic military waste (TRU) from the sites where it was produced to the WIPP (Waste Isolation Pilot Plant) in New Mexico resumed in April 2017.

As a reminder, WIPP began operations in 1999 and is the only site in the United States for the disposal of transuranic waste from the military programme. WIPP storage operations had been suspended in February 2014 following two unrelated incidents (see report No. 9).

e) Canada

At present, two geological repositories are planned in Canada: one for low- and intermediate-level waste, under investigation at the Bruce nuclear site by Ontario Power Generation (OPG), and the other for spent fuel in a location as yet unknown, which the Canadian nuclear waste management organisation (NWMO-SGDN) must identify.

The Canadian government had asked OPG to evaluate the performance of three potential sites for low- to intermediate-level waste disposal: one in granite, one in clay rock and last but not least, one in limestone. Under the Bruce site there is a layer of limestone at -680 m. Although three sites were identified, the environmental impact of the other two sites would be greater, mainly because of the 22,000 transport operations required to bring the waste that is currently stored at Bruce. Moreover, they would entail an additional cost of between €850m and €2,500m. The OPG proposal is therefore to create a geological repository at the Bruce nuclear site.

With regard to the choice of a disposal site for spent fuel, NWMO-SGDN has implemented a process that requires communities in a geologically relevant region to initiate an application. At present, 8 potential sites have been identified, including Bruce. The final decision is only expected in a few years.

6.3 CRITICAL MOCK-UPS

Critical mock-ups are reactors with near-zero power for the neutron study of core configurations and for the validation of neutron calculation codes. As their power is very low, their fuel retains its isotopic characteristics.

The number of critical mock-ups available for research is in sharp decline. From 2018 and for at least three years, only two reactors will be available for research: BFS-1 and Venus.

BFS-1, installed in Obninsk, Russia since 1962, is a 200 W fast reactor designed to simulate cores of research or generator reactors in several configurations, with several types of fuel and blankets, and several coolants/moderators: sodium, lead, lead-bismuth, air and water. The BFS-1 reactor can host transmutation experiments. It is a remarkable R&D tool.

Venus is a reactor installed at SCK•CEN in Mol since 1964. Very flexible, it has been reconfigured several times, both for water-moderated core studies, as well as for critical or sub-critical FNR cores in the case of ADS. In the European Guinevere experiment , the sub-critical core (metallic uranium and lead) was coupled to a proton accelerator and allowed the study of sub-criticality in different configurations, simulating the future Myrrha reactor. The fuel used is that of Masurca and the accelerator was designed and made available by the CNRS. Venus is available for research.

The other critical mock-ups are:

Eole, commissioned in 1965 in Cadarache. It will be definitively shut down at the end of 2017. It allowed the study of experimental cores of light water reactors, including cores with MOX fuel.

Masurca is a mock-up of a fast reactor with a power of less than 5kW, designed to carry out breeding experiments. It allowed the study of neutronics for FNRs, as well as research on the transmutation of minor actinides and fission products. Commissioned in 1966 in Cadarache, Masurca has been undergoing renovation since 2006. The restart in support of the Astrid project is planned for 2021.

Minerve, located in Cadarache, is a 100 W pool-type reactor, mainly used for neutron experiments, for the development of light water reactor cores and for teaching. It was commissioned in 1959 and will also be definitively shut down at the end of 2017.

Zéphyr is a project for a very low power reactor intended to replace Eole and Minerve. The project has yet to be confirmed.

6.4 MAIN ACTIVITIES ON ADS

ADS (Accelerator Driven Systems) are offered as alternatives to fast critical reactors for the transmutation of long-lived actinides. ADS research continues.

a) Germany

The Karlsruhe Technology Institute (KIT), the Institute of Applied Physics at Frankfurt University (IAP-FU) and the Helmholtz Zentrum Dresden Rossendorf (HZDR) are participating in the Belgian Myrrha project.

b) Belarus

Belarus is developing an experimental ADS programme. It has built the Yalina (low power and thermal spectrum) and Yalina-Booster (moderate power with central fast spectrum zone) sub-critical assemblies, used since 2005 in international physics validation programmes for ADS cores.

c) Belgium

Since 1998, SCK•CEN, has been developing the Myrrha project (Multi-purpose hYbrid Research Reactor for High-tech Applications).

The project will take place in three phases:

1) construction of a linear accelerator of 100 MeV and 4 mA and related scientific facilities for 2024;
2) increase in accelerated proton energy up to 600 MeV after 2024;
3) construction of the spallation source and sub-critical reactor cooled with a lead-bismuth liquid alloy.

The CNE visited the facility in March 2017.

d) China

The Chinese Academy of Sciences (CAS) has decided to build an ADS for transmutation research.

e) South Korea

The Nutreck Institute (Nuclear Transmutation Energy Research Centre of Korea) and the Seoul National University (SNU) are developing a programme based on the transmutation of minor actinides by ADS and reprocessing of transmutation fuel by pyrochemistry.

f) United States

The Department Of Energy (DOE) and national laboratories (Oak Ridge, LANL, ANL, Jefferson Lab, Fermi Lab, etc.) have shown interest in ADS.

g) France

As a reminder: the CNRS and, to a lesser extent Areva and the CEA, are working together on the Belgian Myrrha project.

h) India

The ADS programme, started in 2000, was intended to fast-track setting up of the thorium cycle by the production of fissile uranium 233, from non-fissile thorium 232. More recently, the Bhabha Atomic Research Centre (BARC) has put the emphasis on the potential role of ADS in transmutation of minor actinides.

i) Italy

Several research centres (ENEA, INFN, CRS4, etc.), universities (Cirten) and industries (Ansaldo Nucleare) are participating or have participated in European projects on ADS.

j) Japan

The Omega project, started in 1988, concerns separation-transmutation of minor actinides, in order to reduce the footprint of a disposal site. It includes ADS construction.

6.5 DISMANTLING OF NUCLEAR FACILITIES

The number of nuclear facilities in the world reaching the end of operations is increasing. Consequently, the number of installations being dismantled will increase. As a result, in the near future, very large volumes of low-level waste are to be expected.

However, several aspects related to the dismantling of old nuclear facilities remain of concern to sections of society. Are dismantling technologies adequate? Can all the waste be managed? Are the estimated costs realistic? Can the duration of dismantling be estimated? Is the recycling of materials likely to reduce the volume of waste?

Several dismantling projects have been successfully completed or are in progress and provide guidance on the answers to be given to the questions asked. The most documented are described below.

6.5.1 Belgium

a) Belgonucleaire, Mox fuel plant

Following the joint development by SCK•CEN and Belgonucleaire of MOX fuel in 1962, as well as its introduction in the core of the BR3 reactor in 1963 and the construction of the Eurochemic reprocessing plant, Belgonucleaire undertook the construction of a MOX plant in 1969. Between 1973 and the end of its production activities in 2006, Belgonucleaire produced 660 tons of MOX fuel for light water reactors as well as for the SNR-300 fast reactor.

Meanwhile, Belgonucleaire developed the MIMAS fabrication process to make MOX fuel reprocessable and worked on technology transfer to the Melox plant in Marcoule.

The plant included all the facilities of a fuel fabrication plant, with the exception of assembly, which was carried out at the nearby FBFC plant. It included production facilities (mills, mixers, presses, ovens, grinding/sorting, rod-making, etc.), measuring and quality control facilities, all in 170 glove boxes distributed on one level.

Dismantling started in 2009. Cutting of the glove boxes lasted until the end of 2014, followed by dismantling of the infrastructure in 2015 and 2016. The final radiological characterisation of the buildings is expected to be completed in mid-2017, with the ultimate objective being the unconditional release of the site to greenfield status.

The quantity of waste included in the dismantling plan was 1019 tonnes of waste to be released and 401 tonnes of LLLW divided into 182 m^3 of β-γ solid waste, suspected α solids and liquids, as well as 315 m^3 of solid α waste.

The quantity produced is in accordance with the forecast, except for α solid waste for which the final volume is 370 m^3.

In 2010, the estimated dismantling budget came to €130m$_{2008}$ or €175m$_{2017}$. This estimate had been validated in 2009 by Ondraf. This cost will overrun by 30% due to an extension of the project caused by the initially unplanned treatment of waste at the site and by an incident that contaminated premises.

b) BR3, PWR reactor

Belgian Reactor 3, BR3, was the first pressurised water reactor outside the United States and the first power reactor in Belgium. Its power was 40 MWth and 10.5 MWe net on the network. BR3 was the forerunner of Chooz A, both being derived from the Shippingport (USA) plant, the world's first civil power PWR.

The proposal to build a reactor with an electrical power of the order of 10 MW in Brussels in order to supply the 1958 universal exhibition with electricity, dates from 1954. A Westinghouse pressurised water reactor was chosen.

The reactor started production in 1962 and was definitively shut down in 1987. Decommissioning/dismantling studies started in 1989. No PWR reactor has been dismantled to date. In order to develop optimal methodology and technology and to obtain a good estimate of costs for the future dismantling of other Belgian or European PWR power plants, the dismantling of

BR3 was chosen by the European Commission as a pilot project, along with three other projects concerning other types of nuclear installations. The dismantling of the reactor made it possible to adapt and test different technologies, for example for cutting the reactor vessel under water. Dismantling was therefore organised as an applied research project, rather than as an industrial project.

The dismantling plan is a legal obligation in Belgium. At least three years before the final shut-down of a nuclear facility, each owner of such a facility must submit the final dismantling plan to Ondraf. For any new installation, an initial dismantling plan must be prepared and submitted for approval. It must be updated at least every five years.

The plan describes the following points in particular:

- a complete physical and radiological inventory of the components of the installation;
- the methods and evaluation of the techniques for decommissioning and dismantling;
- the management of waste products;
- a description and evaluation of decommissioning strategies;
- costings and the funding scheme for the decommissioning.

Given that the dismantling of BR3 was a world first, emphasis was placed on methodologies, such as dose reduction for workers, decontamination and monitoring, rather than an *a priori* cost estimate. Nevertheless, for each element of the inventory, up to a very detailed scale, a record was drawn up with the physical and radiological characteristics, the expected quantity and nature of the waste, the expected working time, etc. A total of about 1600 records were prepared.

All this data, as well as radiation doses, waste quantities and costs of all the elements associated with dismantling, were entered into a database in order to be able to estimate in a much more precise way the economic costs and radiological exposures for facilities to be dismantled.

In 1989, no date for the end of dismantling had been defined, given the priority allocated to developing methods and scenarios rather than dismantling per se. The first stage was the chemical decontamination of the primary circuit in 1989-1992. The heat shield, vessel and internal structures followed; then the steam generator and all the auxiliaries.

Dismantling is currently more than 80 % complete. The last phase of decommissioning is now scheduled for 2023, the date of commissioning of a storage facility for Category A (low activity) waste. The final objective is the conditional release of the site.

To date, dismantling has so far produced the following volumes of conditioned waste: 27.2 m^3 of LLHLW, 6.4 m^3 of LLILW and 168.8 m^3 of LLLW.

Very large quantities of contaminated metal (primary circuit including the steam generator and pressuriser, auxiliary circuits, etc.) have been decontaminated, either by sandblasting or by the MEDOC® process. This is a process developed specifically to decontaminate the non-activated primary circuit of the reactor up to the release level. MEDOC® is based on the use of Ce(IV) as a powerful oxidant in a sulphuric acid environment with continuous Ce(IV) regeneration by ozone. The facility can process up to 1 tonne of contaminated metal per day with a decontamination factor of 10^4. Thus 95% of the metal has been able to be released either directly or after melting, mainly at Studsvik in Sweden.

A first estimate of the total cost of dismantling, excluding spent fuel management, was €183m$_{2015}$; it is currently estimated at €257m$_{2015}$.

The experience gained during the dismantling of the BR3 made it possible to evaluate the influence on the cost of scenarios that correspond to significant decay periods:

- dismantling that begins immediately after the plant closes;
- dismantling after a decay period of 30 years.

Immediate dismantling would be most cost-effective. Variations between scenarios depend on many local factors and the costs of waste management. The maintenance of knowledge and the protection of workers, the population and the environment will play a decisive role in the selection of dismantling scenarios; they generally favour the choice of immediate dismantling.

c) Eurochemic, reprocessing plant

In 1957, 12 member states of the OECD (Germany, France, Belgium, Italy, Sweden, the Netherlands, Switzerland, Denmark, Austria, Norway, Turkey and Portugal), followed by Spain in 1959, decided to set up an international company “Eurochemic”, with the task of building a pilot plant for reprocessing fuel and researching reprocessing processes. The Mol/Dessel site was selected because of the proximity of SCK•CEN.

The plant came into operation in 1966. Between 1966 and 1974, when it ceased operations, the plant reprocessed 182 tonnes of non- or low-enriched fuel and 30.6 tonnes of highly enriched fuel. The plant ceased operations following the withdrawal of Germany, England and France, who preferred a national approach to reprocessing.

In 1978, Belgium decided to take over the facilities in order to keep open the possibility of continuing operations. This option was finally abandoned in 1986 and studies for dismantling began in 1987.

The reprocessing plant consisted of a cumbersome rectangular concrete construction 90 metres long, 27 metres wide and 27 metres high. The seven-storey building consisted of 106 cellular structures containing about 1500 tonnes of metal components in the form of equipment, approximately 12,500 m^3 of concrete and 55,000 m^2 of contaminated concrete surfaces.

Dismantling began with a small pilot project for decontamination and demolition between 1987 and 1990 in order to:

- demonstrate that the dismantling of nuclear installations is feasible,
- acquire practical information on dismantling methods and techniques,
- test or develop dismantling equipment,
- train personnel in these new techniques,
- assess the cost-benefit of the dismantling work,
- confirm or re-examine the results of the studies carried out for the dismantling of the existing Eurochemic infrastructure.

Dismantling of the plant per se began in 2000, followed by conventional demolition as of 2008. Since 2015, dismantling is complete.

In total, 1494 tonnes of metal and 2913 tonnes of concrete have been decontaminated.

Metal pipes, parts and tanks were thoroughly rinsed so that they could then be processed manually. For their decontamination, Belgoprocess (Ondraf's industrial subsidiary) developed an abrasive blasting facility capable of dry eroding a few microns of the metallic surface.

Concrete structures were scraped at contaminated sites, then crushed into aggregates and sampled prior to release.

Until the end of 2013, 26,771 tonnes of waste was produced, 15,777 tonnes were directly classified as uncontaminated material; 1963 tonnes were considered as radioactive waste and 8860 tonnes of the remaining 9031 tonnes were decontaminated.

More than 95% of the radioactive waste is LLLW intended for surface disposal and less than 5% is LLILW for geological disposal.

At the end of demolition, 92 % of the waste had been recycled.

The cost was initially estimated at €175m_{2000}. The final figure was €210m_{2014}. This amount corresponds to the initial estimate, taking into account inflation of around 26% during the clean-up period. In this amount, the costs of dismantling operations count as much as those of the extensive decontamination with a view to releasing the waste. Treatment of the radioactive waste, intermediate warehousing and disposal represent barely one-third of the initial estimate. Radical decontamination at a high price was chosen, which generated significantly less waste to be conditioned. The initial estimate for the overall clean-up of Eurochemic mentioned 403 man-years. The final result was 600 man-years.

d) FBFC - AREVA, fuel assembly fabrication plant

In 1958, the company Métallurgie mécanique nucléaire, MMN, created a plant to supply fuel assemblies for reactors BR2 and BR3. In 1973, the shareholding was expanded and Franco-Belge de Fabrication de Combustibles, FBFC, was created. The main shareholder is now Areva.

The production of UOX fuel for BR2, BR3 and Chooz A and of MOX for BR3 started in 1961. In 2012, AREVA officially informed the Federal Agency for Nuclear Control, AFCN, that it had decided to cease all its activities at the FBFC site over the next few years. The capacity of the plant was then 700 tonnes/year including 200 of MOX fuel. The last assembly left the plant in 2015.

In all, 920 tonnes of equipment are to be dismantled: 655 t contaminated and 265 t uncontaminated, an area of 35,000 m^2 is to be released, including 25,000 m^2 to be surface-stripped and 10,000 m^2 uncontaminated to be checked.

The first decommissioning authorisation was issued in 2010. The end of dismantling is planned for 2020.

In addition to waste from the 655 tonnes of equipment and 4100 m^3 of sand to be decontaminated, 100 m^3 of sanding residues, 250 m^3 of stones and pipelines will end up as waste. The quantities predicted by Ondraf in 2012 were 515 m^3 of ß- σολιδ ωαστε, 15□μ^3 suspected ανδ 1637 μ^3 of liquid waste.

The decontamination of the large quantities of sand remains a challenge.

The costs cited by Ondraf are approximately €50m$_{2010}$ if the waste can be stored on the surface and €65m$_{2010}$ if a large portion is to be stored at depth.

6.5.2 Spain

a) José Cabrera, first Spanish PWR

This 142 MWe reactor, commissioned in 1968, was shut down in April 2006. Dismantling was to last six years (2010 - 2016) at an estimated cost of €135m.

At the end of dismantling and decontamination, approximately 4% of the planned 104,000 tonnes of waste will be considered as radioactive waste.

b) Vandellos 1, gas-graphite reactor

This 480 MWe reactor, commissioned in 1972, was shut down in 1990. In 1994, the fuel was unloaded and the waste was stored in silos for graphite. In 2003, ENRESA finished the dismantling except for the reactor itself, which released 80% of the site. By 2027, when the residual activity will have decreased by 95%, the reactor structure will be dismantled.

Of the 300,000 tonnes of waste material, 2,000 remain classified as low- or intermediate-level waste.

The cost of partial dismantling in 2003 was €93m.

6.5.3 France

a) Areva Mox, Cadarache workshops

Areva has just finalised the clean-up and dismantling of the facilities of the former MOX fuel fabrication plant in Cadarache.

Two basic nuclear facilities, the “Plutonium Technology Workshops” (ATPu) and the “Chemical Purification Laboratory” (LPC), were dismantled after 40 years of operation (1962-2003).

ATPu was built in 1959. The production of MOX fuel for fast reactors began there in 1962 and in 1978 ATPu manufactured fuel for Phenix and Super-Phenix. The production of PWR MOX started there in 1989. The production capacity was 42 tonnes per year. In 40 years of operation, ATPu recycled more than 50 tonnes of Pu. The operation of LPC started in 1966.

Dismantling, which started in 2009, was finished this year. During these 8 years, the project required thousands of interventions (of the order of 6000) in pressurised suits in the active area (zone IV) and simultaneously mobilised up to 300 employees of Areva and its partners. The exposure of the workers was minimised by limiting interventions to 2 hours per operator. The average individual exposure was thus less than 1 mSv/year.

In total, more than 462 glove boxes, 30 vessels and 4 km of piping were disassembled, cut, then conditioned and removed. For the most part, the waste was conditioned in the form of 30,000 barrels compatible with storage at the CSA.

The entire dismantling was carried out at a cost of approximately €500m.

In February 2017, about sixty cleaned-up premises were formally transferred to the CEA, operator of the site.

b) Chooz A, first French PWR

The construction and operation of the Chooz A reactor, the first PWR in France, was a Franco-Belgian project. The Chooz site is located on the banks of the Meuse, 3 km from the Belgian border. The CNE visited this dismantling site.

The reactor, with a power of 305 MWe, is a 4 to 5 times more powerful copy of the Shippingport reactor. The design comes from a Westinghouse licence, transferred in 1959 to Framatome. The reactor started in 1967 and was operated until 1991.

The reactor, the four heat exchangers, the deactivation pool and the emergency circuits were installed in two rock caverns 150 m underground, accessible through a tunnel. The hillside layout offered natural protection to the plant.

The decree authorising the final shut-down dates from 1993. At the time, EDF chose deferred dismantling as its strategy, with removal of the spent fuel and operating waste, drainage of all the pipes and the dismantling of the machine room before closure under supervision. EDF changed this strategy in 1999, opting for “immediate” dismantling. The new decree authorising dismantling was published in 2007. In the meantime, the non-radioactive parts had been dismantled.

The installation of the reactor in caverns poses technical difficulties because dismantling was not foreseen at the design stage. Thus, the extraction of the steam generators, which were stored at Cires after decontamination, required the development of a dedicated infrastructure. The very active vessel is being cut under water; the operation is scheduled to last until 2019. Dismantling is expected to be completed by 2022.

The total planned waste is 40,000 m^3, of which 80% is conventional non-radioactive waste. Approximately 10,300 tonnes of LLW-SL or VLLW waste is destined for surface disposal. Pending a decision on their ultimate disposal, 30 tonnes of long-lived waste will be stored in the activated waste conditioning and storage facility (ICEDA), which is currently awaiting commissioning at the site of the Bugey plant.

The cost of decommissioning is estimated at €400m.

6.5.4 Sweden

a) Ågesta, pressurised heavy-water reactor

The pressurised heavy water reactor at Ågesta was permanently closed in 1974. In accordance with the Environmental Code a licence was issued by the local environmental court in November 2008 for the clean-up and maintenance of the site until 2020.

b) Barsebäck, boiling water reactors

The two BWR reactors in Barsebäck were closed in 1999 and 2005 respectively. The facilities have been prepared for a period of clean-up and maintenance while awaiting dismantling.

The decontamination of the primary circuits of both units was carried out during the winter of 2007/08. Processing of waste from operations and from decontamination is under way. Overall dismantling activities are planned as from 2022.

c) Studsvik, research reactors

Two materials research reactors located in Studsvik were closed permanently in 2005. Preparations for the dismantling of the reactors are under way. The decontamination of two experiment loops was carried out in early 2008. The dismantling of the reactors, the first phase of which is being carried out by Areva, is to be completed by 2020. Radioactive waste will be processed and stored on the site awaiting disposal.

6.5.5 Cost estimation methods

Two approaches are used to estimate the cost of dismantling: bottom-up and top-down. The method chosen is 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.

a) Bottom-up estimation

This method estimates costs by adding up all current and anticipated costs of the equipment used, waste management, total working time, and any other miscellaneous costs related to the activity. All costs are distributed over time, hence the need to establish a timetable with deadlines. The next step is to update the expected costs for each year at a fixed date. The grand total corresponds to the total cost of the project at current prices (current net value at the chosen fixed date).

The advantage of this method, if carried out with precision and if the dismantling is foreseen for the near future, is that a small divergence can be expected between the estimated costs and the real costs provided that the physical and radiological inventory are well known. This method is therefore particularly suitable when decommissioning/dismantling activity is expected in the coming years.

This method is well suited to recurring activities, because it is easy to base it on feedback from previous work. It can lead to errors of 15 to 20%.

In the case of the dismantling of an old nuclear facility, the first estimates are often difficult and approximate (lack of knowledge, difficulty in evaluating all the necessary tasks, estimation of the variable duration of operations, degree of contamination difficult to evaluate). This method is thus less appropriate in principle in the case of dismantling a plant of a type not yet dismantled, as there is a high probability that costs will be recalculated regularly as new information becomes available and as radiological contamination is better estimated. The experience described above shows, however, that it gives adequate results.

b) Top-down estimation

The starting point for establishing the required budget is the administrative cost estimate, broken down between the activities, the types of costs and their distribution over time.

This method may be indicated when the dismantling/decommissioning project is far in the future. In this case, the degree of accuracy is lower, but is nevertheless acceptable (margin of error 20 to 40%).

The weakness of this approach is that it underestimates future costs through lack of knowledge of all future activities required. The advantage is that it allows the setting of financial objectives to be reached over time. This financial calculation can then be progressively updated and refined.

c) Items to be taken into consideration for the determination of dismantling costs

Cost estimation techniques vary from country to country, according to very different national regulations. There are many differences depending on the activities considered, the dismantling planning schedules and the final condition of the site. These differences make comparisons between countries difficult.

In the interest of efficiency, the European Commission, the Atomic Energy Agency of the OECD and the International Atomic Energy Agency (IAEA) have proposed using a standardised cost structure, called the International Structure for Decommissioning Costing (ISDC). The ISDC lists all the costs involved in the dismantling process, so it is a bottom-up method.

The ISDC adopts a three-tiered structure: the main level of activity, level 1, is divided into several level 2 activities, which are more specific, themselves divided into several level 3 activities, which are even more detailed. Each level comprises 4 cost categories: personnel costs, capital, equipment and materials costs, cost of expenses and unforeseen costs.

The ISDC has identified 11 main activities, therefore level 1, required to estimate the costs of dismantling a nuclear facility: research and development; pre-decommissioning activities; facility shut-down activities; additional activities for safe enclosure or entombment; dismantling activities within the controlled area; waste processing, storage and disposal; site infrastructure and operation; conventional dismantling, demolition and site restoration; project management, engineering and support; management of fuel and nuclear material; and a final “miscellaneous” item, including activities not included above.

This costing structure allows for grouping of items to more easily track dismantling operations at different stages of the process, provides a general costing framework applicable to all types of nuclear facilities, and applies the latest IAEA classification for radioactive waste.

6.5.6 Some reflections on experiences of dismantling

> *Examples of several installations of representative type and size (reactors, fuel plants, reprocessing plant, etc.) show that technologies are available for dismantling in compliance with all conventional safety and nuclear safety standards.*

> *Methodologies for estimating waste quantities, durations and costs prior to dismantling have led to results with a margin of 25% or 30% for projects where data is made public, which shows a degree of reliability at least as large as that of other important industrial projects. The cost of dismantling is generally estimated to be*

around 10% of the cost of new construction for a low-polluting industry and 20% for a highly polluting industry. For the nuclear industry, the figure of 15% is often taken as a rough estimate.

Maintaining knowledge, feedback from experience and international collaborations, as well as protecting workers, the population and the environment, play a key role in the selection of dismantling scenarios. They generally favour the choice of dismantling as soon as possible.

6.6 INTERNATIONAL EXPERIENCE ON THE RELEASE OF WASTE

6.6.1 Overall situation

In different countries, the classification of radioactive waste depends on the management channels available, particularly for low and very low-level waste.

Everywhere in Europe, with the exception of Ireland and Portugal, rules define the conditions for the release of very low-level radioactive materials or exemption procedures. Release makes it possible to remove radioactive materials or radioactive objects from any regulatory control of radiation protection by the producers and the regulatory body. As for exemption, it is conditional and is decided on a case-by-case basis but does not change the status of the material.

Release was already practised in many countries with a large nuclear programme, before the introduction of international regulations. These countries tend not to apply international thresholds directly, but to use their own thresholds.

Harmonisation could be useful for the cross-border movement of released materials.

6.6.2 The case of Germany

In the 1960s, Germany decided to store all its radioactive waste at great depth. Old mines were available, and their use did not seem too difficult or costly. In practice, however, the situation has proved more delicate. The former Asse salt mine was closed in 1978 and long-term safety is critical because of brine infiltration. The Morsleben salt mine was reopened in 1994 but had to be closed in 1998 for legal reasons. The former Konrad iron mine will normally only open in 2022 for the storage of low-level and intermediate-level waste.

The estimated cost of future storage is high and the creation of a VLLW channel is not on the agenda in Germany. At the same time, the country is well aware that the space available in a disposal facility is a scarce resource. Germany has therefore introduced release regulations, based on a “trivial” dose of 10 μSv/year, in order to minimise the quantities of VLLW waste. Very strict regulations and extensive decontamination techniques have enabled the last dismantling projects to release about 97% of the materials. The remaining 3% are currently in storage awaiting future disposal.

The regulations require a release plan from the operator. The plan makes clear, for each category of materials, the steps to follow to comply with the regulations. This plan is then submitted to the authorities for approval. In the event of authorisation, the operator can carry out the plan with a detailed report at each step. The report is submitted to the authorities who can then authorise release.

The operational implementation of release sometimes encounters acceptance difficulties, both from the public and from companies that could recover the materials or waste released. This could become a real problem for the dismantling of German nuclear installations.

6.6.3 The case of the United Kingdom

In 2007, the United Kingdom government published a document entitled "Policy for the Long-Term Management of Solid Low Level Radioactive Waste"[1]. The document defines a strategy, confirmed in 2016, focusing on three topics: a hierarchy of good practices for waste management is introduced; optimum use of storage capacity; and better adapted management solutions are sought. The National Decommissioning Agency (NDA) is responsible for implementing this strategy.

The goal of the strategy is to bring about a culture change among producers in the way they manage waste, in order to reduce volumes and cost while respecting safety. The diversification of management routes led to an 85% reduction in the volumes of low-level waste to be stored at the Drigg repository by increasing decontamination, incinerating, releasing, and disposal of VLLW at a specific site. It is estimated that this strategy has already saved taxpayers €160m.

Management options are being considered, such as the construction of disposal centres at dismantling sites.

Considerable importance is attached to the involvement of local, regional and national stakeholders.

6.6.4 Some reflections on release policies

Among the countries that allow the release of materials with residual radioactivity that is so low that they cannot have an impact on health or the environment, some are motivated mainly by financial arguments such as the estimated cost of disposal, and others by the ethical aspect, such as the need for recycling in a context of sustainable development. For all, reducing the volume of waste is an essential factor as the volume available in a repository is a scarce resource.

The experience of countries with a release threshold shows that regulations, combined with strict procedures and controls, can ensure the protection of populations.

The materials thus released can be reused without restriction, even in consumer goods. European and international harmonisation of approaches to the methods for release of VLLW would therefore seem desirable. The Board reiterates its recommendation for an in-depth reflection by France on this issue.

[1] Policy for the Long-Term Management of Solid Low Level Radioactive Waste in the United Kingdom

APPENDIX I:
BOARD ACTIVITY

The Board was renewed by decree on 28 October 2016, and welcomed 4 new members, Mr. Jose-Luis MARTINEZ, Mrs. Anna CRETI, Mr. Vincent LAGNEAU and Mrs. Mickaele LE RAVALEC. Mr. Maurice LEROY and Mr. Gilles PIJAUDIER-CABOT were re-appointed. On the invitation of the chairman, Mr. Emmanuel LEDOUX agreed to be the invited expert on the Board (see Composition of the National Assessment Board).

Since the publication of its previous report in May 2016, the Board presented its report No. 10 to OPECST and to the relevant ministerial departments. A delegation from the Board visited Joinville on 17 October 2017 to present its report to members of the CLIS (local information and monitoring committee) at the Bure laboratory (see Appendix II).

The Board adopted the same working method as in previous years. It conducted 9 day-long hearings and 2 half-day hearings (see Appendix III), and 5 other closed half-day hearings, all held in Paris, in addition to a certain number of supplementary meetings with concerned parties. The Board members – all volunteers – heard 94 people from Andra and the CEA, as well as from French and foreign academic institutions and industrial organisations (see Appendix IV). These hearings brought together around sixty people on average and were also attended by representatives of the Nuclear Safety Authority, Areva, EDF, the Institute for Radiological Protection and Nuclear Safety and the central administration.

The Board devoted two half-days to visits to the Meuse/Haute-Marne centre and the Aube repository as well as two days to visits to the CEA Marcoule site and the Pierrelatte – Tricastin site, Eurodif and Areva's Georges Besse II (see Appendix II).

In preparing this report, the Board held a two-day pre-seminar during its visit to the Mol underground laboratory with SCK•CEN and a visit to Euridice and the surface laboratories, the MYRRHA support experiments, and Guinevere/TCH in Mol - Belgium. It also held numerous internal meetings, including a five-day residential seminar. The list of Board hearings and visits is given in appendix III to this report. The list of documents that it received from the organisations attending hearings is provided in Appendix V.

APPENDIX II:
BOARD PRESENTATIONS AND VISITS

Board hearings

25 May 2016:	Presentation of report No. 10 to OPECST
4 October 2016:	CNE2 hearing by OPECST as part of the PNGMDR assessment
17 October 2016:	Presentation of report No. 10 to the CLIS

Board visits

18 October 2016 – morning:	Andra – Visit to the Meuse – Haute Marne centre (CMHM)
19 October 2016 – morning:	Andra – Visit to the Aube repository (CSA)
15 February 2017:	CEA – Visit to the Marcoule site on the topic of recovery and conditioning of old waste
16 February 2017:	Areva – Pierrelatte – Tricastin site visit – Eurodif visit - GBII sud visit
28 March 2017:	SCK•CEN – Visit to the Mol underground laboratory, Praclay experiment and surface laboratories (clays, bitumens,etc.)
29 March 2017:	International context and visits to experiments in support of MYRRHA (loops, Guinevere, etc.)

APPENDIX III:
HEARINGS HELD BY THE BOARD

PUBLIC HEARINGS

05 October 2016:	Andra – The major challenges of disposal
06 October 2016 – morning:	CEA – State of play of the civil nuclear sector
16 November 2016:	CEA – Fuel cycle: today and tomorrow
17 November 2016:	Andra – Research in progress for the Cigéo DAC
07 December 2016:	Andra/Areva/CEA/EDF – Research in progress on the conditioning/processing of LLLW and LLILW waste
08 December 2016:	CEA – Waste recovery and conditioning activities and historic waste at sites
12 January 2017:	CEA/CNRS/Andra/IRSN - Fundamental research
22 February 2017:	Andra/producers – VLLW – Dismantling and the recovery of dismantling waste
23 February 2017:	CEA –Waste and material for Generation IV
15 March 2017 – morning:	Andra – Tenorm, mining residues and tailings and historic waste – State of play on their management
16 March 2017:	CEA – Astrid

CLOSED HEARINGS

21 September 2016 – morning:	Andra – Latest news – Cigéo project
21 September 2016 – afternoon:	Areva – News/Context, vision and strategy
22 September 2016 – morning:	CEA – Hearing with the General Administrator of the CEA - Research undertaken at the DEN, waste and material management in the French fleet and the clean-up and dismantling programme for CEA facilities
22 September 2016 – afternoon:	EDF – Cigéo - GEN-IV
11 January 2017:	Andra/Producers - Cost and funding of Cigéo

APPENDIX IV:

LIST OF PEOPLE HEARD BY THE BOARD

ANDRA

ABADIE Pierre-Marie
ARMAND Gilles
BOSGIRAUD Jean-Michel
CAMPS Guillaume
DE MEREDIEU Jean
DOUHARD Séverine
DUTZER Michel
ESPIET-SUBERT Florence
GALY Catherine
GOURRAM Hakim
HOORELBEKE Jean-Michel
LANDAIS Patrick
LANES Eric
LAUNEAU Frédéric
LECLAIRE Arnaud
LIEBARD Florence
MUNIER Isabelle
PASTEAU Antoine
PEPIN Guillaume
PETIT Laurence
RENAULD Valérie
ROBINET Jean-Charles
ROUX-NEDELEC Pascale
SEYEDI Darius
SCHUMACHER Stephan
TALLEC Michèle
THABET Soraya
TORRES Patrice
VITEL Manon
WENDLING Jacques

AREVA

DESCOSTES Mickael
FORBES Pierre-Lionel
FOURCY Etienne
LAMOUROUX Christine
LEBRUN Marc
LOY Christophe
LUQUET DE SAINT GERMAIN Victoire
MOREL Bertrand
ROMARY Jean-Michel
STEPHAN Lavinia

CEA

ADNET Jean-Marc
ADVOCAT Thierry
ANGELI Frédéric
BOULLIS Bernard
CACHON Lionel
CANAS Daniel
CAPPELAERE Chantal
CHARLES Jean-Louis
CHAUVIN Nathalie
DELACROIX Daniel
DEVICTOR Nicolas
DUMAZ Patrick
ESCHBACH Romain
FILLION Eric
FIRON Muriel
GABRIEL Sophie
GAUCHE François
GARNIER Jean-Claude
HOURCADE Edouard
JOURDA Paul
MAGNIN Magalie
MIRGUIRDITCHIAN Manuel
PIKETTY Laurence
PIVET Sylvestre
PLANCQ David
POINSSOT Christophe
SABATIER Laure
SAVOYE Sébastien
TOURON Emmanuel
VARAINE Frédéric
VERWAERDE Daniel

CETU

DEFFAYET Michel

CNRS - IN2P3

DAVID Sylvain
PAGEL Maurice

DGEC

LOUIS Aurélien

DGPR

MICHEL-DIT-LABOELLE Nicolas

DMT

FEINAHLS Joerg

ECOLE DES MINES DE NANTES

KALINICHEV Andrey

EDF

BANCELIN Estelle
COÏC Philippe
DUMORTIER François
DUVIVIER Remi
FERNANDES Roméo
GIRAUD Olivier
ISNARD Luc
LAMARRE Olivier
LAUGIER Frédéric
MAURAU Sylvaine
MESSER Nathalie
VAN DER WERF Jérôme

IPHC STRASBOURG

BARILLON Remi
KERVENO Maëlle

INSTN

DANNUS Pascal

NDA

LOUDON David

NUVIA

SMITH Ron

APPENDIX V:

LIST OF DOCUMENTS SUBMITTED TO THE BOARD IN 2016-2017

Andra

- Knowledge dashboard on LLILW and LLHLW packages envisaged for Cigéo – Summary to end March 2016.
- Annual progress report on work carried out in the underground research laboratory in 2015 – 20 April 2016.
- Activity report 2015.
- Journal de l'Andra – summer 2016 – National version – July 2016.
- Considered opinion of the Environmental Authority of the General Council for the Environment and Sustainable Development on the framing of the national plan for the management of radioactive materials and waste (PNGMDR) published on 22 July 2015.
- Considered opinion of the Environmental Authority of the General Council for the Environment and Sustainable Development on the framing of the national plan for the management of radioactive materials and waste (PNGMDR) published on 20 July 2015.
- 2015 R&D activity report – September 2016.
- ANCCLI White Paper IV – Cigéo: The issues of reversibility and recoverability - January 2017.
- Andra note - Why is it necessary to build Cigéo, the deep repository for the most radioactive waste?

CEA

- CEA Various Keys – Technological innovation - 2016.
- CEA Various Challenges – 2016-2017.

APPENDIX VI:

ANALYSIS OF CIGÉO DOCUMENTS 2016

To prepare the DAC submission, Andra has developed a safety options dossier (DOS) in two sections, one for Cigéo in operation and the other for Cigéo after closure. These documents are accompanied by a dossier of technical options for recoverability (DORec) and a proposed operational master plan. In November 2016, the Board produced an analysis of these documents which is available on its website (www.cne2.fr). Only the main conclusions are mentioned below.

The 2006 law provides that the Board should give an opinion on the construction authorisation request (DAC) for Cigéo. Consequently, the Board has analysed the Cigéo 2016 documents with a view to preparing this opinion. It has taken into account the R&D conducted for more than 20 years and the rules laid down by the law of 28 June 2006 and the law of 25 July 2016 which define the reversibility of a high-level nuclear waste repository. For the Board, Cigéo should be designed as a robust, reversible repository intended ultimately to be closed to ensure long-term passive safety; its closure will be progressive, while guaranteeing recoverability. Since it is intended to receive ultimate waste, the recovery of one or more packages can only be envisaged in the case of a malfunction of Cigéo.

Since reversibility is an evolving decision-making capacity, the Board does not consider it desirable to leave the filled cells open. It recommends putting in place, progressively during the operation of Cigéo, a sealed isolation structure enabling each filled cell to evolve in passive mode in relation to the geological environment; these cells would be subject to a continuous monitoring programme. The reversibility reviews will be an opportunity to decide whether or not to isolate the cells. The operational master plan must therefore analyse in depth the methods used in this isolation strategy. Andra must demonstrate that the sealed isolation structures of the cells fulfil all protection functions in incidental or accidental situations, in particular in the event of fire. Certainly, every stage in the progressive closure of Cigéo complicates the retrieval of a waste package but it increases passive safety.

The Board hopes that the next version of the operational master plan, intended to become a public document, should be more instructive, clearly defining the objectives and guidelines of the project and taking into account the contingencies likely to affect its development.

Cigéo is a complex installation because of its size, its dual location at the surface and at depth, its centuries-long operating life, and the simultaneous work on construction and operation. The Board recommends the creation of a three-dimensional interactive digital mock-up of Cigéo, to train operators and finalise the procedures to be implemented. In operation, the traceability of packages must be ensured in the long term and the documentation should be immediately accessible to operators. The industrial pilot phase is intended to demonstrate complete control of the industrial management of the repository. It should last for as long as is necessary to validate the technical options and make it possible to achieve normal operation.

Control of the interface between the works area and the operational area will be essential. The safety dossier must clarify the measures taken to ensure the safety of personnel in these two areas simultaneously and to analyse accidental situations. It should also address the integration of Cigéo's maintenance process, its impact on operational planning and especially on safety in the event of failure or shut-down for scheduled maintenance.

The Board wishes Andra to present a phenomenological scheme of operation of the LLHLW and LLILW cells as well as the near field over time. This scheme will describe all the physical mechanisms involved in the safety assessment. The demonstration of safety relies mainly on modelling of the release and migration of radioactive chemical species through the components of the repository and the geological environment. The Board recommends clarifying the interlinking of the various models used to represent phenomena at different scales; it requests that a sensitivity study should be presented to assess the effect of the variability of the parameters of the materials on the results of the simulations. It considers that the choice of the parameters associated with the altered environments should be better underpinned. It stresses that there is still a need to have a better understanding of certain phenomena (COx overpressure, THM effects, healing of the damaged zone, gas transients, etc.) and to quantify overall the mechanical behaviour of the rock. It recommends paying attention to the transient phases that involve water and gases in complex thermo-hydro-mechanical and chemical mechanisms, which may play a role in the properties of the components of the repository after closure. It asks for clarification regarding the arrangements for closing the repository.

With a view to the DAC being submitted in 2018, the Board considers that Andra should concentrate all its efforts on the preparatory studies for the disposal of ultimate waste as it is currently defined. Andra will have to prove the robustness of the solution it proposes in the DAC.

APPENDIX VII:

CIGÉO REFERENCE AND RESERVE INVENTORIES

The inventory of waste subject to reversible disposal in deep geological strata was specified in the decree issued on 23 February 2017 establishing the PNGMDR requirements. It includes a reference inventory and a reserve inventory. :

“Art. D. 542-90. - The inventory to be used by the National Agency for the Management of Radioactive Waste for the studies and research carried out with a view to designing the repository provided for in article L. 542-10-1 comprises a reference inventory and a reserve inventory.

The reserve inventory takes into account the uncertainties linked in particular to the setting up of new waste management channels or to changes in energy policy.

The repository is designed to accommodate waste from the reference inventory.

It is also designed by the National Agency for Radioactive Waste Management, in conjunction with the owners of the substances in the reserve inventory, to be able to accommodate the substances included in this inventory, subject as the case may be to developments in its design that can be implemented during operation at an economically acceptable cost... ”

Cigéo will be built (and funded) to host the waste reference inventory, which is described in the latest version E of the industrial waste management programme (PIGD). This inventory is based on the reprocessing of all spent fuel. The inventory takes into account uncertainties (due to the waste that will be produced in the performance of waste recovery and conditioning projects).

The PNGMDR 2016-2018 devotes a large forward-looking section to the question of waste in the reserve inventory. This includes firstly LLLW which would currently be impossible to store other than at depth (39,000 m^3 bitumen packages, 10,000 m^3 graphite jackets, 7000 m^3 graphites from La Hague) and those that could result from changes in the production of nuclear energy (new EPR installations) or strategy: spent MOX fuel, other spent fuels reclassified as waste in case reprocessing is stopped. Andra was to submit to the Minister for Energy by 31 March 2017 a proposal for the types and quantities of waste to be included in the Cigéo reserve inventory.[2] In addition, PNGMDR also specifies that Andra should carry out work on the feasibility of disposal of strategic nuclear material under the reserve inventory.

Finally, although Cigéo is not the envisaged path for depleted uranium (Uapp) and reprocessed uranium (Uret), which would result from the decommissioning of nuclear materials and would then be re-classified as waste, these materials are nevertheless listed as potential wastes on the reserve inventory.

Andra must demonstrate that Cigéo can accommodate waste from the reserve inventory, subject to design changes that can be economically implemented. Some waste from this inventory could even be included in the DAC reference inventory, for reasons related to demonstrating the safety of its disposal.

Andra is requested to provide:

For 2018:

- a feasibility study of the disposal of Uret, cost and impact on the draft channels (article 7);
- a feasibility study of the direct disposal of the spent fuel from research reactors and the spent fuel from naval nuclear propulsion (article 14);

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[2] At the date of writing this report, Andra had not yet produced this document.

- a study on the cost of disposal of spent fuel from nuclear power reactors (article 15).

For 2019

- a feasibility study of the disposal of Uapp, cost and impact on the draft channels (article 4 of the order).

Andra interprets these requests as adaptability studies to the project in its current version, and relies on generic knowledge.

Andra should rely at least on data on the behaviour of these materials in a disposal situation before and after closure of Cigéo. The case of spent UOX fuel assemblies from power generation is well documented, as several countries are preparing for their direct disposal. However, for the other "waste" (Uapp, Uret, spent MOX fuel, spent metallic fuel) where the materials are in different physico-chemical forms from spent UOX fuel, the R&D of their disposal in clay is much less advanced.

To meet the demands of the PNGMDR, Andra should further investigate the storage of spent fuel assemblies over the next few years, to clarify what margins of adaptability the current configuration of Cigéo allows for their disposal and what changes in the Cigéo design would be required to accommodate all waste from the reserve inventory.

APPENDIX VIII:

OBJECTIVES AND APPRAISAL OF THE DOS

The safety options dossier (DOS) comes within the scope of the regulatory process defined by article 6 of the decree of 2 November 2007. This article stipulates that:

"Any person who intends to operate a basic nuclear facility may apply to the Nuclear Safety Authority prior to the initiation of the construction authorisation procedure under article 29 of the law of 13 June 2006, for an opinion on any or all of the options it has selected to ensure the safety of that facility. The Nuclear Safety Authority shall, by an opinion given and published under conditions it determines, specify to what extent the safety options presented by the applicant are suitable for preventing or limiting the risks to the interests referred to in article 28 of the law of 13 June 2006, taking into account the current technical and economic conditions. It may define additional studies and justifications that will be required for a potential construction authorisation request. It may fix the period of validity of its opinion. This opinion shall be notified to the applicant and communicated to the ministers responsible for nuclear safety. "

The decision to submit a DOS was made in May 2014 by the Andra Board of Directors following the conclusions of the public debate published in February 2014. The issue was to obtain an opinion before the DAC on the major safety options of the Cigéo project and to feed reflection on the conditions for reversibility. ASN welcomed this initiative and, in December 2014, clarified its expectations on the content of this document, which it considered to be part of an iterative approach appropriate to the design of a project the size of Cigéo.

ASN's key expectations covered the following points:

- addressing all areas with respect to operational and long-term safety after closure;
- making a full presentation of the entire project;
- introducing elements of justification for the flexibility of the facility which could lead to optimisations;
- integrating questions taking account of feedback from international experience, organisational and human factors, and objectives for the radiological protection of staff.

Andra subsequently submitted the various documents making up the DOS in June 2016 and the appraisal procedure began. In accordance with ASN's expectations, the safety options dossier (DOS) is in two sections, one for Cigéo in operation and the other for Cigéo after closure. These documents are accompanied by a dossier of technical options for recoverability (DORec) and a proposed operational master plan.

The DOS appraisal procedure coordinated by the ASN will conclude with an opinion from this authority, based on the appraisal of the dossier by the IRSN, the report of an international expert review and the opinion of the permanent groups for waste and for laboratories and plants which will decide on the basis of the IRSN's appraisal. The missions of permanent groups are defined by referral to the ASN on 2 August 2016.

Meanwhile, the law of 25 July 2016 was passed, very soon after the DOS was submitted; it took note of the changes in the stages of the procedure, in particular the postponement of the DAC to mid-2018, defined the contours of a industrial pilot phase and provided the first elements for defining reversibility.

APPENDIX IX:

TREATMENT OF UNCERTAINTIES IN THE SIZING OF CIGÉO

The description of the future behaviour of a repository site is necessarily subject to uncertainties. Such uncertainties are natural and must in no case preclude a project simply by their existence. On the contrary, it is advisable to integrate them into the definition of the project in order to arrive at a robust solution.

The uncertainties are mainly divided into two groups:

1) Uncertainties in the description of the environment (including the structure and the geological formation hosting the structure in question). At this stage, the uncertainties relating to the measurements themselves may be distinguished from the uncertainties associated with the fact that the measurements are not always representative of the scale characteristic of the environment studied.
    a. Uncertainties in the measurements. Measurements are often carried out on very small-sized samples (a few cm^3) extracted from the geological formation. A first step at this stage is to repeat and compare measurements of the same property on the same sample in order to assess the uncertainties of the measurement technique. Such an analysis can be tricky when the properties studied fall outside the classical range of measurements. This is the case with COx permeability, which is very low. The techniques used in this case do not necessarily lead to reproducible results. Moreover, when a sample is extracted from the geological environment to be analysed in the laboratory, its environment changes (pressure, stress, temperature), which can modify the properties to be measured. Finally, some experiments designed to assess the permeability of a sample may involve the swelling of particles at certain points in the porous network, not always the same ones, which directly influences the estimated permeability value. Moreover, sample properties can be modified because its extraction from the geological formation involves damage. Some properties are not measured directly, but deduced from other measurements: for example, impedances are deduced from seismic data. Since the measurement is not direct, an inversion is required which leads to the impedance values as well as an estimation of the associated uncertainties;
    b. Uncertainties related to measurement scales. The description of the properties of the geological formation cannot be exhaustive, since the sampling is carried out at a limited number of points. It is therefore necessary to go from measurements made on small samples to the much larger scale of the geological formation. The uncertainties observed at the sample scale then spread to the large scale: a property measured on a sample may be relatively different from the value obtained for a neighbouring sample. This is compounded by uncertainties dependent on the change of scale: for example, they are due to the lack of spatial uniformity of the properties studied. Indeed, behaviour observed at a small scale is not always directly transposable to a large scale. For example, we can investigate the estimation of permeability or Young's modulus at small and large scale. Some properties are measured only on a small scale, others on different scales. It is necessary to be able to relate the properties to each other and to quantify the uncertainties in both in order to extend them to the large scale;
2) Uncertainties linked to models that do not exactly take account of the different physical-mechanical-chemical processes affecting the environment. Once the environment is described, the modelling chain that will simulate its behaviour following certain stimulations must then be constructed. The mechanical behaviour of the environment must then be determined or the propagation of radionuclides in this environment monitored. Taking into account the different physical-mechanical-chemical processes in the modelling chain is subject to uncertainties: the description of these processes is necessarily approximate, it involves couplings that are sometimes difficult to process numerically, space and time scales are very important and changes in scale, often difficult to define numerically, are

necessary to move from the sample on which measurements are made to the mesh of the numerical grid that represents the environment.

These uncertainties are compounded by a third type which refers more to hazards or risks, i.e. rare events for which it is difficult to determine the probabilities of occurrence.

In Cigéo, the role of seismic data as a source of information has appeared to be fundamental for the construction of 3D models representative of the geological formation comprising the repository site. However, the uncertainties associated with seismic inversion still need to be taken into account and propagated along the modelling chain. The implementation of a sensitivity analysis would facilitate a clear-cut decision on the essential parameters, such as Young's modulus.

APPENDIX X:
MIGRATION OF RADIONUCLIDES

Chemical couplings and co-storage in LLILW cells

The 79 families of LLILW in the PIGD are classified, from the point of view of their physico-chemical properties, into 7 categories. In determining suitability for co-storage in the same cell, Andra has identified packages of the waste families belonging to LLILW categories 4 and 5 (no organic matter, no salts, whether or not cemented) and is examining other possibilities, such as co-storage of LLILW categories 5 and 6 (waste with compacted and vitrified structure, non-exothermic).

The co-storage of packages of different categories in the same cell is being studied with a view to optimising Cigéo. Co-storage can only take place if there is no interaction between the radionuclides and the degradation products of the materials of the packages in the vicinity, which would induce effects not taken into account in a scenario without co-storage. The phenomenon feared in the very long-term is greater migration of radionuclides due to increased solubility of the chemical species and a lower retention of these species in the components of the repository. It depends, within the cell, on the extensions (and the speeds) of the migration plumes of radionuclides from one package and degradation products from another package. If these plumes overlap at a given time, there may be interaction if permitted by the concentrations of the organic molecules.

The spacing between two cells containing different categories of waste, not suitable for co-storage, is also part of optimising the repository. Andra is studying this new possibility by using the same criterion of superposition of the plumes to optimise cell spacing.

Packages capable of generating products disturbing the diffusion of radionuclides are those containing organic substances (LLILW category 3) or salts (LLILW 1). Organic molecules resulting from the degradation of organic matter by radiolysis form complexes with actinides and other radionuclides, the salts changing the ionic strength, which can modify the migration parameters.

For a decade, Andra has been acquiring data to simulate the phenomena involved: predominant organic molecules produced by radiolysis, source terms of packages in radionuclides and organic molecules, parameters controlling their migration and parameters controlling the formation of complexes between radionuclides and the organic molecules. There are also international thermodynamic and kinetic databases for these parameters and R&D projects are under way to complement them.

Andra has identified isosaccharic and phthalic acids from the degradation of cellulose and PVC as molecules with a strong complexing power for the tetravalent (Pu, U) or hexavalent (U) actinides. These acids are sorbed on cement-like materials or the COx and form complexes above a critical concentration in solution estimated at $10^{-3}/10^{-4}$ M. Andra considers that the available data must be extended to include sound and degraded cement-like materials and that it is necessary to test possible competitive effects between carboxylic acids. It wishes to have this data for the DAC.

Andra has simulated intra-cell migration of acids and uranium and migration between two cells filled with packages rich in cellulose and PVC (CEA-080 packages) and uranium-rich packages (COG-120). The calculations are made according to the methodology used in the safety scenarios (see the DOS) with mostly measured parameters. In co-storage, the plume of phthalic acid is very extensive and the migration of uranium is greater than in a situation without co-storage. It remains limited to a few metres in all three dimensions. For storage of the packages in two adjacent cells filled with different packages, a spacing of 30 m prevents any interaction. These simulations also provide information on plumes according to the cross-sections of the cells (65 or 110 m^2).

The problem of the co-storage of LLILW packages and, more generally, the problem of optimising the diameter and the spacing of the Cigéo LLILW cells are very important questions. They can only be addressed by simulating the migration of the species carrying the radioactivity.

Migration of radionuclides during flow transients

The quantification of radionuclide fluxes (in the surface-to-bottom connections and in the COx) is based on a set of simulations. Usually, these simulations ignore transient phases by considering that the system is in saturated conditions, with hydraulic equilibrium as soon as it is closed. These studies provide indications on typical migration flows and times. The objective of the research presented by Andra is to refine these predictions by taking into account the transient phases of resaturation.

The complementary studies carried out by Andra have thus tested the impact on the migration of radionuclides from the consideration of large transients: the return to hydraulic equilibrium of the formation, the thermal transient (and coupled effects on hydraulics and mechanics), the hydraulic-gas transient (resaturation), chemical degradation of packages and interfaces. These impacts are quantified by comparing the flows through different outlets (exit from areas, exit from the Callovo-Oxfordian and surface-to-bottom connections) to the flows calculated according to the reference scenario for which saturation intervenes as soon as the repository is closed.

In steady state, transfers of radionuclides are limited by diffusion in the COx. The characteristic transfer time of the repository through the protection of the COx is of the order of 800,000 years. Transfers within the infrastructure of the structure after closure are also limited by diffusion, except in the central area where convection is of the same order of magnitude as diffusion. Under these conditions, the transfer time from the cells to the surface-to-bottom connecting structures are of the order of a million years. Due to the characteristic transfer times and contact surfaces developed with the host rock, Callovo-Oxfordian transfer is predominant.

The characteristic times of the thermal (and thermo-hydro-mechanical) transients lead to a peak temperature and pressure well before the release of the radionuclides. The increase of the transfer properties under the effect of temperature (diffusion coefficient, permeability) is therefore limited and homogenised. The THM impact on transfers of radionuclides is therefore not significant.

The resaturation transient creates convergent water movements towards the structure, which oppose the transfer of radionuclides to the host rock. The presence of technological voids, mainly in the LLILW cells, further prolongs the duration of this effect. In spite of the strong gradients between the already saturated cells and the galleries being resaturated, the transfers remain limited due to the reduction of permeabilities under unsaturated conditions at the head of the cell. The integration of all the effects of the resaturation transient delays the transfer of radionuclides (about 300,000 years) and decreases the flux (by a factor of 2 to 3).

Finally, most chemical degradations have a limited effect on areas very close to the structure and have no major effect on transfers. The saline perturbation associated with the high ionic charge of certain LLILW packages has a greater effect with a change in water flows (hygroscopic and osmotic effect), limited to a few hundred years. The increase of the anionic diffusion is limited to a few metres in the argillites over a duration of less than 10,000 years. The impact on the outflow of the Callovo-Oxfordian is negligible. Taking into account the complexation with organic compounds modifies the mobility of the solutes about fifteen metres around the structure and therefore does not modify the retention of radionuclides at the scale of the repository.

APPENDIX XI:

ASTRID AND THE GENERATION IV DEPLOYMENT SCENARIO

Studies carried out in the framework of the 2006 law by CEA, EDF and Areva show that Separation and Transmutation (ST) is only possible if France acquires a fleet of Gen IV FNRs. The implementation of the Astrid programme is imperative to initiate such a deployment. This can only be progressive and will extend over many decades, as indicated below.

R&D for the detailed preliminary design of Astrid

The R&D for the detailed preliminary design is essentially aimed at increasing (2017-2019) the level of maturity of the components of the Astrid configuration fixed at the end of 2016 and bringing the version of the nitrogen energy conversion system (ECS) to the same level as the water-steam ECS. The CEA has reviewed the assembly handling chain, the installation of the gas ECS, the installation of the hot cell, the cooling systems, the configuration of the core for a Pu fuel from PWR MOX and the optimisation of all the components. The technical specifications are amended accordingly. On this basis, the CEA has begun to study the methodology of cost evaluation.

The CEA has established an R&D roadmap that goes beyond the detailed preliminary design. The collaborative R&D in France can be relocated (materials, instrumentation and measurement systems, modelling of the core and serious accidents), and it can also be carried out with experimental resources available abroad. Under the detailed preliminary design, it is scheduled with EDF, Ardeco and French academic laboratories and foreign bodies (Japan, India, Korea, Russia, USA, Kazakhstan). However, R&D for component qualification requires testing platforms, some of which will only be functional after 2020. Only Giseh for water thermohydraulic studies (water at 50°C simulates liquid Na at 450°C) and Papirus for tests with Na in small quantities are operational. Cheops, Plinus 2 and Masurca will be after 2023. For the time being, the CEA is using Giseh and Papirus and is finalising the design of Cheops (qualification of the Na-N_2 heat exchanger), Masurca (FNR core mock-up) and Plinus 2 (serious accidents). Trials on these platforms will take place during the Astrid consolidation phase.

The design of the Na-N_2 exchangers (ECSG) and the gas ECS, that of the corium collector and the development of sensors for Astrid's continuous in-process monitoring have been the subject of important advances. The CEA has not identified any obstacles in the design and integration of the gas ECS in Astrid's design.

The collector is a component which, in case of meltdown of the core, must accommodate material at a very high temperature (> 3000°C). It should be coated with very pure zirconia protective parts. However, for the detailed preliminary design, the CEA is examining a simpler implementation using refractory metals (tungsten, molybdenum). The coupling between the core of Astrid and the collector to conduct the corium (transfer tubes) has been sized and the whole assembly is optimised.

Instrumentation for in-service monitoring is critical for safety. The CEA is developing new sensors or improving existing sensors (with signal acquisition and signal processing chains and related measurement systems) to improve the monitoring of core characteristics and to monitor all ex-core parameters (mechanical deformations, sodium leakage, etc.).

R&D for the PWR and FNR Mox cycle

For reprocessing, the PWR and FNR MOX assemblies will be transported to the shearing workshop as a first step. In 2014, waste producers therefore launched the “Transport and storage” project to understand the evolution of the properties of assemblies (UOX and MOX) during sudden changes in temperature followed by maintenance at constant temperature: T of the sheaths around 50°C, internal pressure 80 bar in the pool and temperature of the sheaths 420°C, internal pressure180 bar during their dry transport (lasting about 1 month). For these latter conditions, there is no risk of creep under internal pressure or of weakening of the sheaths (by reorganisation of the zirconium hydrides) during cooling. In the pool, no corrosion of the sheaths or assembly structures is to be feared because the chemistry of the water is controlled. Production of helium over the long term increases the internal pressure in the sheaths, but it is still far from their nominal breaking limit. Tests to recover assemblies under accident-handling conditions are in progress, to evaluate how many sheaths would be broken. Finally, the rate of leaking sheaths (0.4% since 1987) has not changed over at least fifteen years. Measurements are continuing to verify that the phenomena of alteration of the $UPuO_2$ oxide by pool water are in accordance with well-known models (Precci programme conducted between 1999 and 2006 and international data on the direct storage of spent fuel). The effect of oxide formation could lead to swelling of the sheath at the point of defect.

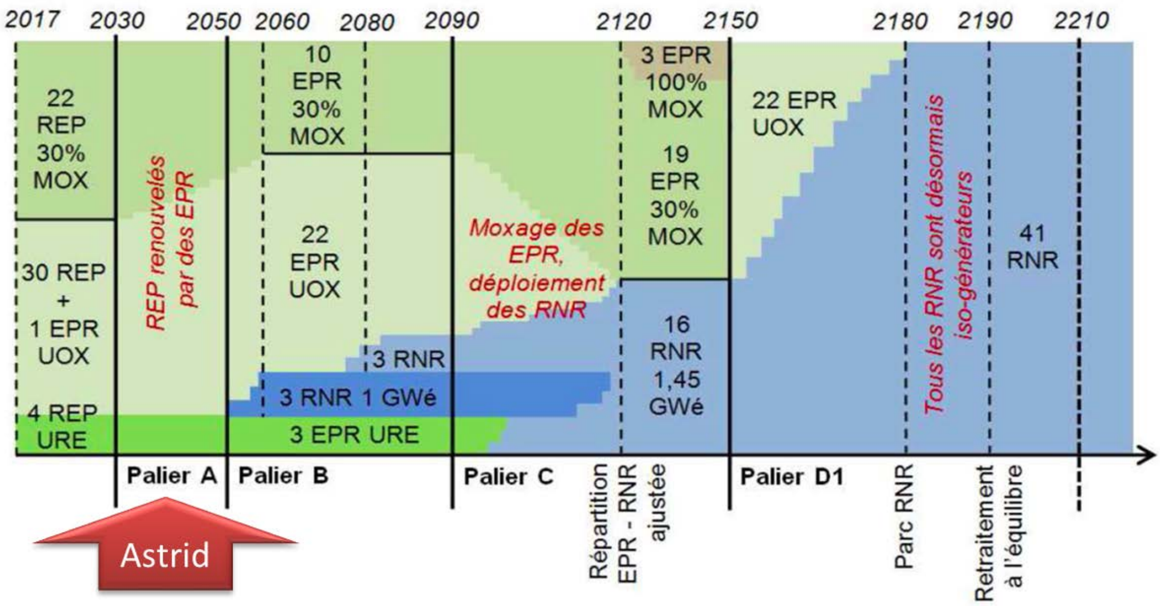
Replacing the Purex process with a single extraction cycle using a single Pu(IV) and U(VI) extracting agent is a key point. Indeed, the Purex process applied to solutions rich in plutonium requires the use of large quantities of reducing agent such as hydrazine, which increases the chemical risk by release of nitrogen oxides. The CEA is experimenting with the use of a complexing solvent mix of DEHBA, DEHiBA and TPH which extracts Pu(IV) and U(VI) from a 4M HNO_3 solution and extracts only U(VI) for an HNO_3 concentration of 0.5M. Thus, by simply experimenting with the acidity of the solutions, plutonium can be separated from uranium as well as fission products and minor actinides. The CEA has studied the main parameters of the extraction system in the laboratory and successfully carried out a test in 2014 and 2015 on a dissolved solution of spent UOX fuel (Atalante CBP cell, patent application). The plutonium and the uranium obtained are very pure, namely 99.96 and 99.99%, respectively. Uranium and plutonium can also be recovered together in adequate proportions to fabricate MOX directly. Numerous improvements and optimisations have yet to be made to qualify this process (MOX passage, resistance of the solvent to radiolysis, column hydraulics, various modellings, etc.). Other mono-amides, alone or in mixtures, would be capable of extracting uranium and plutonium even more efficiently and a selection based on molecular screening and theoretical calculations is under way.

Scenario

The advantage of FNRs over thermal neutron reactors lies in their ability to function with Pu regardless of its isotopic composition. They also make facilitate fission of the americium and the curium because the ratio $\sigma_{fission}/\sigma_{capture}$ for the fast neutrons is superior (regardless of the isotope of U, Pu and Am) to that for the thermal neutrons; this is supplemented by better fertilisation of ^{238}U because of the number of secondary neutrons emitted by the fission of ^{239}Pu. The multi-recycling of the Pu in the FNR makes the isotopic composition of Pu uniform.

For EDF, the transition to a Gen IV FNR fleet as defined in 2015 with the CEA and Areva will last a century and will be mainly linked to changes in economic conditions. EDF supports Astrid, which must prepare this transition, both for the reactors and for the installations of the cycle, and which will ensure continued know-how in these areas. The first FNR can only be commissioned about 25 years after the start-up of Astrid, that is to say around 2060/65, and this development is conditioned by the access to Pu and U recycling capacities. To deploy the FNRs, these recycling capacities will have to evolve towards multi-recycling of the FNR MOX. In the case of americium, its transmutation will require mobilising all the FNRs of a future fleet (assuming a 63 GWe fleet).

The following figure describes, according to the CEA, EDF and Areva, the stages involved in the transition of the current fleet to an FNR fleet via a mixed fleet comprising EPRs and FNRs.



The increase in the percentage of FNRs corresponds to:

- a reduction in the need for natural U,
- an increase in production of Pu at the rate of 7.1 t/year, this production is stabilised if the fleet comprises 40% of FNRs,
- the need for an annual increase in Pu recycling (up to 75 t/year for 100% FNR),
- a continuous increase in the production of minor actinides at the rate of around 3.1 t/year then 2.2 t/year for 100 % FNR.

The closure of the cycle implies consumption of plutonium. The use of FNRs as Pu burners could modify these data at the cost of several major changes:

- an increase in the Pu content in the FNR MOX (CAPRA fuel, 35-45%),
- the modification of the rods (replacement of the ^{238}U fertile zone by MgO),
- the reorganisation of the CFV core assemblies and the management of the FNR.

The difficulties in showing that this is possible with Astrid are real. Various scenarios are under study.

The Board points out that Astrid's place on the above figure would be just before 2050. If it is not built at this time, the roll-out of the succession of fleets will be delayed. The scenario of deployment of an FNR fleet increasingly loses its credibility as the construction of Astrid is delayed.

From Astrid to commercial FNRs

In the deployment of an FNR fleet, EPRs and FNRs will coexist for a long time, in the same way as the reactors in the current fleet will coexist with EPRs. Furthermore, it is essential that France can continue to reprocess spent fuel. In other words, there must be continuity in the fleets. It is important that the existing fleet and the associated fuel cycle retain their characteristics to ensure the transition. This is why the Board monitors the R&D that is being carried out on this fleet.

The Board notes that by 2045 the last 900 MWe PWR put into service will be 60 years old. The La Hague reprocessing plant, meanwhile, should operate until 2040 when it will be 50 years old. In 2040, France's strategic reserve in the pools at La Hague will be 5000 t of PWR MOX (300 t of Pu, 40 t of Am). In the case of new fuel, the Melox plant (50 years old by 2045) is authorised to produce 195 t/year of MOX at 8% Pu for civil use, but it produces only 145 t/year (it had produced 2400 t by end 2014). These dates are to be put into perspective with the option of recycling Pu. After the decommissioning of the 900 MWe PWRs, the use of plutonium-containing fuels can only continue in EPRs and then in the longer term in FNRs. But the evolution of the composition of the nuclear fleet is not fixed, only its power is limited to the current value.

The current R&D for the fleet is essentially aimed at improving safety post Fukushima. R&D on nuclear safety and radiation protection was re-launched after Fukushima by initiating 20 RSNR projects (7 led by IRSN, 5 by the CEA) on the PIA fund of the Commissariat Général d'Investissement. The call for tenders used using the structure and terms of the ANR. Furthermore, Europe and the IAEA have launched their own programmes. The R&D topics focus on the following improvements:

- evaluating extreme natural events,
- preventing core meltdown accidents,
- understanding and monitoring the progression of core meltdown,
- monitoring and stopping the piercing of the concrete slab by the corium,
- preventing the risks of hydrogen explosion and corium-water interaction,
- including accidents in spent fuel storage pools,
- understanding the containment of radioactive products,
- improving crisis management software,
- improving human and organisational capacities for the management of serious accidents.

All these studies and results help to preserve the knowledge and tools for the next generation of reactors.

Due to the decreasing budgets, the civil nuclear activities of the CEA are constrained and are being reprogrammed. The short-term view of the authorities, which strongly impacts R&D and prohibits long-term planning, is to be deplored.

APPENDIX XII:

SEPARATION OF AMERICIUM AND TRANSMUTATION

All the studies converge today (see report CEA 2012) in considering only the transmutation of americium. Curium transmutation is almost impossible because of the radiological protection imposed by its handling on an industrial scale. Neptunium is of little interest because its thermal impact on the repository and its radiological impact are low. Finally, it has long been recognised that the transmutation of fission products is unrealistic, in particular because prior isotopic separations are required.

Interest

Front the point of view of disposal of high-level waste, the interest of separation and transmutation (S&T) lies in the reduction in the thermal load of the CSD-V waste packages of a future FNR fleet through the elimination of ^{241}Am by fission. This implies reprocessing the assemblies of spent FNR MOX fuel, beyond the extraction of U and Pu, to extract the Am. In principle, the use of U and Pu to self-power the FNR fleet will be fast (5 years) and this will limit the Am quantities to transmute, because the longer it takes to reprocess the FNR MOX, the more ^{241}Am and ^{243}Am it contains. The S&T will lead to CDSD-V packages containing only fission products, Np and Cm and the losses in Am from the EXAm process and in U and Pu from the Purex (or other) process. The footprint of the high-level waste for a repository will go from 150/170 to 20 m^2/TWhe (deduced from studies for Cigéo). The doses for scenarios of accidental intrusion into the repository would be significantly reduced.

The S&T strategy involves many stages:

- the implementation of the EXAm process, the conversion of Am oxalate to AmO_2, the fabrication of $UAmO_2$ pellets for the AmBB, etc.
- then, after irradiation, the reprocessing of the irradiated AmBB, the re-fabrication of the AmBB...

The EXAm process can only be implemented after the reprocessing of the FNR MOX, which itself requires modifications to Purex or the development of another process. The priority is therefore to develop an industrial process for reprocessing the FNR MOX, taking into account the constraints linked to the fact that it will be followed by the EXAm process. On the industrial level, one and more probably two plants will be needed to implement these separations.

Where is the R&D to demonstrate the feasibility of S&T in Astrid?

To demonstrate the feasibility of transmutation in Astrid it is necessary to fabricate and irradiate, according to the CEA, 1 assembly of $UPuAmO_2$ fuel (2 % Am, homogeneous) and 3 assemblies of $UAmO_2$ (10% Am, heterogeneous). The design of Astrid allows the introduction of such assemblies without disturbing the neutronics. The Astrid programme, as conceived, allows going as far as examination of the rods as well as qualification of the behaviour of the AmBB. This implies having facilities for recovery of Am (extension of the FNR MOX reprocessing plant) and for fabrication of the AmBB (extension of the fuel fabrication plant). Atalante allows for the preparation of experiments up to rod level.

To demonstrate the capacity for “destruction of Pu”, it is necessary to implement 7 to 8 Capra fuel assemblies. This is theoretically possible in Astrid, but at the cost of drastic modifications. It is necessary to change the assemblies (rods without fertile material and of small diameter, inert rods); this has an impact on the control of the reactor and therefore on the safety. Similarly, the fuel cycle will be profoundly modified and in-depth preliminary studies will be necessary. The qualification of an assembly type can take 30 years. Under optimum conditions, an FNR of 1GWe burns 50 kg Pu/TWhe.

In terms of R&D, it has developed efficiently over more than 20 years (fundamental research and R&D) and there is an important knowledge base.

a) EXAm process and Conversion

The “full EXAm” experiment in Atalante has been under way since 2010, to recover 2 to 3 g of Am in the form of $UAmO_2$. In principle, by the end of 2018 the CEA will have sufficient U-Am mixed oxide, by combining mixed oxide from the full EXAm experiment (58.4% of ^{241}Am and 40.9% of ^{243}Am) and 3 batches of mixed oxide already prepared (with ^{241}Am) to make 4 mini-rods. These mini-rods, foreshadowing the AmBBs, will be irradiated in the ATR (USA) after 2019. This will be Technical Readiness Level (TRL) 5/6. In 2013/2014 the CEA planned to recover 7 g of Am to prepare 20 to 30 $UAmO_2$ pellets (Appendix VIII of report No. 8 page 71).

The EXAm process was developed in Atalante. The full EXAm experiment began in 2010 with the dissolution of 3 kg of UOX and 1.6 kg of MOX. At the end of 2011, the CEA had, after extraction of U and Pu by the Purex process, 24 L of 3.4M HNO_3 solution containing Am, Cm and the fission products including the lanthanides. In the EXAm process, Am and Cm are extracted from an 8M HNO_3 solution, which required concentrating the solution. The CEA carried out this difficult operation in 2015 by steam distillation of HNO_3 and obtained 4 L of 8M HNO_3 solution. The EXAm process was piloted to obtain very pure americium (95.5%) to the detriment of the Am recovery factor and Am/Cm and Am/lanthanide decontamination factors. The resulting 2.4 g should be converted to AmO_2 this year. The fabrication of $UAmO_2$ oxide with a high density (> 94% of the theoretical density) and a controlled porosity (15%), by various processes, is controlled by the CEA.

b) AmBB fuel

Numerous experiments have been carried out on the irradiation of various oxide candidates to become transmutation fuels of the minor actinides in homogeneous mode (FNR fuel, U, Pu, low content of minor actinides) or heterogeneous (FNR fuel, U, minor actinides, or ADS Pu fuel, minor actinides, high content of minor actinides). The aim was to test their suitability for irradiation (amorphisation, restructuring) and helium release for fuels with a high minor actinide content. Today, only transmutation of Am is envisaged, in the AmBB ($UAmO_2$) fuel at 10-20% Am. The AmBB assemblies will be placed at the periphery of the FNR core for a period currently expected to be 7 years. Under these conditions, the temperature would not exceed 1500°C. The Germinal code, developed taking into account changes in the properties of the $UPuO_2$ pellets and the rods of the FNR MOX assemblies, is adapted to the $UAmO_2$ oxides. It allows prediction of AmBB behaviour and the design of a rod and its behaviour under various conditions. Several oxides in the series $U_{(1-x)}Am_xO_{2\pm\delta}$ with d > 95% of the theoretical density and a good porosity (15%) have been prepared, irradiated and are under examination. As for the AmBBs, the latest experiments are Marios (irradiation of $UAmO_2$ at 15% Am in HFR in 2011-2012) and Diamino ($UAmO_2$ at 7.5 and 15% Am, irradiation in Osiris in 2014-2015). The Marios mini-rods and the Diamino discs are under examination at LECA (Cadarache). The CEA expects data from these experiments on oxide swelling and gaseous release as a function of temperature (600 to 1200°C). The Marine experiment which targets the behaviour of $UAmO_2$ oxide at 13% Am in the form of pellets or spherules is under way (irradiation in HFR). Experiments planned in ATR on $UAmO_2$ oxide mini-rods target the coupled behaviour of the pellets and rods. They will run in parallel with American experiments on metallic UAm fuel. R&D on transmutation fuels is carried out in numerous Euratom programmes (Fairfuels, Pelgrimm, etc.) and in the numerous collaborations that the CEA has established (ITU, DOE, etc.).

c) AmBB reprocessing

The EXAm process must be adapted. For the moment, there is no R&D on this subject.

APPENDIX XIII:

CLEAN-UP, DISMANTLING AND WASTE RECOVERY

This appendix provides an update on changes in the long-term or projected clean-up and dismantling (C&D) operations of CEA and Areva facilities, as well as recovery and conditioning operations of waste stored in these facilities.

Clean-up and dismantling (C&D)

C&D presents two essential aspects, the definition of the objectives and the strategies implemented to achieve them.

The reference objective is that of the ASN, which aims to achieve a final state of complete C&D of facilities and total decontamination of the soil. In relation to this ideal situation of “decommissioning to greenfield”, the CEA, EDF and Areva are looking for a multicriteria optimum (technical, safety, socio-economic) that is acceptable for the intended uses of the buildings. In fact, the C&D is therefore partial or more or less extensive according to the objectives sought. The common minimum objective is the decommissioning of the basic nuclear facilities. Strategies depend on the organisations. Each C&D programme is unique and there is no serial effect.

Areva shows the fastest possible C&D strategy for decommissioning basic nuclear facilities, with industrial re-use of buildings or, in addition, soil remediation, both with easements if necessary. Areva follows a continuous C&D logic with a choice in the priority of actions. Areva's big project is UP2 400 (4 facilities in C&D at la Hague) then the GB1 plant (2 facilities in C&D at Tricastin).

The CEA targets the shortest possible times for C&D after shut-down of facilities. Depending on the complexity of the sites, the operations can take place in several stages with intermediate reuse of the buildings. The CEA has more than 100 C&D projects (DEN and DAM) spread over 40 years which require 20 billion euros and several decades of work. For budgetary reasons, the CEA has established priorities in C&D actions. Currently the budget is 0.74 bn/year for 32 installations (basic nuclear facilities and others) in the course of C&D. The diversity of the cases treated or pending has led the CEA to develop multiple skills in C&D. In the absence of an LLLW channel for graphite, the G2 and G3 sites have been stopped.

Since 2001, EDF has also had the fastest possible C&D strategy for the first PWRs. EDF has set up a deconstruction and waste project management and has commissioned Iceda (a dismantling waste conditioning and storage installation). Iceda is to accommodate LLILW waste from UNGG and Chooz A and LLILW from plants in operation (control clusters). The dismantling of the Chooz A reactor (300 MWe) will be complete by 2025 with a period of monitoring of the percolation water until the caves are filled. The dismantling of the UNGG reactors has been postponed due to significant changes in the dismantling techniques for these reactors. The UNGGs will all be dismantled under air, beginning with Chinon A2 (ending around 2060). The pressure to find an outlet for the LLLW graphite stacks of the UNGG reactors has thus become less strong than it was last year. The dismantling of Brennilis and Superphénix continues.

Andra is undertaking remediation of orphan sites polluted with radium. They are very numerous and, because of insufficient credits, Andra cannot go as far as total de-pollution for all of them. After consultation with the authorities, its actions seek to control the health impact by eventually establishing easements.

Waste generated by C&D operations is reduced to a minimum (zoning, sorting, recycling, volume reduction). The dismantling waste still to come is VLLW (60% of all VLLW at termination, concrete, scrap metal, rubble), LILW-SL (40%, reactor core or process equipment), LLLW (30%, graphite stacks) and LLILW (10%, metallic parts from near the cores). The balance of future waste corresponds, in all categories, to process waste.

Waste recovery and conditioning (WRC)

The CEA and Areva have given a very detailed account of the numerous WRC operations they conduct at the Cadarache, Marcoule, Fontenay aux Roses, La Hague and Pierrelatte sites. These operations are in progress and at very different stages of advancement. In order to restore coherence to its overall WRC (and C&D) strategy, the CEA has introduced a new governance by creating a dismantling directorate for civil plants. This new directorate should define priority projects and strengthen their implementation. It is the non-conditioned waste that has to be recovered as a priority, either to ensure the safety of new storage or to be disposed of through existing channels. Waste already conditioned in packages, even temporarily, does not compromise the safety of the facilities.

a) Cadarache

VLLW, LILW-SL and LLILW are at INB 56, they are either liquid or solid, in bulk or in packages. The liquid effluents will all be removed to Agate and the STEL at Marcoule. Packages are to be recovered and placed in larger containers. Bulk waste will be cemented into INB 37. The conditioning planned for the LILW-SL and LLILW is ultimately cementing according to specified procedures. There is no real new R&D. On the other hand, there are significant improvements to be implemented for recovery and characterisation/sorting before conditioning of 350 m^3 of bulk irradiating waste in 4 pits and the characterisation/control of 10,000 packages stored in hangars. The end of operations is planned for 2040.

b) Marcoule

All the waste to be packed comes from the operation of the UP1 reprocessing plant (1958-1983), other facilities (APM) and reactors (Célestin, Phénix). The CEA distinguishes the WRC of the bitumen packages from the STEL from the WRC of the other wastes.

♦ Bitumens

All the LLLW bitumen packages from the north zone (38 pits) have been reconditioned in 6048 packages and have been at EPI 1 since 2008. The 14 hot cells from the south zone contain 28,853 LLLW packages, 28,481 LLILW packages and 2300 LILW-SL packages (called FR packages, which can be stored at CSA as from 2018). For the moment, the WRC concerns the packages from hot cells 1 and 2: 375 FR packages have been recovered and stored in Casemate 14 and 4350 LLLW packages have been stored in EPI 1. 600 packages have yet to be recovered by the end of 2017. EPI 1 contains 10,500 LLLW packages for a capacity of 12,000. The Board visited these facilities.

The continuation of WRC of the bitumen packages of the other hot cells (52,000 packages) requires major installations (guided recovery machinery, packaging enclosures, transport machinery) and the construction of the EPI 2 and 3 storage facilities. It is programmed as from 2020 and should be finished in 2040.

♦ Other waste

This is process, structural and operational waste, stored in the north zone (mixed waste from various sources, 2740 packages, 60 containers, bulk, 10% LLILW), in the south zone of the former UP1 plant and, in 3 decladding workshops, several UNGG fuel models (G1, G2/G3, MAR 400). Other than the magnesian waste (see above), these workshops contain powdered LILW-SL and LLILW waste (379 and 222 t), non-magnesian metallic waste (103 t) and LLLW graphite (759 t).

A WRC operation is under way in the former UP1 plant where a UCDA facility permits the packaging of alpha waste for storage in Cedra: 274 drums out of 1240 have been recovered. The other radioactive materials in the building are recyclable and will be sent to Magenta (Cadarache) or Valduc. The end of the operation is planned for 2020. With regard to the decladding workshops, a recovery installation (conditioning by cementing powdered LILW-SL waste from G2 / G3, UDH) is operational. The recovery of magnesian waste from G1 requires the installation of a recovery unit, planned for 2020. The end of the recovery of all waste from the decladding workshops which requires the creation of several (programmed) conditioning units is planned for 2030/2040. The Board visited the decladding workshops of G2/G3 and UDH, UCDA and room 60 (MAR200 workshop of UP1).

Another operation is under way in the north zone, concerning the recovery of packages from 2 STEL pits out of the 32 pits still filled. In order to continue these operations, the CEA has to build recovery units and a new processing plant (commissioning in 2025).

Finally, the 12,500 LLLW and LLILW packages expected from the WRC will be stored in a new EIP (without bitumen) programmed for 2025.

c) Fontenay aux Roses

All the drums of LLILW (1200 drums), more or less irradiating (MI - Moderately Irradiating and HI Highly Irradiating), resulting from activities on the reprocessing of irradiated or spent fuel (1960-1995) and the clean-up of armoured enclosures, are stored in building 58. The CEA is recovering the MI to send for conditioning by cementing to Cadarache in INB 37 (with some going temporarily to Cedra). The highly irradiating drums (HI) will go to Marcoule in Diadem (under construction). To undertake these operations, the CEA will build, in the extension of building 58, dedicated measurement and conditioning equipment. Studies are at the end of the detailed preliminary design stage. Recovery of all the drums is planned for 2026. The end of dismantling the facilities in building 18 is scheduled for 2030, and building 58 for 2035.

d) Pierrelatte

Over 14 years (1996-2010), the gaseous diffusion plants were brought to level 2 on the IAEA dismantling scale (walls and metal structures of the buildings decontaminated). The equipment (diffusers, pipework, valves, cables) and process waste have been processed and conditioned. Since 2004, the VLLW: large metallic components and compressed metallic waste including 8000 t of aluminium alloys, big-bags of various waste, have been and will be sent to Cires. Process waste, diffusers and cement diffusion barriers were and will be stored at the CSA. 16,000 t of VLLW (20,000 m^3) and 1,100 t of LILW-SL have been removed. There remains 15,000 m^3 (drums, valves, big-bags) in place and 5,000 t to be conditioned.

e) La Hague

The priority WRC projects are the waste from the HAO and SOD silos (1500 t shells, tips, metallic debris, dissolving and shearing fines and resins), in bulk in HAO (850 t) and in drums under water in SOD, waste from STE2 (sludge) and waste from silos 115 and 130 (graphite and magnesian). The WRC is closely linked to the search for the best package according to Areva criteria (see report No. 10).

The recovery cell for HAO and SOC will be operational in 2020. The waste will be sorted. The shells and tips, silicone gaskets, alumina pellets and shearing fines will be sent to the compacting workshop at UP2 800 and put in CSD-C HAO packages according to the new specifications with regard to CSD-C packages (LLILW category 3 packages, rich in organic matter). The dissolving fines and resins will be cemented in CFR HAO packages (LLILW category 4 packages, without organic matter) according to a process still under review. The limitation of hydrogen production per package must be controlled. Areva estimates that there are 125 packages releasing 120 L/H_2/year

Trials to recover STE2 sludge and its conditioning in C5 packages (LLILW category 1, salt without organic matter) in STE3 have started, after creating recovery and conditioning facilities. Recovery is expected to take place between 2021 and 2030. The process of drying/compacting the sludge as intended is proving difficult to implement. It must be completely reviewed and submitted to the ASN. The expected number of packages will depend on the final process (between 10,000 and 30,000).

Equipping of silo 130 for the recovery of waste is programmed for the end of 2017. The silo will be emptied as quickly as possible and the waste in drums will be transferred to building 115 for conditioning. The magnesian LLLW from silos 130 and 115 will be conditioned according to a process that will be chosen in 2017 (destruction of the alloys by chemical means or cementing) according to specifications that do not yet exist.

R&D for conditioning of new waste

Areva is working on a universal all-in-one LLILW package based on transport packaging that can be accommodated directly in the repository.

The R&D covers 6 classes of waste containing the following components: graphites, bitumens, polymers, reactive metals, non-reactive metals and cement-type materials. The first four present risks that the processing/conditioning aims to minimise by different means to make them inert, the last two pose only a problem of volume reduction. R&D for the first 3 classes is evaluated below.

a) Bitumens

The R&D concerns the fate of 5500 t (or 14,000 m^3) of LLILW waste (28,831 packages, 2900 t, 38% of the LLILW) and 12,500 m^3 of LLLW waste (32,901 packages, 2600 t, 60% of the LLLW). They are stored at the STEL at Marcoule and after reconditioning will go to EIP storage (WRC operation). STEL is still producing around one hundred drums/year of bitumen mixes. During construction of the bitumen package, the main risk is fire.

- **Bitumen fire risk**

In its reports Nos. 9 and 10 the Board evaluated the results obtained by the CEA on the reactivity of bituminous mixes with a temperature rise (absence of ignition up to 300°C., no overall self-heating by exothermic reactions) and the resistance to fire of primary packages placed in their storage packages (maximum mix temperature of 110°C on the periphery). These experiments were conducted on freshly prepared mixes. The CEA has extended its studies to mixes aged under radiolysis, which differ from fresh mixes through their higher viscosity and the presence of hydrogen bubbles.

The LLILW bituminous packages from Marcoule are subject to an average dose rate of 2 Gy/h, which leads to a cumulative dose over 100 years of a few MGy. Gamma irradiation of 1 MGy at dose rates 100 and 4500 Gy/h increases the viscosity by a factor of 4 but a temperature increase of 20°C decreases it by a factor of 30. An increase in temperature thus rapidly brings the mix back to its initial configuration. The H_2 bubbles migrate to the surface at a rate which depends on their volume and the viscosity of the bitumen. CEA's theoretical calculations show that an increase in the temperature of the mix above 110°C allows all the bubbles of a 100-year-old drum to be evacuated in 1 hour.

All quantitative data on the behavior of bituminous waste in response to a rise in temperature, due either to self-heating, to an external contribution of heat or to their combined effects, relate to inactive synthetic mixes. The behavior of the actual bituminous mixes waste, prepared since several decades and then stored, could be different from that of fresh mixes.

The Commission recalls that CEA, Andra, EDF and Areva undertook experiments on the temperature and fire resistance of bitumen packages at its explicit request (report No. 6). Indeed, the Commission had questions about their behavior (report No. 3) which remained unanswered.

- **Bitumen incineration**

The CEA studied on the Shiva model installed in Marcoule the incineration by a double plasma torch (Ar/O_2) of 4 inactive batches of 20 kg of bituminous mixes and the incorporation of the residues in 50 kg of glass frit. Each batch was representative of the characteristics of the actual mixes. The experiments provide information on 1) the introduction of the mixes into the reactor, 2) the destruction of the salts (10 salts mostly comprising $BaSO0_4$) with installation temperatures of one to several thousand °C, 3) incorporation of combustion residues into the glass, 4) treatment of volatile compounds and dusts.

The introduction of the mixes requires prior heating. However, the high viscosity complicates operations. The combustion of the bitumen is incomplete. The salts are more or less converted into poorly defined solid or gaseous oxides (metal oxides, SO_x, NO_x, CO_x) and the redox reactions of these oxides give metal sulphides. In particular, $BaSO_4$ is poorly decomposed into BaO and SO_2. Sulphates, oxides and sulphides are poorly incorporated into the glass, and sulphides disturb the heating of the glass by induction in the cold crucible. The gas treatment is incomplete and would require post-combustion. Sulpho-nitric oxides are highly corrosive and their recovery leads to acidic solutions to be neutralised.

In these experiments the behaviour of radionuclides has not been studied. In particular the mixes contain more or less enriched uranium and plutonium in variable proportions depending on whether they are LLLW or LLILW, respectively, in ranges of several hundred grams to a kilogram and from a gram to a few tens of grams. The material balance of the process is very difficult to control.

From these tests, considered as prospective, the CEA concludes that it is impossible to envisage an industrial process of incineration-vitrification of bitumens, all the phenomena involved: combustion, decomposition of salts and redox reaction in the gas phase, incorporation of residues in the vitreous network, are incomplete and very difficult to control. The essential difficulty arises from the presence of refractory sulphates (such as $BaSO_4$) carrying radioactivity which cannot be decomposed thermally (to the oxides BaO and SO_2) or which can react with the silica of the glass (transformation to silicate and SO_2) only at very high temperatures.

The CEA nevertheless conducted an economic analysis of an installation which, by continuing R&D, could treat all Marcoule bitumens to lead to LLILW waste destined for Cigéo. The additional cost over storage without treatment is a deterrent for the CEA. Moreover, the CEA considers, on the basis of a BAT study (using the Best Available Techniques for waste management), that a process of incineration-vitrification has considerably more disadvantages than advantages over storage of bitumens as planned (out of 12 criteria considered, 9 are negative, 1 is positive and 2 are neutral). Consequently, the CEA wished to abandon the R&D.

The tests on the Shiva facility were conducted mainly between 2003 and the end of 2005. The CEA submitted a report to the ASN/ASND in 2015 on "technical and economic analysis concerning the evaluation of the chemical or thermal treatment of bituminous compounds in an attempt to immobilise in other matrices the radioactive waste they contain" in accordance with the decree PNGMDR 2013-2015 (article 19. II). ASN issued an opinion in September 2016 on the technical aspects by asking the CEA to continue the studies on bitumen LLLW and LLILW.

The draft decree PNGMDR 2016-2018 (article 48) repeats this request in the same terms, with a delay to 30 June 2018. In addition, article 48 requires the CEA, Areva, EDF and Andra to submit by the end of 2019 a report on "technical, economic and safety assessment comparing the different treatment and packaging methods envisaged for waste bitumen (geological storage and alternative solutions)". This study must integrate all stages of waste management and the impact of different choices on the design and sizing of Cigéo: "transport, storage and operational safety, environmental impacts, long-term radiological impact".

b) Polymers

The R&D involves 1800 t of waste, 50% of which is conditioned but the chemical nature is only fully established for 30%. The risks are associated with the production, by radiolysis of polymers, of hydrogen and corrosive species (HCl) or leading to complexing agents of the radionuclides contained in the waste that modify their migration in the long term.

LLILW waste known as "alpha-contaminated", mixtures of metal waste and organic waste, originates mostly from MOX fuel fabrication plants. They are and will be stored at La Hague (about 3600 m^3, 30,000 drums of 120 L). They contain U and Pu. The CEA, in partnership with Areva and Andra, is developing the Pivic process (Process for Incineration Vitrification In Can) to process and condition them: combustion at 800°C of the organic matter by plasma torch (Ar/O_2) giving CO_2 (and Cl_2) and residues, melting of metals and a nuclear glass with incorporation of the residues in the glass, processing of the gases. Metals and glass are melted in a multi-walled crucible, heated by induction. The waste resulting from the treatment is the Can: the crucible and the two-phase glass-metal solid, the radioactivity being confined in the glass. One Can corresponds to the treatment of approximately 10 waste drums, which leads to a reduction in waste volume by a factor of 8. The Pivic primary waste package consists of two Cans in a stainless-steel container (1 m high, 0.6 m diameter).

The technologies used include: incineration of organic matter, vitrification of radionuclides, processing of gases and metal melting, and the Pivic process is based on feedback from the Valduc alpha incinerator. The R&D began in Marcoule in 2011 with inactive experiments with Shiva. The feasibility of glass-metal fusion in the crucible is being studied on the Erebus module (induction furnace). A Pivic prototype (Erebus plus combustion chamber) is planned for 2018 and a pilot for 2022, which will allow the process to be qualified (TRL 7). The processing capacity of the final installation (around 2035 at La Hague) would be 30 kg/h with a production of one package per day; 1200 packages are expected. Andra considers, based on the preliminary knowledge dossier and an evaluation of the THM disturbance of the COx, that this package could be hosted in Cigéo. In September 2016, ASN gave a favourable opinion on this processing/conditioning on the basis of a preliminary DOS.

The R&D is focused on 1) the behaviour of the actinides (U, Pu, Am) in the reactor and more particularly on the Pu stock in order to comply with the criterion of criticality of this radio element. Experiments on Shiva with CeO_2 in place of PuO_2 show that 3% of the oxide is retained during incineration. This percentage is that expected for the Pu (feedback from the Valduc incinerator), 2) the neutralisation of $ZnCl_2$, a corrosive chloride, by reaction with various sensors (zeolites, phosphates). The Zn and Cl_2 come from organic matter, 3) the feasibility of concomitant glass-metal fusion. The tests in Erebus are satisfactory and show a limit of 50 kg of glass, 4) the formulation of the glass to incorporate mineral residues and additives. It takes place in the laboratory, 5) the corrosion of the walls.

The Pivic package contains ^{241}Pu and ^{241}Am which leads to a low thermal power of a few watts (6.4 W/package) which slowly decreases with the ^{241}Am period. Under these conditions, the thermal transient in the concrete storage package of the primary packages (T <65°C) and the THM (zero effective stress) disturbance that the storage of the Pivic package could induce in the COx should be examined. Andra plans to co-store the 1200 primary Pivic packages with CSD-C packages (physico-chemical compatibility). It has conducted several simulations of storage of 4 primary packages, based on the methodology already used for CSD-V packages, for various powers of Pivic package and several distances between LLILW cells. These simulations show that, in all possible cases, the power of the Pivic packages can be up to 12/13 watts and the temperature of the concrete does not exceed 50°C. Part of the project is financed through the PIA (2010 Investment in the future programme).

c) Reactive metals

Waste containing reactive metals is stored at La Hague and Marcoule.

- In Silos 115 and 130 at La Hague. Approximately 1050 t of graphite (940 t), Mg alloy (MgMn, 64 t), Zr alloy (ZrMg, 14 t), U metal (2.5 t) and steel (1.5 t). They are UNGG fuel reprocessing waste from the UP2 400 plant, not sorted, in dry bulk or under water. They are of LLLW type.
- In pits at Marcoule. Approximately 1650 t of sheaths in Mg alloy (MgZr and MgMn), 850 t are LLILW, 250 t are LILW-SL and 560 t are still poorly identified (mixture of the two categories) As a precautionary measure, 1160 t of LLILW (70%) is counted. These are essentially UNGG fuel sheaths (separated from the graphite) from UNGG fuel reprocessing, whole, crushed or compacted, in dry bulk. In this waste there may be traces of U (or UH3) and graphite.

The risks that could come from improperly conditioned waste packages could be due to excessive hydrogen production.

The chemical production of hydrogen originates from the oxidation of Mg by cement water ($Mg + 2 H_2O = Mg(OH)_2 + H_2$) or by spontaneous electrochemical reaction between Mg and the graphite or various metals (Fe, Zr, etc.) and the cement water. This latter “galvanic” effect may be greater than the first. In addition, the formation of $Mg(OH)_2$ or other hydroxides on the Mg surface can have a mechanical effect on the blocking matrix due to the local increase in volume.

- **Marcoule waste**

The CEA has selected a “geopolymer, Mgéo” mineral binder which is a sodium fluoro-silicoaluminate (without Ca) prepared from metakaolin (aluminium silicate) and a fluorinated solution of sodium silicate. The fluorine prevents the formation of $MgO/Mg(OH)_2$ and H_2. It transforms the oxides/hydroxides appearing on the Mg surface into fluoride and magnesium silicate. This forms a protective layer on the metal. Its composition is to be determined. The rate of formation of H_2 then becomes very low: 0.03 L/m^2 of Mg/year. The same applies for the corrosion of U and for the galvanic Mg-steel and Mg-Zr effects. The geopolymer is easy to use, has excellent mechanical strength, and is resistant to alpha and gamma irradiation and leaching. The study of its short- and long-term behaviour is the subject of a Needs project.

The CEA is testing the fabrication of packages for the magnesium LILW-SL in a concrete container (2.7 m^3, 700 kg of Mg): blocked bulk sheaths. 730 packages (1970 m^3) are planned. Primary packages for LLILW would consist of 3 bundles of compressed sheaths blocked in 220 L (150 kg of Mg) drums. It estimates 7500 packages (1640 m^3)

- **Waste from La Hague**

The R&D is being conducted by Areva and CEA along two avenues: chemical destruction of reactive metals (Mg and U) or control of the reactivity of metals by cementing with a suitable binder.

The first concerns waste processing with aqueous solutions to dissolve magnesium and oxidise uranium to U(IV) oxide/hydroxide. The graphite is not altered. This process requires decontaminating the solutions and generating secondary waste (resins and effluents). The solid residues and graphite are then cemented separately and the effluents discharged into the sea. The advantage is to be able to direct the packages towards the various channels.

The second concerns the formulation of a conventional cement. Since graphite predominates, it is mainly the galvanic effects between carbon and magnesium of the MgMn and ZrMg alloys that are to be feared. They have been systematically examined for carbon but also for steels and for various cement formulations or the pure phases of which it is composed. Ultimately, a slag (40% SiO_2, 45% CaO, 10% Al_2O_3, 5% MgO) mixed with a sodium hydroxide solution was chosen as the hydraulic binder. It limits the production of hydrogen to a very low level expressed in terms of volume per metal surface per year, i.e. at 1 L/m^2 of Mg/year. Given the low proportion of Mg in the waste, the binder contains no corrosion inhibitor. The uranium (0.25%) will be corroded but Areva considers that the hydrogen produced will not alter the mechanical performance of the package. The binder is being qualified for the production of packages at the m^3 scale.

d) Other waste

The Board has not received any new information on the packaging of other wastes containing reactive metallic materials: sodium (carbonation), aluminium (phosphate-magnesium cements) or graphite waste.

APPENDIX XIV:

LOW-LEVEL AND VERY LOW-LEVEL WASTE

Mining waste

Management of tailings is governed by the Mining Code and by several circulars dealing with the use of tailings (on-site and off-site) and on-site return for those that have been dispersed. Processing residues fall under the Environmental Code (EC 511). The quantities and places where they are stored *in situ* and their status are given in the Mimausa inventory drawn up by the IRSN (166 Mt of tailings, 51 Mt of residues from 52 Mt of ore having generated 80,000 t of uranium).

The uranium content of the tailings is up to 200 ppm and this level distinguishes them from the granites (15 to 20 ppm) and ores (300 to 500 ppm). Areva is examining the radiological impact of dumps of tailings at mining sites and has identified the locations of tailings from mining sites following pre-1985 dispersal. Areva has carried out work to restore radiological standards when the impact was greater than 0.6 mSv/year. The cases where it is between 0.3 and 0.6 and some extreme cases are under discussion between Areva and the authorities.

With regard to tailings, the PNGMR 2016-2018 asks Areva to carry out the following:

- Before the end of 2017, to continue the inventory of tailings from mining sites (article 66);
- Before the end of 2019, to complete the processing of tailings (article 70).

Mining residues (from 40 to 310 Bq/g, of which 4 to 30 Bq/g of ^{226}Ra depending on the uranium extraction method) are stored in 17 ICPE-classified repositories. The residues (clay sands or blocks of ores leached by H_2SO_4) are placed on a trellis and are under cover (2 m of tailings, 0.4 m of soil). The percolation water is treated (6000 m^3 / year) either before discharge or to recover part of the uranium. Areva is monitoring the physico-chemical changes of the residues in 4 repositories. The analysis of core samples shows stabilisation of the phenomena of alteration by water and by diagenesis. In addition, U and Ra are trapped by certain mineralogical phases. The uranium is sorbed on the clay minerals and oxy-hydroxides of Fe(III) and also forms insoluble U(VI)/U(IV) mixed phosphates, the radium co-precipitates with $BaSO_4$ and is also sorbed on the clay. Areva has modelled the distribution of U (10^{-5} M) and Ra (1 Bq/L) between the solid phases and the acidic and oxidising water (pH 3 to 6, Eh 800 at 130 mV). In total, U and Ra are not very mobile. This information is essential for modelling the long-term behaviour of mining residues.

Tenorm

Tenorm management is governed by the Environment Code and several circulars and orders, including that of 25 May 2005, which sets the conditions for acceptance in ISDD, ISDND and ISDI facilities. Very little VVLLW is stored in the 4 authorised ISDD and ISDND facilities (10% of 50 million tonnes), 1,600 m^3 should go to Cires and 21,000 m^3 is classed as LLLW. The vast majority is deposited at the production sites. In addition, much material with a low impact during its use (less than 2 microSv/year) has been recovered. The management of Tenorm will be amended following a Euratom directive (2013/5 of 5 December 2013) on protection against ionising radiation.

The BSS decree under preparation indicates:

- The industries involved; all manufacturers must verify the mass activity of their waste;
- The exemption values; 1 Bq/g for each radionuclide of natural uranium chains, whether or not at equilibrium, and 10 Bq/g for 40K. Below these values the waste is conventional. It can be recovered;
- The classification of Tenorm: any material above the exemption value is a radioactive substance of natural origin (SRON). SRONs fall under the ICPE regime if their quantity exceeds 1 tonne; depending on their activity, they will be go to ISDD and ISDND facilities (below 20 Bq/g) or to facilities for VLLW or even LLLW; they cannot be recovered;
- Various provisions for waste from dismantling facilities.

It is expected that the BSS will be examined by the Council of State in 2017 and then applied. The application of the provisions of the decree will require the modification of all existing regulations.

The Board has taken note of the management of Tenorm in England, which mainly comes from the oil industry. It is stocked in two centres.

Historic waste

This is VLLW waste with a non-natural radioactivity of a few Bq/g or Tenorm. They come under the responsibility of the producers, not Andra. Except for Tenorm, they are governed by the Environment Code and several decrees, and are listed in the 2015 national inventory. They can be found in 13 conventional waste repositories (deposited before the prohibition decrees), in 11 repositories at INBs and in several non-ICPE (Tenorm) in-situ repositories. At the end of 2017, all historic repositories must be identified and their management must be described (PNGMDR order - article 19).

The CEA has characterised all the deposits of historical VLLW waste from the INBs and INBSs in its charge, with the requests of the successive PNGMDRs (about ten). The large-volume deposits (100 to 150,000 m^3) have been undergoing environmental monitoring for decades and the water analysis of the monitoring piezometers shows no impact. The CEA wishes to leave them in place after BAT (Best Available Techniques) analyses. The others, in smaller quantities, will be recovered during clean-up and dismantling operations. The same applies to EDF for a single deposit that will remain in situ.

VLLW

The management of VLLW waste is defined by an order of 1999 and complies with ASN guides (updated in 2016). There is no release threshold for VLLW from the nuclear zones of INBs, many of which have very low residual radioactivity. About 60% of VLLW waste comes and will come from clean-up and dismantling operations.

The forecast for 2015 for VLLW is 2.2 million m^3 at completion. It is based on VLLW production forecasts (in package m^3) by the producers for the National Inventory. The last exercise to evaluate the total production of VLLW took place in 2014 as part of the Preliminary waste management programme (PPGD TFA2). For EDF, a massive inflow of VLLW from the dismantling of PWRs (on average 20,000 to 25,000 m^3/year over 30 years) will take place after 2033. The same applies to Areva with the arrival of VLLW from GB1 (on average 10,000 m^3/year over 10 years). For the CEA, the opposite is true, the trend is to decrease after 2033 (on average 12,000 m^3/year over 20 years between 2014 and 2034). The CEA will thus remain the main contributor to Cires. The time series up to 2030 are detailed both for the origin of the waste and for its quantity and they take into account the feedback from each producer. Producers' forecasts go to around 2070 (except for Areva, which has not evaluated post-2040 dismantling waste (UP2800, UP3, etc.). Finally, the recovery possibilities are not taken into account.

Uncertainties at completion are related to the level of clean-up of the facilities to be carried out. In this respect, ASN's guide (guide No. 24 September 2016) for clean-up of polluted soils for an INB is intended to provide de-pollution objectives and the methodology for assessing the resulting VLLW. The reference state defines a given potential exposure to be achieved and therefore decontamination by removing more or less contaminated soil. Radiological diagnosis must be based on surface and depth measurements and modelling of the migration of radionuclides or toxic chemicals must be carried out.

Andra predicts that after the 900,000 m^3 extension of Cires (full in 2030), a second VLLW repository will be required even if: the waste is sorted more efficiently, its administrative output is reduced by improved zoning of INBs, their volume is reduced, metals and rubble are recycled and some are reclassified as very low-level waste (VVLLW). Andra estimates the amount of metals that could be recycled without radiological protection constraints at 15 to 20,000 t/year.

In view of this data (storage capacity of existing sites, search for new storage sites, consistency of management of low-level waste, diversity of waste and multiplicity of production sites), Andra and the producers considered establishing the comprehensive, coherent and proportionate management of radioactive waste from VLLW to LLLW, consistent with their radiological and chemical hazards, both for humans and for the environment. New organisational decisions will facilitate exchanges between the stakeholders.

Thus the planning of the repository for LLLW and VLLW at CCS is extended and envisaged in discussions relating to a “third repository in the Aube”. Andra, IRSN and the producers are setting up a programme to define the safety principles, to deal with the social acceptance of the risk and to define the governance for deployment over fifty years.

Andra estimates that 40 to 50% of the VLLW volume actually corresponds to VVLLW. They could go to simplified (compared with Cires) *in-situ* storage, such as ISDD-INDND, located at large C&D centres. The concepts of VVLLW and its storage are under consideration. Waste deposits could be managed simply by controlling the dose rate (compliance with less than 10 microSv/year or compliance with the background noise) or according to the source of the waste (zoning). This point is not disconnected from the practice of a release threshold.

These strategies must be supported by open studies to 1) qualify the radiological and chemical hazards of VLLW and VVLLW and adapt the isolation and confinement times accordingly, with respect to the biosphere(hundreds, thousands, hundreds of thousands of years), 2) define the social acceptability of the risks (dialogue, costs).

The Board has been informed of the practices for the release of VLLW in Germany and the United Kingdom. In Germany everything from uncontrolled areas of INBs is exempt from radiological control and release is based on the IAEA's trivial dose concept of 10 microSv/year of activated or contaminated waste. To comply, there are limits of alpha, beta, gamma activity, total and specific activity, surface activity, quantity and nature of waste in accordance with the IAEA and European Directives. The release is subject to release plans approved by the safety authority. Except for metals, the waste goes into conventional storage. If the current thresholds were lowered to take account of the new EU BSS (European Basic Safety Standard) guidelines, the volume of waste would increase 3- to 4-fold and this would create storage problems in addition to the current socio-political problems. Germany is reconsidering this point. The batches of released metals are recycled in the conventional channels without constraints but it is also necessary to show that their radioactivity does not exceed the background noise.

In the United Kingdom, VLLW waste is below 4 GBq/t in alpha and 12 GBq/t in beta/gamma. There is a category of VVLLW that can go into special storage facilities accommodating up to 200 Bq/g. Since 2009, every effort has been made to reduce sending VLLW to storage (decontamination, incineration) and the stored quantity of VVLLW has fallen from 95 to 15% of the total volume between 2009 and 2015. The release threshold and practices are based on the IAEA guidelines (IAEA Directive RSG 1.7) and the European Union (EC Radiation Protection 122). The United Kingdom will take account of the new EU directives (EU BSS).

VLLW release is a considerable problem (10th International Symposium on the Release of radioactive materials, Provisions for Clearance and Exemption, November 2017, Berlin).

The PNGMDR 2016 – 2018

With regard to LLLW, PNGMR 2016-2018 requires:

Of Andra:

- for 2018, to define the safety requirements for storage of LLLW, and to submit a presentation of the methodology for searching for a second LLLW storage site on or near INB/INBSs (article 41);
- for 2019, the outline of a storage facility and to submit the industrial plan for the management of LLLW including: reference inventories (taking into account waste from the reserve inventory), possible sites (CCS and another site) (article 40);
- for 2021 to submit the DOS of a preliminary draft for an LLLW storage facility (article 37). It will be preceded by preliminary reports for 2018 and 2019 (articles 35 and 36) and the inventory of storage capacities (article 42).

Of Areva:

- for 2019, to determine the fate of Malvési LLLW;
- for 2020, to draw up a report on the storage of Ecrin waste (article 63).

Concerning VLLW, the PNGMR 2016-2018 requests for 2020:

- of the CEA and EDF, a study of the storage of VLLW *in situ* rather than dispatch to Cires (article 26);
- of Andra and waste producers, the overall industrial scheme for management of VLLW (article 31);
- Of the waste producers, their waste generation forecasts taking into account all options to reduce their production (articles 20 and 21).

In this context, the recovery of steels from Areva GB1 (136 000 t), EDF GV (80 000 t) and other steels, by melt-decontaminating, is still relevant. In 2018, EDF and Areva are planning to submit a preliminary level DOS to ASN for a demonstrator including secondary waste management and plan the completion of the demonstrator in 2027. The RD focuses on the qualification of the process (Profusion programme): direct melting with grading. They are conducting several studies in parallel: steel markets for the nuclear sector and for the conventional sector, possible postponement of the dismantling of GB1, traceability of the steel obtained. Socodei was chosen to study the implementation of the demonstrator at the preliminary level and for the recycling channels. Two other recycling programmes are under way: Cyber for VLLW concretes and Orcade for polymeric materials. Andra has studied the recovery of VLLW rubble up to a few thousand cubic m^3/year for filling voids in Cires. There is no technical obstacle, even if the radiation protection measures are restrictive, but economic interest is not demonstrated. The recovery of lead from La Hague is effective (contract with Studsvick).

The PNGMR 2016-2018 requires:

Regarding LLLW:

- EDF and the CEA to carry out R&D of characterisation, processing of LLLW graphites, and to specify their inventories before 2019 (article 39). For 2019, to produce the DOS at the preliminary level for a graphite processing plant (article 37 and 38);

Regarding VLLW:

- Areva and EDF to define the technical and safety options for a metallic VLLW processing facility (preliminary level) for 2018 and to provide the associated recovery paths with the application for exemption (article 24);
- Waste producers and Socodéi to define for 2018 the feasibility of melting VLLW metal waste (article 28);
- Andra to deploy for 2018 a recovery channel for VLLW rubble in Cires (articles 22 and 23).

APPENDIX XV:
FUNDAMENTAL RESEARCH

There can be no question here of analysing all basic research with an impact downstream of the electro-nuclear cycle. In this appendix the Board presents and evaluates research developed in the Needs programme and by a few national stakeholders brought to its attention this year.

CNRS – NEEDS

The Board has been evaluating the research carried out under Needs since this structure was set up in early 2014 (reports Nos. 8, 9 and 10). Needs is managed by the CNRS (IN2P3 and INC) in association with the major nuclear stakeholders (Andra, Areva, CEA, EDF) and other organisations, BRGM and IRSN. The aim is to bring together a national community of researchers in 7 unifying projects (PFs) covering the major scientific issues related to nuclear energy. Research projects are in response to calls for tenders. Some calls are targeted to lead to prioritising topics. Most often, several laboratories of the partner organisations are involved in the different WPs (Working Packages).

Needs presented several structuring projects to the Board within three PFs.

a) Nuclear systems PF

The NACRE materials structuring project concerns the acquisition of missing or imprecise nuclear data (cross-section of nuclear neutron-induced reactions for U/Pu and Th/U systems, characteristic of fission products). This data is crucial for the simulation of reactors (Gen IV in particular). NACRE brings together experimenters, theorists and evaluators involved in the development of European or global databases. For example, the neutron elastic scattering cross-section on ^{238}U, which is a major process in the reactor core, is only known to within 10%. It is being re-evaluated by a new approach (experiment at Geel using the Gelina neutron source).

b) Processing conditioning of waste PF

The NMC (New Containment Matrices) structuring project concerns old waste containing reactive metals (Mg and Al). The principle of these matrices is to place the reactive metals in their passivity domain (pH of the matrix in the passivity domain of the metal, formation of a protective layer). Previous research has resulted in the formulation of cement binders (fluorine-doped sodium aluminosilicate for Mg - called MgéO - and lithium-doped phosphato-magnesium cement for Al), which reduce the rate of hydrogen production in packages to less than 1 L/m^2/year. The NMC project focuses on the interaction mechanisms of Mg and Al and other waste components in the presence of the pore water of the materials (passivation, corrosion layer, galvanic effects, inhibitors) and on chemical, micro- and macro-structural evolutions of the matrices.

c) MIPOR PF

The aim of this PF is to improve knowledge of the THMCG behaviour of the porous media (clay and cement-like) involved in geological storage: in clay materials used as closures and covering structures; and in the cement-like materials, concrete and mortar, constituting packages and structures, from the micro to the macroscopic. The abbreviation THMCG stands for behaviour of media under Thermal, Hydraulic, Mechanical, Chemical loads and pressure of Gas. The numerous MIPOR projects have already generated novel micrometric results on high-resolution digital imaging of poral space, THM phenomena and gas behaviour. Four structuring projects are starting including one on diffusion processes.

Other national stakeholders

a) Andra and ANR

Andra launched two calls for tenders (2014 and 2015), using the structure of the ANR, to initiate innovative solutions in the management of VLLW. The first call for tenders allowed the selection of 4 projects (Profusion, Cyber, Orcade and Valofusion) under the “Recycling and Recovery” topic; the second call for tenders resulted in 4 projects under the “Processing and Conditioning” topic and 3 projects under “Measurements and controls”. All these are multi-partner projects. The calls for tenders are supported by PIA funds, of which only 10% goes to basic research. There are some similarities between projects selected by the ANR and Needs projects such as, for example, the Needs NMC project and projects under the ANR “Processing and Conditioning” topic. Andra wishes to maintain the momentum created by these two first calls for tenders and to reinforce the synergy between the research supported through different sources of funding.

b) CEA

The CEA is continuing research on the properties of nuclear glasses. It has presented the results of two long-term behavioural studies to the Board.

The first concerns the characteristics of very highly irradiated glasses by various particles in order to simulate the characteristics of the vitreous matrices of high-level packages when they have received, by auto irradiation, a dose corresponding to 10^{19} alpha/g disintegrations (in some 100,000 years). This dose (energy deposited per cm^3 for example) comes from the recoil nuclei and the alpha particles. In external irradiation studies, the recoil nuclei are simulated by accelerated heavy ions (Kr, Au, Li) and the alpha decay of ^{244}Cm is used for internal irradiation. Regardless of the origin of the dose, all the results show that the macroscopic properties (density, mechanical properties, longevity of the glasses stabilise above 4.10^{18} alpha / g. The glass then reaches a new disordered structure due to the modification of the short-range orders of the boron and oxygen atoms (increase in BO3 clusters measured by ^{11}B NMR and Raman). The two-beam Au and He^{2+} irradiation coupled with the molecular dynamics of ballistic effects in a glass show that this state results from the annealing of the damage caused by the recoil nuclei by the energy provided by the alpha particles. The diffusion coefficient of the helium atoms is unchanged and they fit into the gaps without ever forming bubbles or causing mechanical damage.

The second study concerns the alteration of the glass in the presence of water containing various compositions of elements (Ca, Mg, K, Cs, Fe,) to simulate the water coming from the COx. It provides complementary results

- on the one hand, on the coupled variations of V_0 as a function of the pH and of the contents of elements present in the water and,
- on the other hand, on the nature of the 3 layers of alteration imposed by the other leaching kinetics (Vresidual, Vr). The first layer, near the glass, is passivating and limits the diffusion of water, the second is porous and the third comprises silicate secondary phases. The integration of near field elements in the alteration layers modifies them and influences Vr. The recovery of a rapid alteration at high pH is promoted by the presence of zeolite and leads to destabilisation of the passivating layer. Finally, the irradiation increases the leaching of the glasses.

These studies are continuing in order to achieve a better understanding of the structure and formation of alteration layers.

The CEA has also presented the Board with the results of its research in the field of radionuclide migration. They aim to understand and predict the transport of radionuclides across the geosphere, in support of demonstrating the safety of a geological repository. Numerous studies have been and are being conducted in collaborations on programmes developed jointly by all research organisations and nuclear stakeholders. The CEA has adopted a conventional approach. It involves the speciation of radionuclides in solution, their interaction with surfaces and their transport in water. The CEA has established and validated predictive migration models for complex clay media. All spatial scales are concerned.

The CEA has an important nuclear toxicology programme, which is too broad to be evaluated by the Board. It aims to understand how radioactive or non-radioactive but chemotoxic substances interact with the constituents of cells at the molecular level (micro-organisms such as bacteria and micro-algae, plants, man). In keeping with this objective, the programme addresses areas such as the remediation of contaminated soils, the decorporation of toxicants by chelation (inorganic molecules or peptides) and the study of biomarkers in response to stress caused by the toxin. A large multidisciplinary community from the CEA (DRF, DEN, ICSM, etc.) and all French organisations involved in radiation protection or toxicology participate in this programme. The CEA has presented results to the Board on uranium.

c) Subatech radioactivity group, Chaire storage of radioactive waste, European Foundation for the energies of tomorrow, EDF

A major programme for modelling the interactions of ions and organic molecules with clays, COx argillites and cement-type materials was launched in 2010. It focuses on the sorption and mobility of these entities as well as those of water. Conventional molecular dynamics calculations are carried out with a semi-empirical ClayFF model to determine the partial loads carried by the atoms. The latter are calculated directly by quantum mechanics (DFT in the GGA approximation). Modelling can illuminate the roles of basal surfaces, edges and intercalation layers of crystals on sorption/mobility of ions. Crystals of illite and smectite as well as silicoaluminates with more complicated structures (simulating COx) have been examined for Cs^+, K^+, and other multi-charged M^{2+} ions. Numerous results have been obtained and modelling on the crystals composing the cements is in progress. These very theoretical studies are conducted simultaneously with macroscopic experiments carried out in the same laboratory on the same minerals and elements. This is a major focus of the approach.

d) IRSN

The Environmental Risks Group (GRE - 7 laboratories) of the Radiation Protection Division of the IRSN is carrying out research for the assessment of the radiological risk for humans and the environment in any exposure situation (chronic, accidental) and is also developing a programme for the management of contaminated soils. Improved knowledge of the mechanisms of radioactivity transfer in the biosphere thus underlies these activities.

Experimental resources have been installed on several sites contaminated by nuclear accidents or placed under radiological supervision. IRSN collects data from French environmental observation networks, participates in European programmes (AMORAD/CONTYL/CYCL, etc.) and collaborates with operators and researchers in the field. GRE has recently developed models for Cs operational transfers between soils/sediments and plants. They are based on a conventional thermodynamic approach of Cs exchange between specific sites of the solid phase and the liquid phase; they take into account the exchange kinetics of the labile fraction.

e) Andra OPE

Since 2009, Andra has set up a Sustainable Environment Observatory (OPE) around Cigéo in order to assess the initial state of the environment before constructing any facilities and then to monitor environmental changes over time.

The aim is to create an environmental data bank and to integrate the OPE into the national (SOERE, IR) and international (ERIC) observation networks.

This will be a major element for future fundamental studies on the impact of Cigéo.

Human resources in national research

The Board confirms that the quality of the research and its compatibility with the legislation are excellent. However, the human resources allocated to it are decreasing both at the CEA and in the academic world. The Board reiterates that fundamental research upstream of R&D is essential in order to establish the scientific knowledge on which the French strategy for the management of radioactive materials and waste is based. The underlying issues require a substantial amount of basic research.

Academic radiochemistry attached to the CNRS and to universities outside the CEA has recently been investigation in a study.

Today, six units are affiliated to IN2P3 (23 researchers and 20 teacher researchers) and 7 units are affiliated to INC (18 researchers and 39 teacher researchers).

Approximately 60% of the activities concern the physico-chemistry of the fuel cycle (upstream and downstream, storage of waste, environment and impact on living things) and 40% concern nuclear materials. Some researchers also work in the field of health with particular emphasis on the synthesis and development of new radio-pharmaceuticals.

Laboratories, outside the CEA, where it is possible to work on significantly radioactive material or use irradiation resources have become sparse (Subatech in Nantes, IPNO in Orsay, etc.). In addition, there are about ten CNRS laboratories that maintain ad hoc partnerships with nuclear stakeholders (without involving radioactivity). In total, the potential for non-CEA fundamental research for the 2006 law is low and is distributed across a dozen laboratories.

Nevertheless, the research conducted by this community is coherent: it participates in targeted, European (such as JOPRAD) or national federations (Needs, GDR, Labex, biennial symposiums, Radiation Chemistry and Radiofrequency Group of the SCF, etc.). This participation gives it visibility and provides it with a large portion of the funding required for its projects. It is a classic situation today for all basic research but is penalised by additional constraints: targeted calls for tenders for “nuclear energy” are rare; in the case of “white calls”, they systematically exclude projects related to nuclear energy.

Today there are two university courses: Master's in Nuclear Energy (Paris Sud) and Master's in Separation Chemistry, Materials and Processes (Montpellier), in which the INSTN and chemistry schools (Chimie ParisTech and ENS Chimie Montpellier) participate. The CEA participates extensively in these courses and INSTN offers its own training courses (Atomic Engineering, Short Courses). However, uncertainty about the future of nuclear power leads to a certain disenchantment amongst students regarding “nuclear” education, which weakens these courses.

The close collaboration between the CEA teams and the academic world is stimulating high-level research that is still competitive on an international level. However, their sustainability remains to be seen.

APPENDIX XVI:

INTERNATIONAL FRAMEWORK AND SOURCES OF FAST NEUTRONS

The international framework

Management of the fuel cycle and, by extension, the related E&R, come under both a national and international legal framework.

As a reminder, the international framework consists mainly of:

- the Euratom Treaty, Article 37 (1957) which obliges each Member State to provide the [European] Commission with general data on any radioactive waste treatment projects;
- the Espoo Convention (1991) on evaluation of the impact on the environment (EIE) in a cross-border context;
- the international Ospar Convention (1992) on the prevention of marine pollution;
- the nuclear safety convention (1994), in order to reach and maintain a high level of nuclear safety;
- Directive EC/97/11 (1997) concerning the evaluation of the effects of certain public and private projects on the environment;
- the joint convention on the safety of management of spent fuel and radioactive waste (1997);
- the Åarhus Convention (1998) which governs public participation in decision-making processes and access to environmental justice;
- Directive 2011/70/Euratom on the management of spent fuel and radioactive waste;
- Directive 2013/59/Euratom setting out the basic requirements for health protection from hazards resulting from exposure to ionising radiation.

Directives and conventions have for the most part been transposed into national legislation. Directive 2013/59 / Euratom should be transposed into French law before 6 February 2018.

Fast spectrum sources

There are no major new elements.

As a reminder:

a) Germany

The FRM II reactor in Garching (2004 - …), 20 MWth, enables fast spectrum irradiation.

b) Belgium

The BR2 research reactor (1963-2026-2036?), 120 MWth, allow irradiation of small (diameter 1.5 to 3 cm x h=80 cm) samples in the fast neutron spectrum.

c) China

The sodium-cooled CEFR 65 MWth (20MWe) test reactor was commissioned in July 2010. Since then it has been shut down for prolonged periods. In December 2014, the CEFR operated at full rating for three days.

d) United States

The United States does not have any available fast spectrum sources.

e) France

Since Phénix was stopped, there are no more fast spectrum reactors in France. The Jules Horowitz test reactor, in construction, will be used to irradiate a small volume at high flux in a fast spectrum. It is due to be commissioned in 2022.

f) India

Since 1985, India has the FBTR, Fast Breeder Test Reactor, 40 MWth in Kalpakkam. The Prototype Fast Breeder Reactor (PFBR) of 500 MWe is in the final construction phase. One of the objectives is to study the thorium cycle.

g) Japan

The Japan Atomic Energy Agency, JAEA, hopes to restart the Jōyō reactor in 2021. Following incidents and the consequences of the Fukushima accident, the government decided to stop the Monju project.

h) Netherlands

The HFR (1961-2024) in Petten allows limited and very small volume fast spectrum irradiation.

i) Russia

The Bor-60 (1969-2020) of 60 MWth is a sodium-cooled test reactor. The characteristics of the fast spectrum power reactor BN-600 (600 MWe) would be suitable for qualifying FNR fuel.

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