Results obtained in the context of investigations on separation and transmutation make it possible to contemplate the gradual deployment – taking the opportunity provided by a coming renewal of the nuclear power generation fleet – of systems having the capability of fully recycling actinides. With a consequent drastic reduction in the constraints relating to ultimate waste, and enhanced proliferation resistance.

New **systems** to **curb** waste **at source**?

CEA/DEN

An instance of integration of a fourth-generation reactor – in this case, a very-high-temperature reactor (VHTR) – into an electricity and hydrogen cogeneration complex. The cylindrical structure positioned next to the reactor is a heat exchanger. The interposition of a hillock between the hydrogen plant and the buried reactor bears witness to the precautions used to manage both the nuclear and the hydrogen risks.

CEA

Design project for a fourth-generation gas-cooled fast-neutron reactor (GFR).

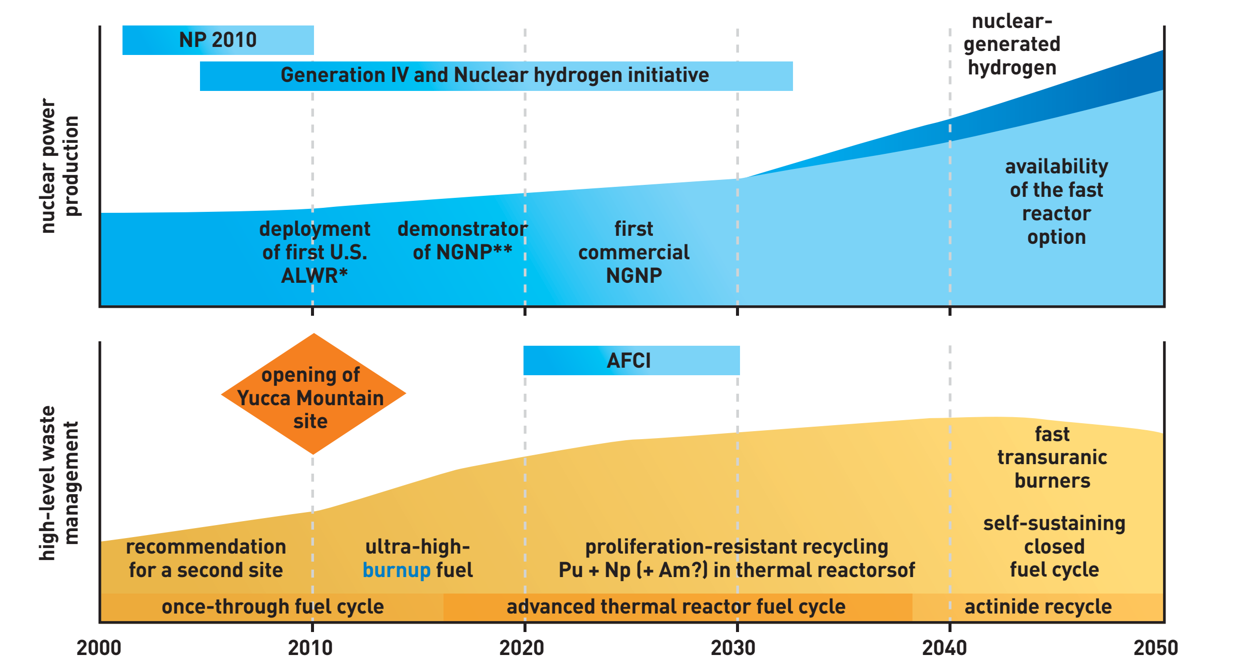
A fourth generation of nuclear systems, for what purpose?

Management of **high-level radioactive** waste stands as a major target for advances, for the nuclear systems of the future. Making good use of **uranium**, and not merely 0.5% of that element (uranium 235), as is the case for the reactors of today, stands as the prime goal, if nuclear power is to be able to make a significant contribution to meeting the world's energy requirements, and thus help curb greenhouse gas emissions. **Recycling** all **fuel** to minimize the presence, in waste, of materials active over very long timespans – **minor actinides** (**MAs**) – is the second major challenge. Such challenges, just as the necessary advances in terms of economic competitiveness, safety, reliability, proliferation resistance, and physical protection, stand among the goals assigned, by 2000, to the reactors of the latter half of the 21st century, by the **Generation IV International Forum**. To achieve this, these reactors must use **fast neutrons**, and recycle all of their fuel – the characteristics of four of the six systems selected, at the end of 2002, by the Forum (*sodium-, gas-, lead-, and* ***supercritical****-water-cooled fast reactors*). The term "system," used to refer to these, marks the importance attaching to the fuel cycle and its optimization, for the reactors of the future, conceived as "reactor, fuel and cycle" systems, to be optimized as a whole. The other major goal of fourth-generation systems is to open up the field of nuclear power applications, to the production of fuels for transportation (hydrogen, and synthetic hydrocarbons), and supply of high-temperature heat to industry. This goal should, in the longer term, also include such advances as may be achieved as regards waste management (see Box 1). In France, while recognizing the validity of these various goals for the nuclear systems of the future, the Atomic Energy Committee (**Comité de l'énergie atomique**), in its meeting of 17 March 2005, considered priority should be given to investigations on fast-neutron reactors (or fast reactors: **FRs**) and advanced cycle processes, emphasizing the importance attached to the use of uranium, and the form of ultimate waste, for the second stage of renewal of the reactor fleet, around 2040.

Which nuclear technologies for a sustainable development?

As a whole, advances anticipated for the fuel cycle of fourth-generation systems concern three areas: raising the energy potential of nuclear fuel reserves, first of all, with the ability to convert, and burn, **fertile materials**, by means of fast neutrons, and recycle the fuel. Optimum use of **storage** and/or **disposal** sites, second, by minimizing nuclear waste volume and thermal load. A simplification of the demonstration of performance for such sites, finally, through a significant reduction in the lifespan and potential toxicity of radioactive waste, through the **transmutation** of long-term contributor elements, taking advantage of recycles inside reactor cores.
Fast-neutron nuclear systems are more effective than **thermal-neutron** systems, such as **PWRs**, as regards use of **plutonium**, and transmutation of minor actinides. This advantage is due to the physical characteristics of neutron reactions with **transuranic** elements. Indeed, the probability of inducing **fission** (which destroys the nucleus) rather than a **neutron capture** (which yields higher **isotopes**) is much higher in fast-neutron systems than in thermal-neutron reactors, by a factor

Fuel cycle strategies adopted in other countries



* Advanced Light-Water Reactor, of the EPR or AP1000 type. ** Next-Generation Nuclear Plant (gas-cooled thermal reactor)

Figure E1.
The United States' long-term nuclear power development strategy.

In the United States, since 1997, a number of official committees (PCAST, NERAC...) have been at work on a strategy to provide renewed impetus for nuclear power. These initiatives led DOE to take a number of major decisions: the setting up, in 2000, of the Generation IV International Forum; the announcement, in 2002, of a number of measures to reinvigorate construction, in the short term, of third-generation power plants (Nuclear Power 2010 Program); and the launching, in 2003, of the AFCI (Advanced Fuel Cycle Initiative) Program, covering reprocessing, recycling, and transmutation (see Figure E1). In Japan, JNC published, in 2000, in collaboration with the Japanese power utilities, a feasibility study for various fast-neutron reactors, showing the need, for that country, to turn to these technologies, using a closed fuel cycle, both in order to optimize waste management, and to improve the security of its energy supply (see Figure E2). The study anticipates, in particular, a demonstration, between 2015 and 2030, of recycling, on a significant scale, of actinides (both major and minor) in this type of reactor. Negotiations on the modalities of such an international demonstration have begun, with active participation from CEA, under the aegis of the Generation IV International Forum.

Among other initiatives relating to systems of the future, the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), launched by IAEA in September 2000, is mainly concerned with defining an evaluation methodology for nuclear systems, in particular with respect to safety, proliferation, the environment, and waste management.

In Europe, a program on systems of the future, organized along the lines of the six pathways identified by the Generation IV International Forum, was initiated in the 6th Framework Program (FP6); further, a proposal has been submitted for a "Fission" Platform, under FP7 (2007–11), to cover current research on water-cooled reactors, and two directions for innovation for future systems: fast neutrons and recycling for sustainable nuclear power, and high temperature for cogeneration and heat-generation applications.

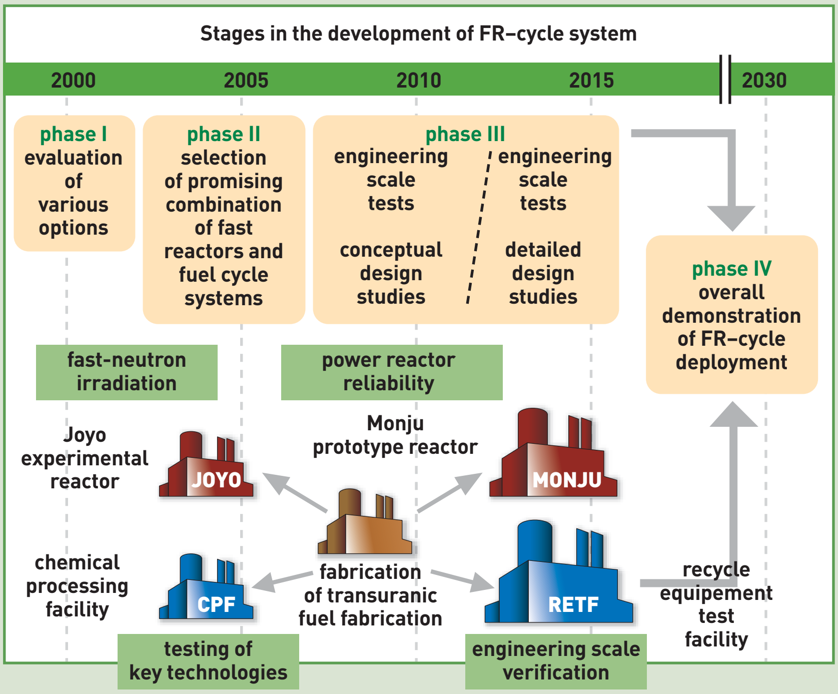


Figure E2.
Japan's long-term nuclear power development strategy.

TVO/Hannu Huovila

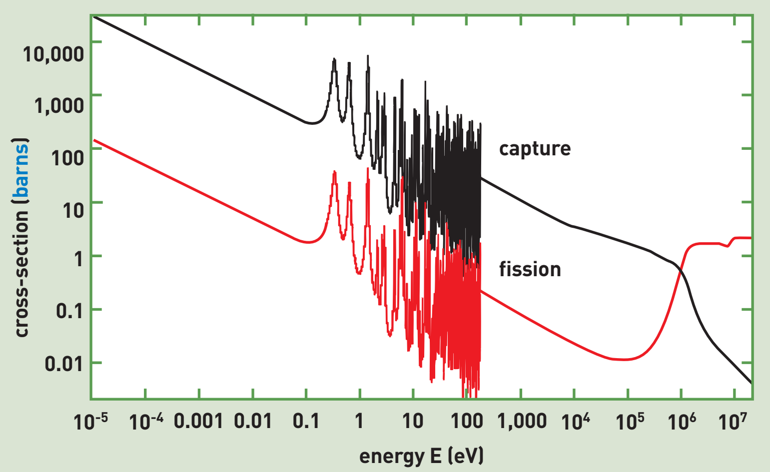
Construction site, in Finland, for the Olkiluoto 3 reactor, the first EPR reactor to be built. EPRs will form, in France, the third generation of nuclear power generation reactors.

of around 10 for the first interaction (see Figure 1) (see Boxes E and F).

This characteristic results, in fast-neutron systems, in arriving naturally at an equilibrium concentration for higher isotopes (Cm, Bk, Cf…) in the multiple recycling case, thus allowing full recycling of actinides. Fully recycling the fuel makes it possible to restrict, essentially, ultimate waste solely to **fission products** (see Figure 2), and to reduce quite significantly its long-term harmfulness and **residual** thermal **power**.

Waste of this quality reverts to a potential harmfulness comparable to that of the **natural uranium** from which the initial fuel was drawn, over 300–500 years, rather than 10,000 years, as is the case with current practice, consisting in leaving minor actinides in the waste (see Figure 3). This change in timescale, combined with

Figure 1.
Capture and fission cross-sections for americium 241 as a function of neutron energy.



confinement having a lifespan of several hundred thousand years, stands as a very marked advance.

Full recycling of all actinides in fast-neutron systems calls for novel processes for spent fuel reprocessing, and refabrication of the fuel to be recycled. Development of such processes is to pass through three major milestones. The first should be, around 2012, a demonstration of the **Ganex** grouped actinide extraction process (in the "reprocessing" step in Figure 2), deriving from investigations on enhanced separation, carried out in the **Atalante** laboratory. In 2015–20, an international laboratory should be set up at La Hague, to enable demonstrations of the fabrication of fuels including transuranic elements. Around 2020–25, finally, an international program should be set up, for the demonstration of overall actinide management, including the separation, by that laboratory, of several tens of kilograms of minor actinides, fabrication of fuel **assemblies** including transuranic elements, in the United States (**ORNL** or **INL**) or France, and trials of such fuel in the Monju reactor, operated in Japan by **JAEA**. These investigations can draw on a collaborative agreement with the French **CNRS** on separation and transmutation processes (research groups, under the aegis of the **PACE** Program).

These demonstrations, involving three partner countries (France, Japan, United States), will stand as an international experimental platform for all processes involved in full fuel recycling in sodium-cooled FRs (with a broad common core of demonstrations, equally valid for other fast-neutron systems). This platform should also contribute to arriving at a unified view, on the international scene, of overall, optimized actinide management, and as to the processes that should be used.

Towards recycling of waste in fourth-generation systems?

The availability, in France, around 2040, of fast-neutron reactors, together with a spent fuel reprocessing process allowing the separation of minor actinides, makes it possible to contemplate deployment of a transmutation strategy that would take advantage of the replacement of industrial installations for nuclear power generation (FRs), and reprocessing (the La Hague plant). This approach allows optimization of current facilities, and taking the opportunity of their replacement to switch processes and technologies.
A reference scenario (see Figure 4) anticipates, concurrently, around 2040, initial deployment of fast-neutron systems in the French fleet (whether sodium- or gas-cooled; see Box 2), and the coming on stream of a new spent fuel reprocessing plant (UP4), supplanting the current one at La Hague. This scenario leaves open the options of solely recycling U and Pu, or going for the full recycling of actinides (U–Pu–MAs), depending on the targets set for advances with respect to ultimate waste quality, and overall techno–economic optimization of the cycle back-end. Ongoing investigations tend to show this strategy is both flexible and robust, in many respects.
It leaves the options open, first of all, regarding management of minor actinides. The existence of considerable common ground, as regards the various actinide separation and conversion processes, should allow the new reprocessing plant to be so dimensioned as to continue carrying out sole U–Pu recycling, if required, while making possible, in the longer term, recycling of all actinides, whether in two separate streams (U–Pu, and MAs), or in a single stream, with joint management of all actinides (U–Pu–MAs), if this option proves compatible with the cycle optimization referred to above. Recycling of minor actinides may be contemplated as soon as fast reactors are deployed, provided by tha [illegible] cation worksh [illegible] fuels, corresponding to the technology of the selected pathway, and designated recycling mode (**heterogeneous**, with U and Pu in the fuel, and MAs in the blanket, or **homogeneous**, with U–Pu–MAs in the fuel).
Fast-neutron systems can clear the accumulated actinide stock, currently stored either in cooled spent fuel, [illegible]. Such a clearance is pos- [illegible] neutron reactors to accept 2.5–5% minor actinides in their core.
This strategy may further be adjusted to cater for deferred deployment of FRs. The new reprocessing plant

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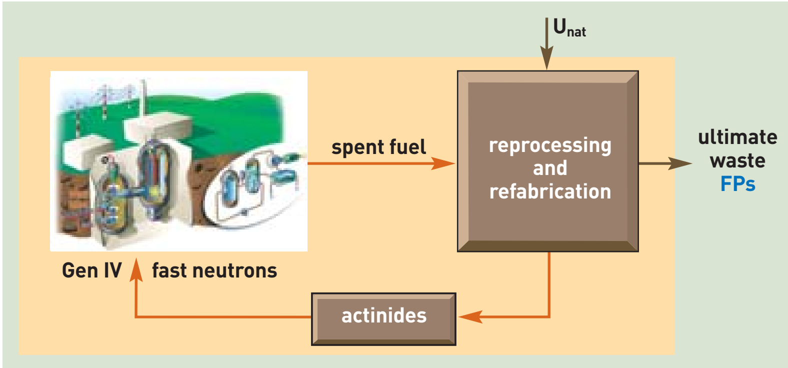


Figure 2.
Principle schematic of full actinide recycling.

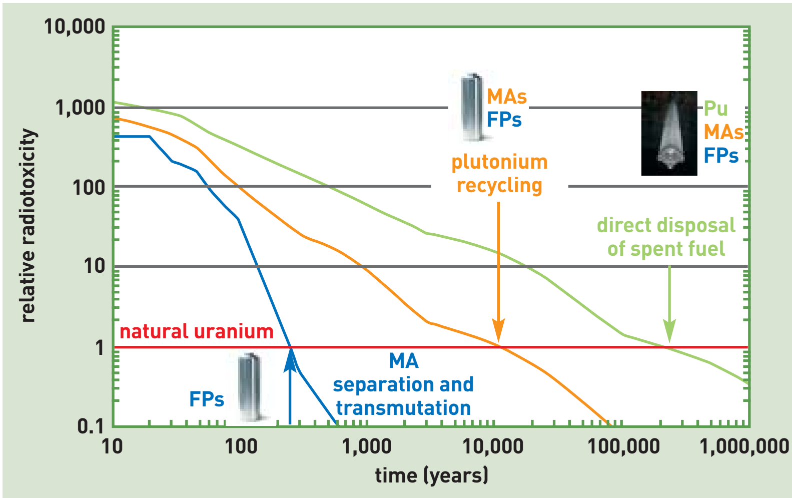


Figure 3.
Evolution of the potential harmfulness of ultimate waste according to various recycling modes.

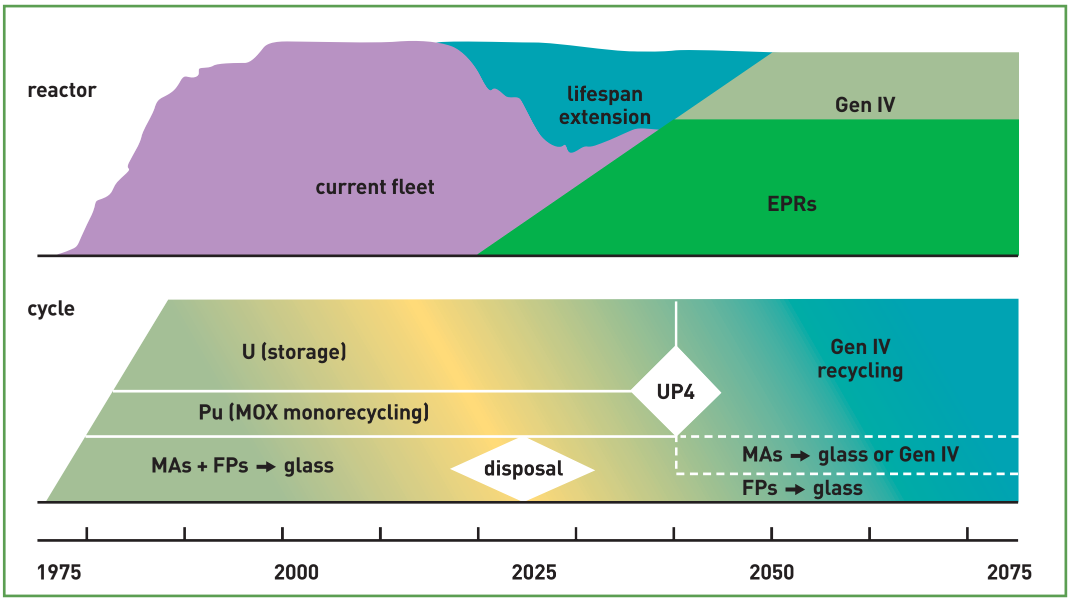


Figure 4.
Replacement scenario for the French nuclear power generation fleet, and the La Hague reprocessing plant. The current fleet is comprised of second-generation reactors, EPRs standing as the third generation.

"Cool" fuel for high temperatures

2

Feasibility of the gas-cooled fast-neutron reactor (GFR) entails development of a fuel exhibiting particularly high performance, in terms of behavior in the presence of fast neutrons and at high temperature, affording a true field of innovation for design, and materials. In particular, the size of the coolant gas passage section calls for selection, for the fuel, of fertile/fissile materials with a high density of heavy nuclei; while the concern to restrict heating up, and thereby release of gaseous fission products, rising pressure of which stresses the cladding, leads to considering a compound exhibiting very high heat conductivity. The high operating temperature (warranting high efficiency, and an opening up to cogeneration), which may even become very high in accident conditions (< 1,600 °C), requires the selection of refractory materials, exhibiting reasonable heat conduction, along with the development of suitable fabrication and reprocessing processes.
As regards the actinide compound, carbides and nitrides exhibit markedly better heat conductivity than oxides (# 20 W · m^{-1} · K^{-1} at 1,000 °C, i.e. 3 times higher than for oxides). As regards the other materials for a composite fuel, since the latter is often called upon to ensure a function of fission product confinement (see Figure E1), the high temperature leads to looking to ceramic materials, particularly composites featuring a fiber-based structure, for good mechanical resistance. Compounds being investigated include SiC, TiC, TiN, and Ti_3SiC_2, the latter exhibiting very high ductility.
Fuel development is basically focusing on two design concepts: one being a pin design, featuring a simple, well-known geometry, making it possible to provide for expansion spaces at the extremities, for fission gases (see Figure E2), the other involving honeycomb plates, featuring a large surface area for heat exchanges, confinement of fission products around each pellet, in the cavity holding it, and, probably, good control of the mechanical interaction between fuel and confinement structure (see Figure E3).

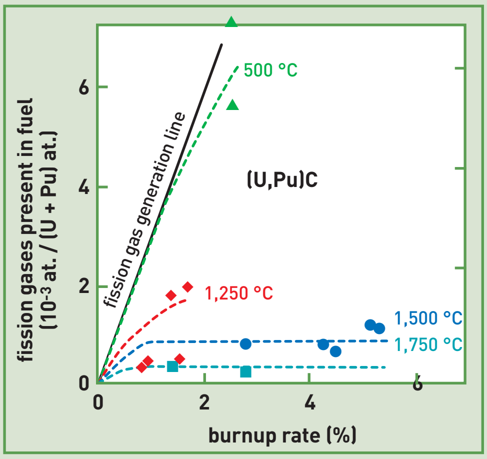


Figure E1.
Retention of fission gases by a uranium- and plutonium-carbide-based fuel, as a function of burnup rate.

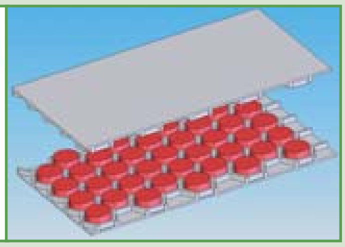


Figure E2. (à gauche)
Pin fuel for the gas-cooled fast-neutron reactor (GFR).

Figure E3. (à droite)
Plate fuel for GFR.

may be fitted by 2040 with functionalities allowing it to cater, as and when required, for the various recycling modes possible with later fast-neutron systems, while not closing the option of switching, at a later date, to full actinide recycling, which would allow a reduction by two orders of magnitude of residual quantities of actinides in waste.
One possible strategy would be to deploy, around 2040, an initial series of fast-neutron systems using plutonium from the PWR fleet, which might stand as precursors for a fourth generation of reactors, to be deployed as and when use of breeder reactor technology becomes a requisite.

A gradual transmutation strategy?

The scientific and technical findings obtained in the context of investigations on separation and transmutation, and the opportunities provided by the replacement of nuclear installations mean it is possible to contemplate a gradual transmutation strategy. With the ability to adjust to the conditions governing deployment of fast-neutron systems in the French fleet, this would preserve the capability, if this proves compatible with overall techno–economic cycle optimization, for full actinide recycling, affording the ability to achieve a radical reduction in the potential radiotoxicity and thermal load of ultimate waste, while making the entire fuel cycle more proliferation resistant.

> **Frank Carré** and **Philippe Brossard**
Nuclear Energy Division
CEA Saclay Center

A What is radioactive waste?

According to the International Atomic Energy Agency (IAEA), radioactive waste may be defined as "any material for which no use is foreseen and that contains radionuclides at concentrations greater than the values deemed admissible by the competent authority in materials suitable for use not subject to control." French law in turn introduces a further distinction, valid for nuclear waste as for any other waste, between waste and final, or "ultimate," waste (*déchet ultime*). Article L. 541-1 of the French Environmental Code thus specifies that "may be deemed as waste any residue from a process of production, transformation or use, any substance, material, product, or, more generally, any movable property left derelict or that its owner intends to leave derelict," further defining as ultimate "waste, be it the outcome of waste treatment or not, that is not amenable to further treatment under prevailing technological and economic conditions, in particular by extraction of the recoverable, usable part, or mitigation of its polluting or hazardous character."

Internationally, experts from IAEA and the Nuclear Energy Agency (NEA) – an OECD organization – as those in the European Commission find that long-lived waste produced in countries operating a nuclear power program is stored securely nowadays, whilst acknowledging a final solution is required, for the long-term management of such waste. They consider burial in deep geological structures appears, presently, to be the safest way to achieve final disposal of this type of waste.

What constitutes radioactive waste? What are the volumes currently involved?

Radioactive waste is classified into a number of categories, according to its level of radioactivity, and the radioactive period, or half-life, of the radionuclides it contains. It is termed long-lived waste when that period is greater than 30 years, short-lived waste otherwise. The French classification system involves the following categories:

– very-low-level waste (VLLW); this contains very small amounts of radionuclides, of the order of 10–100 Bq/g (becquerels per gram), which precludes considering it as conventional waste;

– short-lived low and intermediate level waste (LILW-SL); radioactivity levels for such waste lie as a rule in a range from a few hundred to one million Bq/g, of which less than 10,000 Bq/g is from long-lived radionuclides. Its radioactivity becomes comparable to natural radioactivity in less than three hundred years. Production of such waste stands at some 15,000 m^3 per year in France;

– long-lived low-level waste (LLW-LL); this category includes radium-bearing waste from the extraction of rare earths from radioactive ore, and graphite waste from first-generation reactors;

– long-lived intermediate-level waste (ILW-LL), this being highly disparate, whether in terms of origin or nature, with an overall stock standing, in France, at 45,000 m^3 at the end of 2004. This mainly comes from spent fuel assemblies (cladding hulls and end-caps), or from operation and maintenance of installations; this includes, in particular, waste conditioned during spent fuel reprocessing operations (as from 2002, this type of waste is compacted, amounting to some 200 m^3 annually), technological waste from the operation or routine maintenance of production or fuel-processing plants, from nuclear reactors or from research centers (some 230 m^3 annually), along with sludges from effluent treatment (less than 100 m^3 annually). Most such waste generates little heat, however some waste of this type is liable to release gases;

– high-level waste (HLW), containing fission products and minor actinides partitioned during spent fuel reprocessing (see Box B), and incorporated at high temperature into a glass matrix. Some 120 m^3 of "nuclear glass" is thus cast every year. This type of waste bears the major part of radioactivity (over 95%), consequently it is the seat of considerable heat release, this remaining significant on a scale of several centuries.

Overall, radioactive waste conditioned in France amounts to less than 1 kg per year, per capita. That kilogram consists, for over 90%, of LILW-SL type waste, bearing but 5% of total radioactivity; 9% of ILW-LL waste, less than 1% HLW, and virtually no LLW-LL waste.

What of the waste of tomorrow?

From 1991, ANDRA compiled, on a yearly basis, a geographical inventory of waste present on French territory. In 2001, ANDRA was asked by government to augment this "National Inventory," with the threefold aim of characterizing extant stocks (state of conditioning, processing traceability), predicting future waste production trends to 2020, and informing the public (see *An inventory projecting into the future*). ANDRA published this reference National Inventory at the end of 2004. To meet requirements for research in compliance with the directions set out in the French Act of 30 December 1991 (see *Radioactive waste management research: an ongoing process of advances*), ANDRA, in collaboration with waste producers, has drawn up a Dimensioning Inventory Model (MID: Modèle d'inventaire de dimensionnement), for the purposes of arriving at estimates of the volume of waste packages to be taken on board in research along direction 2 (disposal). This model, including as it does predictions as to overall radioactive waste arisings from the current reactor fleet, over their entire lifespan, seeks to group waste types into families, homogeneous in terms of characteristics, and to formulate the most plausible hypotheses, with respect to conditioning modes, to derive the volumes to be taken on board for the purposes of the investigation. Finally, MID sets out to provide detailed stocktaking, intended to cover waste in the broadest possible fashion. MID (not to be confused with the National Inventory, which has the remit to provide a detailed account of actual waste currently present on French territory) thus makes it possible to bring down the variety of package families to a limited number of representative objects, and to specify the requisite margins of error, to ensure the design and assessment of disposal safety will be as robust as feasible, with respect to possible future variations in data.

To ensure consistency between investigations carried out in accordance with direction 2 and those along direction 3 (conditioning and long-term storage), CEA adopted MID as input data. MID subsumes waste packages into standard package types, then computes the number and volume of HLW and ILW-LL packages, according to a number of scenarios, all based on the assumption that current nuclear power plants will be operated for 40 years, their output plateauing at 400 TWhe per year.

Table 1 shows the numbers and volumes for each standard package type, for the scenario assuming a continuation of current strategy, with respect to spent fuel reprocessing: reprocessing of 79,200 UOX fuel assemblies and storage of 5,400 MOX

MID standard package types	Symbols	Producers	Categories	Number	Volume (m³)
Vitrified waste packages	CO — C2	Cogema*	HLW	42,470	7,410
Activated metal waste packages	B1	EDF	ILW-LL	2,560	470
Bituminized sludge packages	B2	CEA, Cogema*	ILW-LL	105,010	36,060
Cemented technological waste packages	B3	CEA, Cogema*	ILW-LL	32,940	27,260
Cemented hull and end-cap packages	B4	Cogema*	ILW-LL	1,520	2,730
Compacted structural and technological waste packages	B5	Cogema*	ILW-LL	39,900	7,300
Containerized loose structural and technological waste packages	B6	Cogema*	ILW-LL	10,810	4,580
Total B				**192,740**	**78,400**
Total overall				**235,210**	**85,810**

* renamed Areva NC in 2006

Table 1.
Amounts (number, and volume) of waste packages, as predicted in France for 40 years' operation of the current fleet of reactors, according to ANDRA's Dimensioning Inventory Model (MID).

assemblies discharged from the current PWR fleet, when operated over 40 years.

What forms does it come in?

Five types of generic packages (also found in MID) may be considered:

- **cementitious waste packages**: ILW-LL waste packages employing hydraulic-binder based materials as a conditioning matrix, or as an immobilizing grout, or yet as a container constituent;
- **bituminized sludge packages**: LLW and ILW-LL waste packages, in which bitumen is used as confinement matrix for low- and intermediate-level residues from treatment of a variety of liquid effluents (fuel processing, research centers, etc.);
- **standard compacted waste packages** (CSD-C: *colis standard de déchets compactés*): ILW-LL packages obtained through compaction conditioning of structural waste from fuel assemblies, and technological waste from the La Hague workshops;
- **standard vitrified waste packages** (CSD-V: *colis standard de déchets vitrifiés*): HLW packages, obtained mainly through vitrification of highly active solutions from spent fuel reprocessing;
- spent fuel packages: packages consisting in nuclear fuel assemblies discharged from reactors; these are not considered to be waste in France.

The only long-lived waste packages to be generated in any significant amounts by current electricity production (see Box B) are vitrified waste packages and standard compacted waste packages, the other types of packages having, for the most part, already been produced, and bearing but a small part of total radioactivity.

What is happening to this waste at present? What is to be done in the long term?

The goal of long-term radioactive waste management is to protect humankind and its environment from the effects of the materials comprised in this waste, most importantly from radiological hazards. Any release or dissemination of radioactive materials must thus be precluded, through the lasting isolation of such waste from the environment. This management is guided by the following principles: to produce as little waste as practicable; limit its hazardous character as far as feasible; take into account the specific characters of each category of waste; and opt for measures that will minimize the burden (monitoring, maintenance) for future generations.

As for all nuclear activities subject to control by the French Nuclear Safety Authority (Autorité de sûreté nucléaire), fundamental safety regulations (RFSs: *règles fondamentales de sûreté*) have been drawn up with respect to radioactive waste management: sorting, volume reduction, package confinement potential, manufacturing method, radionuclide concentration. RFS III-2.f, in particular, specifies the conditions to be met for the design of, and demonstration of safety for an underground repository, and thus provides a basic guide for disposal investigations. Industrial solutions (see *Industrial solutions for all low-level waste*) are currently available for nigh on 85% (by volume) of waste, i.e. VLLW and LILW-SL waste. A solution for LLW-LL waste is the subject of ongoing investigation by ANDRA, at the behest of waste producers. ILW-LL and HLW waste, containing radionuclides having very long half-lives (in some cases, greater than several hundred thousand years) are currently held in storage installations coming under the control of the Nuclear Safety Authority. What is to become of this waste in the long term, beyond this storage phase, is what the Act of 30 December 1991 addresses (see Table 2).

For all of these waste types, the French Nuclear Safety Authority is drawing up a National Radioactive Waste Management Plan, specifying, for each type, a management pathway.

<table>
<tr><th></th><th>Short-lived
Half-life < 30 years
for the main elements</th><th>Long-lived
Half-life > 30 years</th></tr>
<tr><td>Very-low-level waste (VLLW)</td><td colspan="2">Morvilliers dedicated disposal facility (open since 2003)
Capacity: 650,000 m³</td></tr>
<tr><td>Low-level waste (LLW)</td><td rowspan="2">Aube Center
(open since 1992)
Capacity: 1 million m³</td><td>Dedicated disposal facility under investigation for radium-bearing waste (volume: 100,000 m³) and graphite waste (volume: 14,000 m³)</td></tr>
<tr><td>Intermediate-level waste (ILW)</td><td>MID volume estimate: 78,000 m³</td></tr>
<tr><td>High-level waste (HLW)</td><td colspan="2">MID volume estimate: 7,400 m³</td></tr>
</table>

Table 2.
Long-term management modes, as currently operated, or planned, in France, by radioactive waste category. The orange area highlights those categories targeted by investigations covered by the Act of 30 December 1991.

(1) According to the Dimensioning Inventory Model (MID)

B Waste from the nuclear power cycle

Most high-level (high-activity) radioactive waste (HLW) originates, in France, in the irradiation, inside nuclear power reactors, of fuel made up from enriched uranium oxide (UOX) pellets, or also, in part, from mixed uranium and plutonium oxide (MOX). Some 1,200 tonnes of spent fuel is discharged annually from the fleet of 58 pressurized-water reactors (PWRs) operated by EDF, supplying over 400 TWh per year, i.e. more than three quarters of French national power consumption.

The fuel's composition alters, during its irradiation inside the reactor. Shortly after discharge, fuel elements contain, on average,[1] some 95% residual uranium, 1% plutonium and other transuranic elements – up to 0.1% – and 4% of products yielded by fission. The latter exhibit very significant radioactivity levels – to the extent this necessitates management safety measures requiring major industrial resources – of some 10^{17} Bq per tonne of initial uranium (tiU) (see Figure 1).

The *uranium* found in spent fuel exhibits a makeup that is obviously different from that of the initial fuel. The greater the irradiation, the higher the consumption of fissile nuclei, and consequently the greater the extent by which the uranium will have been depleted of the fissile isotope 235 (^{235}U). Irradiation conditions usually prevailing in reactors in the French fleet, with an average fuel residence time inside the reactor of some 4 years, for a burnup rate close to 50 GWd/t, result in bringing down final ^{235}U content to a value quite close to that of natural uranium (less than 1%), entailing an energy potential very close to the latter's. Indeed, even though this uranium remains slightly richer in the fissile isotope than natural uranium, for which ^{235}U content stands at 0.7%, the presence should also be noted, in smaller, though significant, amounts, of other isotopes having adverse effects in neutronic or radiological terms (^{232}U, ^{236}U), that had not figured in the initial fuel (see Table 1).

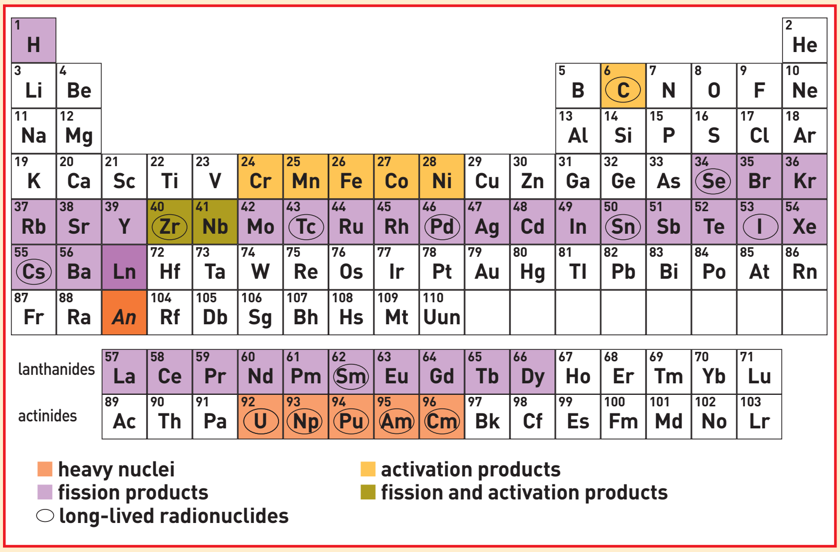


Figure 1.
The main elements found in spent nuclear fuel.

(1) These figures should be taken as indicative values. They allow orders of magnitude to be pinpointed for enriched-uranium oxide fuel, taken from the main current French nuclear power pathway; they do depend, however, on a number of parameters, such as initial fuel composition and irradiation conditions, particularly irradiation time.

The *plutonium* present in spent fuel is yielded by successive neutron capture and decay processes. Part of the Pu is dissipated through fission: thus about one third of the energy generated is yielded by "*in situ* recycling" of this element. These processes further bring about the formation of heavy nuclei, involving, whether directly themselves, or through their daughter products, long radioactive half-lives. These are the elements of the actinide family, this including, essentially, plutonium (from ^{238}Pu to ^{242}Pu, the odd-numbered isotopes generated in part undergoing fission themselves during irradiation), but equally neptunium (Np), americium (Am), and curium (Cm), known as minor actinides (MAs), owing to the

element	isotope	half-life (years)	UOX 33 GWd/tiU (E ^{235}U: 3.5%)		UOX 45 GWd/tiU (E ^{235}U: 3.7%)		UOX 60 GWd/tiU (E ^{235}U: 4.5%)		MOX 45 GWd/tihm (Ei Pu: 8.65%)	
			isotope content (%)	quantity (g/tiU)	isotope content (%)	quantity (g/tiU)	isotope content (%)	quantity (g/tiU)	isotope content (%)	quantity (g/tihm)
U	234	246,000	0.02	222	0.02	206	0.02	229	0.02	112
	235	$7.04 \cdot 10^8$	1.05	10,300	0.74	6,870	0.62	5,870	0.13	1,070
	236	$2.34 \cdot 10^7$	0.43	4,224	0.54	4,950	0.66	6,240	0.05	255
	238	$4.47 \cdot 10^9$	98.4	941,000	98.7	929,000	98.7	911,000	99.8	886,000
Pu	238	87.7	1.8	166	2.9	334	4.5	590	3.9	2,390
	239	24,100	58.3	5,680	52.1	5,900	48.9	6,360	37.7	23,100
	240	6,560	22.7	2,214	24,3	2,760	24.5	3,180	32	19,600
	241	14.4	12.2	1,187	12.9	1,460	12.6	1,640	14.5	8,920
	242	$3.75 \cdot 10^5$	5.0	490	7.8	884	9.5	1,230	11.9	7,300

Table 1.
Major actinide inventory for spent UOX and MOX fuel after 3 years' cooling, for a variety of enrichment and burnup rates. Burnup rate and quantity are expressed per tonne of initial uranium (tiU) for UOX, per tonne of initial heavy metal (tihm) for MOX.

B (next)

lesser abundance of these elements, compared with that of U and Pu, the latter being termed major actinides.
Activation processes affecting nuclei of non-radioactive elements mainly involve structural materials, i.e. the materials of the tubes, grids, plates and end-fittings that ensure the mechanical strength of nuclear fuel. These materials lead, in particular, to formation of carbon 14 (^{14}C), with a half-life of 5,730 years, in amounts that are however very low, much less than one gram per tonne of initial uranium (g/tiU) in usual conditions.
It is the *products yielded by fission* of the initial uranium 235, but equally of the Pu generated (isotopes 239 and 241), known as fission products (FPs), that are the essential source of the radioactivity of spent fuel, shortly after discharge. Over 300 radionuclides – two thirds of which however will be dissipated through radioactive decay in a few years, after irradiation – have been identified. These radionuclides are distributed over some 40 elements in the periodic table, from germanium (^{32}Ge) to dysprosium (^{66}Dy), with a presence of tritium from fission, i.e. from the fission into three fragments (ternary fission) of ^{235}U. They are thus characterized by great diversity: diverse radioactive properties, involving as they do some highly radioactive nuclides having very

Magnum/Harry Gruyaert

After discharge, spent fuel is stored in cooling pools, to allow its radioactivity to come down significantly.
Shown here is a storage pool at Areva's spent fuel reprocessing plant at La Hague.

family	UOX 33 GWd/tiU (E ^{235}U: 3.5%)	UOX 45 GWd/tiU (E ^{235}U: 3.7%)	UOX 60 GWd/tiU (E ^{235}U: 4.5%)	MOX 45 GWd/tihm (Ei Pu: 8.65%)
	quantity (kg/tiU)	quantity (kg/tiU)	quantity (kg/tiU)	quantity (kg/tihm)
rare gases (Kr, Xe)	5.6	7.7	10.3	7
alkali metals (Cs, Rb)	3	4	5.2	4.5
alkaline-earth metals (Sr, Ba)	2.4	3.3	4.5	2.6
Y and lanthanides	10.2	13.8	18.3	12.4
zirconium	3.6	4.8	6.3	3.3
chalcogens (Se, Te)	0.5	0.7	1	0.8
molybdenum	3.3	4.5	6	4.1
halogens (I, Br)	0.2	0.3	0.4	0.4
technetium	0.8	1.1	1.4	1.1
Ru, Rh, Pd	3.9	5.7	7.7	8.3
miscellaneous: Ag, Cd, Sn, Sb...	0.1	0.2	0.3	0.6

Table 2.
Breakdown by chemical family of fission products in spent UOX and MOX fuel, after 3 years' cooling, for a variety of enrichment and burnup rates.

short lifespans, and conversely others having radioactive half-lives counted in millions of years; and diverse chemical properties, as is apparent from the analysis, for the "reference" fuels used in PWRs in the French fleet, of the breakdown of FPs generated, by families in the periodic table (see Table 2). These FPs, along with the actinides generated, are, for the most part, present in the form of oxides included in the initial uranium oxide, which remains by far the majority constituent. Among some notable exceptions may be noted iodine (I), present in the form of cesium iodide, rare gases, such as krypton (Kr) and xenon (Xe), or certain noble metals, including ruthenium (Ru), rhodium (Rh), and palladium (Pd), which may form metallic inclusions within the oxide matrix.
Pu is recycled nowadays in the form of MOX fuel, used in part of the fleet (some 20 reactors currently). Residual U may in turn be re-enriched (and recycled as a substitute for mined uranium). Recycling intensity depends on market prices for natural uranium, the recent upturn in which should result in raising the current recycling rate (about one third being recycled at present).
Such U and Pu recycling is the foundation for the reprocessing strategy currently implemented in France, for the major part of spent fuel (some two thirds currently).

For the 500 kg or so of U initially contained in every fuel element, and after partitioning of 475 kg of residual U and about 5 kg Pu, this "ultimate" waste amounts to less than 20 kg of FPs, and less than 500 grams MAs. This waste management pathway (otherwise know as the closed cycle), consisting as it does in reprocessing spent fuel now, to partition recoverable materials and ultimate waste, differs from strategies whereby spent fuel is conserved as-is, whether this be due to a wait-and-see policy (pending a decision on a long-term management mode), or to a so-called open cycle policy, whereby spent fuel is considered to be waste, and designated for conditioning into containers, and disposal as-is.
In the nuclear power cycle, as it is implemented in France, waste is subdivided into two categories, according to its origin. Waste directly obtained from spent fuel is further subdivided into minor actinides and fission products, on the one hand, and structural waste, comprising hulls (segments of the cladding tubes that had held the fuel for PWRs) and end-caps (fittings forming the end-pieces of the fuel assemblies for these same PWRs), on the other hand. The process used for spent fuel reprocessing, to extract U and Pu, also generates technological waste (operational waste, such as spare parts, protection gloves...) and liquid effluents.

C What stands between waste and the environment?

Raw, solid or liquid radioactive waste undergoes, after characterization (determination of its chemical and radiological makeup, and of its physical–chemical properties), conditioning, a term covering all the operations consisting in bringing this waste (or spent fuel assemblies) to a form suitable for its transport, storage, and disposal (see Box D). The aim is to put radioactive waste into a solid, physically and chemically stable form, and ensure effective, lasting confinement of the radionuclides it contains. For that purpose, two complementary operations are carried out. As a rule, waste is immobilized by a material – whether by encapsulation or homogeneous incorporation (liquid or powdered waste, sludges), or encasing (solid waste) – within a matrix, the nature of, and performance specification for which depend on waste type (cement for sludges, evaporation concentrates and incineration ashes; bitumen for encapsulation of sludges or evaporation concentrates from liquid effluent treatment; or a vitreous matrix, intimately binding the nuclides to the glass network, for fission product or minor actinide solutions). This matrix contributes to the confinement function. The waste thus conditioned is placed in an impervious container (cylindrical or rectangular), consisting in one or more canisters. The whole – container and content – is termed a package. Equally, waste may be compacted and mechanically immobilized within a canister, the whole forming a package.
When in the state they come in as supplied by industrial production, they are known as primary packages, the primary container being the cement or metal container into which the conditioned waste is ultimately placed, to allow handling. The container may act as initial confinement barrier, allotment of functions between matrix and container being determined according to the nature of the waste involved. Thus, the whole obtained by the grouping together, within one container, of a number of primary

Cross-section of an experimental storage borehole for a spent fuel container (the lower part of the assembly may be seen, top right), in the Galatée gallery of CECER (Centre d'expertise sur le conditionnement et l'entreposage des matières radioactives: Radioactive Materials Conditioning and Storage Expertise Center), at CEA's Marcoule Center, showing the nested canisters.

C (next)

ILW-LL packages may ensure confinement of the radioactivity of this type of waste. If a long-term storage stage is found to be necessary, beyond the stage of industrial storage on the premises of the producers, primary waste packages must be amenable to retrieval, as and when required: durable primary containers must then be available, in such conditions, for all types of waste.

In such a case, for spent fuel assemblies which might at some time be earmarked for such long-term storage, or even for disposal, it is not feasible to demonstrate, on a timescale of centuries, the integrity of the cladding holding the fuel, forming the initial confinement barrier during the in-reactor use stage. Securing these assemblies in individual, impervious cartridges is thus being considered, this stainless-steel cartridge being compatible with the various possible future management stages: treatment, return to storage, or disposal. Placing these cartridges inside impervious containers ensures a second confinement barrier, as is the case for high-level waste packages.

In storage or disposal conditions, the waste packages will be subjected to a variety of aggressive agents, both internal and external. First, radionuclide radioactive decay persists inside the package (self-irradiation process). Emission of radiation is concomitant with heat generation. For example, in confinement glasses holding high-activity (high-level) waste, the main sources of irradiation originate in the alpha decay processes from minor actinides, beta decay from fission products, and gamma transitions. Alpha decay, characterized by production of a recoil nucleus, and emission of a particle, which, at the end of its path, yields a helium atom, causes the major part of atom displacements. In particular, recoil nuclei, shedding considerable energy as they do over a short distance, result in atom displacement cascades, thus breaking large numbers of chemical bonds. This is thus the main cause of potential long-term damage. In such conditions, matrices must exhibit thermal stability, and irradiation-damage resistance.

Stored waste packages will also be subjected to the effects of water (leaching). Container canisters may exhibit a degree of resistance to corrosion processes (the overpacks contemplated for glasses may thus delay by some 4,000 years the arrival of water), and the confinement matrices must be proven to exhibit high chemical stability.

Between the containers and the ultimate barrier provided, in a radioactive waste deep disposal facility, by the geological environment itself, there may further be interposed, apart, possibly, from an overpack, other barriers, so-called engineered barriers, for backfill and sealing purposes. While these would be pointless as backfill in clay formations, they would have the capability, in other environments (granite), of further retarding any flow of radionuclides to the geosphere, notwithstanding degradation of the previously mentioned barriers.

Technological demonstrators of ILW-LL packages for bituminized sludges.

CEA

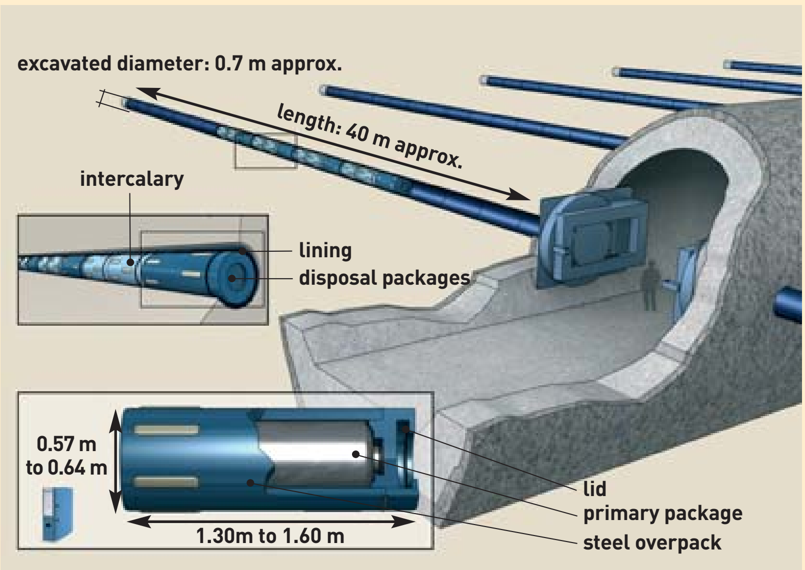
D From storage to disposal

The object of nuclear waste storage and disposal is to ensure the long-term confinement of radioactivity, in other words to contain radionuclides within a definite space, segregated from humankind and the environment, as long as required, so that the possible return to the biosphere of minute amounts of radionuclides can have no unacceptable health or environmental impact.
According to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, signed on 5 September 1997, "storage" means "the holding of spent fuel or of radioactive waste in a facility that provides for its containment, with the intention of retrieval." This is thus, by definition, an interim stage, amounting to a delaying, or wait-and-see solution, even though this may be for a very long time (from a few decades to several hundred years), whereas disposal may be final.
Used from the outset of the nuclear power age, industrial storage keeps spent fuel awaiting reprocessing, and conditioned high-level waste (HLW), or long-lived intermediate-level waste (ILW-LL) in conditions of safety, pending a long-term management mode for such waste. Retrieval of stored packages is anticipated, after a period of limited duration (i.e. after a matter of


A. Gonin/CEA

CEA design study for a common container for the long-term storage and disposal of long-lived, intermediate-level waste.

years, or tens of years).
Long-term storage (LTS) may be contemplated, in particular, in the event of the deferred deployment of a disposal facility, or of reactors to carry out recycling–transmutation, or simply to turn to advantage the natural decay of radioactivity (and hence the falling off of heat release from high-level waste), before putting the waste into geological disposal. By "long term" is meant a timespan of up to 300 years. Long-term storage may take place in a surface or subsurface facility. In the former case, the site may be protected, for instance, by a reinforced-concrete structure. In the latter case, it will be located at a depth of some tens of meters, and protected by a natural environment (for instance, if buried in a hillside) and its host rock.
Whichever management strategy is chosen, it will be imperative to protect the biosphere from the residual ultimate waste. The nature of the radioelements the latter contains means a solution is required that has the ability to ensure their confinement over several tens of thousand years, in the case of long-lived waste, or even longer. On such timescales, social stability is a major uncertainty that has to be taken on board. Which is why disposal in deep geological strata (typically, 500 m down) is seen as a reference solution, insofar as it inherently makes for deployment of a more passive technical solution, with the ability to stand, with no increased risk, an absence of surveillance, thus mitigating a possible loss of memory on the part of society. The geological environment of such a disposal facility thus forms a further, essential barrier, which does not exist in the storage case.
A disposal facility may be designed to be reversible over a given period. The concept of reversibility means the design must guarantee the ability, for a variety of reasons, to access the packages, or even to take them out of the facility, over a certain timespan, or to opt for the final closure of the disposal facility. Such reversibility may be envisaged as a succession of stages, each affording a decreasing "level of reversibility." To simplify, each stage consists in carrying out one further technical operation bringing the facility closer to final closure, making retrieval more difficult than at the previous stage, according to well-specified criteria.



ANDRA

ANDRA design for the disposal of standard vitrified waste packages in horizontal galleries, showing in particular the packages' various canisters, and some characteristics linked to potential reversibility of the disposal facility.

E What is transmutation?

Transmutation is the transformation of one nucleus into another, through a reaction induced by particles with which it is bombarded. As applied to the treatment of nuclear waste, this consists in using that type of reaction to transform long-lived radioactive isotopes into isotopes having a markedly shorter life, or even into stable isotopes, in order to reduce the long-term radiotoxic inventory. In theory, the projectiles used may be photons, protons, or neutrons.

In the first case, the aim is to obtain, by bremsstrahlung,[1] through bombardment of a target by a beam of electrons, provided by an accelerator, photons able to bring about reactions of the (γ, xn) type. Under the effects of the incoming gamma radiation, x neutrons are expelled from the nucleus. When applied to substances that are too rich in neutrons, and hence unstable, such as certain fission products (strontium 90, cesium 137…), such reactions yield, as a rule, stable substances. However, owing to the very low efficiency achieved, and the very high electron current intensity required, this path is not deemed to be viable.

In the second case, the proton–nucleus interaction induces a complex reaction, known as spallation, resulting in fragmentation of the nucleus, and the release of a number of particles, including high-energy neutrons. Transmutation by way of *direct* interaction between protons is uneconomic, since this would involve, in order to overcome the Coulomb barrier,[2] very-high-energy protons (1–2 GeV), requiring a generating energy greater than had been obtained from the process that resulted in producing the waste. On the other hand, *indirect* transmutation, using very-high-energy neutrons (of which around 30 may be yielded, depending on target nature and incoming proton energy), makes it possible to achieve very significantly improved performance. This is the path forming the basis for the design of so-called hybrid reactors, coupling a subcritical core and a high-intensity proton accelerator (see Box F, *What is an ADS?*).

The third particle that may be used is thus the neutron. Owing to its lack of electric charge, this is by far the particle best suited to meet the desired criteria. It is "naturally" available in large quantities inside nuclear reactors, where it is used to trigger fission reactions, thus yielding energy, while constantly inducing, concurrently, transmutations, most of them unsought. The best recycling path for waste would thus be to reinject it in the very installation, more or less, that had produced it…

When a neutron collides with a nucleus, it may bounce off the nucleus, or penetrate it. In the latter case, the nucleus, by absorbing the neutron, gains excess energy, which it then releases in various ways:

- by expelling particles (a neutron, e.g.), while possibly releasing radiation;
- by solely emitting radiation; this is known as a *capture reaction*, since the neutron remains captive inside the nucleus;
- by breaking up into two nuclei, of more or less equal size, while releasing concurrently two or three neutrons; this is known as a *fission reaction*, in which considerable amounts of energy are released.

Transmutation of a radionuclide may be achieved either through neutron capture or by fission. Minor actinides, as elements having large nuclei (heavy nuclei), may undergo both fission and capture reactions. By fission, they transform into radionuclides that, in a majority of cases, are short-lived, or even into stable nuclei. The nuclei yielded by fission (known as fission products), being smaller, are only the seat of capture reactions, undergoing, on average, 4 radioactive decays, with a half-life not longer than a few years, as a rule, before they reach a stable form. Through capture, the same heavy nuclei transform into other radionuclides, often long-lived, which transform in turn through natural decay, but equally through capture and fission.

(1) From the German for "braking radiation." High-energy photon radiation, yielded by accelerated (or decelerated) particles (electrons) following a circular path, at the same time emitting braking photons tangentially, those with the highest energies being emitted preferentially along the electron beam axis.

(2) A force of repulsion, which resists the drawing together of same-sign electric charges.

E (next)

The probability, for a neutron, of causing a capture or a fission reaction is evaluated on the basis, respectively, of its capture cross-section and fission cross-section. Such cross-sections depend on the nature of the nucleus (they vary considerably from one nucleus to the next, and, even more markedly, from one isotope to the next for the same nucleus) and neutron energy.
For a neutron having an energy lower than 1 eV (in the range of slow, or thermal, neutrons), the capture cross-section prevails; capture is about 100 times more probable than fission. This remains the case for energies in the 1 eV–1 MeV range (i.e., that of epithermal neutrons, where captures or fissions occur at definite energy levels). Beyond 1 MeV (fast neutron range), fissions become more probable than captures.
Two reactor pathways may be considered, according to the neutron energy range for which the majority of fission reactions occur: thermal-neutron reactors, and fast-neutron reactors. The thermal neutron pathway is the technology used by France for its power generation equipment, with close to 60 pressurized-water reactors. In a thermal-neutron reactor, neutrons yielded by fission are slowed down (moderated) through collisions against light nuclei, making up materials known as moderators. Due to the moderator (common water, in the case of pressurized-water reactors), neutron velocity falls off, down to a few kilometers per second, a value at which neutrons find themselves in thermal equilibrium with the ambient environment. Since fission cross-sections for 235U and 239Pu, for fission induced by thermal neutrons, are very large, a concentration of a few per cent of these fissile nuclei is sufficient to sustain the cascade of fissions. The flux, in a thermal-neutron reactor, is of the order of 1018 neutrons per square meter, per second.
In a fast-neutron reactor, such as Phénix, neutrons yielded by fission immediately induce, without first being slowed down, further fissions. There is no moderator in this case. Since, for this energy range, cross-sections are small, a fuel rich in fissile radionuclides must be used (up to 20% uranium 235 or plutonium 239), if the neutron multiplication factor is to be equal to 1. The flux in a fast-neutron reactor is ten times larger (of the order of 1019 neutrons per square meter, per second) than for a thermal-neutron reactor.

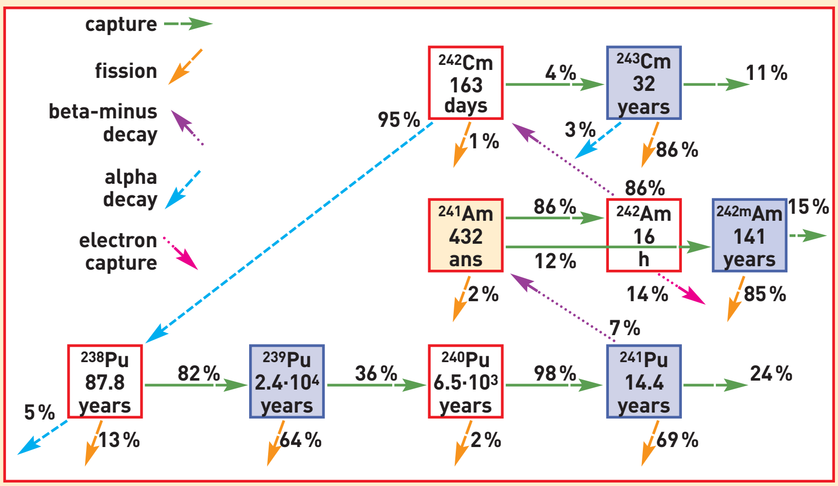
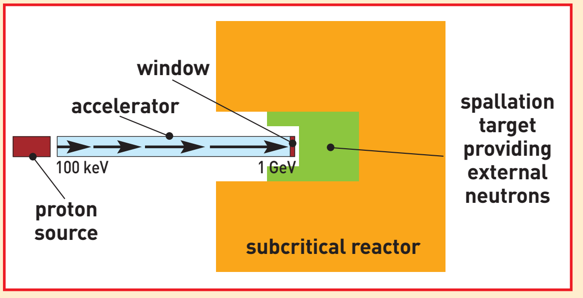


Figure.
Simplified representation of the evolution chain of americium 241 in a thermal-neutron reactor (shown in blue: radionuclides disappearing through fission). Through capture, ^{241}Am transforms into ^{242m}Am, this disappearing predominantly through fission, and into ^{242}Am, which mainly decays (with a half-life of 16 hours) through beta decay into ^{242}Cm. ^{242}Cm transforms through alpha decay into ^{238}Pu, and through capture into ^{243}Cm, which itself disappears predominantly through fission. ^{238}Pu transforms through capture into ^{239}Pu, which disappears predominantly through fission.

F What is an ADS?

An ADS (accelerator-driven system) is a hybrid system, comprising a nuclear reactor operating in subcritical mode, i.e. a reactor unable by itself to sustain a fission chain reaction, "driven" by an external source, having the ability to supply it with the required complement of neutrons.[1]

Inside the core of a nuclear reactor, indeed, it is the fission energy from heavy nuclei, such as uranium 235 or plutonium 239, that is released. Uranium 235 yields, when undergoing fission, on average 2.5 neutrons, which can in turn induce a further fission, if they collide with a uranium 235 nucleus. It may thus be seen that, once the initial fission is initiated, a chain reaction may develop, resulting, through a succession of fissions, in a rise in the neutron population. However, of the 2.5 neutrons yielded by the initial fission, some are captured, thus not giving rise to further fissions. The number of fissions generated from one initial fission is characterized by the effective multiplication factor k_{eff}, equal to the ratio of the number of fission neutrons generated, over the number of neutrons disappearing. It is on the value of this coefficient that the evolution of the neutron population depends: if k_{eff} is markedly higher than 1, the population increases rapidly; if it is slightly higher than 1, neutron multiplication sets in, but remains under control; this is the state desired at reactor startup; if k_{eff} is equal to 1, the population remains stable; this is the state for a reactor in normal operating conditions; and, if k_{eff} is lower than 1, the neutron population dwindles, and becomes extinct, unless – as is the case for a hybrid system – an external source provides a neutron supply.



Principle schematic of an ADS.

From the effective multiplication factor, a reactor's reactivity is defined by the ratio $(k_{eff} - 1)/k_{eff}$. The condition for stability is then expressed by zero reactivity. To stabilize a neutron population, it is sufficient to act on the proportion of materials exhibiting a large neutron capture cross-section (neutron absorber materials) inside the reactor.

In an ADS, the source of extra neutrons is fed with protons, generated with an energy of about 100 keV, then injected into an accelerator (linear accelerator or cyclotron), which brings them to an energy of around 1 GeV, and directs them to a heavy-metal target (lead, lead–bismuth, tungsten or tantalum). When irradiated by the proton beam, this target yields, through spallation reactions, an intense, high-energy (1–20 MeV) neutron flux, one single incoming neutron having the ability to generate up to 30 neutrons. The latter then go on to interact with the fuel of the subcritical neutron multiplier medium, yielding further neutrons (fission neutrons) (see Figure).

Most hybrid system projects use as a core (of annular configuration, as a rule) fast-neutron environments, since these make it possible to achieve neutron balances most favorable to transmutation, an operation that allows waste to be "burned," but which may equally be used to yield further fissile nuclei. Such a system may also be used for energy generation, even though part of this energy must be set aside to power the proton accelerator, a part that is all the higher, the more subcritical the system is. Such a system is safe in principle from most reactivity accidents, its multiplication factor being lower than 1, contrary to that of a reactor operated in critical mode: the chain reaction would come to a halt, if it was not sustained by this supply of external neutrons.

A major component in a hybrid reactor, the window, positioned at the end of the beam line, isolates the accelerator from the target, and makes it possible to keep the accelerator in a vacuum. Traversed as it is by the proton beam, it is a sensitive part of the system: its lifespan depends on thermal and mechanical stresses, and corrosion. Projects are mooted, however, of windowless ADSs. In the latter case, it is the confinement constraints, and those of radioactive spallation product extraction, that must be taken on board.

(1) On this topic, see *Clefs CEA*, No. 37, p. 14

The industrial context

1

The characteristics of the major part of the radioactive waste generated in France are determined by those of the French nuclear power generation fleet, and of the spent fuel reprocessing plants, built in compliance with the principle of reprocessing such fuel, to partition such materials as remain recoverable for energy purposes (uranium and plutonium), and waste (fission products and minor actinides), not amenable to recycling in the current state of the art.

58 enriched-uranium pressurized-water reactors (PWRs) have been put on stream by French national utility EDF, from 1977 (Fessenheim) to 1999 (Civaux), forming a second generation of reactors, following the first generation, which mainly comprised 8 UNGG (natural uranium, graphite, gas) reactors, now all closed down, and, in the case of the older reactors, in the course of decommissioning. Some 20 of these PWRs carry out the industrial recycling of plutonium, included in MOX fuel, supplied since 1995 by the Melox plant, at Marcoule (Gard *département*, Southern France).

EDF is contemplating the gradual replacement of the current PWRs by third-generation reactors, belonging to the selfsame pressurized-water reactor pathway, of the EPR (European Pressurized-Water Reactor) type, designed by Areva NP (formerly Framatome–ANP), a division of the Areva Group. The very first EPR is being built in Finland, the first to be built in France being sited at Flamanville (Manche *département*, Western France).

The major part of spent fuel from the French fleet currently undergoes reprocessing at the UP2-800[1] plant, which has been operated at La Hague (Manche *département*), since 1994, by Areva NC (formerly Cogema,) another member of the Areva Group (the UP3 plant, put on stream in 1990–92, for its part, carries out reprocessing of fuel from other countries). The waste vitrification workshops at these plants, the outcome of development work initiated at Marcoule, give their name (R7T7) to the "nuclear" glass used for the confinement of long-lived, high-level waste.

A fourth generation of reactors could emerge from 2040 (along with new reprocessing plants), a prototype being built by 2020. These could be fast-neutron reactors (i.e. fast reactors [FRs]), either sodium-cooled (SFRs) or gas-cooled (GFRs). Following the closing down of the Superphénix reactor, in 1998, only one FR is operated in France, the Phénix reactor, due to be closed down in 2009.

(1) A reengineering of the UP2-400 plant, which, after the UP1 plant, at Marcoule, had been intended to reprocess spent fuel from the UNGG pathway.