# THE NUCLEAR REACTORS OF THE XXI$^{st}$ CENTURY

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*The nuclear fission reactors that will expectedly be developed in the next century are presented with special emphasis on the basic features of the involved physics and on the most innovative technological characteristics. In particular, the so-called hybrid systems are presented. For the sake of presentation, a short reminder of the main features of the current nuclear reactors is given.*

## 1 Introduction

Although it can be imagined that nuclear reactors could be abandoned on the basis of the present world energy consumption (the part of nuclear origin, of about 5 % [1], could easily be shifted to other types of energy production, and there is today no international consensus on the future role of nuclear power), this is very unlikely. Indeed, the actual trends indicate that the population will still be increasing for decades and that the demand of the developing countries (which actually count for 75 % of the population and only 31 % of the energy consumption) will keep growing. The World Council for Energy foresees that the energy demand will increase by something between 50 and 300 % in the next century [1, 2]. Furthermore the proven reserves of oil and natural gas correspond to about 50 years of nowadays consumption. For coal, the relevant number is about 200 years. Hydro-electricity is close to its maximum potential. Traditional (mainly wood) and renewable energy sources are not expected to be increased sizeably before a few decades for various reasons (price, available land, environment,...). Note, however that the uranium resources can ensure nowadays consumption for 400 years with usual nuclear reactors. But the resources can be increased by at least one order of magnitude if new technologies are used (see below).

The role of nuclear power can also be important in the future for two other reasons. First, the economy of many developed countries relies heavily on this energy source, which thus cannot be replaced so easily. Second, the use of fossil sources might be reduced drastically (at least it is the hope of some opinion makers) in order to match environmental concerns about global climate changes, and in particular to reduce the yearly 22 Giga-tonnes of $CO_2$ coming from energy production.

Even though it is highly probable that nuclear energy will be needed to cover the needs, at least for the first half of the next century, there exists no international con-

sensus on the future role of this technology. Some countries are definitely against any development, some are positively in favour, but most of the governments adopt a passive or wait-and-see attitude. This results very likely from the contemplation of the potentials of nuclear energy and the reluctance of public opinion in some countries. The potential expansion of nuclear energy covers several features : extension of the resources (uranium and mainly thorium), the developments of other than current applications (urban heating, desalination of sea water, sea transport,...), limitation of the incidences on the environment, safety of energy supply,... To be prepared to fulfill the needs for the next century and to demonstrate the advantages of nuclear energy, the nuclear community has defined (through the International Atomic Energy Agency, IAEA) the perspectives for the future:

- conception of nuclear installations and of their running modes in view of a development of the field

- improving efficiency and safety

- setting a safe and acceptable treatment of nuclear wastes.

In particular, the IAEA has decided to create a forum in order to discuss the informations on new nuclear reactors. Many ideas have been proposed and the rest of this article will summarize some of them, which either are very promising or have attracted the interest of the nuclear physics community (which has not been the case since the building of the first "nuclear reactor" by Enrico Fermi and his team in 1944). The IAEA, for itself, has defined four categories of new developments :

- advanced reactors with passive safety features

- reactors based on thorium

- rapid neutron reactors cooled with lead or lead/bismuth

- hybrid reactors containing several innovative features.

# 2 A Reminder of Nuclear Reactor Physics

## 2.1 Introduction

For those who are not accustomed with the physics of nuclear reactors, I present the most salient features of fission and of an ordinary nuclear reactor.

Fission of some heavy nuclei can be induced by absorption of slow neutrons. In this process, the available energy, about 200 MeV, is transformed mainly into kinetic energy of the fission fragments. An important property is the concomitant emission of neutrons, whose average number $\nu$ lies between 2 and 3, with an average kinetic energy of the order of 1 MeV. The fact that $\nu$ is larger than unity leaves the possibility of having a chain reaction by regeneration (or even multiplication) of neutrons. Many fission fragments are radioactive and decay by $\beta^-$ and $\gamma$ emission.

Most common nuclear reactors are using slow neutrons to induce fission of $^{235}U$ nuclei. Each reactor is an assembly containing $U$ nuclei, usually a mixture of $^{235}U$ and $^{238}U$ (the

fuel), light nuclei (the moderator) able to slow the neutrons by successive elastic (nuclear) collisions down to thermal energies ($\sim 1/40$ eV) and a fluid (the coolant) which can extract from the reactor the energy released by fission. The largest part of it corresponds to the kinetic energy of the fission fragments. The latter, being charged, are slowed down by Coulomb collisions with electrons, generating heat close to the seat of the fission itself.

## 2.2 Neutronics and chain reaction

Let us assume for the moment an infinite homogeneous reactor with a fuel composed of $^{235}U$ and $^{238}U$ nuclei. The neutrons can undergo two kinds of reactions : non capture reactions, mainly elastic scattering in the moderator, and capture reactions, which are classified into :

- fissile captures leading to fission, basically of $^{235}U$

- fertile captures, transforming $^{238}U$ into $^{239}Pu$, which in turn can also undergo fission induced by thermal neutrons

- sterile captures, mainly (n, $\gamma$), by which neutrons are lost.

It is customary to consider a cycle starting with a neutron issued from a fission, which is then slowed down and/or captured and eventually leads to a subsequent fission delivering new neutrons. The key quantity is the (average) number $k$ of these new neutrons. If $k > 1$, the chain reaction can take place with a multiplication of neutrons. If $k < 1$, the chain reaction eventually fades down. If $k = 1$, the assembly is said to be critical and corresponds ideally to a reactor. The quantity $k$ is given by the four-factor formula

$$k = \eta \epsilon p f. \tag{1}$$

The quantity $\eta$ is the neutron multiplicative factor in a capture by the fissile material ($^{235}U$). It is the product of $\nu$ by the probability that the capture is fissile (so $\eta < \nu$). Roughly speaking, the quantities $f$ and $p$ represent the probability that the neutron does not make a sterile capture in the fertile material ($^{238}U$) and the moderator, respectively. The factor $\epsilon$ takes account of the unfrequent supplementary fissions undergone by $^{238}U$, which is not totally only fertile: $\epsilon$ is larger than, but close to 1. The factors $\eta$ and $\epsilon$ are determined by the composition of the fuel ($U$), whereas $p$ and $f$ depend also upon the geometry and composition of the rest.

The time evolution of the neutron distribution is a classical diffusion problem and can be handled by some generalized Boltzmann-like equations. Disregarding the distribution in velocity (the so-called one velocity approximation), the following transport equation can be written for $n(\vec{r}, t)$, the number of neutrons by unit volume at point $\vec{r}$ and time $t$

$$\frac{1}{v_0} \frac{\partial n}{\partial t} = D\Delta n - \Sigma_a n + q(\vec{r}, t). \tag{2}$$

The terms on the r.h.s. describe the drift, the absorption and the creation of neutrons, respectively. The quantity $D$ is the diffusion length linking the current and the gradient of density $\left( \vec{j} = -D\vec{\nabla} n \right)$ as in ordinary diffusion. Typically the conditions are those of

weak absorption and weak creation (due to the fact that the neutron density $n$ is much smaller than the density of nuclei $\overline{N}$) and $D$ assumes the usual formula in this regime : $D \approx \frac{1}{3} \left( \overline{N} \sigma_{tot} \right)^{-1}$, where $\sigma_{tot}$ is the total (non absorbing) neutron-nucleus cross-section. In eq. (2), $\Sigma_a = \overline{N} \sigma_a$, where $\sigma_a$ is the weighted absorption cross-section.

I will here discuss the very crude approximation of a constant neutron density, which nevertheless helps to understand the basic features of a reactor. Multiplying eq. (2) by $D^{-1}$ and integrating over a finite volume gives, after introduction of the customary parameters $\ell = (\Sigma_a v_0)^{-1}$, $L = \sqrt{\dfrac{D}{\Sigma_a}}$ :

$$L^{-2} \ell \frac{dN}{dt} = -L^{-2} N + \frac{1}{D} \int q\,(\vec{r}, t)\, d^3 r. \tag{3}$$

According to the above discussion, the quantity $q$ is given, for a very short (instantaneous) generation of neutrons, by $q = k \Sigma_a n$ and then eq. (3) transforms into

$$\ell \frac{dN}{dt} = -N + kN. \tag{4}$$

The stationary solution $\dfrac{dN}{dt} = 0$ requires $k = 1$, as expected. If $k > (<) 1$, $N$ will increase (decrease) exponentially. Before discussing further the nonstationary solutions, I will say a few words about finite size effects which, of course, are natural in a reactor. In that case, the first term in the r.h.s. of eq. (2), after integration, would give a leakage term in eq. (4) of the form $-SjD$, where $S$ is the external surface area and $j$ is the average current through this surface. In the simplest cases, this term can be absorbed into the last term of eq. (4) where $k$ is replaced by $k_{eff}$, smaller than $k$. In an actual reactor, the stationary condition refers to $k_{eff} = 1$. In the following, and for didactic purposes, I will disregard finite size effects. In many circumstances, the latter require the mere replacement of $k$ by $k_{eff}$.

## 2.3  Control of the chain reaction

Solution of eq. (4) is simply $N = N_0\, exp\left[(k-1)\, t/\ell\right]$, with $\ell$ defined as before being the lifetime of a neutron in the medium. Its value is typically of 1 $ms$ and even if $k$ is slightly larger than 1, the number of neutrons can grow by an enormous factor in one second. In fact, the predicted explosive growth does not happen because a small fraction $\beta$ of the neutrons produced by fission are delayed, being emitted by the fission fragments, in contrast to the prompt neutrons that are emitted during the fission itself. Therefore, if we call $C$ the number of these fission fragments, eq. (4) because (after multiplication by $\ell^{-1}$)

$$\frac{dN}{dt} = -\frac{N}{\ell} + \frac{k_p}{\ell} N + \frac{C}{\tau_d}, \tag{5}$$

where $\tau_d$ is average decay time of these fragments and where $k_p$ is the multiplicative factor for the prompt neutrons only ($k = k_p + k_d$). The quantity $C$ in turn obeys the following obvious equation

$$\frac{dC}{dt} = \frac{k_d}{\ell} N - \frac{C}{\tau_d}. \tag{6}$$

Eqs. (5)-(6) admit as solution $N = N_0 e^{t/T}$, $C = C_0 e^{t/T}$ with

$$\rho = k - 1 = \frac{\ell}{T} + \frac{k\beta\tau_d}{T + \tau_d}. \tag{7}$$

In practice $\ell \approx 1\ ms$, $\beta = 0.65\ \%$; $\tau_d$ being of the order of minutes, $T$ can be as large as one hour for values of the reactivity $\rho$ much smaller than $\beta$ ($2 \times 10^{-4}$ is easily achievable). The physical interpretation is that criticality is ensured by the delayed neutrons without which the reactor would basically be a subcritical assembly. If necessary, there is a comfortable "period" $T$ to act on the reactor in order to get the reactivity down to its appropriate value, basically by the control rods.

Variations of $k$ may have various origins. The slowest ones are due to the slow variation of the chemical and/or isotopic composition. These can easily be handled by control rods. Rapid variations may be due to changes of core temperature. It is then very useful to have a negative coefficient of reactivity

$$\alpha = \frac{1}{k} \frac{dk}{dT} < 0, \tag{8}$$

in order to have a positive feed-back avoiding the overheating of the core. This is mainly ensured by the Doppler broadening of the resonant absorption of epithermal neutrons (giving $\alpha \approx -1.8 \times 10^{-5} K^{-1}$) and by adopting an incomplete moderation in the moderator (see [3] for more details).

Another important parameter is the void coefficient, which can be described as the variation of the flux of heat extracted from the core with modifications of the properties of the coolant, basically the appearance of bubbles. It is also important that this coefficient is negative.

Finally, the power of the reactor is determined by the neutron density built in the reactor. The latter is fixed by keeping $k > 1$ (but smaller than $1 + \beta$) for some time after which $k$ is fixed to unity (in the average). The power is then given by

$$P \simeq V n v_0\ N_f\ \sigma_f E_f, \tag{9}$$

where $V$ is the volume of the reactor, $N_f$ is the number of fissile nuclei, $\sigma_f$ the fission cross-section and $E_f$ is the energy delivered by a fission event ($\sim 200$ MeV).

## 2.4   Types of reactors

I have considered up to now the most representative type of reactor used worldwide, namely a thermal reactor, using thermal neutrons to induce fission. There is another category of reactors, the rapid neutron reactors (RNR), where the neutrons issued from fissions are not moderated. Furthermore, fast neutrons may be used to convert a fertile material (like $^{238}U$) into a fissile material ($^{239}Pu$ in this case, by the following process $n + ^{238}U \rightarrow\ ^{239}U \xrightarrow{\beta}\ ^{239}Np \xrightarrow{\beta}\ ^{239}Pu$, the last two arrows indicating a $\beta$-decay). A reactor in which this process is important is named a breeder. When the formation of new fissile nuclei ($^{239}Pu$) is larger than the consumption of original fissile nuclei, it is named as a fast breeder (see discussion below).

The number of types of reactors is, in theory, considerable. One can indeed chose a priori independently the fissile isotope ($^{233}U$, $^{235}U$ or $^{239}Pu$), the fertile isotope ($^{232}Th$, $^{238}U$), the energy of the neutrons (thermal, rapid, or even spallation), the moderator ($H_2O$, $D_2O$, graphite, none), the geometry of the assembly (homogeneous, heterogeneous, intermediate), the coolant ($H_2O$, $D_2O$, $CO_2$, $He$, liquid $Na$, liquid $Pb$, molten salts), the number of confining barriers (1-3), which makes 4536 possibilities. Of course, many of them are either impossible, non recommended or simply without industrial interest.

The basic option implies the choice of the fuel, between $U$ and $Th$, and of the neutron energy at which fission is induced. $Th$ has also been used experimentally and requires a fast breeder, as $^{233}U$ is not present on Earth. $U$ is most often used with an enriched isotopic composition ($\sim 4$ % of $^{235}U$) and with thermal neutrons. A small amount of $^{238}U$ is converted into $^{239}Pu$, which is also fissile. Using rapid neutrons leads to the fission of $^{239}Pu$ mainly, the breeding from $^{238}U$ being possibly adjusted to barely consume $^{239}Pu$ or to produce it. Most of the choices are made possible by a judicious combination of the fuel and an adequate arrangement.

The choice of the moderator is in practice linked with the one of the coolant. Combinations where the fuel, the moderator and the coolant are distinct should be discarded, as a loss of coolant does not affect the production rate of energy (this was the case in the RMBK reactor of Tchernobyl). Using a gas as coolant may be less critical, as there is no sharp change of cooling properties as in a liquid/gas transition.

Coolant is quite often taken as water, which also plays the role of the moderator. Heavy water is better for the neutron balance (less absorption). Another possibility is a gas ($He$), in which case the moderator is usually graphite. Many others (chemical or technological) constraints are such that only a few possibilities have been used in practice. The most common reactor is the PWR (Pressurized Water Reactor), using enriched $U$, thermal neutrons and water as coolant and moderator at the same time. A sketch is given in fig. 1. Table 1 summarizes the types of reactors which have been used as power reactors on an industrial scale. The BWR reactor differs from the PWR's in that vapor
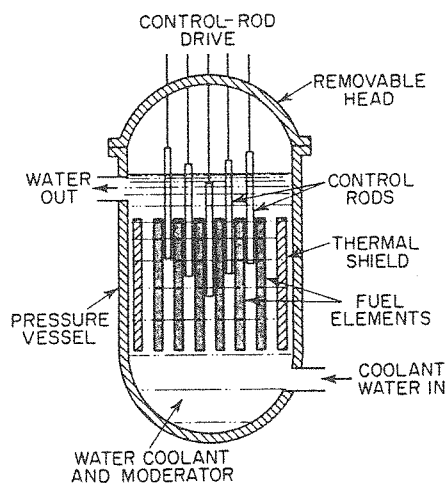


Fig. 1. *Schematic representation of a PWR reactor.*

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| Type of Reactor | Fissile Isotope | Fertile Isotope | Neutron Energy | Coolant | Moderator | Abbre- viation | % of Installed Power |
|---|---|---|---|---|---|---|---|
| Pressur- ized Water Reactor | $^{235}U$ | $^{238}U$ | thermal | $H_2O$ | $H_2O$ | PWR | 63 |
| Boiling Water Reactor | $^{235}U$ | $^{238}U$ | thermal | $H_2O$ | $H_2O$ | BWR | 25 |
| Natural Uranium Heavy Water | $^{235}U$ (nat) | $^{235}U$ (nat) | thermal | $D_2O$ | $D_2O$ | HWR (CANDU) | 6.4 |
| Heavy Temper- ature Gas Cooled | $^{235}U$ | $^{238}U$ | thermal | $CO_2, He$ | graphite | HTGR | 4.5 |
| Liquid Metal Fast Breeder | $^{239}Pu$ | $^{238}U$ | rapid | liquid Na | none | LMFGR | 0.8 |

Table 1. *Characteristics of the most representative nuclear reactors.*

appears in the core (but above the fuel rods) and can thus possibly be used without a heat exchange system.

# 3   The future reactors

## 3.1   Introduction

Advances in technologies are usually coming as a response to some strong motivations. As far as nuclear reactors are concerned, the most obvious one is safety. The latter covers three aspects : (i) nonproliferation of materials necessary to make atomic bombs (mainly $^{239}Pu$); (ii) operational safety of reactors; (iii) treatment of nuclear wastes. Even if the

construction of reactors is stopped tomorrow, these problems will remain for the presently operating reactors. Another motivation arises if nuclear power is kept at its present level: this is the search for a more efficient use of nuclear fuel. If the nuclear power is asked to grow, new motivations will certainly appear. In the rest of this section, I will discuss the projects of new reactors, in relation with their motivations, and indicate the status of their development. I will pay a special attention to the so-called hybrid systems or hybrid reactors.

## 3.2   Advanced reactors with passive safety systems

These kinds of reactors are merely based on some improvements of the existing PWR and BWR reactors, as far as the working principles are concerned. Safety features have been added. One can distinguish the evolutionary reactors, and the advanced reactors. The first ones will develop a power of $\sim$ 1.3 GWe[1] (compared to $\sim$ 1 GWe for the most powerful actual reactors) and are more oriented to economical requirements. They will nevertheless include new safety features as a more automatic operating mode to reduce human failures and a design according to more conservative thermal and stress criteria. The advanced reactors are more innovative. They will be of smaller size and power ($\sim$ 600 MWe), containing less radioactive material. More importantly, they will employ semi-passive safety features based on gravity and natural convection: a reservoir of water is placed above the core, so that water can be delivered to the core without pumps (which may fail), a high-pressure core-cooling system and a system that cools the containment building by the evaporation of water delivered by an elevated reservoir, through sprinkles, and a chimney to cool the confinement building by circulation of outside air convecting naturally. A typical design is contained in fig. 2. Several prototypes exist in the world and have or are going to obtain certification. These include the "system 80+" and the AP-600 in the USA, the EPR in Europe, the VVER-1000 and the VVER-600 in Russia, and the JAERI in Japan, among others.

Other advanced projects deal with HWR's, namely the improvements of the Canadian CANDU reactor, and with HTGR's, like the Chinese HTR-10 and the Japanese HTTR reactors (see ref. [4, 5, 6] for more details).

## 3.3   The fast neutron reactors

In Europe, the technology of the fast neutron reactors is presently not considered as very promising, due to the relative failure and the mishaps of Superphénix. However, several industrial RNR's have been operating satisfactorily in the world, like Phénix in France, BN-600 in Russia and MONSU in Japan. Several technical studies and experimental reactors are existing in the world. Most of them are based on the use of liquid $Na$ as coolant. Investigations are made for a short-term or intermediate-term development of RNR's based on $Th$. The advantages are the lesser radio-toxicity of the nuclear waste, and the abundance of $Th$ on Earth: about 4 MT for the "high content" ores only. This corresponds roughly to 12 centuries at the present world's total power consumption. If

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[1]This means an electric power of 1 GW $= 10^9$ W. The thermal power produced by the reactor is about three times this value.

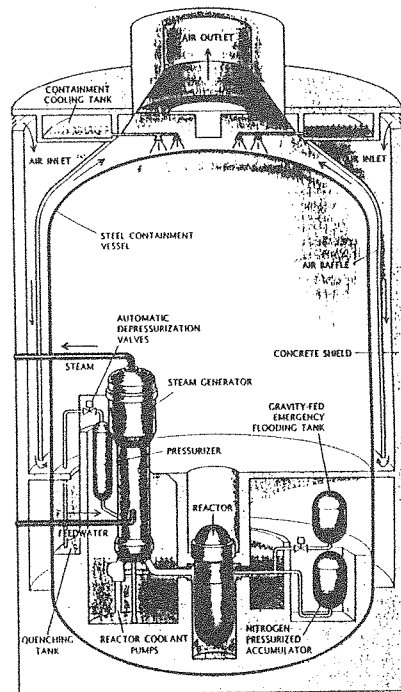lower grade ores can be used, this figure might be boosted by a factor 100 at least [7].



Fig. 2. *Representation of an advanced PWR reactor with semi passive safety devices, as explained in the text (adapted from ref. [4]).*

Other studies envisage using liquid $Pb$ as a coolant, in replacement of liquid $Na$ because of the high chemical reactivity of the latter. The fusion temperature of $Pb$ is 327° C. A lower temperature can be considered by using a $Pb/Bi$ mixture which has an eutectic at 212° C, although $Bi$ is not so much available.

## 3.4   Hybrid reactors with innovative features

### 3.4.1   The transmutation of nuclear wastes

In a typical PWR (1 GWe), the chemical composition of the original fuel material is substantially changed during operation. Several radioactive species are produced, basically actinides ($Np$, $Pu$ and heavier nuclei, denoted as minor actinides) built by neutron captures on $^{238}U$ and a large fraction of fission products. Typically 21 T of fuel generate yearly $\sim 200$ kg of actinides (among which $\sim 120$ kg of $^{239}Pu$) and $\sim 50$ kg of radioactive long-lived fission products (mean lifetime larger than 30 years) per year. After removal from the reactor, the fuel is kept for some years in a water pool, in situ, after which it is either stored or reprocessed, i.e. $U$ and $Pu$ are extracted from the rest, which is stored as a waste. Accumulation of nuclear wastes is beginning to pose a serious problem. Basically,

two solutions are considered: either storage in deep-level repository or transmutation of the waste. The first one consists in storing the radioactive material (after compactification and fixing in a glass matrix) in a geologically stable and dry underground site. The second one supposes the transformation, by nuclear reactions, of the unstable isotopes in stable or very short-lived ones. For the actinides, the simplest method is to use neutrons, as most of them are easily fissile. This idea is already more or less applied in reactors using MOX fuel, which is nothing but a mixture of $U$ and reprocessed $Pu$. The situation is less evident for radioactive fission fragments, as they do not share a similar property. On the other hand, only a few are really long-lived: $^{93}Zr$, $^{99}Tc$, $^{107}Pd$, $^{129}I$, $^{135}Cs$.

### 3.4.2  The hybrid assembly

This is the most innovative feature. It has been triggered by the wish to eliminate reactivity accidents (divergence of the reactor), but is more or less imposed if one wants to transmute (or incinerate) nuclear wastes in a reactor. The basic idea, already mentioned a long time ago [8], has been revived and improved by C.D. Bowman [9] and C. Rubbia [7] and is illustrated in fig. 3: a subcritical ($k < 1$) assembly is fed by neutrons obtained from the bombardment of a so-called spallation target (basically a piece of heavy element) by a high-energy proton beam. I will briefly discuss the basic features of this assembly.
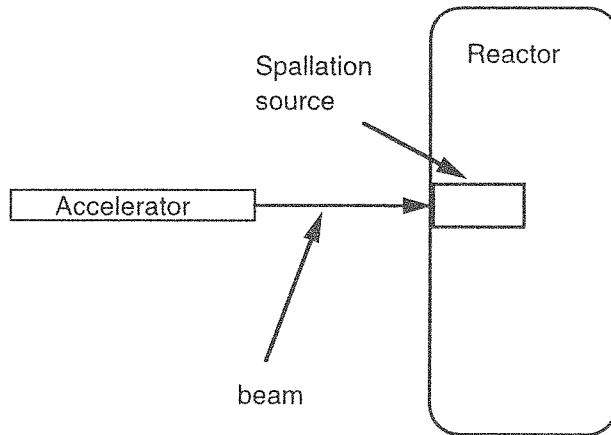


Fig. 3. *Basic parts of a hybrid system. The spallation source is inside the reactor in a subcritical configuration.*

a) Neutron multiplication

If $n_{sp}$ neutrons are introduced in the reactor by the spallation source, the total number of produced neutrons will be, according to the discussion of section 2.2,

$$n_{tot} = n_{sp}\left(1 + k + k^2 + k^3 + ...\right) = \frac{n_{sp}}{1-k}, \tag{10}$$

where $k$ ($< 1$) is the multiplicative factor introduced previously. Except for the first $n_{sp}$ neutrons, the other ones are produced by fission. Therefore, the corresponding

number of fissions is given by

$$N_{fiss} = \frac{n_{tot} - n_{sp}}{\nu} = \frac{n_{sp}}{\nu}\frac{k}{1-k}. \tag{11}$$

and the number of neutrons escaping fission is[2]

$$N_D = n_{tot} - N_{fiss} = \frac{n_{sp}}{1-k}\left(1 - \frac{k}{\nu}\right). \tag{12}$$

b) Power

The power $P$ of the reactor is expressed by

$$P = \frac{dN_{fiss}}{dt}E_f = \frac{dn_{sp}}{dt}\frac{k}{\nu}\frac{1}{1-k}E_f, \tag{13}$$

where $E_f$ is the energy released in a fission. It can also be written as

$$P = I\left[\frac{n_{sp}}{p}\right]\frac{k}{\nu}\frac{1}{1-k}E_f, \tag{14}$$

where $I$ is the intensity of the beam and $\left[\dfrac{n_{sp}}{p}\right]$ is the number of neutrons produced in the spallation source per incident proton, to be discussed below. Now, the power of the beam being $P_{beam} = IE_p$, where $E_p$ is the energy of the protons, one has

$$G = \frac{P}{P_{beam}} = \frac{1}{E_p}\left[\frac{n_{sp}}{p}\right]\frac{k}{\nu}\frac{1}{1-k}E_f. \tag{15}$$

The quantity $G$ is the (theoretical) gain in energy of the assembly. To be clear, in an ordinary reactor, $G = \infty$ (corresponding mathematically to $k = 1$), since there is no need to inject energy. But as we have said, the main advantage of this assembly is that it can operate in a subcritical regime.

c) The spallation source

Inside the spallation source, an impinging proton makes first a nuclear reaction (after being partially slowed down by Coulomb interactions with electrons), producing a certain number of light particles ($p$, $n$, $d$, $t$,...), which in turn will produce subsequent nuclear interactions. Charged particles are largely stopped inside the target, whereas for a suitably chosen material (heavy with small neutron absorption cross-section, like $Pb$), many neutrons are coming out. As eq. (15) indicates, the important quantity is $E_p^{-1}\left[\dfrac{n_{sp}}{p}\right]$. Fig. 4 shows that this quantity reaches a maximum for $E_p \approx 1$ GeV. Actual investigations aim to determine what is the best choice. As it is difficult to make a thorough experimental study, they are based on theoretical calculations calibrated on specific (benchmark) measurements. An important ingredient is the INC model used to describe elementary high-energy nucleon-nucleus

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[2] We disregard leakage and absorption for simplicity.

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interactions (also called spallation reactions). This model assumes that the interaction is separated into two steps; in the first one, the incident particle makes a few hard nucleon-nucleon collisions, ejecting some fast particles; in the second one, the remaining excitation energy is released by softer processes, mainly evaporation of neutrons. A detailed account of the model and of the spallation reactions can be found in refs. [10, 11].

d) The accelerator

Fig. 4 shows that $E_p^{-1}\left[\dfrac{n_{sp}}{p}\right]$ can be as high as 20 and even with $k \approx 0.95$, eq. (15) indicates that a gain $G = 40$ is easily obtained. However, if the assembly is to be used as a power reactor ($P > 1$ GWTh), a beam intensity of at least 25 mA is needed. This demands a major step in the development of the accelerators, as the most powerful cyclotron in the world barely reaches 2 mA at 600 McV (of course, linear accelerators may be considered, but they have other drawbacks).
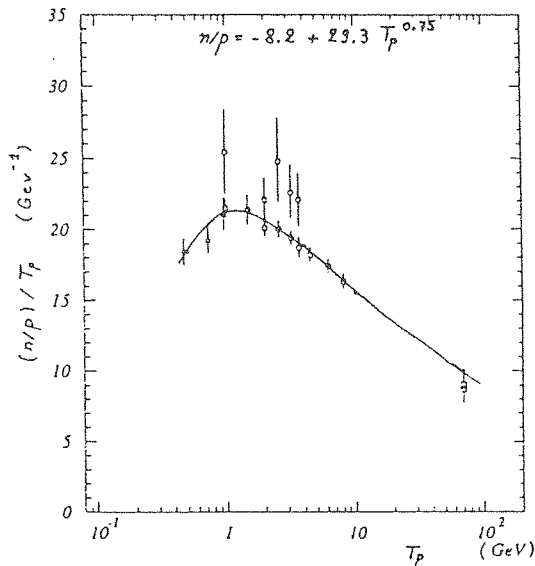


Fig. 4. Number of neutrons per proton and per energy $(E_p^{-1}\left[\dfrac{n_{sp}}{p}\right]$ in the text) as a function of the proton energy for a cylindrical lead target of 20 cm diameter and 60 cm length.

### 3.4.3  Incinerator-reactor

In an ordinary PWR, some amount of $^{238}U$ is transformed by neutron absorption into the fissile element $^{239}Pu$. The conversion ratio CR expresses the number of formed $^{239}Pu$ nuclei per consumed nucleus of $^{235}U$. If the capture in the moderator can be neglected, then

$$CR = \eta\varepsilon - 1. \qquad (16)$$

The values of CR range from zero for a pure "burner" to numbers in the range 0.5-0.7 in typical "converter", to unity or larger for a "breeder". Thus if CR = 1, the amount of fuel remains constant, whereas if CR > 1, it is increasing.

Similar considerations can be made if some amount of waste (transuranians for instance) is introduced into a reactor. Then one can define a transmutation ratio TR as the number of transmuted radioisotopes by consumed fissile nucleus. In an ordinary reactor, it would be roughly given by

$$TR \approx \eta P_{abs},\tag{17}$$

where $P_{abs}$ is the neutron absorption probability in the waste material. This ratio is limited by technological constraints, as one has to keep $k = 1$ after introduction of the waste material. This is one of the reasons why RNR's based on $^{239}Pu$ (like Phénix) or on $^{233}U$ are often presented as good candidates since the $\eta$ values are the largest for these isotopes. For an hybrid assembly, as discussed above, one rather considers the transmutation rate for a given waste material, i.e. the number of nuclei of this sort transmuted per unit time. Coming back to eq. (12), one easily gets

$$\frac{dN_{tr}}{dt} = \frac{dn_{sp}}{dt}\frac{1}{1-k}\left(1 - \frac{k}{\nu}\right)P_{abs},\tag{18}$$

where, once again, $P_{abs}$ is the average absorption probability on the waste material. Owing to eqs. (14)-(15), one gets

$$\frac{dN_{tr}}{dt} = P_{beam}\frac{1}{E_p}\left[\frac{n_{sp}}{p}\right]\frac{1}{1-k}\left(1 - \frac{k}{\nu}\right)P_{abs},\tag{19}$$

and

$$\frac{\frac{dN_{tr}}{dt}}{P} = \frac{E_f}{E_p}\frac{1 - \frac{k}{\nu}}{\frac{k}{\nu}}P_{abs}.\tag{20}$$

One sees that the key parameter (aside to $E_p$ which cannot be changed too much in order to maximize the gain) is the quantity $k/\nu$. If this parameter is large, the assembly will run basically as a reactor. If it is small, it will run as an incinerator; of course, a limiting value is given by requiring a gain $G$ equal to unity, corresponding to a pure incinerator.

### 3.4.4 Projects of reactor-incinerators

The design of these devices depends crucially on the wastes to be transmuted. One can roughly divide the latter in $Pu$, minor trans-uranic actinides ($Np$, $Cu$, $Am$) and typical fission fragments ($^{99}Tc$, $^{126}Sn$, $^{129}I$). One can show that the latter are efficiently transmuted with thermal and epithermal neutrons. The minor actinides demand a high flux of preferably rapid neutrons. $Pu$ is a special case. It can be transmuted by thermal as well as by rapid neutrons (by fission), with a better efficiency in the last case. Actually, in Belgium, France and other countries, $Pu$ is already transmuted in ordinary PWR's. However, the amount of $Pu$ is limited by safety reasons: apparently, too large a fraction of $Pu$ leads to a positive void coefficient (for a large amount of voids), although this could be cured to some extent [13]. The incineration of Pu could be better realized in RNR's of type Superphénix, where the $Pu$ fraction of fuel can be increased from $\sim$ 20 % to $\sim$ 45 % in replacement of $^{238}U$. This is the CAPRA concept studied by the CEA in France.

Several projects around the world are presently studied. In the USA, there is a Brookhaven project (PHOENIX) and a Los Alamos project, of which there exist several versions (ABC, ATW, ADEP). I will describe briefly the latter. It is a hybrid system, based on the use of a high flux of thermal neutrons in a subcritical reactor with molten salts. The fuel salt is the same as in the only experimental molten salt reactor even built, the Oak Ridge MSRE. The original design, schematized in fig. 5, uses a 1.6 GeV proton beam hitting directly (i.e. without any window) the spallation source, made of liquid $Pb$, circulating at 1 m/s. The subcritical assembly is divided in three concentric zones: the first one contains heavy water (the moderator) and some fission fragments, which are to be transmuted by epithermal neutrons, the second one is made of the molten salt transporting the fuel and the minor actinides, under the form of fluorides, and finally the third one is constituted of heavy water and possibly $Th$, which is to be transformed in $^{233}U$ to be used later in zone 2. The high neutron flux ($5 \times 10^{15}$ n/cm$^2$/s) allows the transmutation of $^{237}Np$ (which requires a double neutron capture) and some other minor actinides. The molten salt is circulating outside the core in order to allow a continuous chemical extraction of the fission fragments to separate the incinerated elements from the ones which have to be transmuted and which are reinjected. The authors of the project even propose a strategy where these reactors would progressively replace the PWR's. In 30 years, the stock of minor actinides would be basically resorbed. It is then proposed to pass to a technology based on the $Th/U$ cycle.
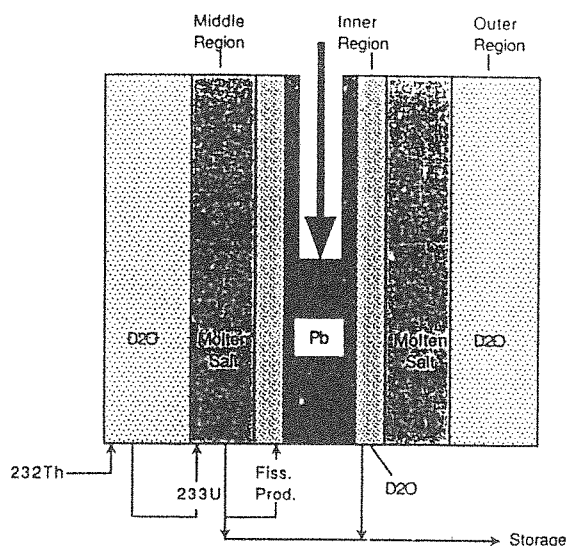


Fig. 5. *Schematic representation of the original Los Alamos project of a hybrid system.*

In Europe exists the most advanced project, initiated by Carlo Rubbia (and then designated as the Rubbia project) [7]. Its original form is schematically described in figs. 6 and 7. It is based on a hybrid system, where the accelerator complex is constituted

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by cyclotrons delivering a 1 GeV proton beam with a maximum intensity of 50 mA. The reactor contains a solid core with the fuel ($Th$) and a large quantity of liquid lead, which plays the role of the spallation source and of the coolant. The large height of the $Pb$ column (20-30 m) allows a circulation by natural convection, avoiding the problem of pumps. The proton beam (fig. 7) is injected down to the core region through a vacuum pipe and hits the spallation source after crossing a tungsten window, $\sim$ 2 mm thick. The reactor uses rapid neutrons ($\sim$ 1 MeV): $Pb$ shows poor neutron absorbing and moderating properties. This concept, popularized by its charismatic author, contains three main features which are not strictly innovative, but which here received a strong impetus: the $Th/U$ cycle, the very concept of the hybrid systems and liquid $Pb$ as a coolant. Several safety features are remarkable: no pump to circulate the coolant, the shutdown of the power delivered in the spallation source (i.e. of the beam) can be very fast (see below however), circulation of cooling air on the external face of the vessel. There is also another interesting feature, related to the use of $Th$, already mentioned, namely the strong reduction of the production of $Pu$ and other minor actinides.
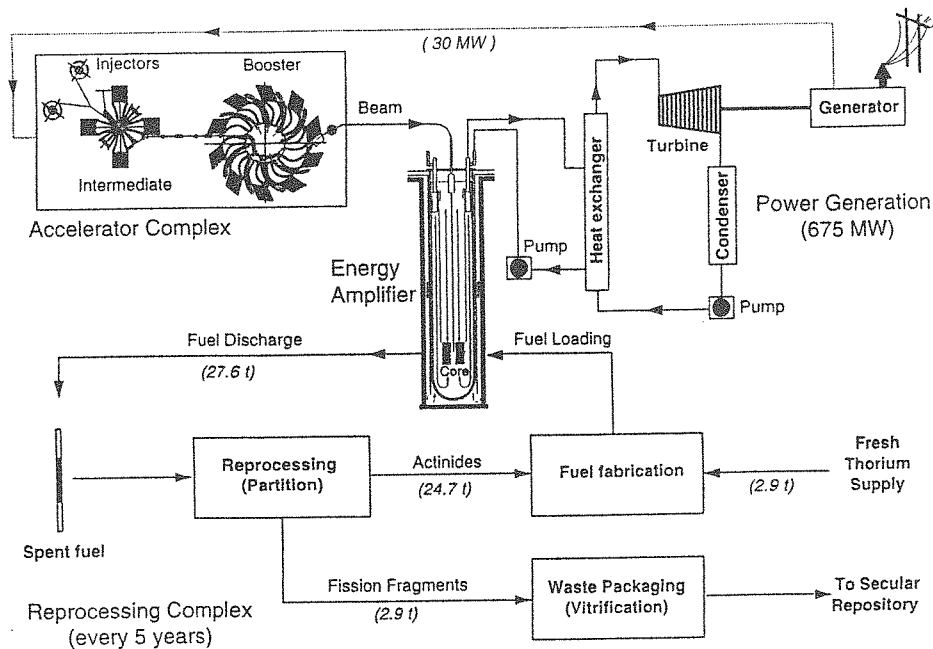


Fig. 6. *Layout and flow diagram of the Rubbia project (adapted from ref. [7]).*

Several, sometimes acute, technological problems have still to be solved. As we already said, no accelerator can deliver, by far, the required maximum intensity. The manipulation of large amounts of $Pb$, 10000 T per reactor, is not obvious (there is apparently no related economical problem, as lead is very cheap). The behaviour and the aging of the beam window, that has to sustain a large energy deposition, are not well known. Finally, even if the beam is shut down, heat continues to be generated in the core, because the

transformation of thorium gives birth to a radioactive protactinium isotope[3], with an half-life of 27 days.
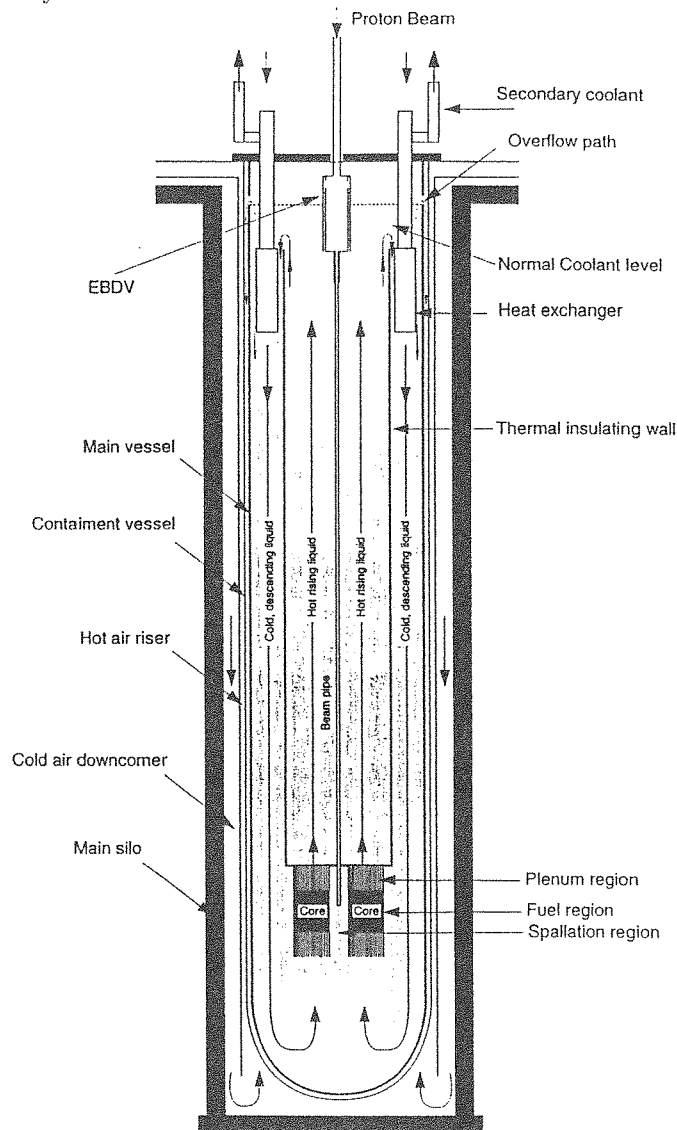


Fig. 7. *Detail of the reactor part of the Rubbia project (adapted from ref. [7]).*

Originally, this project was presented as devoted to energy production (the name "Energy Amplifier" was dubbed for this device). The necessity of consuming energy to power the accelerator (compared to ordinary reactors) is largely compensated by the

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[3]Through the sequence $n + {}^{232}Th \rightarrow {}^{233}Th \xrightarrow{\beta^-} {}^{233}Pa \xrightarrow{\beta^-} {}^{233}U$ .

interest of using $Th$ as a fuel. The proponents of the project speak of having so economical energy resources for 2000 centuries at twice the present rate of total energy consumption [7]. Emphasis was also put on safety features, principally the subcritical regime, and on reduction of waste generation. In a second version of the project [14], it is proposed to use this device as an incinerator as well, for fission fragments at least. This is based on a very nice idea. Neutrons emitted by the spallation source diffuse throughout the system (mainly $Pb$) by some kind of random motion. On the average, the energy of the neutrons is a function of time[4]

$$E = \frac{C}{(t - t_0)^2},\qquad(21)$$

where $C$ is a constant and $t_0$ is the time at which the neutron is emitted. Correlatively, the average displacement is, as in Brownian motion, a simple function of time, too. Therefore, the average neutron kinetic energy is a simple function of the distance of the source. Of course, there are fluctuations, but the latter do not seem to be very important. Advantage can be taken from the fortune circumstance that many fission fragments show strong resonances in the 0.1-10 KeV range, in the neutron absorption cross-section. Positioning each fission fragment at the right place, they can be incinerated with a high rate by neutron resonant capture. In the device (see fig. 7), the fission fragments will be positioned outside the core. One has to remind that lead is not efficient to slow down neutrons (contrarily to ordinary water). An experiment (TARC) has recently been conducted to study the feasibility of this process with a proton beam shooting on a lead target surrounded by a few cubic meters of (solid) lead blocks [15]. The results are very promising.

Other countries have launched other projects. There are two projects of hybrid systems in Japan, conducted by the JAERI Laboratory, whose aim is to incinerate $Pu$ and minor actinides. The first one is based on a solid core of metallic fuel, cooled with liquid $Na$, and a tungsten spallation source. The other one uses a molten salt as fuel, spallation source and coolant at the same time. In France and in Russia, many studies are in progress, although no project has attained the same status of development as the previous ones. Let us cite however the CEA-Cadarache project, a hybrid system with molten salts and rapid neutrons using $Th$ as fuel, and the CEA-INCA project, aiming to transmute minor actinides with a dedicated system.

It should be stressed that all these projects are still on the theoretical stage. The future major step in this study is the construction of a "demonstrator", a prototype of hybrid systems, whose characteristics is presently under debate.

# 4 Conclusion

This paper briefly describes the main trends and ideas concerning future nuclear fission reactors, putting the emphasis on the presentation of the physics involved and on the basic innovative technological features. I have tried to make the presentation as didactic as possible. For this reason, a reminder of nuclear reactor physics has been introduced, the presentation and sometimes the vocabulary have been simplified and perhaps oversimplified (I apologize to the purists), and the bibliography has been limited, as far as possible, to publications of general interest. For those who want to learn more

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[4]Of course, this formula has to be corrected for the very first instances of the neutron lifetime.

about nuclear reactor physics or about innovative developments, we recommend refs. [16, 17, 18] and [19, 20], respectively.

As we said in the Introduction, it is very hard to assess the role of nuclear energy in the next century. It is hardly conceivable that the worldwide energetical needs can be fulfilled without nuclear energy in the near future. Besides the uncertainties in the future population growth, a key issue is the time necessary for the technology based on renewable energy (solar, wind, biomass) resources to reach a high level of productivity at a reasonable cost. The experts do not foresee such a situation before the mid of the XXI$^{st}$ century. It is therefore expected that at least some of the projects I described will be pursued. This is, I believe, conditioned by a better image of nuclear energy in the public opinion. Such a change seems very unlikely nowadays, but may appear more plausible in a few years, if the public concerns about global climate change and the very limited oil reserves are increasing.

It is quite remarkable that many of the new ideas have been proposed or studied by physicists. The latter did not pay attention to issues of nuclear reactors for more than 40 years. In my opinion, this renewed interest has been triggered by the Tchernobyl accident and the rising problem of nuclear wastes. Of course, the physicists have investigated these new problems in close contact with the engineers. But, there is little doubt that the contributions of the physicists will be recognized as important.

It is not at all sure that really innovative nuclear reactors will be operating one day. The time scale for the full development of the hybrid systems is larger than 20 years, probably 40 years. By that time the renewable energy technics might be sufficiently developed. Even if it is not the case, there are economical constraints, that I did not discuss at all. It is rather clear that the economic operators (the electricity companies mainly) would like to adjust and improve the present technology rather than looking for another probably more sophisticated technology, which would require costly investments.

# References

[1] Conseil Mondial de l'Energie, "L'Energie pour le monde de demain", Editions Technip, 1993

[2] V.M. Mourogov, IAEA Bulletin, vol. 30, N° 2, (1997), 2

[3] R.A. Knief, Nuclear Engineering : "Theory and Technology of Commercial Nuclear Power", Taylor and Francis, Bristol, PA, 1992

[4] M.W. Golay and N.E. Todreas, Scientific American, April 1990, 58

[5] R.J. McCandless and J.R. Redding, Nuclear Engineering International, vol. 34, N° 424, (1989), 20

[6] P.E. Juhn, J. Kupitz and J. Cleveland, IAEA Bulletin, vol. 30, N° 2, (1997), 21

[7] C. Rubbia et al., Report CERN/AT/95-44 (ET), (1995)

[8] W.B. Lewis, Atomic Energy of Canada report DR-24, (1952)

[9] C.D. Bowman et al., NIM **A320** (1992) 336

[10] J. Cugnon, Spallation Reactions, preprint CEA DAPNIA/SPhN-96-38 and lectures to the 1996 Ecole Joliot-Curie

[11] J. Cugnon, C. Volant and S. Vuillier, Nucl.Phys. **A620** (1997) 475; **A625** (1997) 729

[12] H. Nifenecker, A. Giorni and J.M. Loiseaux, La Recherche, N° 301, (1997), 729

[13] H. Nifenecker and O. Méplan, Atelier Gédéon, France, 6-7 mai 1996

[14] C. Rubbia, Report CERN/LHC/97-04 (EET), 1997

[15] E. Belle et al., to be published

[16] R.L. Murray, Nuclear Energy, $4^{th}$ ed., Pergamon Press, (1993)

[17] J. Ligou, Installations Nucléaires, Presses Polytechniques Romandes, (1982)

[18] I.R. Cameron, Nuclear Fission Reactors, Plenum Press, New York, 1982

[19] J.P. Dufour et al., "Production d'Energie Nucléaire et Traitement des Déchets", Proc. of the 1996 Ecole Joliot-Curie de Physique Nucléaire, ed. by the IN2P3, Paris, France

[20] J.P. Schapira, Proc. of the Workshop on "Innovative Options in the Field of Nuclear Fission Energy", Les Houches, April 1998, to be published.