

Radioactive waste: what **health effects**, or **risks**?

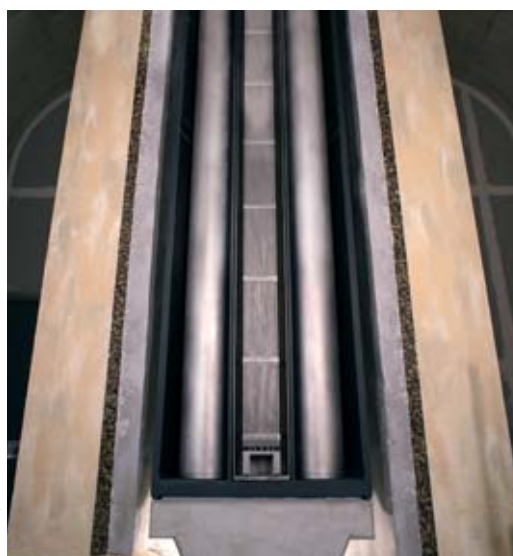
The whole issue of nuclear waste, as seen by public opinion, is grounded in the fear of detrimental effects on human health, and the environment. Two major questions sum up this uncertainty: have all risks been properly identified, including those concerning effects that might arise several thousand years from now? Do radiation protection regulations, according to current recommendations, really make it possible to vouchsafe real protection over timespans of many millennia? The answers to these queries are in the affirmative, even though some biological mechanisms do still require to be more precisely detailed.

The production of nuclear power is inescapably bound up with the handling, and disposal, of **radio-active** elements, liable to have detrimental effects on humans, and their environment. As regards effects on, or risks for, health, the issues, related to the long **half-life** of some **radionuclides**, concern the extent to which they are innocuous, with respect to the environment, and future generations.

The answer involves the one principle: the measures deployed for **waste conditioning** and management have the purpose of restricting, and delaying, *for the time required for their radioactivity to disappear*, the dispersion of radionuclides in the **biosphere**. Operational feedback on management of this type of waste, and computations of **doses** and concentrations corresponding to the various management modes being considered, do indeed show that intake of radionuclides by humans, or human irradiation, would only occur at low concentrations, and small doses, undistinguishable from those due to natural radioactivity.

Radionuclides and biological effects

Investigations on the biological effects of ionizing radiations, indeed, are not directly linked with the issue of radioactive waste. Radionuclides, yielding as they do, through **decay**, one or more daughter nuclides (daughter product **chains**), involve, through their interaction with biological matter, three types of outcomes: irradiation, which varies according to the type and energy of the emitted radiation; concomitant **transmutation** into one or more elements, of a different chemical and radioactive nature; finally, the effects related to the chemical properties of the given radionuclide and its daughter products. For all radionuclides, these effects are approached in a global fashion, by way of investigations on their metabolism and **dosimetry**, apparently leaving in the dark more specific aspects, relating to transmutation or chemical toxicity, which are not specifically addressed in this context. The latter issue, for radionuclides and their daughter products, relates to that arising for other chemical elements, metals in particular: toxicity levels, and exposure limits, for workers and the population, are published on a



Section of an experimental spent fuel storage well, in the Galatée gallery, constructed at CEA's Marcoule Center, to investigate the behavior of the components of a subsurface long-term waste storage facility. For this storage option, as for the other options investigated, the ultimate goal remains that of delaying any dispersion of radionuclides into the biosphere, for the time required for their radioactivity effectively to disappear.

regular basis by various organizations.⁽¹⁾ Particular reference should be made to those elements that are indispensable to the physiological balance, a deficit of which manifests itself in a state of deficiency (selenium, iodine). Indeed, a preexisting deficiency alters distribution and/or concentration in tissues, in the event of an inflow of radioactive **isotopes**, thus increasing the doses received by the retaining organs, as found in populations deficient in stable iodine. As regards radiotoxicology, international organizations such as **UNSCEAR** and the **ICRP** regularly update their data on the health effects of the various sources of radioactivity, focusing on the effects of irradiation, and the physical-chemical properties of the radionuclide, which determine which organs and cells are affected by contamination.

Intrinsically, every radioactive element is characterized by its half-life, specific **activity** (expressed in **Bq/g**), and emission characteristics (type, and energy). Its behavior in the environment depends on physical-chemical properties, which determine its ability to undergo diffusion: chemical class, isotopic distribution and presence of analogs, **oxidation-reduction** potential of the medium... Diffusion further depends on the characteristics of transport carriers. Intake by humans is the outcome of all of these properties together, making each radionuclide more or less accessible, directly (in

(1) The WHO (World Health Organization), and two US federal agencies, EPA (Environmental Protection Agency) and NIOSH (National Institute for Occupational Safety and Health).



air, or water) or through the food chain. As a rule, the elements that are most mobile in the biosphere are also those for which the rate of absorption in the digestive tract (gut transfer factor f_1) ⁽²⁾ is highest (chlorine, iodine, technetium...), whereas those elements that are considered to be poorly mobile in the biosphere exhibit a low gut transfer factor (**plutonium**, niobium). The biological outcomes, and the specific character, in terms of pathology, of an internal contamination are linked to the target organs for the isotope involved. Simply, these are the entrance paths – the lungs and digestive tract – and the various organs in which the isotope is retained, depending on its chemical properties: a fairly uniform distribution throughout the organism (chlorine, **tritium**), or concentration in one or more organs (the thyroid for iodine, the liver and skeleton for plutonium...). Inside the organism, radionuclides exhibit a behavior similar to that of the corresponding stable isotopes (iodine, carbon...), or of elements deemed to be chemical analogs (strontium and cesium, for instance). Internal irradiation persists until the radionuclides are eliminated, through the combined effects of radioactive decay, and biological turnover. Further to these findings, the possible dispersion of radionuclides from nuclear waste packages would result in human exposure, mainly through ingestion; inhalation, as external irradiation, being seen as a lower-level risk (see Table).

Radiation-induced health effects

The specific character of the health effects of radionuclides manifests itself, on the one hand, according to the organs and tissues in which they are retained, and, on the other, depending on the characteristics of the radiation emitted. For **alpha** emitters, irradiation occurs *in situ* in the cells incorporating the isotope. For **beta** emitters, and even more so in the case of **gamma** emitters, ionizations could occur at a distance from the radionuclide, since the higher the radiation energy, the longer the track will be. A second factor is the **dose rate**, which, for an equivalent incorporated activity, will be all the lower, the longer the physical half-life. Once these characteristics are taken into account, effects are of the same kind as those induced through external irradiation. The deterministic effects from high

doses are avoided through implementation of radiation protection standards. Risk rises with the dose for non-specific random effects such as cancers and hereditary consequences. In the area of waste management, implementation of the recommended limits (less than 1 millisievert per year) focuses risk studies on the onset of random effects for low doses. Two aspects must be taken on board: that of risks in general for low doses, and the relevance of the dose calculation, enabling risk assessment.

The issue of low doses

Cancers, just as hereditary effects, are not specifically induced, subsequent to irradiation. The spontaneous frequency is high enough to make it virtually impossible, below a dose of 100–500 mSv, to identify the excess risk due to radiation. For low doses, risk is calculated by **linear-no-threshold** extrapolation, from data obtained at higher doses. In humans, no excess of radiation-induced hereditary effects are found, even in exposed populations showing an excess of cancers. The risk of hereditary effects, estimated at less than one fifth of the carcinogenic risk, is obtained from indirect data. The extrapolation is wholly based on epidemiological, or experimental, carcinogenesis studies, most of the findings from which, for low doses, are consistent with various types of extrapolation (linear-no-threshold, linear quadratic, or quadratic...). The risk level computed from linear-no-threshold extrapolation is deemed to correspond to a prudent position, with no underestimation of effects. This would appear to be evidenced by recent findings in cell and molecular biology, at lower doses (less than 10 mGy or so), suggesting an elimination, rather than conservation, of mutated cells. Likewise, epidemiological studies, of which there have been many over the past few years, on populations dwelling in high-natural-radioactivity areas, as, on the other hand, investigations of the risk of bone cancer in women bluecollar workers chronically exposed to radium 226, suggest a lower risk in the event of chronic exposure at low dose rates. However, contrary to external irradiations, for which numerous data allow the risk level to be identified, according to the characteristics of the irradiation, and of the irradiated tissue, the quantification of risk, subsequent to internal contamination, yields less certain estimates. The findings from epidemiological surveys on workers and populations from military-industrial

Table.
Main biokinetic
characteristics for
some long-lived
radionuclides.

(2) The value of f_1 stands for the fraction of the ingested substance that is directly absorbed and passes into body fluids.

radionuclide	main emission	physical half-life (years)	biological half-life (time after which one half of the element is eliminated by the biological pathway)	gut transfer factor f_1	distribution in body target organ(s)
³⁶ Cl	beta	3·10 ⁵	10 days	1	uniform
⁷⁹ Se	beta	1.1·10 ⁶	triexponential elimination, with half-lives of 3 (10%), 30 (40%), and 200 (50%) days	0.8	fairly uniform liver, kidneys
⁹⁴ Nb	beta, gamma	2·10 ⁴	biexponential elimination, with half-lives of 6 (50%) and 200 (50%) days	< 0.01	lungs, skeleton, liver
⁹⁹ Tc	beta	2.1·10 ⁵	0.5 day in the thyroid 75% of the technetium retained in the thyroid is eliminated, with a biological half-life of 1.6 days	0.5	fairly uniform thyroid, salivary glands, gastrointestinal tract (GIT)
¹²⁹ I	beta	1.57·10 ⁷	80 days in the thyroid	1	thyroid
¹³⁵ Cs	beta, gamma	2.3·10 ⁶	biexponential elimination, with half-lives of 2 and 110 days	1	fairly uniform
²³⁸ U	alpha, gamma	4.47·10 ⁹	100 days	0.02	skeleton, kidneys
²³⁹ Pu	alpha	24,130	10 years in the liver, up to several tens of years in the skeleton	5·10 ⁻⁴	liver, skeleton, gonads

sites in the former Soviet Union, as from studies of molecular mechanisms in response to a radionuclide intake, are expectantly awaited, in this respect.

The relationship between dose and risk

For the purposes of radiation protection, the ICRP specifies, for each isotope, in the event of internal exposure, the committed **effective dose**, corresponding to the overall risk for the duration of the contamination, according to the quantity incorporated, and intake pathway. **Dose coefficients** give, after inhalation or ingestion, the effective dose, expressed in sieverts (with effective doses per unit intake [DPUIs] expressed in **sieverts per becquerel**). Appropriate use of such dose coefficients entails a good knowledge of the physical-chemical form of the isotopes, at the time of their incorporation, since the overall risk evaluation provides no direct information as to risk at the level of the isotope's target organs, or tissues. Additional criteria are required, if an estimate is to be arrived at, as to the risk incurred by the target organs of the radionuclide and isotope. Since the organs at risk, and duration of exposure, will vary, depending on the radionuclides involved, health outcomes will be altogether different for a plutonium isotope, having a biological half-life of several tens of years, which is fixed in the lungs, liver, or bones; for an iodine isotope, with a biological half-life of 80 days, fixed in the thyroid; or for a carbon isotope, with a biological half-life of 40 days, involving a relatively uniform distribution. Some of the assumptions used to characterize the biological behavior of an isotope, or to arrive at dosimetric parameters, should be viewed with caution, whether it be chemical analogy, transposition of data obtained on rodents, or the extrapolation to minute concentrations of findings from large amounts, in weight terms. Factors of highly diverse kinds are not directly included in dose calculations, such as age-related sensitivity, or the maximum amount, by weight, that may be absorbed for each element of biological concern, this being a limiting factor for the effective dose. For radionuclides issuing from waste, an estimate of dilution by natural elements in the environment is an essential parameter, for the evaluation of the corresponding dose, and possible toxic effects. For many isotopes, data are available, to complement the dose calculation by way of relevant qualitative or quantitative criteria. For some classes of radionuclides (**lanthanides**), or specific physical-chemical forms, due consideration should be given to the benefit of further targeted investigations.

What other effects should be monitored?

The measures implemented in radioactive waste management reduce to a minimum the exposure undergone by populations, with annual dose estimates, for each radionuclide (well below 1 mSv) corresponding to quantities, by weight, that entail a concomitant protection against their chemical toxicity. Does this level of protection against the risk of cancer also vouchsafe populations are protected against other types of effects, that may be found for low doses? A number of points of information may prove helpful in the debate. Over the past few years, monitoring of atom-bomb survivors has yielded findings suggesting that certain

late noncancer effects, in particular vascular or pulmonary effects, may occur in random fashion, affecting a fraction of the exposed groups (10%). More directly related to internal contamination, unconfirmed findings have noted a relationship between chronic exposure to cesium 137 in the child and increased symptoms indicating cardiovascular alterations. These studies raise the issue of a possible risk of noncancer effects for low doses. However, the survey of epidemiological studies carried out by UNSCEAR does not appear to corroborate such findings, effects occurring for cumulative doses of several hundred millisieverts.

Does chronic exposure over many generations permit an accumulation of DNA lesions, liable to manifest itself, in some tens of generations, through *delayed hereditary effects*? Recent molecular biology findings indicate preferential induction, by irradiation, of *recessive-type* mutations, which are only expressed if the other allele of the same gene is mutated likewise. It is thus conceivable that such mutations might be transmitted over several generations without being expressed, accumulating in the genetic inheritance, thus contributing to what the ICRP calls the *genetic burden* of a population. Experimental methods did not allow, until recently, directly addressing this issue. However, studies of populations dwelling for tens of generations in high-natural-radioactivity areas show no visible increase in hereditary disorders that might be ascribed to irradiation. A few isolated findings suggest the existence of biological markers indicating an accumulation of lesions, assumed to be radiation-induced, however with no associated pathological effect. Such chronically exposed populations provide a particularly favorable model for investigation, for the analysis of the accumulation, selection, and counterselection of recessive mutations, which contribute, in any event, whatever their origin might be, to evolution. More widely, investigations on the delayed effects of ionizing radiations, both somatic and hereditary, indicate that protection against the risk of cancer also ensures, for low doses and concentrations, protection against other types of delayed effects. For the exposure levels considered, understanding of the biological mechanisms involved, together with the analysis of response specificity at low doses or concentrations, must be expanded, on the basis of molecular and cell biology, if it is desired that, not content with arbitrary extrapolations, there be an ability to compare the mechanisms, and consequences, of chronic exposure to various environmental agents.

More mundanely, as regards waste conditioning, **storage**, or **disposal**, environmental radioactivity monitoring, complemented by impact studies providing estimates, according to the population's way of life, of maximum expected doses, provide the best indicators with respect to evolution in terms of exposure, and risk. In the absence of any alteration in radioactivity (air, water, plants...), no further measure need be contemplated. In the event of an alteration, remedial actions would have to be adjusted, depending on migration times from the **waste package**, in order ultimately to preclude any increased exposure for the populations.

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Gametogenesis investigations, at CEA's Fontenay-aux-Roses Center. One of the aims is to investigate the transmission of mutations by germ cells exposed to a genotoxic stress.

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 (*déchets ultimes*). 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 m³ 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 m³ 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 m³ annually), **technological waste** from the operation or routine maintenance of production or fuel-processing plants, from nuclear reactors or from research centers (some 230 m³ annually), along with sludges from effluent treatment (less than 100 m³ 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 m³ 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 (m ³)
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 high 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.

	Short-lived Half-life < 30 years for the main elements	Long-lived Half-life > 30 years
Very-low-level waste (VLLW)	Morvilliers dedicated disposal facility (open since 2003) Capacity: 650,000 m ³	
Low-level waste (LLW)	Aube Center (open since 1992) Capacity: 1 million m ³	Dedicated disposal facility under investigation for radium-bearing waste (volume: 100,000 m ³) and graphite waste (volume: 14,000 m ³)
Intermediate-level waste (ILW)		MID volume estimate: 78,000 m ³
High-level waste (HLW)	MID volume estimate: 7,400 m ³	

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 10¹⁷ **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** ²³⁵U. 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

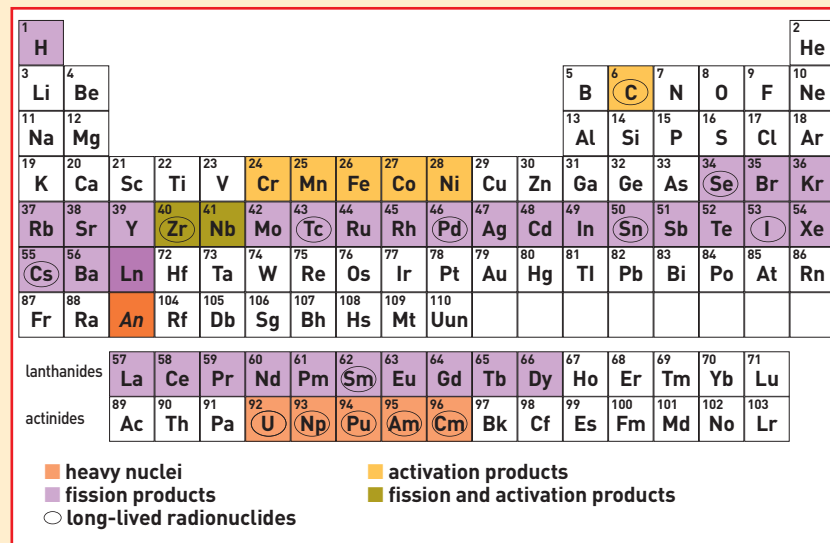


Figure 1.
The main elements found in spent nuclear fuel.

burnup rate close to 50 **GWd/t**, result in bringing down final ²³⁵U 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 ²³⁵U 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 (²³²U, ²³⁶U), that had not figured in the initial fuel (see Table 1).

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 ²³⁸Pu to ²⁴²Pu, 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

(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.

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	7.04·10 ⁸	1.05	10,300	0.74	6,870	0.62	5,870	0.13	1,070
	236	2.34·10 ⁷	0.43	4,224	0.54	4,950	0.66	6,240	0.05	255
	238	4.47·10 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	3.75·10 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/tiU (Ei Pu: 8.65%)
	quantity (kg/tiU)	quantity (kg/tiU)	quantity (kg/tiU)	quantity (kg/tiU)
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 known 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 model), 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 **fusion product** or **minor actinide** solutions). This matrix contributes to the confinement function. The waste thus conditioned is placed in an impervious **contai-**



A. Gomin/CEA

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.

ner (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 pri-

mary 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

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.

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



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).

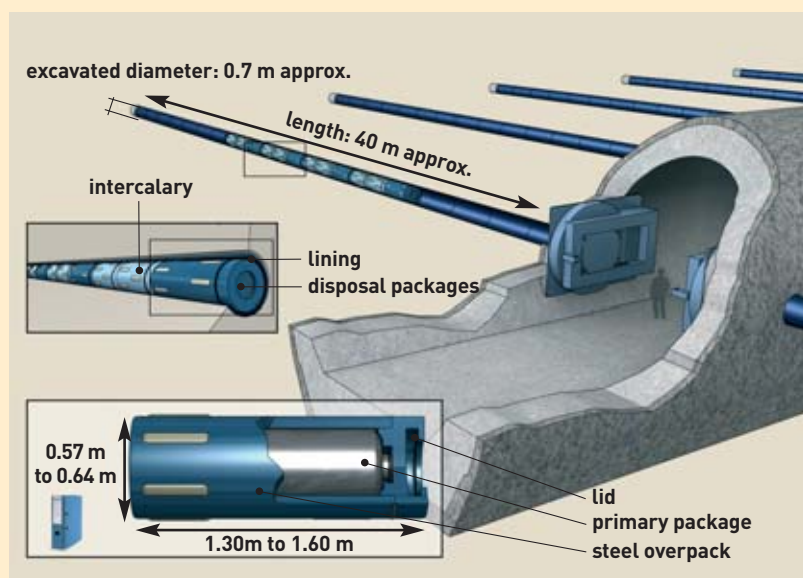
Long-term storage (LTS) may be contemplated, in particular, in the event of the deferred deployment of a disposal facility, or of reactors to carry out

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 **sub-surface** 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 hill-side) 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 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 ther-

mal 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 ^{235}U and ^{239}Pu , 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.

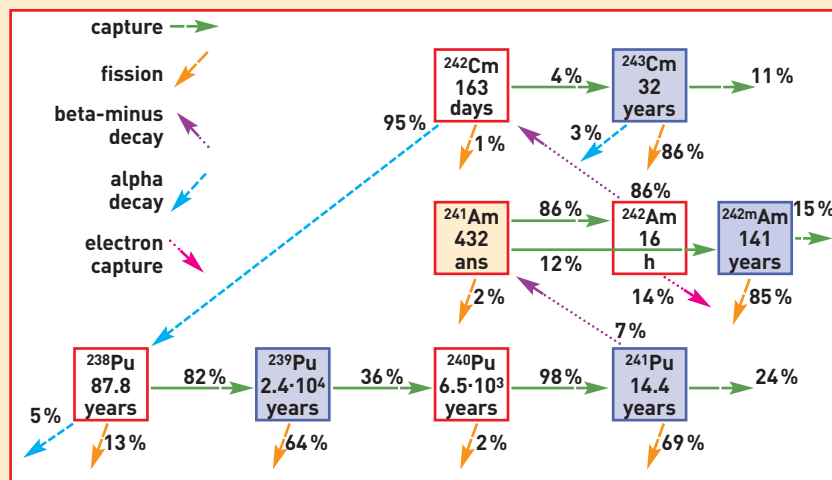


Figure.

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.

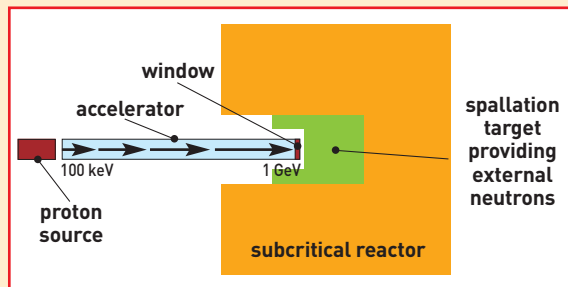
F What is an ADS?

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 $(k_{\text{eff}} - 1)/k_{\text{eff}}$. 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 lat-

ter 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.