*Nuclear Development*

# Accelerator-driven Systems (ADS) and Fast Reactors (FR) in Advanced Nuclear Fuel Cycles

## A Comparative Study

*NUCLEAR ENERGY AGENCY*
*ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT*

# 4. ACCELERATOR-DRIVEN SYSTEM (ADS) AND FAST REACTOR (FR) TECHNOLOGIES

## 4.1 Introduction

The previous chapters have indicated that advanced nuclear fuel cycles incorporating P&T may include fast neutron spectrum reactors, whether of FR- or ADS-type. While both share, in principle, the characteristics of a fast spectrum and consequently share also fuel, material and coolant technology, distinctive technological differences occur between FRs and ADSs. These differences relate to:

- *Fuel*: as ADS would use more fertile-free fuel and would also allow higher concentrations of minor actinides in the fuel, this influences the fabrication and the reprocessing potential of such fuels.
- *Materials*: while the fast neutron spectra are quite comparable in a FR and in the core lattice of an ADS, the harder neutron spectrum in the source region of an ADS as well as the emission of energetic charged particles from the spallation source impose additional constraints on the choice and behaviour of the target materials, the adjacent reactor structures, and especially the beam window. Activation by high-energy particles is also a new issue.
- The *target and sub-critical lattice* in an ADS present new challenges for technology and is one of the major differences between ADS and FR.
- Finally, the need for an *accelerator* to drive the ADS is an additional component which needs further development towards higher performance and reliability.

This chapter will deal with these technological differences and especially the new requirements for ADS development. Chapter 5 will deal with the question of safety of ADSs versus FRs and will highlight additional technological aspects to complement those mentioned in the following chapter. Chapter 3 has already introduced the fuel fabrication and reprocessing issues and we shall therefore focus in this chapter on the reactor technology for FR and ADS.

In the absence of any specifically agreed international design, this chapter will discuss the main technological aspects which are design-independent and will highlight the following three questions:

- To what respect would the required ADS-technology differ from the already developed FR-technology and what are the additional developments needed?
- Can the existing FR technology basis be of specific use for ADS-development and is there scope for synergy between both developments? In other words, what is the extra effort needed?
- What are the main technological bottlenecks in ADS or FR developments for deployment of an industrial-scale waste transmutation system?

After a short history of FR-development and the current status of FR-technology, this chapter will give an overview of some ADS concepts and detail the technological challenges related to the development of ADS in general. A summary of the comparison of these technological differences will conclude this chapter.

## 4.2 Common grounds of ADS and FR technology

### *4.2.1 History and current status of existing FR technology*

The FR history is as old as that of thermal reactors. For the first 20 years of their existence, these two systems advanced side by side. The first FR was Clementine at Los Alamos (USA) in 1946 with a power of 150 kW. The first nuclear reactor in the world to generate electricity was a FR, the EBR-1 in the United States, in 1951. Around the sixties, as shown in Figure 4.1 and Table 4.1, four experimental fast reactors of about the same power went critical, and the oldest and largest of them, DFR (72 MWth), was successfully operated over 18 years. So also was Rapsodie later on. BOR-60 is still in operation now. After DFR, which used sodium-potassium, sodium was adopted as primary coolant.

The first prototype fast reactor for power generation was the US Enrico Fermi reactor (1964, 66 MWe). After three years of operation, this reactor suffered a fuel melting incident and was finally shut down in 1972. From 1972 to 1974, three prototypes of comparable size were successively brought into operation: BN-350 in the USSR (now in Kazakhstan), Phénix in France and PFR in the UK. The second is still in operation. BN-350 and PFR were finally shut-down in 1999 and 1994, respectively. The cores of these two plants operated satisfactorily, but the plants experienced steam generator problems. The SNR-300 prototype was built in Germany by a German-Belgian-Dutch consortium; plant and fuel were ready in 1985, but owing to a political impasse, the plant was never allowed to start up. The first criticality of the Japanese prototype Monju occurred in April 1994. Monju experienced a sodium leakage in the secondary loop in December 1995.

The stage of the large (pre-industrial) demonstration plants began with the start-up of BN-600 (600 MWe) in the USSR (Russia) in 1980 and Superphénix (1 240 MWe) in France in 1985. These achievements are further discussed below, together with those in Japan and other countries. Due to technical difficulties associated with the use of sodium as a coolant and economic problems in a saturating rather than expanding nuclear energy market, BN-600 and Superphénix remained the only industrial-scale fast reactors, meaning that the experience base for such reactors is much smaller than that for thermal reactors.

The motivation for building fast reactors has progressively changed. At the outset, the main objective for developing the FR was breeding in order to conserve uranium resources. It is easy to see the advantage of such a technology in an era of uranium shortage and price increase, as was forecast in the nineteen-seventies. In reality, however, uranium remained abundant and cheap, mainly because the growth rate of nuclear energy was lower than had been expected. Consequently, the use of FRs in a "burner" mode for managing excess plutonium gained in importance and remains today a particular focus of fast reactor R&D activities. Moreover, the desire to further optimise the back-end of the fuel cycle including the disposal of high-level waste has recently been stimulating an increasing interest in extending the application of the FR from the burning of plutonium to the burning (transmutation) of all transuranic actinides.

Table 4.1a. **Main features of constructed fast reactors**

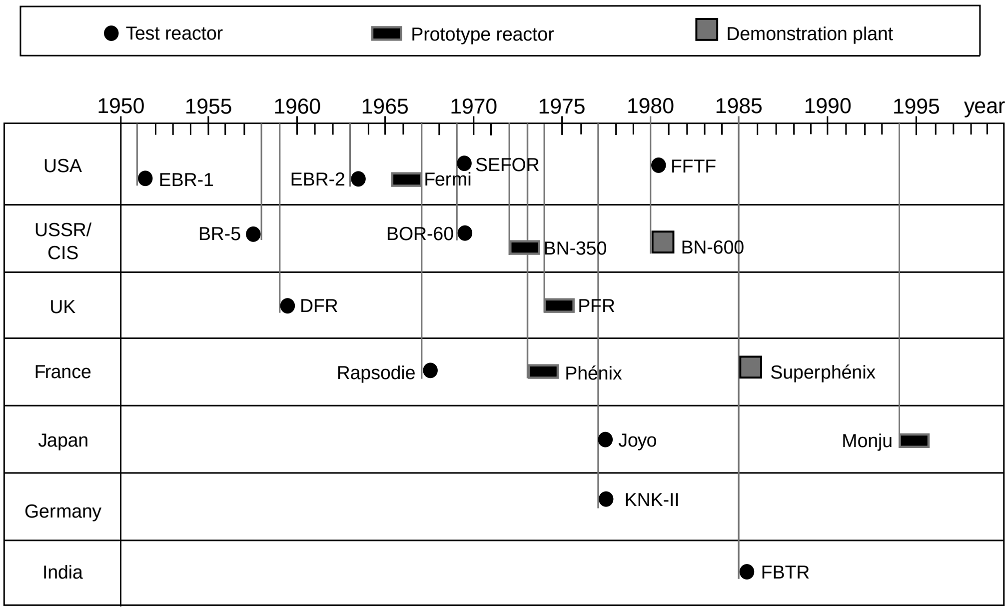
| Reactor name | Country | Location | First criticality date | Stutdown date | Thermal capacity (MW) | Electric capacity (MW) | Fuel | Primary circuit configuration | Primary coolant | Primary coolant temperature (°C) In/Out |
|---|---|---|---|---|---|---|---|---|---|---|
| Clementine | USA | Los Alamos | 1946 | 1953 | 0.025 | | Pu metal | | Mercury | 140/40 |
| BR-2 | CIS | Obninsk | 1956 | 1957 | 0.1 | | Pu metal | | Mercury | 70/40 |
| EBR-1 | USA | Argo (Idaho) | 1951 | 1963 | 1.4 | 0.2 | U | | Sodium/ potassium | |
| BR-5 | CIS | Obninsk | 1959 | 1971 | 5 | | $PuO_2$, UC | | Sodium | 450/375 |
| BR-10 | | | 1971 | | 10 | | MOX, UN | Loop | | |
| DFR | UK | Dounray | 1959 | 1977 | 72 | 15 | U-Mo | Loop | Sodium/ potassium | 350/230 |
| EBR-2 | USA | Argo (Idaho) | 1963 | 1994 | 62 | 20 | U-Zr, U-Pu-Zr | | | 482/370 |
| E. Fermi (EFFBR) | USA | Detroit | 1963 | 1972 | 200 | 66 | U-Mo | Loop | Sodium | 427/268 |
| Rapsodie | France | Cadarache | 1966 | 1982 | 20/40 | | MOX | Loop | Sodium | 510/404 |
| BOR-60 | CIS | Dimitrovgrad | 1969 | | 60 | 12 | MOX | Loop | Sodium | 550/360 |
| Joyo | Japan | Oarai | 1977(mark-I) | | 100 (Mark-II) | | MOX | Loop | Sodium | 500/370 |
| FBTR | India | Kalpakhan | 1985 | | 40 | | (U,Pu)C | Loop | Sodium | 518/400 |
| KNK-11 | Germany | Karlsruhe | 1977 | 1991 | 58 | 21 | $MOX/UO_2$ | Loop | Sodium | |
| SEFOR | USA | Arkansas | 1969 | 1972 | 20 | | MOX | Loop | Sodium | 430/370 |
| FFTF | USA | Hanford | 1980 | 1994 | 400 | | MOX | Loop | Sodium | 590/370 |
| PFC | Italy | Brasimone | Aband. | | 125 | | MOX | Loop | Sodium | 525/375 |
| BN-350 | CIS | Chevenko | 1972 | 1999 | 1 000 | 150 and desalinisation | $UO_2$ | Loop | Sodium | 500/300 |
| PFR | UK | Dounray | 1974 | 1994 | 600 | 270 | MOX | Loop | Sodium | 560/400 |
| Phénix | France | Marcoule | 1973 | | 560 | 250 | MOX | Loop | Sodium | 552/385 |
| SNR-300 | Germany | Kalkar | Aband. in 1991 | | 770 | 327 | MOX | Loop | Sodium | 560/380 |
| BN-600 | CIS | Beloyarsk | 1980 | | 1 470 | 600 | $UO_2$ | Loop | Sodium | 550/550 |
| CRBR | USA | Clinch River | Aband. in 1983 | | 975 | 380 | MOX | Loop | Sodium | |
| Monju | Japan | Tseruga | 1994 | | 714 | 280 | MOX | Loop | Sodium | 529/397 |
| Superphénix | France | Creys-Malville | 1985 | 1996 | 3 000 | 1 240 | 1 240 | Loop | Sodium | 545/395 |
| BN-800 | CIS | Beloyarsk | Suspended | | | 800 | 800 | | Sodium | 550/350 |

Table 4.1b. **Main features of fast reactors which have been operated**

| | **Joyo (Mark II) Japan** | **Phénix France** | **Monju Japan** | **BN-350 Kazakhstan** | **BN-600 Russia** | **Superphénix France** |
|---|---|---|---|---|---|---|
| **CAPACITIES** | | | | | | |
| Thermal capacity (MW) | 100 | 560 | 714 | 1 000* | 1 470 | 3 000 |
| Gross electric capacity (MW) | 0 | 250 | 280 | 150 | 600 | 1 240 |
| Net electric capacity (MW) | 0 | 233 | 246 | 135 | 560 | 1 200 |
| **CORE** | | | | | | |
| Active height/active diameter (m) | 0.55/0.72 | 0.85/1.39 | 0.93/1.8 | 1.06/1.5 | 1.02/2.05 | 1/3.66 |
| Fuel mass (tHM) | 0.76 | 4.3 | 5.7 | 1.17 $^{235}U$ | 12.1 ($UO_2$) | 31.5 |
| Number of assemblies | 67 | 103 | 198 | 226 | 370 | 364 |
| Maximum power (kW/l) | 544 | 646 | 480 | – | 705 | 480 |
| Average power (kW/l) | 475 | 406 | 275 | 400 | 413 | 280 |
| Expected burn-up (MWd/t) | 75 000 | 100 000 | 80 000 | 100 000 | 100 000 | 70 000 (first core) |
| **FUEL** | | | | | | |
| Fissile material | MOX | MOX | MOX | $UO_2$ | $UO_2$ | MOX |
| Enrichment (%) first core | 30 Pu | 19.3 Pu | 15/20 $Pu_f$ | – | – | 15.6 Pu |
| Mass of plutonium (t) first core | | | | | | 6 |
| Enrichment (%) reloads | 30 Pu | 27.1 Pu | 16/21 $Pu_f$ | 17/21/26 | 17/21/26 | 20 Pu |
| Mass of plutonium (t) reloads | | | | | | 7 |
| Assembly renewal rate | 70 days | 3 months | 20% of core every 5 months | 80 efpd | 160 efpd | 100% of core every 3 years |
| Form | pellet | pellet | pellet | pellet | pellet | pellet |
| Number of pins per assembly | 127 | 217 | 169 | 127 | 127 | 271 |
| Assembly geometry | Hexagonal | Hexagonal | Hexagonal | Hexagonal | Hexagonal | Hexagonal |
| Average linear power (kW/m) | | | 21 | 36 | | |
| Maximum linear power (kW/m) | 40 | 45 | 36 | 48 | 48 | 48 |
| Maximum clad temperature (°C) | 650 | 700 | 675 | 700 | 620 | 620 |
| Maximum temperature at centre (°C) | 2 500 | 2 300 | 2 350 | 2 200 | | |

* Real thermal capacity is 520 MW.

Figure 4.1. **Fast reactor programmes: start-up of reactors (first criticality)**




*4.2.1.1 Fuels for FRs*

From the outset, many types of fuel were tested: enriched uranium or plutonium in metallic, nitride, oxide or carbide form, or a mixture of plutonium and uranium oxides. It is worth noting that the past fifty years of fast reactor fuel development have witnessed changes in popularity of the various fuel types from the initial use of metal to emphasis on oxide fuels, then to ceramics (mostly oxide) and finally back to both oxide and metal, as performance demands and priorities have changed.

Because of fuel swelling at high burn-up and for compatibility with the cladding, pure metallic Pu/U alloy is no longer used. Today, fuels are essentially composed of ceramics obtained by sintering. The fuel most widely used at present is in the form of a mixture of plutonium and uranium oxides, $(U,Pu)O_2$ with up to 30% Pu-content (higher Pu-contents being problematic for aqueous reprocessing) (See Table 4.2). The fertile material is natural or depleted uranium oxide. It is located in the matrix of the fuel itself, and, as a breeder, in axial and radial blankets surrounding the core.

FR-MOX fuel manufacture is similar to the manufacture of MOX fuel for LWRs, including high-temperature sintering where the fuel is in the form of solid pellets, annular pellets or vibro-fuel. Major differences are that the fuel stack is housed in a steel cladding tube, and that pin clusters are placed within a hexagonal steel wrapper tube. Since the clad of the pins acts as a first barrier, it must be compatible with the fuel and the coolant (molten sodium) to guarantee mechanical strength and tightness for as long as possible. The fuel material has presented few limiting factors, even when performance targets have been extended by a factor of three. It has been the cladding material rather than the fuel itself which has had the greater influence. Stainless steel is the material that best meets these requirements today and, as demonstrated by various tests, allows to reach burn-ups of between 100 000 and 200 000 MWd/t.

Table 4.2. **Irradiation performance of MOX fuel in fast reactors (major achievements)**

| **Country or group of countries** | **Standard MOX fuel**[1] | | **Experimental fuel** | | |
|---|---|---|---|---|---|
| | N°. of pins irradiated | Burn-up reached MWd/t | Maximum burn-up MWd/t | Main reactors[3] | Type of fuel[3] |
| Western Europe | 265 000 | 135 000 | 200 000[2] | Phénix, PFR, KNK-II | Solid and annular pellets |
| United States | 64 000 | 130 000 | 200 000 | FFTF | Leading pins |
| Japan | 50 000 | 100 000 | 120 000 | Joyo | Solid pellets |
| CIS | 13 000 | 135 000 | 240 000 | BOR-60 | Vibro-pac fuel |
| | 1 800 | 100 000 | – | BN-350 | Solid and annular pellets |
| | 1 500 | 100 000 | – | BN-600 | Solid and annular pellets |

1. The distinction between "standard" and "experimental" fuel is not obvious. "Standard" refers to the bulk of fuel pins comprised in full sub-assemblies and irradiated without special management measures.
2. The figure of (approximately) 200 000 MWd/t of heavy metal corresponds to pins loaded in PFR.
3. This summary is not comprehensive. Neither all reactors, nor all fuel types are listed.

Table 4.3 lists the main manufacturing facilities, as well as the reactors that they have supplied. Manufacturing capacity was relatively low, and only the Cadarache plant in France had been designed for an industrial-scale capacity, i.e. that required for the Superphénix cores.

Table 4.3. **Fuel manufacturing facilities for fast reactors**

| **Country** | **Manufacturing plant** | **Capacity (tHM/y)** | **FR supplied by the plant** |
|---|---|---|---|
| Belgium | Dessel, Belgonucléaire | 5 | SNR-300 |
| France | Cadarache, COGEMA | 20 | Rapsodie, Phénix, Superphénix |
| Germany | Hanau, SIEMENS | 10 (b) | KNK-II, SNR-300 |
| Japan | Tokai-mura, PNC | 10 | Joyo, Monju |
| United-Kingdom | Windscale, BNFL | 5 | PFR |
| USA | Apollo, Babcock-Wilcox (ex: NUMEC) | 5 (a) | FFTF |
| Russia | Chelabinsk, Paket at Mayak | 0.3 | BN-350, BN-600 |
| | Dimitrovgrad, RIAR | 1 | BN-600 |

(a) Now dismantled.
(b) Now permanently shutdown.

Today, technological maturity has been attained based on mixed oxide for the fuel, fertile blanket of depleted uranium oxide (in the breeder mode), and stainless steel for the pin cladding and the assembly wrapper tube.

However, other avenues are still being explored, such as carbide or nitride fuels, as will be discussed below.

*4.2.1.2. FR-fuel reprocessing*

While the reprocessing of FR fuels makes use of the same process as is used for thermal reactor fuel (the PUREX process), a number of special factors must be taken into account: the presence of sodium, use of stainless steel cladding and structural components, high residual power, and high plutonium content.

While the experience gained is much smaller than that for reprocessing of thermal reactor fuels, France and the United Kingdom possess experience in reprocessing FR assemblies. In addition, lots of fuel from BOR-60 were also recycled. The FR fuel has been reprocessed either in specialised installations (e.g. the AEA plant at Dounreay in the United Kingdom, or the Marcoule site in France), or diluted with fuel from thermal power plants (La Hague, France). Part of the fuel from the Phénix and PFR prototypes was reloaded into the core after two successive reprocessing operations, thus demonstrating the complete fuel cycle.

*4.2.1.3. Reactor coolant choices*

Given the high power density, molten metals selected for their high thermal conductivity are used as coolant. This type of coolant allows the reactor to be operated at low pressure, thus reducing the probability of a loss-of-coolant accident. The liquid metal used in the first test reactors was mercury, but this was soon replaced by sodium (Na) which is common and cheap. Sodium melts at 98°C and boils at 880°C, giving it a wide service range. Its density at these temperatures is comparable to that of water, so that well-known pumping technology can be used. Yet, sodium presents certain drawbacks: at operating temperature, it ignites spontaneously in contact with air; also, it reacts violently with water, meaning that sodium-water reactions must be considered in designing the steam generators. Today, the technology for controlling these problems is well developed.

Another unfavourable effect is the activation of the sodium in the core. This entails the construction of a secondary, inactive sodium circuit which separates the active sodium from the steam generator, and hence a cost disadvantage. The primary circuit may be of the loop type (e.g. Monju) or the pool type, i.e. fully integrated in the reactor vessel (e.g. Superphénix). The latter concept allows all the active sodium to be confined in the main reactor vessel.

### *4.2.2 Current trends in FR technology development*

*4.2.2.1 Alternative coolant choices for fast reactor systems*

Sodium has so far been universally adopted as the coolant for prototype and demonstration reactors; it has the desired, favourable heat transport and neutronics characteristics, is compatible with steel structural materials, and is available at a relatively low cost. On the other hand, it forms a radioactive activation product, $^{24}Na$, and reacts chemically with water and air.

More recently, the difficulties with sodium reactions experienced at prototype and demonstration reactors have led to a renewed interest in alternative liquid metals which may have a number of advantages such as [63]:

- Increased economic competitiveness, if the plant can be simplified, e.g. by suppressing the intermediate heat transfer circuit.
- Increased inherent safety.
- Resistance to proliferation, e.g. through a very long reactor lifetime without refuelling.

Three developments are presently stimulating research on alternate liquid metal coolants:

- The opening of the nuclear sector of Russia, which gave access to information on the lead-bismuth eutectic (LBE) technology used in reactors for submarine propulsion [64,65], and projects for lead-cooled fast reactors [66,67].
- The programme on new and innovative technologies supporting the nuclear option in the 21$^{st}$ century launched by the US DOE ("Generation IV") [68,69] and also a similar programme in Japan [70,71], both including sodium, lead and lead-bismuth cooled fast reactors.
- The research on transmutation of nuclear waste in sub-critical, accelerator-driven reactors (the subject of this report): consideration of lead or lead alloy for spallation targets prompted their consideration as the reactor coolant, too.

Table 4.4. lists the advantages and disadvantages of Pb and Pb-Bi as compared with sodium.

Table 4.4. **Comparison of Pb(-Bi) versus sodium as reactor coolant**

| Advantages of Pb and Pb-Bi | Disadvantages of Pb and Pb-Bi |
|---|---|
| Lead- and lead-bismuth-cooled cores have a **smaller positive void reactivity effect** (the positive void reactivity effect is a much-criticised feature of sodium-cooled fast reactors).<br><br>A smaller positive void reactivity effect allows an ADS core to be operated at a higher $k_{eff}$ and correspondingly **lower proton current**.<br><br>**Lead**: In contrast to sodium, Pb is **not activated** in critical reactors (does not apply to an ADS in which Pb is activated by high-energy (n,p) reactions).<br><br>High boiling temperature (1 743°C for Pb and 1 670°C for Pb-Bi vs. 880°C for sodium) implies **reduced potential for boiling-induced accidents**.<br><br>Hot liquid Pb and Pb-Bi does not violately react with air.<br><br>The **absence of a chemical reaction with water** may allow loop-type reactors to be designed with a simplified heat transport system.<br><br>**Lead**: In the event of a hypothetical fuel melting accident, frozen lead may provide an **effective barrier against radiation and a radioactivity release** due to the high melting temperature (328°C for Pb vs. 98°C for sodium and 123°C for Pb-Bi). | **Lead-bismuth**: Neutron capture in $^{209}$Bi produces the alpha emitter $^{210}$**Po**.<br><br>**Lead**: The high melting temperature (328°C for Pb vs. 98°C for sodium and 123°C for Pb-Bi eutectic) implies an **increased potential for coolant blockage accidents**.<br><br>Important functions in Pb and Pb-Bi cooled systems are jeopardised by **erosion and corrosion** (this item has been solved for sodium). In combination with the poor inspectability of liquid-metal cooled systems, this could pose significant **safety problems** (structure failures, blockages by sludge).<br><br>The **high density** of Pb and Pb-Bi (or the associated high static pressure)<br>– complicates the accelerator-reactor interface design;<br>– calls for design measures against floating of core structures;<br>– complicates the seismic design of the plant;<br>– increases the probability of loss-of-primary-coolant accidents;<br>– increases pumping power needs.<br><br>In primary systems with natural circulation, the properties of Pb and Pb-Bi may make the **response of the system to heat balance disturbances very sluggish** (favours a start-up accident).<br><br>**Less experience exists** than for Na as most FRs are sodium-cooled (Pb-Bi used in submarine reactors, Pb not used in any operating reactor). |

Lead, as a coolant, has three important advantages over sodium: it boils at high temperature (1 743°C), does not react with water or air, and is not activated by fission neutrons. However, lead has the disadvantages of high density and low heat conductivity, is corrosive for steels and has a high melting temperature (328°C) with the risk of freezing.

The lead-bismuth eutectic (LBE), which has been used as coolant for submarine reactors in Russia [64,65], can be operated at lower temperature (melting temperature 123°C, close to that of sodium), which improves, in comparison with lead, the compatibility with structural materials and reduces coolant freezing risks. However, bismuth produces the radioactive, volatile nuclide $^{210}Po$ ($t_{1/2}$ = 138 days) which becomes a radiological hazard in case of coolant leakage.

Other liquid metals are also possible coolants, but are less suited for large-scale application (e.g. mercury which is used as a target in pulsed neutron sources) or have never been used in reactors (e.g. tin).

Fast reactor concepts using gas coolants (helium, carbon dioxide) were studied in the past [72,73]. Such concepts are now being revisited, mainly because the transparent gas atmosphere facilitates in-service inspection and maintenance and the heat transport circuits can be simplified.

Table 4.5 lists the advantages and disadvantages of He and $CO_2$ as compared with liquid metals.

Table 4.5 **Comparison of gas versus liquid metals as reactor coolant**

| **Advantages of He and $CO_2$** | **Disadvantages of He and $CO_2$** |
|---|---|
| The **steam entry reactivity effect** of gas-cooled fast reactors, which can be positive with normal fuels, is of **less concern** than the coolant void reactivity effect of liquid-metal cooled fast reactors.<br><br>Gas-cooled fast reactors are **neutronically suited for transmutation** because minor actinides in the fuel have a beneficial influence on the steam entry reactivity effect which can be positive with normal fuels (the opposite is true for the coolant void reactivity effect of liquid-metal cooled fast reactors).<br><br>Transparency of gas coolants **simplifies inspection** of internal structures.<br><br>**Helium**: Is **inert and not corrosive**.<br><br>**Heat transport system can be simplified** (no need for intermediate circuits, possibility to use a helium turbine).<br><br>**Simplified handling of irradiated fuel** (no cleaning from rests of frozen coolants).<br><br>**Experience available** from $CO_2$-cooled commercial thermal reactors (like Na, He has only been used in prototype reactors). | High coolant pressure implies **potential for loss-of-coolant accidents**.<br><br>High pressure difference across reactor-accelerator interface constrains **design options** regarding e.g. the pressure vessel (choice of a steel pressure vessel of moderate size rather than the proven large pre-stressed concrete pressure vessel, which limits the power of the core).<br><br>Lacking thermal inertia of unmoderated cores implies **fast transients** in off-normal conditions.<br><br>Less favourable heat transfer characteristics imply a **limitation in fuel power density** and hence suitability for transmutation.<br><br>High flow rates of hot dense gas induce dynamic loads and hence **noise and vibration**.<br><br>**$CO_2$**: May require **gas chemistry and corrosion control**. |

Helium is particularly attractive as it is a chemically and neutronically inert single-phase gas which is not activated. Disadvantages of gas coolants are the less favourable heat transfer characteristics and the higher operating pressure which requires depressurisation accidents to be dealt with in the safety analysis.

Finally, attention should also be drawn to new initiatives for using water as an advanced reactor coolant:

- In Japan [73], the usual BWR concept is being adapted towards a tight lattice fuel bundle and a higher void fraction, to obtain a breeding ratio of unity or slightly above.

- In Europe [74], an LWR operating in a thermodynamically supercritical regime, i.e. water enters the reactor as liquid and exits as high pressure steam without a phase change, is being studied.

These recent developments, based on LWR technology, are not especially aimed at actinide transmutation, but rather at improving resource utilisation and plant optimisation.

*4.2.2.2 Perspectives for fast reactors*

In essence, a major slow-down in FR development occurred in almost all OECD countries since the mid-1980s. In addition, in the past five years, R&D on fast-spectrum systems has shifted from new FR to ADS concepts. Nevertheless, FR concepts have been continuously studied and developed as, for example:

- The Self-consistent Nuclear Energy System (SCNES) in Japan.
- The Integral Fast Reactor system (IFR) in the USA.
- The European Fast Reactor (EFR) in Europe.

The following paragraphs will give an overview of the FR activities in some Member countries and briefly describe the respective developments.

*Japan*

Test and prototype reactors

The development of fast reactors in Japan has been based on Joyo and Monju. The experimental reactor Joyo, which went critical for the first time in 1977 with a Mark-I core at an initial power of 50 MWth (later increased to 75 MWth), was equipped in 1982 with a Mark-II core of 100 MWth. This core, in which the blankets are replaced by stainless steel reflectors (to become a burner), has been operated for 35 duty cycles to test fuels and materials. Since June 2000, the reactor is being upgraded to the Mark-III core, that will provide enhanced irradiation capabilities (about 30% higher neutron flux, increased plant availability, upgraded irradiation technology). The maximum fuel burn-up achieved so far in Joyo is 71 000 MWd/t (fuel pin average).

Monju, a prototype reactor of 714 MWth (280 MWe), went critical for the first time in April 1994, and supplied electricity to the grid for the first time in August 1995. The first core of Monju was loaded with MOX fuel at a plutonium enrichment of 20 and 30% in the inner and outer core zones, respectively. At equilibrium, one fifth of the core fuel is to be discharged at each refuelling; the discharged burn-up will reach 80 000 MWd/t (average). The successive core loads of Monju are planned to consist of the same MOX type fuel. Higher burn-up values will progressively be attempted.

In December 1995, during pre-operational testing at 40% power, Monju experienced a sodium leak in the secondary, inactive sodium circuit, caused by a rupture of a temperature detector. Since then, the reactor is shut down for repair work. After authorisation to proceed with the design modifications, at least 4 years are needed before plant operation can be resumed.

Self-Consistent Nuclear Energy System – SCNES

In Japan, a self-consistent nuclear energy system (SCNES) [75-80] has been defined as a system satisfying four objectives, i.e. energy generation, (fissile) fuel breeding, confinement of minor actinides and radioactive fission products and, last but not least, guaranteeing nuclear safety. The term "self-consistent" means that even if the system produces materials dangerous to human beings and the environment, it can eliminate these materials within the system itself. The potential of large fast breeder reactors with MOX, nitride and metallic alloy fuels has been studied in relation to nuclear safety and their ability to transmute radioactive nuclides .

Some long-lived fission products (LLFP) with relatively large capture cross-sections, namely $^{79}Se$, $^{99}Tc$, $^{107}Pd$, $^{129}I$, $^{135}Cs$ and $^{151}Sm$, are selected for transmutation, based on their having effective half-lives of under 10 years, and the others are confined in the system. The LLFPs can be transmuted in the radial blanket and part of the axial blanket regions. In order to enhance transmutation efficiency, neutrons in the radial blanket region are moderated by solid hydride ZrH1.6. Minor actinides are recycled and transmuted as a fuel and can be confined to the system without any significant impact on nuclear and safety characteristics. The hazard index level of the LLFPs per ton of spent fuel from the SCNES after 1 000 years is as small as that of a typical uranium ore.

In case of a metallic fuel, a slug of U-TRU-10%Zr alloy with a few percent MA would be used in a homogeneous recycling scheme. The proposed reprocessing method for this metallic fuel would be a sequence of electro-refining, retorting, filtration, electro-migration and finally injection casting. Metallic technetium would be precipitated at the bottom of the cadmium pool in the electro-refining cell while metallic palladium is dissolved in the pool. Selenium would be recovered from the off-gas system in the pre-treatment process. Finally, iodine could be recovered as NaI in the pre-treatment process before electro-refining where caesium remains as chloride in the electro-migration process for salts.

A nitride fuel actinide recycle system coupled with nitride FBRs and pyrochemical reprocessing was also investigated in order to establish a confinement and transmutation system for long-lived radioactive nuclides. The results of these studies are summarised as follows:

- The use of nitride fuel permits an excellent fast reactor core performance; i.e. small or negative void reactivity and nearly zero burn-up reactivity changes.
- A transmuter with 5 wt% MA-content can support more than 6-10 units of PWR spent fuel a year. After several recyclings, the MA compositions almost reach equilibrium, except for $^{246}Cm$.
- The toxicity of $^{14}C$ produced by using $^{14}N$ (natural nitride) becomes almost equivalent to that of americium and curium after 5 000 years, and the toxicity for 90% enriched $^{15}N$ is similar to that of FPs after 1 000 years. Therefore, the recovery of $^{14}C$ becomes a very important issue to which pyrochemical reprocessing could be a solution.

- The pyrochemical process can be adapted to reprocess nitride spent fuel. Evaluation of the process shows that actinides are reasonably well separated from fission products, and that the high level wastes are nearly actinide-free.
- Preliminary studies for the plant design also showed that the fuel cycle cost of this coupled system could be substantially reduced by employing pyrochemical reprocessing, owing to its simplicity and compactness.

*France and Western Europe*

Prototype and demonstration reactors

Research work has been on the way for nearly 30 years, not only in France, but also in the neighbouring European countries Belgium, the Netherlands, Germany, Italy, and the United Kingdom. Work, centred on DFR, KNK-II, SNR-300 and PFR has now been terminated, while more generic R&D work is still being pursued in some laboratories (especially linked to plutonium incineration). The plan is to build a European Fast Reactor (EFR) as a successor to Superphénix, but the respective decision has now been postponed to 2010 or later.

As for the 250-MWe Phénix prototype, plans are to restart it and to operate it for another 6 irradiation cycles. Phénix had been shut down from 1990 to 1994, and again from 1995 up to now, first to take measures to avoid unintended reactor shut-downs as observed in 1989 and 1990, and secondly to refurbish the secondary sodium circuits. However, the maintenance, inspection, and repair work proved to be much more extensive than originally planned, with the safety authority asking, among other requests, for seismic upgrade measures, repairs to all the steam generators, visual inspection of the upper internal structures of the reactor block, and ultrasonic inspections of the welds of the reactor core shell. Upon completion of the renovation work, resumption of operation at 2/3 nominal power is planned for the summer of 2002. At this power level, the planned 6 irradiation cycles would correspond to a period of approximately 5 years which would keep the reactor in operation until 2008.

The major achievements on MOX fuel in the prototypes Phénix and PFR can be summarised as follows:

- Pin cladding materials consisting of austenitic steels and nimonic alloys have been optimised and tested in high burn-up irradiations at PFR to more than 135 000 MWd/t for full sub-assemblies. Average burn-ups above 180 000 MWd/t have been achieved with experimental fuel.
- Ferritic wrapper tubes, virtually non-swelling under the impact of fast neutrons, have been developed to accommodate these high burn-ups.

The 1 200 MWe Superphénix (SPX) plant, owned by the European electricity utility NERSA, a joint venture between utilities from six countries, had been restarted in 1994 after a four year interruption initiated by a pollution of the primary sodium, which was followed by a public hearing. It was then re-licensed to be progressively converted from a plutonium breeder to a burner, following the recommendation of a governmental commission (the so-called Curien Commission). To that end, steel reflector assemblies have been fabricated to replace the radial, fertile blanket. Three test assemblies have also been manufactured: two of CAPRA type, differing in the origin of their plutonium, either from first or second generation, and one NACRE assembly containing 2% neptunium added to the usual MOX, as part of the SPIN programme.

During 1996, Superphénix went critical for 266 days, i.e. 95% of the scheduled operation time. The reactor was shut down at the end of December, having reached 320 equivalent full power days, which corresponds to a burn-up of 35 000 MWd/t (maximum) for the core.

In 1997, the decision by the French government, confirmed in February 1998, to definitely shut down Superphénix, lead to a marked reorganisation of the fast reactor programme in France. Concerning the investigations on enhanced TRU burning, a partial redeployment in Phénix of former Superphénix experiments is planned.

The European Fast Reactor collaboration

The European Fast Reactor (EFR) collaboration was established in 1988 when the participating organisations launched a program of design and validation activities, which was pursued for ten years. During the first two-year phase – the Conceptual Design phase – the best features of the national commercial fast reactor projects were integrated into a compact EFR "first consistent design" along with alternative and fallback options. This was followed by a three year Concept Validation phase in which the system engineering for EFR was completed and the R&D results were integrated into the design.

The initial major objectives set by the utilities for the EFR were:

- An up-to-date safety standard, comparable with that of future LWRs, and licensability in the participating countries without significant design changes.
- Potentially competitive electricity generating costs compared with future LWRs.

These objectives were supplemented by the recommendations which resulted from the 1993 safety and economic assessments, the feedback of Superphénix operating experience, and the prospects for new missions for fast reactors in the nuclear fuel cycle. As a result, emphasis was placed on:

- Demonstration demonstrating that a high load factor could be achieved in combination with ambitious operating and safety standards, notably with consideration of severe accidents in the containment design.
- Progress on provision for in-service inspection and repair.
- Flexibility regarding the missions in the fuel cycle and in particular possible integration of CAPRA core designs into the Reference EFR.

To meet the goal for economics it was essential, in addition to minimising the plant investment cost, to produce a design which would ensure both high plant availability and a lifetime target similar to that for future LWRs (possible extension to 60 years). This necessitated special attention to components and structures where failure would lead to prolonged outage for repair (of which the permanent reactor structures, the heat exchangers and the steam generators are particularly important), and to developing efficient in-service inspection and repair methods. The approach adopted was to use, as far as possible consistently with the other requirements, technologies which were already verified or which could be expected to be fully endorsed by R&D. Considerable attention was given to the development of well founded and validated design rules.

EFR safety approach

A prime feature of the safety design of EFR is the extensive application of "defence-in-depth" principles. The successive protection levels include:

- Careful selection of appropriate materials, sound basic design backed up by extensive R&D, strict application of quality assurance procedures.
- The provision of systems to detect failures or deviations from normal operation and to prevent such failures from escalating into fault conditions.
- Protective systems and engineered safety features incorporated into the design to cope with classical and other initiators.
- Preventing failures of equipment or human error from leading to accidents.
- Providing several sequential physical barriers to prevent any hazardous radioactive release to the environment.
- Ultimate risk minimisation measures to enhance further the reliability of shut-down and decay heat removal and the retention capability of the containment.

Because the reactor is not pressurised and is surrounded by a close-fitting guard vessel, uncovering of the core due to loss of coolant is precluded and measures to prevent core melting are concentrated on enhanced shut-down and decay heat removal. Through these preventive measures the risk of core melting is reduced to an extremely low level and beyond the objectives generally set for future reactors. Nevertheless, according to the most demanding safety requirements, the prevention level is supplemented by a mitigation approach in which consequences of core melting are considered in the design of the containment system.

EFR core design

The EFR core design has been optimised for safety through:

- The choice of fuel pin linear rating, dictated by the prevention of local fuel melting in case of inadvertently withdrawing absorber rods withdrawal.
- The choice of core height to minimise the positive reactivity effect of sodium voiding.
- The Doppler coefficient to provide efficient reactivity feedback in rapid transients.
- The use of annular fuel pellets to prevent escalation of core accident sequences involving fuel melting.

Recent progress has been made, in the framework of international R&D agreements, on developing advanced computer codes for assessing the behaviour of optimised fast reactor cores, like the EFR core, under extreme conditions (core disruptive accidents). Applying these codes has shown that the consequences are benign, and the level of energy liberated is small compared with the strength of the primary system boundary.

Integrating a plutonium and minor actinide burning core can lead to two EFR variants (breeder and burner) to avoid compromising either as an economic power generator, but it is desirable that the differences should not be extensive. In the first CAPRA feasibility studies, this desire to accommodate a breeder or burner core with only minor adaptation of the reactor was demonstrated to be achievable. A burner core with the same power, core envelope and absorber rod arrangement as for EFR was proposed, but with a different pin diameter and fuel subassembly pitch to improve the plutonium burning efficiency. The diagrid and above-core structure therefore have the same overall dimensions for both breeder and burner cores, but the detailed geometry has to be adapted to the sub-assembly pitch, giving an otherwise identical reactor based on the same technology.

*United States*

Past achievements

Between 1953 and 1994, several experimental fast reactors were operated in the USA (see Table 4.1a). Two of these, the Experimental Breeder Reactor 2 (EBR-2) in Idaho and the Enrico Fermi Fast Breeder Reactor (EFFBR) in Michigan, achieved significant electrical outputs (20 MWe and 66 MWe, respectively). The latter suffered a fuel melting incident and was finally shut down in 1972. The Fast Flux Test Facility (FFTF, 400 MWth) started in 1980 with a mixed oxide fuelled core and reached its burn-up target of 100 000 MWd/t for a full core load in 1987. The US fast reactor programme came to a halt in 1994 when the government decided to shut down EBR-2 permanently, to put FFTF in standby conditions, to cancel the Clinch River Breeder Reactor (CRBR) project, and to abandon the Integral Fast Reactor (IFR) project (see below) [81]. At the end of 2001, the US Department of Energy finally announced its decision to terminate also FFTF.

The IFR concept, a new fast reactor initiative promoted by Argonne National Laboratory, is based on a fuel consisting of a ternary alloy of uranium, plutonium and zirconium. This ternary alloy was conceived as a successor to the electrometallurgically reprocessed alloy which had already been recycled in EBR-2. Such alloys remain compatible with the steel cladding up to high burn-up values. The concept was integrated by General Electric into a full plant design called PRISM (Power Reactor Innovative Small Module), consisting of nine reactor modules contributing 135 MWe each. The PRISM design was subsequently modified to raise the electric output per module to about 300 MWe (PRISM Mod. B). All IFR designs were based on full actinide recycling using an electrometallurgical processing plant co-located with the reactor complex.

The IFR concept

The integral fast reactor (IFR) was under development by the US Department of Energy for the decade between 1984 and 1994 [3,82]. Technology development was carried out at Argonne National Laboratory, and an industrial team led by the General Electric company utilised the technology for a specific nuclear power plant design, the Advanced Liquid Metal Reactor (ALMR).

The IFR fuel cycle[36] consists of a fast-spectrum nuclear reactor using liquid metal (sodium) cooling and metallic alloy fuel, coupled with recycle technology based upon an electrometallurgical partitioning of fission products from actinides. The actinides are combined with make-up uranium feedstock and multiply recycled through the reactor for total consumption by fission while the fission products are stabilised in waste forms suitable for long-term disposal.

IFR recycle technology

The recycle technology development was designed to meet two key requirements, i.e.:

- To retain all TRU intimately admixed during every recycle step back to the reactor (alternately, the TRU recycle product was not required to be free from fission products because the minor actinides already made remote handling a necessity).
- To achieve a fission product waste stream essentially free from TRU.

The first key requirement – that of keeping the TRU always intimately mixed – was to assure that the IFR fuel cycle introduced no diversion or proliferation vulnerabilities not already present from the

36. See fuel cycle scheme 5 in Chapter 3.

LWR once-through fuel cycle. The higher actinides (primarily americium and curium), which are intimately mixed with plutonium throughout the recycle, not only serve as fuel contributing to the efficient use of uranium resources, but together with the residual fission products they and the higher plutonium isotopes render the recycle material compositionally unattractive and radioactively unapproachable as is LWR spent fuel. The second requirement was to assure not only that all actinides were recycled to the reactor for total fission (resource utilisation) but also that the fuel cycle waste stream is comprised solely of fission products that would decay to radiotoxicity levels as small as that of the original ore during a time period of only 300 to 500 years. Such a period would be short enough for engineering and institutional measures to assure waste isolation with a high degree of confidence until the hazard had decayed away (waste, environment).

The electrometallurgical recycle technology utilises an electrolyte bath of lithium and potassium chlorides above a cadmium-pool, both at 500°C. Chopped fuel pins are anodically dissolved out of the steel cladding and into the electrolyte at a few volts; the reactive FPs and bond sodium remain in the electrolyte, the noble metal FPs and cladding hulls remain as solids in the chopped-pin basket or dissolve in the lower cadmium-pool, and the uranium is deposited on a solid cathode. All TRU and some lanthanide fission products deposit in a separate liquid cadmium-cathode. The chemical free energies of the chlorides are such as to guarantee that all TRUs co-deposit to avoid the possibility of a pure plutonium product. The uranium and TRU cathodes are removed from the electro-refiner and separately retorted at about 1 300°C to recover entrained electrolyte salt and cadmium for recycle to the electro-refiner. The resulting actinide ingots are blended with a make-up of depleted uranium and zirconium alloy and are injection cast at about 1 300°C into fuel slugs which are sodium bonded inside steel cladding for recycle to the reactor.

The radioactivity of the recycled products requires that all recycle process steps occur behind heavy shielding. This shielding requirement motivated the selection of simple and compact technologies for recycle and refabrication in order to minimise the volume and associated construction costs of hot cell facilities. Electrometallurgical processing and refabrication of metallic alloy fuel by injection casting were selected in part for the sake of the large reduction in criticality-limited process equipment and hot cell volumes attainable in the absence of a neutron-moderating aqueous matrix that would reduce critical masses by a factor of 50. Sodium bonding of fuel to the clad allows for loose dimensional tolerances on the cast pin slugs, and the fast neutron spectrum allows for loose compositional tolerances on fission product carry-over from electrometallurgical processing for recycle back into the reactor. Such loosening of tolerances facilitated remote processing.

IFR fuel technology

The IFR fuel is a metal alloy of 10 wt% zirconium, 15 to 25 wt% recycled TRU, and the remainder of depleted uranium. The zirconium-content was selected to achieve a high enough solidus (1 180°C) for operation at reactor outlet temperatures of 590 to 600°C but a low enough liquidus (1 300°C) to facilitate the injection casting fabrication process. The fuel slugs are sodium bonded inside austenitic or ferritic stainless steel cladding. Smear density (area ratio of fuel slug to inside clad) is set at 75% so as to allow for fission gas induced porosity in the fuel slugs to interconnect by the time the fuel has swelled radially to contact the clad at about 1.5 atom% burn-up. The interconnected porosity provides an escape route for fission gas to the (upper) gas plenum, terminates radial swelling at 30 v%, and thereby precludes mechanical interaction between fuel and clad driven by fission gas induced swelling.

An extensive fuels irradiation program on uranium-zirconium and uranium-plutonium-zirconium IFR fuel was conducted from 1983 to September 1994; it verified favourable fuel pin performance with peak burn-ups up to 20 atom% (200 MWd/kg) at linear heat rates of up to 500 W/cm. Transient testing – both operational tests and accident tests to fuel pin disruption – established a database of properties for use in safety and licensing activities. A safety case was developed to "license" the conversion of the EBR-II core loading to IFR recycled fuel assemblies (containing fission product carry-over), but the IFR programme was terminated before electrometallurgically - recycled fuel could be reintroduced into EBR-II.

IFR passive safety technology

The metallic alloy fuel form not only facilitates a compactness and simplicity in the recycle equipment and process, but yields safety benefits as well. The absence of low-mass-number scattering elements in the alloy and its high density combine to facilitate core layout designs of minimal reactivity loss upon burn-up, thereby passively precluding opportunity for transients induced by control-rod-run-out. The high thermal conductivity of the metallic fuel maintains a small temperature increment of the fuel above the coolant, which in turn minimises stored energy and stored (Doppler) reactivity. These effects combine to counter passively transients due to loss of coolant pumping or loss of heat sink – even in the absence of scram action. Finally, the relatively low melting point of the fuel in relation to the clad creep rupture and eutectic interaction temperatures, and in relation to the sodium boiling point, combined with the homogeneously distributed fission gas in the fuel, together provide a fast acting "fuse" fuel dispersing mechanism which squelches reactivity addition in severe accident situations and precludes the possibility of reaching super- prompt-critical conditions or vapour explosions. This results in minimal energy deposition before dispersal, avoids energetic loads on the vessel, and passively promotes retention within the vessel even under severe accident scenarios.

Removing decay heat by passive means in IFR reactor designs rests on retaining the coolant inventory by the use of sodium cooling at atmospheric pressure and double top-entry tanks containing all of the primary heat transport equipment and fluid. Natural convection carries decay heat from the fuel clad to dedicated secondary circuit decay heat removing loops driven by buoyancy flows and operating continuously at <1% rated power. The heat capacity of the primary coolant in the primary circuit tank is large enough to absorb within safe temperatures the initial excess of decay heat over the capacity of the passive secondary heat removal channel.

These passive safety properties make it possible to remove all safety functions from the balance of plant equipment, and thereby to reduce their cost in fabrication, installation, and maintenance.

IFR waste technology

Transmutation and fission products created from the uranium feed stream but unsuited to recycling as fuel are destined ultimately for a geological repository, and the ecological and safeguards implications of waste management are an important element of IFR technology development. The IFR design objective of multiple recycle for total consumption of the uranium feed stream, which results in a waste stream essentially free of TRU, was motivated not only by the goal of resource conservation but also by the desire to minimise the ecological and safeguards risks associated with the waste from the IFR fuel cycle.

The electrometallurgical recycle technology is designed to discharge less than one part per thousand of the recycled TRU fuel into the waste stream destined for the repository. Noble metal fission products from the IFR cycle are to be incorporated as alloying elements in an iron-zirconium metallic alloy. Active FPs are to be captured in a zeolite ion exchange matrix which is subsequently blended and glass bonded into a ceramic/glass monolith. These waste forms for FPs from the IFR fuel cycle had been selected and fabrication technology development and waste form characterisation was underway but uncompleted in September 1994 when the programme was cancelled.

The virtual absence of actinides from the waste totally eliminates long- and short-term risks of proliferation or criticality from IFR waste disposal. Similarly, the approach for minimising the radiotoxicity risk associated with IFR wastes is straightforward: with transuranics eliminated from the waste, the radiotoxicity source term exiting the fuel cycle derives only from long- and short-lived fission products. The hazard from the LLFPs is broadly comparable with that of the radiotoxicity that was removed from the earth with the ore that was mined to make the fuel in the first place [83]; the radiotoxicity hazard from the short-lived FPs decays to a level below that of the LLFPs within 500 years.

IFR development status

At the time the IFR programme was terminated (September 1994), the fuels irradiation program had demonstrated excellent steady-state and transient performance of the metallic alloy fuel, sodium-bonded to either austenitic or ferritic steel clad at heat ratings up to 500 W/cm and for peak burn-up up to 20% (200 MWd/kg) at TRU/heavy metal enrichments up to 25%. Passive safety performance had been demonstrated at the EBR-II power plant, with benign consequences upon loss of pumping action without scram and loss of heat sink without scram; both tests were performed from full power with absolutely benign consequence and immediate reactor restart to full power. The injection casting fuel fabrication technology was fully developed and the electrometallurgical processing technology had been demonstrated at the industrial (10 kg/batch) scale with all actinides and with surrogate (non-radioactive) FPs. The cathode consolidation processing step had been partially demonstrated at industrial scale. A full set of industrial-sized recycle/refabrication equipment had been designed, built, installed in the recycle hot cell, and checked out. The processing technologies for reducing LWR spent fuel were still under bench scale development, while developing fabrication technology and for the waste form and testing its properties were just getting started. The GE-led industrial team had completed advanced conceptual design of all NPP systems and had just initiated conceptual design for the recycle facility. The USNRC had extensively reviewed the safety of the ALMR and had formally issued a pre-licensing safety evaluation report (SER).

After September 1994, EBR-II was de-fuelled and lay-up activities were started. The GE-led industrial work was discontinued. The equipment which had been built and installed in the recycle hot cell was redirected to the task of electrometallurgical treatment of 100 EBR-II spent fuel assemblies so as to demonstrate a capability to produce fission product, uranium, and TRU products suitable for long-term disposal. (Untreated sodium bonded EBR-II spent fuel does not meet acceptance criteria for the geological repository.) The work on FP waste forms was continued for application to the EBR-II spent fuel treatment, and the electrometallurgical process technology development was continued for treating selected DOE-owned spent oxide fuel.

Electrometallurgical treatment activities on spent EBR-II fuel have recently confirmed that the full spectrum of fission product elements present in the EBR-II spent fuel are indeed separable from the uranium in industrial-scale, remotely operated and maintained electrometallurgical and cathode processing equipment.

*Other countries*

Commonwealth of Independent States

In the Commonwealth of Independent States (the former USSR), BN-600 in Russia is operated successfully, and BN-350 in Kazakhstan was finally shutdown in 1999 after 27 years of service following a government decision. Both reactors have basically been fuelled with enriched $UO_2$; plutonium test sub-assemblies containing so far more than 3 000 fuel pins have been successfully irradiated up to 100 000 MWd/t. This fuel, based on pellet technology, was produced at the Mayak plant at Chelyabinsk. The design lifetime of BN-600 (30 years) expires in 2010. The development of a lifetime extension program till 2020 will start in 2001.

In parallel, the BOR-60 experimental fast reactor at RIAR (Research Institute of Atomic Reactors), Dimitrovgrad, has been loaded with, among other advanced fuels, both pellet and vibro-packed MOX fuel, including fuel with a high plutonium content in standard and advanced claddings, manufactured on site up to an annual production capacity of one tonne of granulated fuel. Thus BOR-60 was able to recycle its own plutonium.

Based on this experience, construction has started on:

- One 800 MWe fast reactor, BN-800, at the Beloyarsk site.
- A large MOX fuel manufacturing plant at Mayak RT-1 (Complex-300).
- A new RT-2 facility to store and reprocess spent fuel from civil reactors and to fabricate MOX fuel.

However, financial difficulties have led to delays over the whole construction programme. Nevertheless, according to the "Programme of Nuclear Power Development in the Russian Federation for the 1998-2005 Period and up to 2010" [84], the construction of the BN-800 power plant is to be completed by 2010 at the Beloyarsk site. Meanwhile, the BN-800 core was redesigned in order to increase plutonium consumption, ensure a negative void coefficient, and reduce costs.

The current situation regarding the Russian R&D activities in the field of fast reactors can be summarised as follows:

- Developing the hybrid core design for the BN-600 reactor, and first design studies of a full MOX core (both the traditional fuel pellet, and RIAR's vibro-packed fuel are being studied). Irradiation tests in BN-600 with experimental sub-assemblies including both MOX fuel types are planned.
- Justifying lifetime extension for BR-10, BOR-60 and BN-600.
- Studies advanced, high safety fast reactor designs, including a large (~1 600 MWe) sodium-cooled fast reactor, and designs with alternative coolants (e.g. lead).
- Developing the basic design of the BREST-300 (lead-cooled) demonstration fast reactor with closed fuel cycle facilities located at the reactor site, and its experimental justification (the Beloyarsk site is considered to be a candidate site for BREST-300).
- Design studies for the large reactor BREST-1200.

India

In India, mixed carbide (U-Pu)C fuel was loaded in the Fast Breeder Test Reactor (FBTR) near Bombay, and the first core of a very low power attained criticality in 1985. The second core should reach the power of 40 MWth; it requires about 200 kg of (Pu 0.55, U 0.45)C fuel pellets. The peak burn up in the fuel reached 71 170 MWd/t. Post-irradiation examination of the fuel sub-assembly discharged at 50 000 MWd/t peak burn up indicates that the gap between fuel and clad is still not closed, and that the fuel is in excellent condition. A revised target peak burn up of 100 000 MWd/t has consequently been set. Detailed design, R&D, manufacturing technology development and the safety review for the 500 MWe Prototype Fast Breeder Reactor (PFBR) are ongoing.

China

In China, the Chinese Experimental Fast Reactor (CEFR) is the first step in the development of fast reactor technology. CEFR is a sodium-cooled pool type reactor; it has a thermal power of 65 MW and is equipped with a 25 MWe turbine generator. After completing the conceptual design and preliminary design in 1993 and 1997, respectively, some additional safety analyses were performed (mainly with regard to sodium spray fires). Presently, the CEFR's reactor building is under construction. The main components, including the reactor block, primary and secondary circuits, and the fuel handling system have been ordered. First criticality of CEFR is scheduled for the end of 2005.

### *4.2.3 P&T-related specific aspects of fuels and coolants*

#### *4.2.3.1 Fuels with increased concentrations of minor actinides*

The fuel types of interest for P&T are the same as those which have been used or proposed for conventional reactors, i.e. oxide, nitride, and metal. The ranking of these fuels by melting point and heat conductivity is oxide >nitride >metal and metal >nitride >oxide, respectively.

Nitride fuel is particularly suited for transmutation applications because different MAs can coexist in the fuel. Highly concentrated $^{15}N$, which does not exist in significant quantities naturally, will be used in order to control formation of long-lived radioactive $^{14}C$; $^{15}N$ must be recovered in the fuel during reprocessing. Basic data on the thermal properties of MA nitrides necessary for fuel design were obtained. It was confirmed that the particle size of TRU nitrides can be adjusted by means of carbothermic reduction and that very fine uranium nitrides can be produced via the sol-gel process. In addition, burn-up tests of mixed uranium-plutonium nitride fuel produced on a trial basis showed that fuel integrity can be maintained up to a burn-up of at least 5.5%. However, it will be necessary to accumulate irradiation data for MA-nitride fuels, devise measures to deal with fuels with very high decay heat, and develop an economical way to produce enriched $^{15}N$. For this purpose, MA-nitride fuels will be produced on a trial basis and irradiation experiments will be carried out.

One FR option has chosen fuel with MAs mixed into conventional mixed-oxide fuel. This extension of conventional fuel makes use of what has been learned in past FR studies. In this case, MAs in the fuel are limited to some 5% in order to maintain the integrity of the fuel and avoid adverse effects on the neutronic characteristics of the core. Nuclear data on MA nuclides were measured and evaluated, and design studies were carried out to determine the acceptable amounts of MAs and rare-earth elements in the fuel. Immediate issues include improving the accuracy of the nuclear data and physical properties of MAs, evaluating the behaviour of MA fuel under irradiation, and developing the technology to produce MA fuel industrially.

Another possibility is the use of metallic fuel – a ternary alloy of uranium, plutonium and zirconium (U-Pu-Zr) – which would be suitable for dry reprocessing, and would help simplify the fuel production process. For the electrorefining process, an important step in the dry reprocessing, feasibility has been confirmed through joint international research and is at the stage of engineering experiments, but the feasibility of the process for reducing oxides, and of technology to treat spent salt, has still to be confirmed. Immediate issues include compilation of data on fuel behaviour based on irradiation experiments, and development of injection-casting technology for fuel production.

#### *4.2.3.2 Effect of fuel and coolant choice on transmutation characteristics*

As discussed in Chapter 2, the amount of TRU or MAs transmuted in an actinide burner with a closed fuel cycle (FR or ADS) is proportional to the fission energy released by the system. The choice of coolant and fuel, however, influences the engineering design of the transmuter and the overall feasibility of the system. A study was recently conducted in Japan (see Annex E) to evaluate the effect of the choice of coolant and fuel on the transmutation characteristics (i.e. the minor actinide balance) of different fast reactor cores.

Since it is difficult to make such a comparison on the specific effect of coolant alone, three realistic reactor designs were chosen:

- A sodium-cooled fast reactor of commercial size.
- A lead-cooled reactor of the BREST-300 type.
- A $CO_2$ gas-cooled reactor of ETGCFR type.

The transmutation characteristics of the respective cores were compared by normalising the MA mass balances of the cores to the thermal power produced.

As gas-cooled reactors have a particularly hard neutron spectrum, it could be speculated that the gas-cooled core might have more favourable MA transmutation characteristics than the sodium- and lead-cooled cores. However, the study showed that, after normalisation, the three cores performed almost identically.

The same study also assessed the influence of the fuel type on the transmutation characteristics. Here, a 1 000 MWe sodium-cooled fast reactor was taken as reference case. The following alternative fuel types were compared with the reference $(U,Pu)O_2$ fuel:

- $(U,Pu)^{15}N$ as nitride fuel.
- U-Pu-10Zr as metal alloy fuel.

The respective analysis indicated that the transmutation characteristics of the nitride and metal cores are similar and slightly better than those of the oxide core. The difference can be attributed to the harder neutron spectrum of the alternative fuel-type cores. However, in terms of MA mass transmuted per year, the difference is rather insignificant.

The study thus confirms that the overall transmutation characteristics are not sensitive to the choice of coolant and fuel-type.

## 4.3 ADS technology

### *4.3.1. Introduction*

Active projects for ADS systems exist in France, Italy, Japan, Korea, USA, and several other European countries. Research in these countries mainly comprise basic studies on the different aspects of an ADS, although some of these projects aim towards a pre-engineering design phase within the next few years. International collaboration is emerging in Europe (Technical Working Group), in the USA (ANL and LANL) and new co-operative arrangements have been established between France and USA in this domain. An overview of these activities is given in Annex F and will also be mentioned in Chapter 7.

As the R&D on ADS relies essentially on two distinct disciplines, i.e. reactor physics and accelerator physics, some specific efforts have been launched, for instance by OECD/NEA, in order to exchange information between both communities. Today, some discussion is beginning on the viability of developing a dedicated accelerator for ADS applications whereas there would be some scope to pursue multi-purpose accelerators first (see also Chapter 7 on R&D needs).

In the following, the discussion of the ADS technology is divided into three sections dealing with the sub-critical reactor, the spallation target, and the accelerator.

### *4.3.2 Sub-critical reactor aspects*

Both the evolutionary and innovative transmutation approaches which incorporate accelerator-driven systems call for sub-critical cores with a fast neutron spectrum and fuels dominated by TRU or minor actinides. As pointed out before, these cores are characterised by a very low fraction of delayed neutrons and by a low (or near zero) Doppler reactivity coefficient. In principle, the physics of the ADS and of its sub-critical core is well understood, and there are several publications which deal extensively with the subject [85,86]. However, several concepts are new and their understanding requires experimental validation.

The following sections focus on a description of the basic physics phenomena in the sub-critical multiplying core, with reference to the coupling phenomena and their impact on the sub-critical core (SC), and discuss how the sub-criticality helps to reduce (or to eliminate) the negative consequences of impaired core characteristics on the safety of the multiplying medium. The areas which need particular care for experimental validation will be indicated, and some ongoing experimental programmes will be quoted. Finally, some relevant design-oriented problems of sub-critical cores and their integration into an ADS will briefly be indicated.

#### *4.3.2.1 Neutron flux distribution*

In a critical system, the condition of balance of neutron production and consumption at each point of the phase space $(E,\ \bar{r},\ \overline{\Omega})$ is expressed by the Boltzman equation, which can be expressed in matrix form:

$$A\overline{\varnothing} = P\overline{\varnothing} \qquad (1)$$

where A is the "consumption" and P the "production" operator, and $\varnothing$ the flux vector.

In the same system, made sub-critical, the condition to have a stationary state is to have an external source $S(E,\bar{r},\overline{\Omega})$ such that, e.g. the Eq. (1) can be written as:

$$A\varnothing_{in} = P\varnothing_{in} + S \qquad (2)$$

$\varnothing_{in}$ is the solution of the inhomogeneous Eq. (2). The distribution in space, energy, angle of $\varnothing_{in}$ is obviously different from that of $\varnothing$. Of course, $\varnothing_{in}$ approaches $\varnothing$ as the level of sub-critically becomes smaller and smaller, approaching the critical configuration.

For an ADS, once defined in material properties, the geometry of the system, the relevant cross-sections and the source intensity (in neutrons per second), the distribution of the inhomogeneous flux is fully determined by Eq. (2).

Relevant integral parameters characterising the sub-critical core (SC), such as reaction rates, can be easily calculated. This allows evaluating the power deposited at each point of the system, the damage rate, the breeding ratio etc. This is done exactly as in critical systems, characterised by $\varnothing$.

*4.3.2.2 The reactivity of the sub-critical core*

It is formally possible to describe a sub-critical system with the introduction of a parameter $k_{eff}$ which allows to "restore" the balance Eq. (2):

$$A\varnothing = \frac{1}{k_{eff}} P\varnothing \tag{3}$$

Since $\varnothing$ has the same distribution as the "critical" flux, this equation is obviously an approximation of the real case, as described by Eq. (2).

In order to improve the definition of sub-criticality and to take into account the change in distribution of the flux, a different definition of the sub-criticality has been proposed, by means of a "k-source" $k_S$. The procedure is to apply the formal balance condition (3) to the inhomogeneous flux Eq. (2):

$$A\varnothing = \frac{1}{k_{eff}} P\varnothing_{in} \tag{4}$$

Integrating and recalling that $A\varnothing_{in} = \frac{1}{k_S} P\varnothing_{in}$ one obtains:

$$k_S = \frac{\langle P\varnothing_{in} \rangle}{\langle P\varnothing_{in} \rangle + \langle S \rangle} \tag{5}$$

*4.3.2.3 Neutron source importance*

Understanding the behaviour of the source-driven sub-critical core depends largely on the evaluation of the relative importance of the source neutrons to the fission neutrons generated in the SC.

One introduces a parameter φ*, which is the ratio of source neutrons and of the average importance of fission neutrons. It can be shown that this parameter φ* is related to $k_{eff}$ as:

$$\frac{\Gamma}{\bar{\nu}} \varphi^* = \frac{1}{k_{eff}} - 1 \tag{6}$$

where $\bar{\nu}$ is the average number of prompt neutrons per fission, and Γ the average number of source neutrons per fission. Relation (6) is given in [87], where the experimental determination of φ* is discussed.

The φ* parameter plays an important role in assessing the ADS performance parameters. In fact in [14], it is shown that the relation between the proton beam current $i_p$, the power in the SC and its sub-criticality is given by:

$$i_p = \frac{\bar{\nu}\left(\frac{1}{k_{eff}} - 1\right)}{\varphi * Z} \frac{W}{\varepsilon_f} \quad \text{(Ampère)} \tag{7}$$

where W is the power of the SC in watts, $\varepsilon_f$ the energy per fission (MeV) and Z is the number of neutrons per incident proton.

It can be seen from Eq. (7) that a value of φ* higher than 1 can reduce proportionally the proton beam current requirement for a given sub-criticality level. Measurements of φ* are made in the CEA facility MASURCA in Cadarache, in the framework of the MUSE programme [88], which will be described shortly in sub-section 5.3.2.10.

*4.3.2.4 Kinetic behaviour of sub-critical cores*

The equations which give the kinetic behaviour of a system driven by an external source are of the type:

$$\begin{cases} \dfrac{dW}{dt} = \dfrac{\rho - \beta}{\ell_{eff}} W + \Sigma \lambda_i C_i + S \\ \dfrac{dC_i}{dt} = \dfrac{\beta_i}{\ell_{eff}} W - \lambda_i C_i \end{cases} \tag{8}$$

where $C_i$ are the precursors of delayed fission neutrons with decay constant $\lambda_i$. $\beta_i$ is the fraction of the total number of delayed neutrons emitted per fission ($\Sigma\, \beta_i = \beta$) due to the precursors [89].

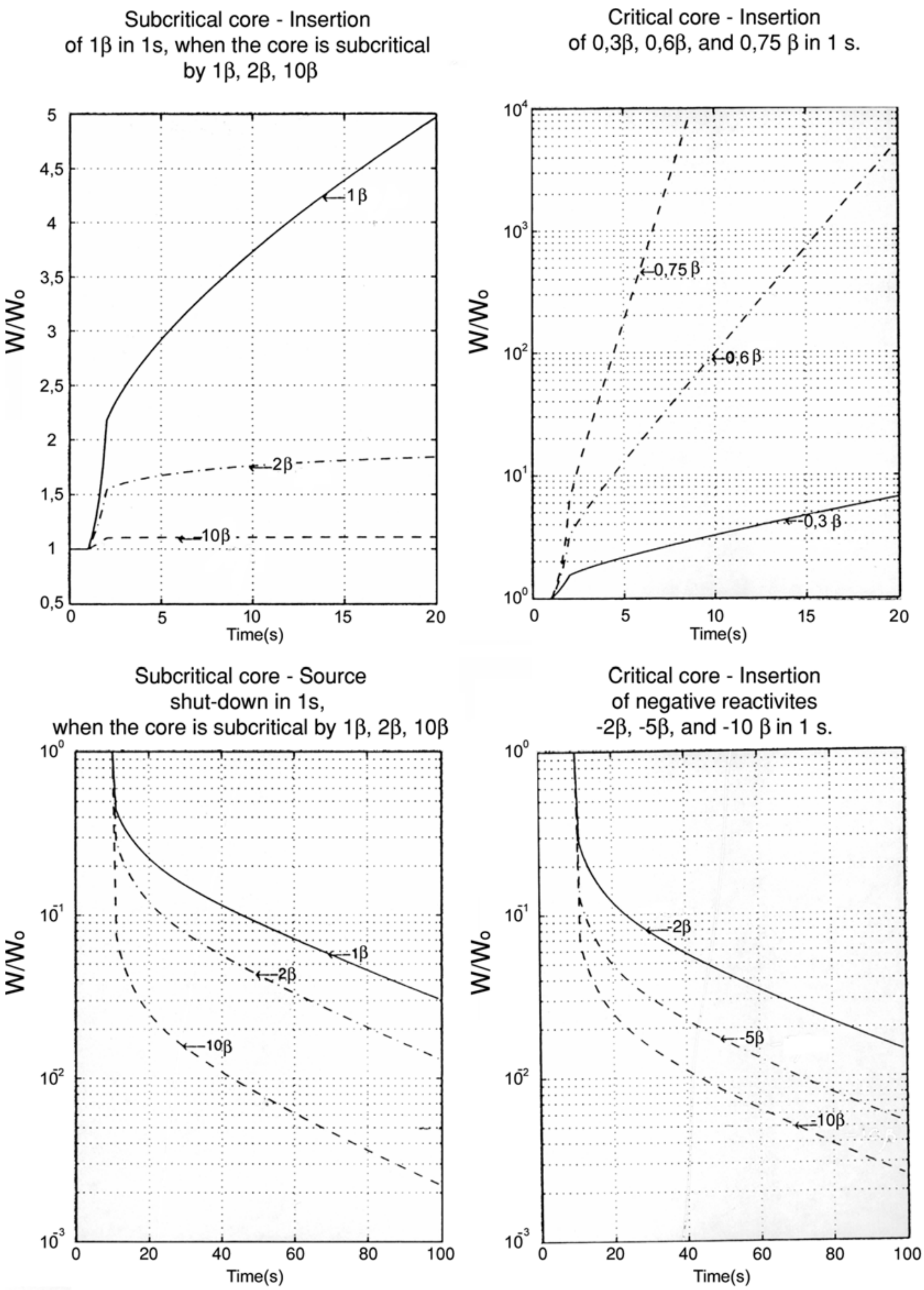
$\ell_{eff}$ is the neutron generation time and ρ is the reactivity $\left(\rho = \frac{1}{k} - 1\right)$.

In steady state (i.e. if $\frac{dW}{dt} = 0$ and $\frac{dC_i}{dt} = 0$), we have:

$$\rho \frac{W}{\ell_{eff}} = -S \tag{9}$$

A decrease by a factor h of the reactivity (ρ' = ρ/h) or an increase by a factor h of the source (S' = hS), induces an instantaneous increase in the power W' = hW. For example, if, the system is sub-critical corresponding to -10β, a reactivity insertion of + 5β causes a doubling of the power (see Figure 4.2). This of course is totally different from the behaviour of a critical system, which becomes prompt critical.

Figure 4.2. **Kinetic behaviour of sub-critical and critical cores**



In more general terms, the kinetic behaviour of a critical system is characterised by delayed neutrons and their time constants (about 10 s.), while the kinetic behaviour of a SC is determined by the time constants related to the external source, in the sense that an instantaneous variation of source has an effect on the time scale of the prompt neutron lifetime (typically of the order of microseconds).

The evolution of the power with time, and the related variation of the temperature, are related to the variation of the reactivity (Doppler reactivity effect, fuel expansion reactivity, reactivity due to the material concentrations in the core, including the coolant etc.). These feed-back reactivity effects are essential for the safety of a critical reactor. In a sub-critical core, the relevance of feedback reactivity effects varies according to the level of sub-criticality. In fact for a deeply sub-critical core, the dynamic behaviour is dominated by the external source and its variation in time. Closer to criticality, the feedback effects become more important and the behaviour of the core is approaching that of the corresponding critical core.

In a very simplified way, if the core is sub-critical by -10β, a feedback reactivity equal to ±1β, induces a ±10% variation of power and a ±50% variation of power if the system is sub-critical by -2β. In a critical reactor +1β reactivity insertion makes the reactor prompt critical and -1β stops the chain reaction. In view of the definition of an “optimal” level of sub-criticality, it is very relevant to verify the transition of the behaviour of the SC from a “source-dominated” to a “feed-back dominated” regime.

*4.3.2.5 Reactivity and loss-of-flow accidents*

Fast external insertions of reactivity give rise to different consequences in critical or sub-critical cores. Examples have been given in [88,90]. In [90] a 0.55 β/s reactivity insertion in a Phénix fast reactor type core, critical or sub-critical at $k_{eff}$ = 0.95, gives rise (at constant external source level) to the following power and average temperature evolutions:

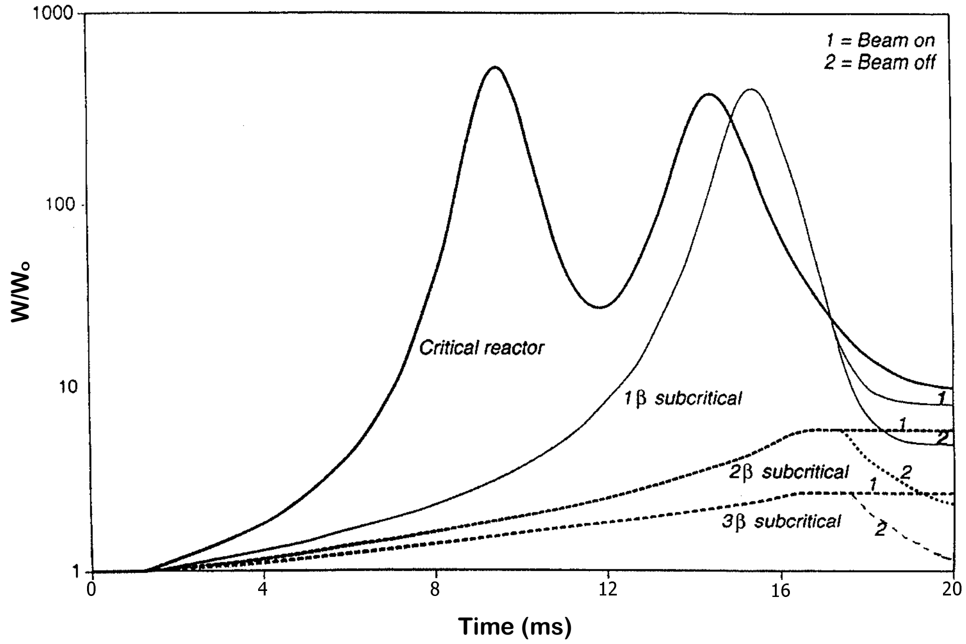
| | **Critical core** | **Sub-critical core (k = 0.95)** |
|---|---|---|
| Delay before fuel fusion | 2 s. | 12 s. |
| Inserted reactivity | 1.1β | 6.6β |
| Power increase W’/W | 2.2 | 1.5 |

In [88], a reactivity of 170 β/s is injected in a critical core ($W_o$ = 1 GW), or in the same core made sub-critical at -1β, -2β, -3β. The results show that prompt criticality is reached in the critical core after 6 ms with a first power peak of 700 GW at 8.5 ms and a second peak of 500 GW at 13.2 ms. In the sub-critical mode, the peaks are respectively of 530 GW at - 1β, 6 GW at - 2β and 2.2 GW at 3β (t = 16 ms) (See Figure 4.3).

The increase in power is considerably slower in a sub-critical system, and the total energy deployed is much smaller.

In the case of loss-of-coolant-flow accidents, References [88] and [90] give simple examples, which show that, in the case of no shut-down of the source, the behaviour of a -10β sub-critical system is less favourable, since in a critical system the increase of the coolant temperature is slower and lower due to the feed-back effects. Again, the choice of the level of sub-criticality is relevant, if one takes into account the potentially beneficial effects of the intrinsic characteristics of the core.

Figure 4.3. **Impact of external reactivity insertion**




This of course, has to be verified for each type of core and associated fuel and coolant. It is obvious from these considerations that the accelerator beam intensity must be coupled through safety grade scram circuits to the power level of the SC, so that it can immediately be shut down in case of a power excursion.

*4.3.2.6 Cores with low Doppler effect*

In the case of an ADS dedicated to transmutation, the fuel will be dominated by MA which will have a low Doppler effect, due to the absence of $^{238}U$.

The effect on the dynamic behaviour of the core will differ according to the level of sub-criticality (see Figure 4.4). At large sub-criticality, the calculations of the effect of reactivity insertion performed with a "standard" Doppler coefficient $k_D$, or with a "low" Doppler ($k'_D = 0.1\ k_D$), show no difference in the power or reactivity behaviour. Close to criticality on the contrary, the effect can be significant.

*4.3.2.7 Choice of the sub-critical level*

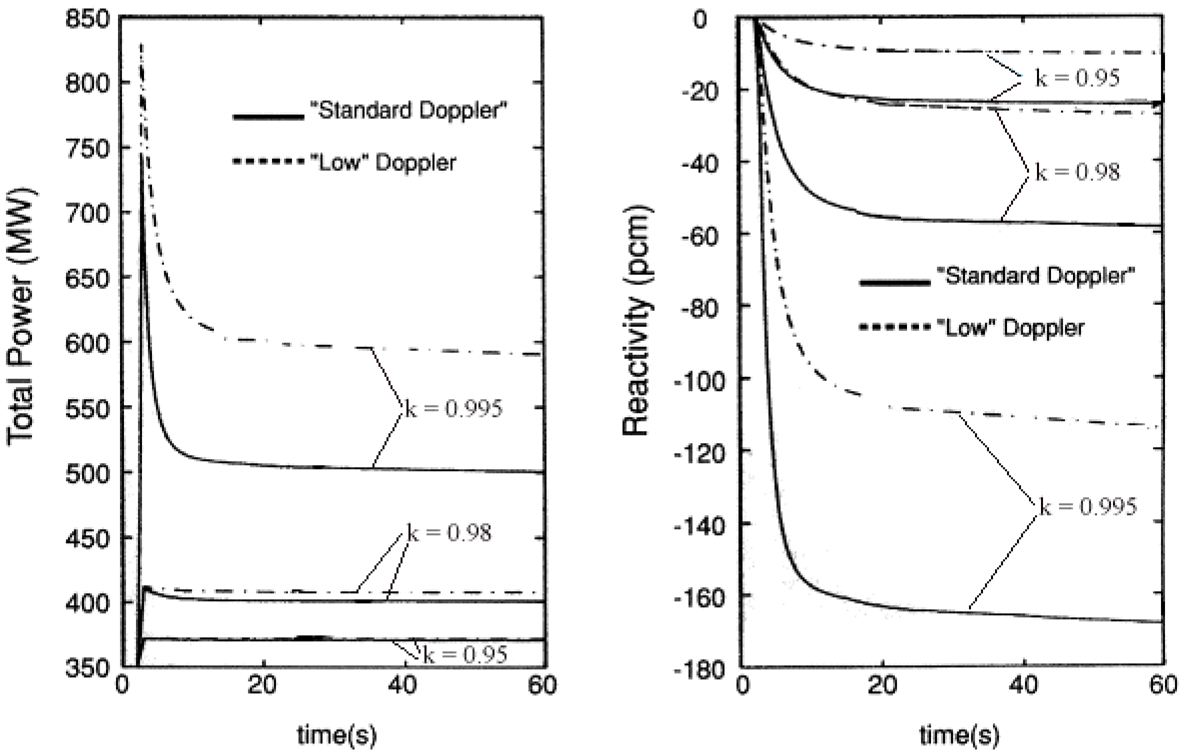
No final criteria have been established up to now in order to define an "optimal" level of sub-criticality. However previous considerations indicate the importance of finding a compromise between the "source-dominated" and the "feed-back dominated" regimes.

More quantitatively, in the case that no control rods are foreseen in the SC, the level of sub-criticality should be such that the core stays sub-critical when going from a "hot" state (i.e. normal operation) to a "cold" state (i.e. reactor shut-down). Since thermal feed-back induces generally (e.g. in standard fast reactors) a positive reactivity effect ($\Delta k_{FB}$) on going from "hot" to "cold", one can require that the "cold" core should stay sub-critical even in the case of an accidental reactivity insertion ($\Delta k_{AC}$), due for example to coolant voidage.

In that case the required "$k_{eff}$" should conform to the following relation:

$$k_{eff} + \Delta k_{FB} + \Delta k_{AC} < 1$$

Figure 4.4. **Insertion of 1/3 reactivity in 1 second.**
**Behaviour of an ADS "Phénix Type" at three different levels of sub-criticality.**




During operation, the maximum reactivity insertion ($\Delta k^{M}_{AC} \langle 1$) can be higher than $\Delta k_{AC}$. In that case one has the requirement that:

$$k_{eff} + \Delta k^{M}_{AC} < 1$$

Moreover, during reactor operation, the reactivity varies owing to the irradiation (burn-up) of the fuel and its isotopic evolution. In general this reactivity variation $\Delta k_{BU}$ is negative, but in some case (e.g. a fuel made essentially of minor actinides, which act as "fertile" materials, since they are transmuted into more "reactive" elements, as it is the case for example of $^{241}$Am), $\Delta k_{BU}$ can be positive. In that case, if the core has no control rods and one does not want to modify the external source, e.g. by changing the current intensity, one should have:

$$k_{eff} + \Delta k^{M}_{AC} + \Delta k_{BU} < 1$$

Looking for a compromise between the different criteria indicated above, one has also to consider that a very large sub-criticality may not be necessarily the optimal solution. In fact, besides obvious considerations on the "cost" of a strong external source, a largely sub-critical core has a peaked power distribution, dominated by the source distribution and therefore very far from flat, as required to optimise the fuel irradiation and, consequently, the fuel transmutation (see also the next chapter on safety issues).

### *4.3.2.8 Reactivity control and monitoring*

The control of reactivity and of the power level in a critical reactor is essentially through control rods. In principle, an ADS can be controlled solely through the external source. As an example, the variation of reactivity with the fuel burn-up can be compensated by an appropriate change of the beam current intensity. A similar system can also be conceived to control the reactivity change between "hot" and "cold" states. However, major variations of the current would be necessary. For example in a SC without control rods, with $k_{eff} = 0.99$ in the "cold" state, $k_{eff} = 0.98$ in the "hot" state at the beginning of an irradiation cycle and $k_{eff} = 0.95$ at the end of the cycle, the source intensity should change by a factor of approximately 5 to account for both the attainment of nominal power and the variation in reactivity during the operational cycle. In this context, it is clear that the use of control rods should be carefully considered to ensure at least some of the functions of reactivity control.

Moreover, if in a SC, in particular in a "source dominated" mode, the shut down of the source has an instantaneous effect to reduce power, the inverse effect, e.g. an "overshoot" due to a sudden increase of the external source, has the consequence of an instantaneous increase in the power. Although more limited than the potential power increase in a critical reactor, such an accidental situation should be examined.

Also, when the reactor is shut down, the consequences of inserting the full "reserve" of beam current should be analysed. In fact, if the insertion of the full "reserve" of beam current cannot be excluded, this accidental event could lead to a power variation given by [91]:

$$\frac{W'}{W} = 1 + \frac{\Delta\rho}{\rho + \beta} = 1 + \frac{\delta_{ip}/i_p}{1 + \beta/\rho}$$

If $W_{Max}$ is the maximum allowable power in a short time interval, one can deduce the maximum allowable sub-criticality level such that $W' < W_{Max}$.

Finally, we should mention that in principle, long term variations of the reactivity can be achieved by an appropriate variation of the $\varphi^*$ parameter. This can be obtained, for example, by changing the geometrical arrangement of the buffer (or of the buffer material) surrounding the spallation source.

As for monitoring the level of sub-criticality, different methods can be envisaged and experimentally validated. Some examples are as follows:

- Using the source of in a pulse mode. Recording the time evolution of the counting rates of in-core neutron detectors can allow measuring the reactivity. In fact, the point kinetics predicts the prompt decay of the neutron population after a pulse to be of the type exp($-\alpha t$) with $\alpha = (\rho - \beta)/\ell$. For known values of $\beta$ and $\ell$ one can deduce $\rho$ from the decay of the neutron population observed experimentally.
- If control rods are foreseen, the modified source multiplication method (MSM, see [92]) can be used provided the calibration of the control rods' reactivity is performed at near-critical level.

### *4.3.2.9 Beam trips*

As far as the coupling of the accelerator to the sub-critical core is concerned, one significant point which has been raised [93] is the effect of frequent beam trips on the SC. Since we have seen that the time scale for power variation (due to source variation) is very short, while the heat transfer time from fuel to coolant is of the order of 0.1 to 1 sec, the heat is stored in the fuel for ~1 s. making high thermal conductivity fuels a possible requirement. In a similar way, thermal stresses in the core structures can be expected owing to the difference in time constants between power increase and temperatures variations in the structures, and in the case of frequent beam trips, fatigue failures of the structures could occur and arouse safety concerns.

*4.3.2.10 Experimental validation*

The physics characteristics and the predicted behaviour of a SC, as outlined in previous paragraphs, need an experimental validation, in order to calibrate the calculation tools and to gain confidence in the prediction of the basic safety features of an eventual future ADS, which will be fuelled with very innovative fuels.

The main fields which need experimental validation are:

- The effects of the relative contributions of the source neutrons and of the neutrons generated by fission, φ* measurements should achieve that objective in stationary conditions.
- Experiments performed at different sub-critical levels, with or without feed back effects, can be essential to understand the transition between a "source-dominated" and a "feed-back dominated" regime.
- Spatial and energy distributions of neutrons and their variations close to the external source.
- Assessment and monitoring of the sub-criticality level.
- The relationship between the external source and the power in the core.

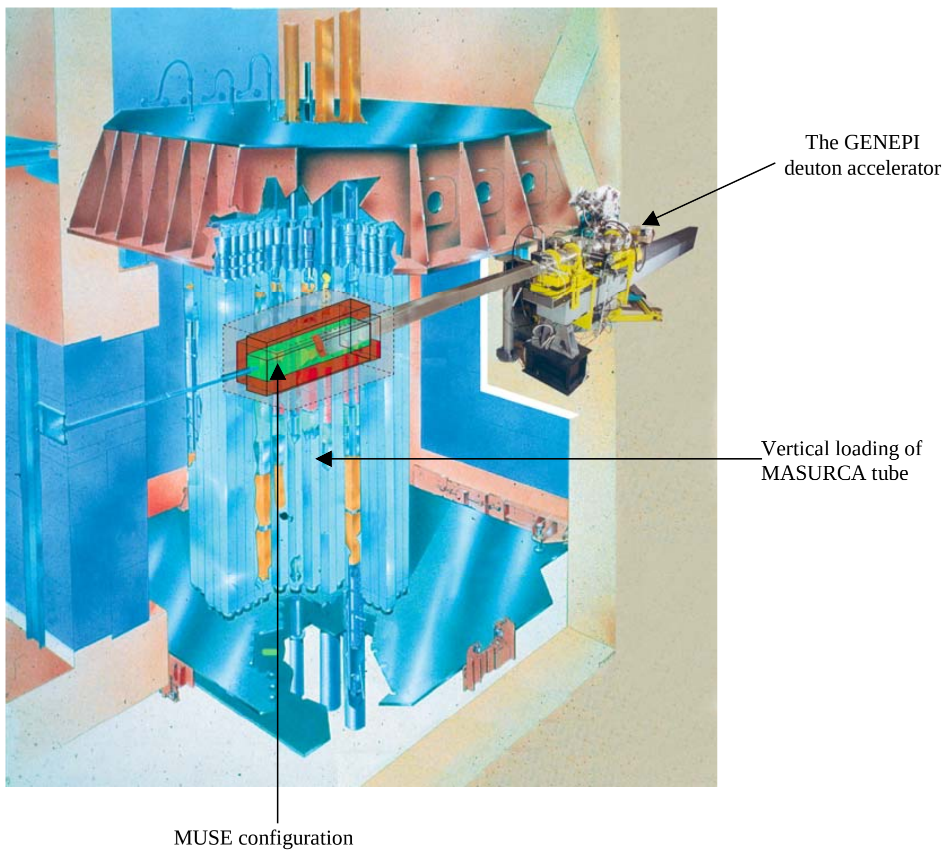
A first experiment related to verifying the physical principles of an ADS was performed by C. Rubbia at CERN (FEAT experiment, [94]). A proton beam struck directly a natural uranium block, and the "energy amplification" was experimentally verified.

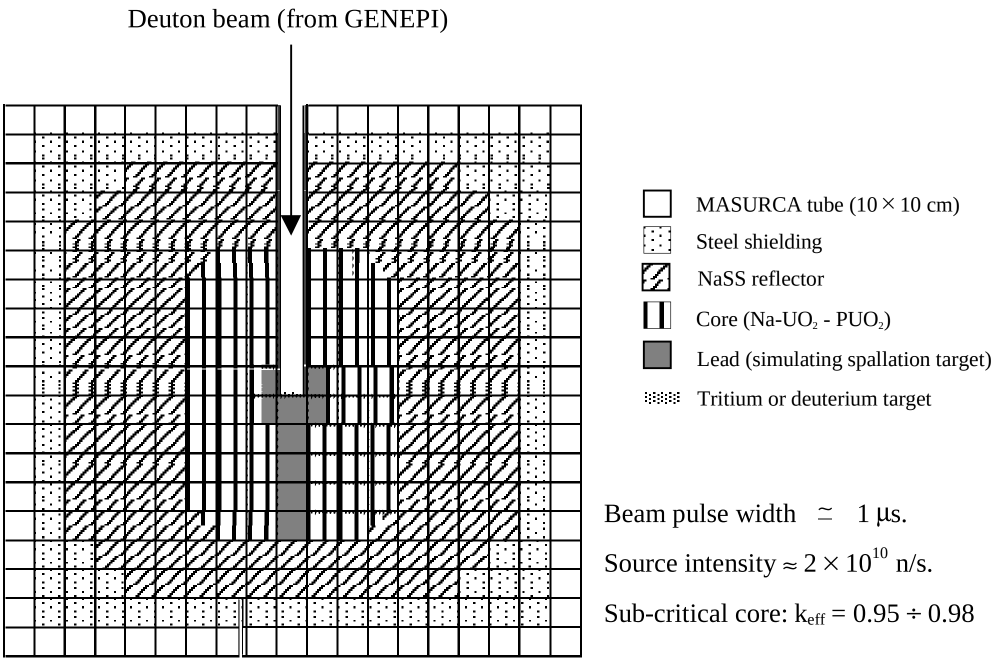
If the SC is not very sub-critical, i.e. with $k_{eff}>0.95$, it is possible to study its neutronics using other well-known external neutron sources substituted for a true spallation source, for instance a $^{252}Cf$ spontaneous fission source or a 14 MeV (d,t) neutron source. Since 1995, such studies have been under way at the MASURCA facility of CEA in CADARACHE, and a series of experiments called "MUSE" (MUltiplication avec Source Externe) has been performed (See Table 4.6) in a collaboration between physicists from Cadarache (CEA) and ISN-Grenoble (IN2P3), now extended to various European partners in the frame of a specific project in the Fifth European Framework Programme. The purpose of these experiments is to separate the effects of the source and of multiplication in the SC [87].

In the present MUSE-4 experiments with a pulsed 14 MeV neutron source called GENEPI, the target is surrounded by a lead buffer, to simulate the neutron diffusion inside an actual lead (or lead-bismuth) target. Numerical simulations have shown the validity of the basic hypothesis of the experiments, namely that using a spallation neutron source or the neutrons issued from the (d,d) or (d,t) reactions, the neutron spectrum in the core close to the buffer region is very much the same, whatever the energy distribution of the neutron source (See Figure 4.5). Additional information on the MUSE experimental programme is given in Chapter 7.

A joint ENEA-CEA working group has recently launched the idea to carry out a pilot experiment to demonstrate the feasibility of stable operation and to analyse the dynamic behaviour as well as to investigate certain safety issues of an ADS. This experiment, actually called TRADE and which would form a first example of ADS component-coupling "at real size" (<1 MWth), would be performed in the TRIGA reactor at the ENEA Casaccia Centre. The reactor would be operated as a sub-critical assembly and externally driven by a proton cyclotron. Additional information on the TRADE experimental programme is given in Chapter 7.

Figure 4.5. **The MASURCA installation for the MUSE programme**




Beam pulse width $\simeq$ 1 µs.

Source intensity $\approx 2 \times 10^{10}$ n/s.

Sub-critical core: $k_{eff} = 0.95 \div 0.98$

A sub-critical MUSE-4 configuration

The MUSE experiments and other experiments which are planned or have recently been started should give most of the demonstrations needed in order to proceed to a sound design of an experimental ADS, as proposed, e.g. in the European Roadmap towards and ADS demonstration [9].

Table 4.6. **The MUSE experiments at MASURCA**

| | **Type of source** | **Range of sub-criticality** | **Diffusing buffer around the source** |
|---|---|---|---|
| MUSE-1 (1995) | $^{252}Cf$ spontaneous fission neutron source | $-1.5\% \frac{\Delta k}{k}$ | None |
| MUSE-2 (1996) | $^{252}Cf$ spontaneous fission neutron source | $-3.0\div3.5\%\ \Delta k/k$ | Sodium<br>Steel |
| MUSE-3 (1998) | Pulsed neutron source from (d,t) | $-0.5\div-6.\% \frac{\Delta k}{k}$ | Sodium<br>Steel |
| MUSE-4 (2000-2001) | Pulsed neutron source from (d,d) and (d,t) | $-1\div0.4\% \frac{\Delta k}{k}$ | Lead |

*4.3.2.11 Technological problems*

All conceptual ADS designs available today are of a preliminary nature and some relevant technological problems are still to be solved in a satisfactory way.

This is the case, for example, of the shielding configurations in the upper part of the systems. The shielding in fact should allow for the potential deep penetration of high-energy neutrons ($E_n \geq$ 100 MeV) released by the spallation of protons (Typically $E_p$ = 0.6 - 1.5 GeV).

High-energy neutron penetration experimental studies performed in Japan, confirm the very large thickness of material (such as concrete or stainless steel) needed in order to reduce to an acceptable level the doses around the structures.

The beam entrance configuration is also a matter of concern. In fact, a simple vertical entrance of the beam can imply a very complicated system for the fuel loading-unloading system and can also be sub-optimal with respect to the need to guarantee the beam tube free from the intrusion of back-scattered neutrons.

These are just a few examples of technological problems that can have impact on the coupling of the different components of an ADS, and which could need substantial efforts in order to develop a robust ADS design.

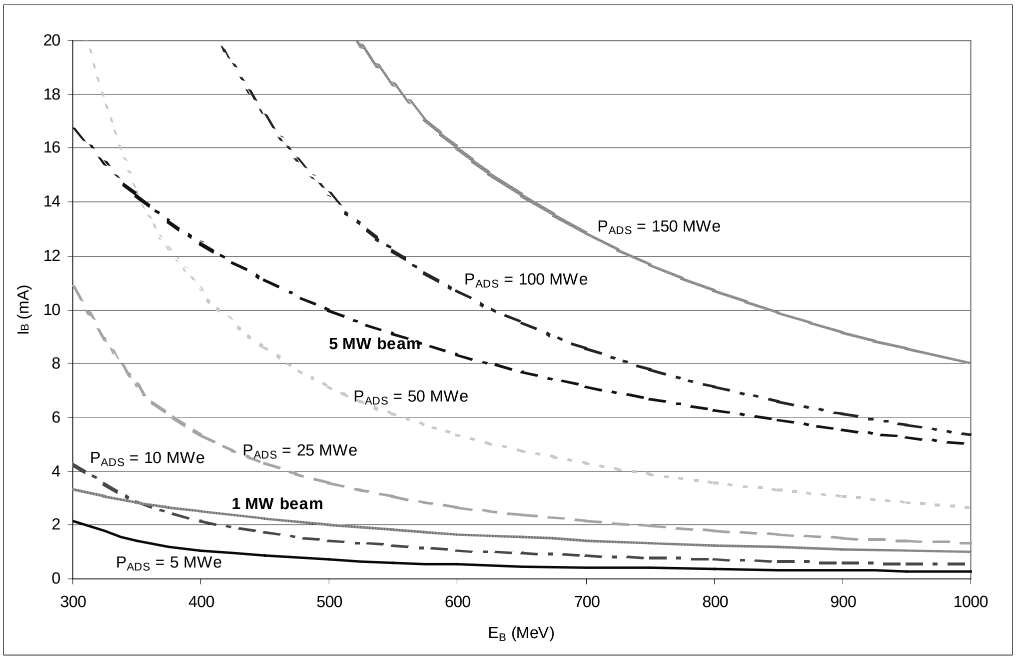
*4.3.2.12 Relationship between sub-critical reactor and accelerator*

A crucial element in the assessment of an accelerator-driven system is the required beam power, and thus accelerator characteristics, for a given sub-critical reactor configuration, i.e. reactor power, level of sub-criticality ($k_{eff}$), etc.

Figure 4.6 shows the relation between the beam parameters (beam current $I_B$ and beam energy $E_B$) and the electric power level of the ADS for a fixed $k_{eff}$ of 0.97. As shown, the total power level of an

ADS is limited to about 100 MWe as currently beam power is limited to at most 5 MW. The chosen ADS characteristics (136 MWe) in the fuel cycle schemes are in that respect already extrapolations of technology to the future.

Figure 4.6. **Beam power for ADS**




### *4.3.3 Spallation target technology*

One of the most stressed components of an ADS is the spallation target which is designed to generate the maximum amount of neutrons while ensuring the removal of the heat released in the spallation process. As the proton beam power being deposited in such a target attains several MW, even up to about 20 MW, very high power densities of several hundreds of kW per litre, occurring in the structure and in the spallation material, need to be safely removed. In addition, the mixed proton-neutron irradiation field in the target imposes very specific conditions on the design and operability of the target and influences strongly all the thermo-mechanical options for such targets. Heavy metals such as tungsten, tantalum, uranium, lead-bismuth, lead, mercury are considered as possible spallation materials for targets. Gas, heavy water or liquid metals are under consideration as coolants for these targets.

Two main options for a spallation target are available, i.e. solid or liquid metal. A wide range of experience with operating facilities exists in the design and use of solid spallation targets, essentially in mostly pulsed spallation neutron sources of lower power. Owing to the extremely high power density, the use of a solid metal target in ADS applications has to be excluded. The main advantages of a liquid metal target in ADS are, first, the perfect cooling of the target and the heat removal capacity which is inherent in the design having a flowing liquid metal coolant, and as a consequence, secondly, the greater achievable and tolerable power densities and the significant reduction in irradiation damage to the target itself and structural materials. However, other problems can arise with liquid metals such as lead or lead-bismuth as spallation material and coolant. These are corrosion and erosion of structural materials which are in direct contact with the liquid metal at high temperatures and high flow velocities, the behaviour of (volatile) spallation products and the need for a beam window between the evacuated proton beam guide and the spallation region.

*Solid targets*

At present, only solid targets are used in operating spallation neutron sources. They are usually assemblies of rods or disks fabricated from tungsten, uranium or tantalum and cooled with heavy water. In certain test cases, steel rods containing lead or lead-bismuth have been introduced as spallation material. The following problems and difficulties should be noted in designing and developing these targets:

- Radiation damage to target and structural materials, i.e. swelling, and the degradation of heat-conducting properties associated with helium accumulation.
- Complexity of target cooling, where high energy deposition is observed (several hundreds kW per litre).
- Radiolysis and activation of the cooling water, where experience in SINQ (PSI, Switzerland) showed that contamination of water with $^{7}$Be and other spallation products as well as deposition of $^{7}$Be on inner surfaces of the circuit can be unpleasant operational problems.
- Cooling of the target after shut-down of the accelerator (proton beam), where residual energy deposition due to decay of spallation products can reach 1-2 kW/L and relatively powerful auxiliary cooling systems are needed.
- Whether sodium (Na) should be used as liquid metal coolant, given that its very favourable thermal and heat-removal characteristics are counterbalanced by its chemical reactivity with air and water.

Solid targets were proposed in some early ADS-projects but this route was rather quickly abandoned in favour of liquid metal targets.

*Liquid metal targets*

Lead (Pb) and lead-bismuth (Pb-Bi) eutectic have been the two primary candidate liquid metal target materials for the production of spallation neutrons in ADS. Lead would be advantageous over Pb-Bi as it would significantly reduce by a factor of $10^{3}$-$10^{4}$ the build-up of the α-emitting $^{210}$Po coming from (p,xn)-reactions on Bi in the target. Nevertheless, an essential disadvantage of lead is the higher melting point (327°C compared to 125°C for Pb-Bi eutectic) which causes many engineering and technology complications. Other liquid metals, such as mercury (Hg), have also been proposed for use in advanced spallation neutron sources as well as for ADS. The main advantage of Hg would be the absence of $^{210}$Po-activity and the possibility of not having to heat up the system before operation which would simplify many engineering problems. But the high volatility of Hg imposes extremely strict requirements on the primary circuit and the integrity of the cover gas system since radioactive mercury must be prevented from leaking into a working environment. Mercury targets with a boiling point of 356°C would also be difficult to use in ADS as the working temperatures are higher than in spallation neutron sources for other purposes. Finally, the high neutron absorption cross-section of Hg is against its use in ADS.

The use of liquid metal targets does not simplify the design and development of spallation targets as multiple problems do appear. First of all, a liquid metal target needs a container, i.e. a liquid metal circuit integrated into the core of an ADS as well as an interface between the liquid metal target and the proton beam guide. This latter may take the form of a solid beam window or a windowless design (see later). The choice of structural materials both compatible with the liquid metal and able to resist the thermo-mechanical loads on the target circuit is a prime focus for R&D. The selection of a container material for the liquid target will therefore greatly affect the lifetime and safety of the target system. Because a beam window is an integral part of the target containment structure, it will be exposed both to a significant flux of high-energy protons and neutrons at a temperature of up to 650°C, and to a very corrosive environment. In a full-scale ATW system, for instance, a total proton fluence between $10^{26}$ and $10^{28}$ p/m$^2$.year can be expected on the target window. It is therefore likely that the beam window will have to be replaced at least yearly because of the expected material damage. In conclusion, the material should have good compatibility with the liquid metal target, good machining and welding capabilities, sufficient mechanical strength at the high operating temperatures, a low-neutron-absorption cross-section, and good performance under an intense proton bombardment.

In most of the target designs, temperature-gradient mass transfers will be the most damaging of all materials degradation phenomena in liquid metal because of the large temperature gradients expected in the system. As a result, both the thinning of the window and potential fouling of the heat exchangers are troublesome. The relative resistance of 24 metals and alloys to mass transfer in liquid lead with a temperature difference of 300°C (500-800°C) was measured by Cathcart and Manly [95]. The results indicate that only niobium and molybdenum showed no mass transfer, which in the other materials was slight to heavy. This liquid metal corrosion behaviour can be significantly reduced by controlling the partial oxygen pressure in the liquid Pb or Pb-Bi [96]. Additional measures such as coatings or surface restructuring and alloying are very promising technologies to minimise liquid metal corrosion of critical components such as the beam window [97].

Besides these mass transfer and oxidation phenomena, spallation products, including hydrogen and oxygen, will build up through the interaction of high-energy protons with the target material. These products will behave as alloying or impurity elements in the liquid lead or lead-bismuth target material where they are formed and thus may have detrimental effects on the containment material. While the production of oxygen seems to be negligible or could even be eliminated with an oxygen control system as mentioned above, further studies on the production rate of oxygen and on its interaction with other spallation target products are required.

Specific features of the coolant technology for liquid metal targets are caused by two factors:

- Accumulation of spallation products which could possibly influence the physical-chemical processes in the coolant and destroy protective oxide films on structural surfaces.
- High activity in the coolant and the cover gas causes high activity in the gas mixture which is removed from the circuit. This makes the coolant loop non-repairable and essentially aggravates analysis of gas compositions.

With respect to these spallation products, benchmark calculations showed that long-lived radioactivity accumulates mainly owing to primary nuclear reactions. Secondary reactions are responsible for producing a small number of long-lived isotopes, $^{207}$Bi, $^{210}$Po and some others generated by radiative capture of low energy neutrons. Neutrons in the energy range 20-800 MeV and protons with energies above 100 MeV make the main contribution to the total activity generation although these parts of the spectra inside the target have a rather small contribution to the total flux. Correctly estimating the activity of short-lived nuclides is the main problem in analysing target behaviour in the case of short accelerator shut-downs. They make the dominant contribution to both activity and heat release in the first moments after the shutdown, creating the intermediate links and additional routes for decay to the long-lived nuclides. The strong dependence of calculated concentrations of short-lived nuclides on the choice of the cross-section data library for determining the reaction rates is to be noted. Recent experiments using tungsten and lead targets indicate that the experimental production rates of spallation products agree with the computer code calculations within a factor of 2 for about 50% of the observed nuclides but differ significantly more, even by several orders of magnitude, in for the others [98].

*Lead targets*

In the LANL ATW pre-conceptual design for a liquid lead target, it is estimated that the maximum temperature of the system will be around 900°C at the point where the proton beam impinges on the window. At this temperature, iron-based alloys are inadequate because of their high-creep rates and poor oxidation resistance. Iron-based alloys are usually limited to a maximum of 650-700°C in service. Similarly, nickel-based and cobalt-based super-alloys are only marginally acceptable because they are limited to a maximum of 900-1 000°C. Another disadvantage of these super-alloys is that nickel is incompatible with liquid lead, and cobalt has a high absorption cross-section. Refractory metals such as niobium, tantalum, molybdenum and tungsten are usually used at service temperatures much above 900°C.

The three refractory metals – molybdenum, tantalum, and tungsten – all have their own problems as containment materials for the ATW system. Molybdenum becomes too brittle after low-fluence proton irradiation. Tantalum has an unacceptably high thermal neutron absorption cross-section and poor oxidation resistance. Tungsten also has a high thermal neutron absorption cross-section and low ductility.

Nb-Zr has been selected as a containment material for the ATW liquid lead target. Although this material has desirable properties for contact with the liquid lead target, one major drawback is its low oxidation resistance at high temperatures. Therefore, the ATW lead target must be designed to be used in a vacuum or at very low partial pressures of oxygen. This low oxidation resistance therefore also defines the minimum wall thickness (about 0.2-0.4 cm) of the window if a 10 or 20% loss in wall thickness is permitted during one year of operation [99].

Another possible way to alleviate the materials problem for the liquid lead target would be to use coolant to lower the window temperature at the cost of a more complex window design. There are two design options to lower the maximum temperature in the beam window material, which is at the stagnation point of the window hemisphere with present target design studies. First is a re-shaping of the footprint of the proton beam (e.g. the two-dimensional power distribution in the proton beam) in such a way that the maximum power is moved out of the centre of the beam. Second is by changing the flow field in the spallation volume and around the beam window, e.g. by introducing an additional jet flow across the window as is going to be realised with the MEGAPIE spallation target at SINQ, PSI [100].

As a consequence of reducing the maximum operating temperature of the beam window, more candidate materials would be available. Reducing the liquid metal temperature below 300°C could favour serious liquid metal embrittlement, which needs further considerations.

*Lead-bismuth targets*

Owing to the lower working temperatures achievable, lead-bismuth eutectic has become the preference in most of the ADS designs with separate targets. A specific feature of a molten lead-bismuth target is the high activity of spallation and fission products in the coolant (approx. 500 Ci/kg). Because of thermal diffusion and evaporation, gaseous (Kr, Xe) and volatile (Hg, Cs, I, Br, Rb) nuclides escape into the cover gas system. The specific activity can reach a few Ci per litre (2.5 Ci/l for the LANSCE conditions), which is 5 orders of magnitude higher than in a reactor with Pb-Bi coolant under normal conditions. This necessitates special shielding for the cover gas system, complicates repair operations and makes a gas system break hazardous. In addition, an important factor influencing radiation safety is the accumulation of $\alpha$-active polonium radionuclidesisotopes. Unlike reactors where only $^{210}$Po is formed as a result of neutron capture by $^{209}$Bi, in spallation targets (p,xn)- and ($\alpha$,xn)-reactions result in the formation of Po nuclides, among which $^{209}$Po ($T_{1/2}$ =102 y) and $^{208}$Po ($T_{1/2}$ =2.9 y) are the most important (the specific activity of $^{208,209,210}$Po reaches about 1 Ci/kg in the target circuit).

For the reliable operation of liquid metal target coolant technology, it is important to ensure purification of the coolant and corrosion resistance in structural materials in the cooling circuit. This technology was developed for Pb-Bi as a coolant in nuclear reactors. It comprises, in particular, the formation and maintenance of oxide scales perhaps with a very thin, stable aluminium or silicon plating on the surfaces of the structural material to protect them against corrosion, as well as reducing lead oxides by means gaseous mixtures including hydrogen. Such an oxygen control system together with in-situ ceramic oxygen meters has been developed for loop systems in the Russian Federation and in the Karlsruhe Lead Laboratory KALLA [101].

### *Structural materials for Pb-Bi targets*

The high solubility of nickel excludes the use of austenitic stainless steel or nickel-based alloys as containment material for Pb-Bi eutectic. The materials 2-1/4Cr-1Mo steel, modified 9Cr-1Mo steel, 12Cr-1Mo steel or HT-9 are suggested as candidate container materials for the Pb-Bi eutectic targets in ADS systems. The general tendency of these materials is that the higher the chromium content in the alloy, the higher the corrosion resistance to the eutectic and the lower the strengths, and vice versa. However, an effective oxygen control system still has to be applied to the spallation target.

Inhibitors are very effective in reducing the corrosion of steel by forming, for example, carbide and nitride films on the surface. Zirconium is perhaps the most effective inhibitor of low-alloy steels and it has also a lower thermal neutron absorption cross-section than titanium. If inhibitors are used during the ADS operation, a variety of spallation products, built up by the interaction of high-energy protons with the target material, will react with the inhibitors in the Pb-Bi eutectic. These reactions may reduce the beneficial effect of the inhibitors and more research on the interaction between the spallation products and the inhibitors is required.

### *Radiation dose to structural materials and beam window*

The most complicated and unknown issue is the radiation stability of the beam window (and other structural materials) related to degradation of its mechanical properties under conditions of mixed proton-neutron exposure. Experimental data are, in principle, available only to damage doses of about 10 dpa. Damage dose in the end of the TC-1/LANSCE lifetime (irradiation of 7.5 mA.month) is about 40 dpa, helium and hydrogen generation is 2 500-4 000 and 18 000-22 000 appm respectively. Maximum stresses in the window are 340 MPa.

By use of a windowless target but having a separate window in the proton guide tube, the radiation dose to this window outside of the target may be approximately 1.5-2 times lower than in a metal-cooled window since neutron irradiation will be negligible in comparison with proton irradiation.

The evaporation of liquid metal and of the spallation products into the proton guide tube, the additional shielding efforts, and the vacuum system have to be considered. In case of a rupture of the guide tube the release of radioactive products has to be taken into account. Another issue with a windowless target is the hydraulic stability of the freely falling liquid metal film, and the formation and possible rupture of waves on it. In principle, in a symmetrical solution, the problem of high temperatures in the stagnation point (lowest position in the cone-shaped film) and evaporation of coolant is not solved for high, ADS relevant proton beam powers.An asymmetrical solution demands more space, but could give solution to this issue.

*Effect of beam trips*

Beam trips have two major consequences for the target: first, thermal shock and considerable pressure waves on target circuit components and, second, the necessity to use on the secondary side of the heat exchanger cooling water at a temperature higher than the melting point of the coolant, in order to prevent coolant in the circuit from solidifying. Applying double-walled heat exchangers with a coupling fluid, which can change the heat transfer surface between primary side and secondary side of the heat exchanger, would be a possible solution and allow direct control of the heat transfer characteristics.

To prevent or mitigate accident consequences it is important to remove the beam quickly from the target in the event of an emergency situation in the heat removal system (pump failure, window rupture, …). For this purpose, obviously, emergency signals should be envisaged using different physical parameters (temperature, energy supply characteristics, …) and duplicate signal channels.

### *4.3.4 Accelerator technology*

To accelerate a high intensity proton beam to an energy of the order of one GeV, two completely different accelerator schemes are possible, a linac or a cyclotron. The choice depends on many factors, but it is important to clarify from the beginning that, to fulfil the beam requirements for ADS applications (specifically a very low frequency of beam interrupts), both machine designs have to be modified and developed, to extend into a new dimension of complexity, cost, and size. The operating mode for ADS will most likely be CW (continuous); although pulsed mode operation could be used for testing, set-up, etc.

A proton linear accelerator (linac) has performance limitations which may be more economic than technological, and provides straightforward solutions to some of the cyclotron’s problem areas.

Strong transverse focusing elements (quadrupoles) placed at frequent intervals along a linac as well as longitudinal focusing, due to the phase stability, set a much higher limit to the charge per bunch that can be accelerated without significant beam loss. Linacs also operate with RF-cavity frequencies typically 10-20 times higher than used in cyclotrons. Taken together, these factors mean that in principle a linac could accelerate a current from one to two orders of magnitude higher than a cyclotron, with no problems at extraction. Given an adequate linac length, the final energy is not limited on dynamical grounds. The electrical efficiency is high at high beam currents, even in a normal-conducting linac. If a superconducting linac is used for the high-energy section, the efficiency is even greater.

The major drawback for a linac is the length, which depends only on the final beam energy and the accelerating gradient, and is independent of the beam current. The length is an important factor in the cost of the facility, since typically most of it will need to be shielded against radiation produced by small beam losses.

Cyclotrons are based on the so called "cyclotron resonance" which states that, in a constant magnetic field, perpendicular to the beam orbit, the particle revolution frequency is fixed and independent of the particle energy; but, in the case of sector focused cyclotrons, at relativistic energies, energy levels beyond 1 GeV become more difficult to obtain. This simple rule, implemented with some clever ideas to improve the transverse focusing, has been the basis of the hundreds of low energy, low beam power, medical cyclotrons, scattered around the world. The main characteristic of such a machine is that just a few accelerating structures, fed by a CW RF generator via a resonant cavity, are required to transfer, step by step, the full energy to the beam. The beam circles isochronously with respect to the RF field in all the hundreds of passages needed to build up the full energy. Cyclotrons generally produce CW beams, since they operate with fixed magnetic fields and a fixed RF frequency.

As the relativistic effect increases the particle mass, transverse focusing has to be effected by spiral shims on the magnet pole with an angle increasing with energy. High intensity cyclotrons use separated magnet sectors and acceleration over two to three cyclotron stages. A proton energy of 1 GeV seems to represent a reasonable limit for a multistage cyclotron design.

A major problem for a high energy, high current cyclotron complex is the beam extraction system. To limit losses and minimise activation, the deflecting system that guides the beam out of the magnetic field, deflecting it with magnetic channels and high voltage electrodes, is permitted to touch only a negligible fraction of the beam. The current limit in the cyclotron is then given by a design requirement to produce a clean beam at the outer radius of the machine, with a radial separation sufficient for a single turn extraction. The latter depends on the voltage capability and number of the RF cavities, because the turn separation is determined by the energy gain per turn.

#### *4.3.4.1 Present status of linear accelerator technology*

Most of the existing large proton linacs have been designed as injectors of large synchrotrons [102], and are short pulse machines with relatively low average beam power. The highest power machine is the LANSCE linac at Los Alamos [103], an 800 MeV accelerator that is capable of delivering an average beam power exceeding 1 MW, with a duty factor of about 10%. All the existing machines are built with room-temperature water-cooled accelerating structures, and are pulsed.

In a linac the maximum current that can be accelerated is dictated by the charge per bunch and this value has been for all the past applications much higher than the required average current which has been of order 1 mA or less. For a given energy, the linac length depends on the average accelerating field, and the power required to excite a room temperature RF structure is proportional to the square of this field. As a consequence, to minimise the mains power and the investment cost, in almost all existing facilities pulsed operation has been chosen. CW operation, however, makes sense economically at very high average currents (of the order of 100 mA). In fact, in this case, the power transferred to the beam is so high (100 MW for one GeV energy) that a good efficiency is obtained even if a power of similar magnitude (about 50 MW) is dissipated on the walls of the RF cavities. The early designs of the large and expensive accelerators intended for tritium production in the USA and France, as developed in the early 1990s, were based on this approach.

Proton linacs are now considered fairly competitive in the 10-20 MW beam-power range mainly because of the very impressive results obtained in the last ten years in the fields of superconducting (SC) cavities and related cryogenics. Hundreds of CW superconducting RF cavities are presently in operation at CERN (LEP2) and Jefferson Lab (CEBAF), with an accelerating field exceeding 5 MV/m. Owing to the very low RF losses in the superconducting regime (5 orders of magnitude lower than for room temperature copper), a very small power is required to create a much higher accelerating field and almost all the RF power is then transferred to the beam. This permits a much shorter and more efficient linac design. Including the cryogenic static losses and the cryoplant conversion efficiency, the mains power required to establish the accelerating field is both at LEP2 (working at 4.5 k) and CEBAF working at 2 k), of the order of a few kW per MeV. This value depends only on the cavity gradient and operating temperature, and not on the linac beam current. The outstanding results recently obtained at DESY in the framework of the International TESLA Project (a superconducting electron-positron linear collider) have demonstrated that much higher accelerating fields can be obtained (up to 25 MV/m) and reliably used. Moreover the improvements to the cryo-module design (and partially in the niobium quality), have greatly reduced the required mains power; as an example the TESLA cryo-module, with cavities operated in CW at 12 MV/m, requires a total mains power for cryogenics, including RF and HOM, of 600 W per MeV.

The linac scheme that is considered in the following is then based on a solution, accepted worldwide, which moves switches to the use of superconducting-cavity technology at an energy of the order of 100 MeV. That means that the low energy part of the linac would be made up of room-temperature copper cavities, while the high- energy part would be a superconducting-cavity accelerator operating at 2 k. The transition energy has been set at 190 MeV rather than 100 MeV in the SNS linac case, because the lowest beta section of the SC linac is the more critical one for the Lorentz force detuning effect, which is an important issue in the case of pulsed operation.

The short schedule of this funded project in the USA and the lack of superconducting cavity prototyping in this beta range also pushed the transition energy in the SNS linac to a higher value.

The chosen reference linac is composed of a sequence of 4 different accelerators. The beam extracted from the last, the superconducting high-energy linac, is directed on to the spallation target. All the discussions on reliability and efficiency have to take into account the particular characteristics of these different accelerator types, individually referred to the present status of the art. The energies chosen for the transition from one accelerator to another should be considered only as a reference case, the precise value being determined by the overall design optimisation. The 4 accelerators are:

- DC injector up to: 100 keV.
- Radio frequency quadrupole (RFQ) up to: 5 MeV.
- Normal conducting linac (DTL, or similar) up to: 100 MeV.
- Super-conducting linac (elliptical cavities) up to: design energy.

The present state of the art of the first two linac components sets a current limit for the accelerator of the order of 100 mA. Higher currents could be obtained by combining outputs from two DTLs if desired, but this process (funneling) has not been demonstrated. It is worthwhile to note that the current limit applies to the peak current if the linac is pulsed, or to the average current if the CW operation is chosen.

Prototypes of 100 mA proton sources are now in operation in various laboratories [104]. Taking as a reference the results from IPHI at Saclay, it appears that this component, for an injector voltage of the order of 80 kV, is well understood and very reliable. In operating the source at a current level 20% below the design value, no beam trips are observed. Extrapolating the preliminary existing data, at the maximum design current, one beam trip (spark on the electrode) per week is expected. With the experience gained and taking a proper margin on the design values this component could be considered as highly reliable.

A working 100 mA CW RFQ at 350 MHz is now in operation at LANL, as part of the LEDA project [105,106]. Other CW RFQs are in the design/construction phase at Saclay, Jaeri, Legnaro (INFN) and LNL. The general impression is that, thank to the LEDA (Low Energy Demonstration Accelerator) experience and always with a proper margin, this part of an accelerator can now be designed as a very reliable component, limiting the reliability problems to the high power RF components, like klystrons and RF coupler windows.

Prototypes of CW DTL (Drift-tube Linac) and CCDTL (Coupled-cavity Drift-tube Linac) have been recently developed in the framework of the tritium and ADS R&D programs (e.g. the Italian TRASCO and the French IPHI programs). The DTL linac scheme is very old and well established. In fact it has been used since the early fifty in all the high peak current pulsed injectors for proton synchrotrons. What is required by the new projects is the CW operation that implies the dissipation of a very high power from the accelerating electrodes. This engineering problem is quite similar to that solved for the development of CW RFQs. Very reliable 3D computer programs now exist for a joint optimisation of the electromagnetic and thermo-mechanical behaviour. Once the RF structure is developed and tested, the limit of this accelerator is expected to be related, as usual, to the high power, standard RF components.

The super-conducting linac design is derived from the experience gained at CERN, TJLab, and DESY, where high performance super-conducting electron linacs are in reliable operation. The switching of the SNS design to this technology, once funded and in spite of the tight time schedule, is strong proof that the expectations of improved reliability, and reduced capital and operational cost have to be considered as fully realistic. In practice this design uses elliptical-cavity technology developed for electron linacs ($\beta = 1$), but compresses the shapes longitudinally to adapt them to the lower beta appropriate for different sections of a proton machine. Working prototypes of these cavities have been built at several beta values, and they behave as expected from extrapolating the electron cavity performances [107,108]. Cavity efficiency, in terms of cryogenic mains power per MeV and real estate gradient, increases as beta approaches 1. Beta values below 0.5 are not considered because the shape compression of elliptical cavities is too extreme for good efficiency. Other kinds of SC cavities (spoke resonators) may be applicable below beta 0.5 in the future, but until now only single-cell versions have been demonstrated. Three beta families are required for energies above 100 MeV. Roughly speaking, the first section ($\beta$ ~0.5) is used to accelerate the beam up to ~200 MeV, the optimisation of the second and the third beta sections (and corresponding energy ranges) depends on the required linac output beam energy. Energies up to 2 GeV are compatible with a three-section scheme.

*Beam power, component reliability, trip rate and duration*

The need to achieve high beam reliability, or a low beam interrupt rate, is a key requirement for ADS applications. High beam power is preferred by an ADS linac in term of efficiency, cost per MW and even reliability. Minimum and maximum current values respectively of 10 mA and 100 mA could be considered as reference numbers. For currents above 50 mA, a duplication of the low energy section (up to ~100MeV) should be considered to achieve high reliability, while a “spares on line” scheme is preferred for the super-conducting linac. A high duty cycle (up to 90%) beam structure is compatible with the present status of the art of the RF controls [109], opening the option to share the beam power among a number of sub-critical reactor experiments. In this case each experiment would see a pulsed beam in the millisecond scale [110].

In principle, modern controls, based on fast digital electronics, make it possible to reduce the duration of most beam trips generated by sparking high voltage components to less than 100 milliseconds. Since most proposed transmuter designs do not undergo significant temperature changes in less than 300 milliseconds, beam trips of this duration or less will have essentially no impact on their integrity. However, beam trips longer than 300 milliseconds, which would be due to real equipment failures, would produce thermal cycling of the transmuters that could cause life-limiting stress damage. The frequency of these longer beam trips depends on the equipment safety margins used in the accelerator design, and also on the degree of equipment redundancy. These factors will be major cost drivers. The estimated annual number of trips that can be achieved with "high-reliability" accelerator design ranges from a few tens to a few hundreds per year, a large reduction from the 10 000 per year that is the performance level of existing accelerators. Further studies are required to refine the permitted and attainable trip rate and its impact on the project cost.

On the assumption that all the standard accelerator components are well designed and built according to "space-qualified" specifications; that is with the required margin and redundancy, predictable long beam trips (minutes, hours or days) should be just those randomly generated by the lifetime of the high power components. For example, the present lifetime of high-power klystrons is in the order of 25 000 hours. The failure problem of the ceramic RF windows, which at present affects a number of accelerators (but not all), should be solved with a better design. In fact there are no fundamental limits preventing a fully reliable operation of this crucial component.

As a preliminary synthesis of the reliability issue, we can state that at present it should be possible to design a linac having from a few tens to a few hundreds of beam trips per year that are driven by equipment- failure (i.e >100 milliseconds). The trips caused by sparking and similar (non-failure) events can be reduced to a time scale <100 milliseconds, and would have practically zero impact on the transmuter. In the case of a multi-user pulsed beam these short beam trips could be practically undetectable.

The problem of the few long beam trips per year that are expected can be solved with equipment redundancy, that is extra money (second low energy linac and spares on line for the super-conducting part). For a discussion of classes (duration & cause), and frequency of beam trips, see [111].

*Beam losses, conversion efficiency*

In a 100-mA CW RFQ, where the continuous beam from the source is bunched and accelerated, LEDA measurements show that the total beam losses can be limited to less than 5% [105,106]. At lower current, 80 mA, 1% to 2% beam losses were measured. For the IPHI design, based on the LEDA design with some improvements, improvement by a factor of two is expected. Because the beam energy is low in the RFQ, such losses do not cause a significant activation problem.

From the LANSCE linac experience, low beam losses are expected in the medium energy part of the ADS linac (DTL or CCDTL) and unrestricted hands-on maintenance should be guaranteed. In addition the longitudinal beam dynamics in the ADS low-energy linac will be much superior to those in LANSCE, because of the replacement of the Cockroft-Walton injector-plus-buncher with a modern RFQ. This step eliminates longitudinal mismatches and greatly reduces the tails in longitudinal phase space.

All the multi-particle beam simulations (up to a few million particles) performed so far by LANL, CEA and INFN, using different codes especially implemented for this purpose, have shown that, with proper optical matching and reasonable error tolerances, no particles are lost in the high-energy accelerator and beam transport systems. In practice, this means that the operational beam loss limit desired for hands-on maintenance (<0.1 nA/m) beam along the high-energy part of the accelerator seems straightforward to attain. Given reasonable matching, the problem of beam halo formation, very crucial in circular machines, should be negligible in a short linac within a few tens of lattice periods.

The conversion power efficiency, defined as the ratio between the beam power and the mains power, increases strongly with beam current. Neglecting the marginal power required for magnets, efficiency is determined by the klystron efficiency, the Joule losses in the normal-conducting accelerating structures and by the cryogenic-losses in the super-conducting linac, RF and static. The last two are current independent so that their proportion of the total required power increases as the beam power decreases.

On the basis of the reference 1 GeV linac design and taking 67% for the klystron efficiency, the estimated required mains power, $P_{mains}$, as a function of the beam power, $P_{beam}$, is approximately given by the simple formula:

$$P_{mains} = 1.9 \times P_{beam} + 10\text{-}15 \text{ MW}$$

About half of the 10-15 MW is the power deposited in the walls of the NC low-energy linac plus its water-cooling pumps, and the other half is due to the liquid helium refrigerator. As a consequence, for a beam power of 1 MW, the efficiency is between 6-8%, while for a beam power of 30 MW the efficiency is 42-45%.

*Operation and maintenance aspects*

In the existing large accelerator complexes, a short (one day or less) maintenance time is scheduled on a weekly or monthly basis and a long one every year. This maintenance scheme reduces the number of unscheduled long beam trips induced by component failures. In the ADS case, with a proper redundancy in the linac design, the short maintenance periods should be suppressed and a single long maintenance period per year (of the order of one month) could be sufficient. This is however a matter of opinion at present, since the required RAMI analysis has not yet been done, but it is believed to be very difficult to operate an ADS linac for a year without the need for significant maintenance. The number of maintenance periods, however, could probably be reduced from one per week to one per month. The transmuter will have to be "refuelled" about once every three months, an activity that takes 10-12 days. Thus it would be easy to obtain about one month's worth of accelerator maintenance, split into three periods over the year.

To obtain this result, one could use the following design criteria:

- All standard power components designed with a suitable margin.
- Two parallel low energy linacs, up to 100 MeV, in two separated tunnels, to maintain one while the other is running.
- 10% of extra modules in the SC linac, to switch off failed component while running the accelerator with a different parameter set. Klystrons have to be in a shielded area for replacement.

Modern fast electronics should guarantee a linac retuning time in the 100-millisecond region, to compensate for failed elements in the acceleration chain, that is compatible with the transmuter thermal response times. A detailed analysis is required to evaluate the optimum compromise between cost and reliability of the accelerator. A similar and parallel analysis should be carried out for the transmuter, in terms of the trade-offs between tolerance to thermal cycling, neutronic performance, and cost.

*4.3.4.2 Present status of cyclotron technology*

The concept of a cyclotron-based accelerator for ADS is, like the linac-scheme, a multi-stage accelerator facility with a final energy of 1 GeV.

A proposed scheme for such a three-stage 1 GeV design would probably employ the following energy ranges and machine types for the individual accelerator stages:

| | | |
|---|---|---|
| • DC proton source | at about: | 60 keV |
| • DC-Pre-accelerator, Cockcroft-Walton or Radiofrequency Quadrupole (RFQ) | up to: | 0.8 to 4 MeV |
| • Injector-cyclotron, 4 to 6 sectors | up to: | 80 to 120 MeV |
| • Final stage ring cyclotron, between 8 and 12 sectors | up to: | 1 GeV |

The accelerator facility at PSI can be seen as a "proof of principle" facility for the generation of high power proton beams using cyclotrons. Since an upgrade program in the years 1990-1995 the 590 MeV Ring cyclotron at PSI routinely produces beam currents of 1.5 mA to 1.7 mA; the highest beam current extracted so far is 2.0 mA. The facility was operated at a beam power of about 1 MW over more than 6 000 h/y in 1999 with the beam being available during 91% of the scheduled beam time. This is considered excellent for the research projects to which the beams are applied. Thus the PSI cyclotron facility is among the accelerators that produce the highest beam power and probably leading in respect to the annual accelerated beam charge. It indicates that a 1 GeV-machine can probably be built based on today's knowledge, experience and technology. The performance, efficiency and costs of such a project can be predicted with fairly high accuracy.

The problem with beam losses at extraction is solved with a design that guarantees well separated turns at extraction. In the case of the PSI ring cyclotron the turn separation equals 8$\sigma$ of the beam profile at extraction. As an alternative solution to achieve low beam losses at extraction M. Craddock [112] considers extraction by stripping $H^-$ -ions, while L. Calabretta *et al.* [113] propose accelerating $H^2$ ions and also extract the beam by stripping.

The main stage of the PSI facility is a separated sector cyclotron (SSC) with an energy of 590 MeV. The concept of separated magnet sectors was introduced by H. Willax in 1963 [114] in order to provide the high energy gain per turn required to minimise extraction losses. Compared to the "classical" cyclotron layout, which employs "Dees" (RF acceleration electrodes) inserted between the magnet pole gaps, in a SSC, the magnets and acceleration structures (the cavities) use separate sectors, such that there is more room available for the RF structures. Such acceleration cavities can be built much larger and are therefore more efficient. In practice up to 10 times higher Q-values and acceleration voltages compared to the classical "Dee" design can be achieved. The high acceleration voltage results in a high-energy gain per turn, which is the most important parameter in order to generate separated turns and hence avoid beam losses at extraction. At the same time it helps to raise the limit on the beam current imposed by space charge forces. By increasing the number of sectors more RF cavities can also be inserted, which further increases the energy gain per turn. Hence using the highest possible acceleration voltage and adjusting the number of cavities allows a cyclotron design with the energy gain per turn necessary to reach any desired beam current.

Several conceptual design studies on cyclotrons to be used as a final stage in an ADS facility have been published [115-119]. Most of the proposals use an energy of 1 GeV and a beam current around 10 mA ($P_{beam}$ >10 MW), and are essentially based on the design of the PSI ring cyclotron (SSC), which at present delivers up to 2 mA at 590 MeV ($P_{beam}$ >1 MW) using 8 sector magnets and 4 cavities with an RF voltage of 730 kV peak. For the 10 MW cyclotron 10 or 12 magnet sectors are proposed, arranged in a ring with about 15 m diameter; and 6 to 8 cavities with a peak voltage of 1 MV. Higher beam currents could be achieved in larger rings with more cavities and hence a higher energy gain per turn [119]. Sector magnets in an SSC can be of super-conducting design; such magnets – comparable to the ones to be used in a 1 GeV cyclotron – are under construction at present at the RIKEN laboratory in Tokyo (Super-conducting Ring Cyclotron for the RI Beam Factory at RIKEN) [120]. This concept allows building particularly compact and energy-efficient SSC's.

The limit on the beam current due to space charge effects has been shown to depend on the cube of the energy gain per turn [121,122]. This law has been used to extrapolate properties and the beam performance of cyclotrons for higher beam power levels. The beam current limit is reached when the beam losses at extraction increase since the broadening of the beam diameter due to space charge forces exceeds the turn separation given by the radius and the energy gain per turn. From experience gained in the upgrade of the PSI facility, in which the peak RF voltage in the cavities was raised from 450 kV to 730 kV, it seems indeed, that such extrapolation is feasible [123,124]. Recent results on a 1:3 scale full power model cavity (measured field gradient: 4.2 MV/m at 150 MHz) show that voltages in excess of 1 MV can indeed be expected for the new cavities now under construction at PSI [124]. No technological breakthrough is required, but some challenging (but rewarding and interesting) R&D will still be called for.

The injector cyclotron at PSI, the Injector 2, accelerates a 72 MeV proton beam up to 2 mA for injection into the main stage cyclotron. Again the concept of a separated sector machine is employed, specially designed for high beam intensities with 4 magnet sectors and 2 RF resonators with a RF voltage of 250 kV peak and 2 acceleration gaps each. The same proven technology is used for beam injection and extraction as in the main stage cyclotron.

Interestingly, this cyclotron is operated in a very special, new scheme where the injected beam bunch is matched into a phase space volume that is stable under high space charge forces. In this matched condition the beam bunch is self-focused in the longitudinal and radial directions and kept together by the space charge forces. In contrast to the phase stability in linear accelerators, halo particles are not spread out over the whole bucket, but return to the matched bunch [125-127].

The injector cyclotron for a future ADS facility should preferably also be operated in this matched condition in order to reduce beam losses in the main stage cyclotron through better beam quality and the fact that particles are well confined in a compact phase space volume with little tailing. A final design of such a cyclotron has not been worked out in detail. Alternative solutions have been proposed by Mandrillion *et al.* [117] and at lower energies by Y. Jongen [118].

The DC proton source and pre-accelerator for an ADS facility are similar to the linac design discussed above. Prototypes of 100 mA ion sources exist. The acceptance of the CW beam into the injector cyclotron is, however, much lower than for the combination RFQ and linac. The pre-accelerator for the PSI Inj.2 is a 870 keV Cockcroft-Walton generator. For the acceleration of a 2 mA beam, a CW proton beam of 10 to 12 mA is bunched into the matched phase space volume of the Inj.2 mentioned above. Using a simple code for calculations of on bunched beams under space charge conditions it may be shown that a beam of 50 mA can also be bunched so that the charge corresponding to 10 mA beam current is contained in the same phase space volume [122]. Presumably an RFQ could also be employed as pre-accelerator, but a design study and prototype work as for the linac case have not been made for the lower RF frequency range (around 50 MHz) used in cyclotrons. The tools for such a study exist.

In the following key fields further development and prototype work might be needed:

- Radio-frequency (RF) systems (used for acceleration), with special emphasis on high power CW amplifiers, high power coupling loops, cavities with high acceleration voltages and low spark rates; and generally, "flat-topping systems". Flat-top systems are used to permit a wider particle phase acceptance during the acceleration process. They decelerate the beam, and stability in voltage and phase becomes difficult to achieve as soon as the power absorbed from the beam exceeds the wall power in the flat-top cavities.

- PSI is pursuing a project to develop and build a RF cavity for >1 MV peak voltage. Relatively simple conditioning will suffice; fortunately, considering the size of cavity, no chemical cleaning, high temperature baking procedures, etc. is needed to reach the relatively modest electric field gradient of ~3.5 MV/m [128].
- Beam collimation at high power, the design of local shielding, as well as installations for remote handling and replacement of highly activated parts.
- Simulation of beam behaviour and longitudinal matching under strong space charge forces for the PSI Inj.2 and an injector cyclotron for an ADS facility, and especially simulating the performance of an RFQ as pre-accelerator. An advanced computer code for this task has been developed in collaboration between CERN, LANL and PSI.
- Injection and extraction systems, now consisting of a combination of electromagnetic and electrostatic (high DC voltage) components, will have to be optimised to handle increased beam losses besides being designed explicitly for low spark discharge rates in electrostatic devices.

*Beam power, trip rate and duration, component reliability*

The success of the PSI cyclotron at beam currents up to 2 mA demonstrate that it is feasible to obtain the desired performance at high beam intensities with a cycloton-based accelerator provided oncesatisfied with today's reliability.

Operating cyclotrons at high beam power and, at the same time, requiring very few beam trips of short duration and pushing the time lost to unscheduled beam interruptions to negligible levels, poses a relatively recent challenge in the development of cyclotron technology. The priorities in accelerators used in nuclear and particle physics were clearly set to push the technological performance envelope to higher currents, higher precision in terms of energy resolution or higher yields in the acceleration of exotic particles and charge states (radioactive beams). Cyclotrons designed for medical applications (isotope production and irradiation therapy) were for the first time faced with extreme demands for availability (>95% of the scheduled beam time, low unscheduled down times) and a high annual beam time (>7 000 h/y; that is: low scheduled maintenance time). An ADS makes even tougher demands requiring no more than a few 10s to a few hundreds of beam interruptions per year [127], i.e. a beam availability of better than 98%. The demand corresponds more to the conditions typical of a nuclear power plant than those in a research facility. Hence existing facilities are generally not well suited to evaluate what can be achieved with respect to reliability, as can be seen in recent summaries for linacs [111], or cyclotrons [127], which report at best ≈100 trips/week.

In statistical terminology, Mean Time Before Failure is the critical parameter, failure in this context meaning a beam interruption (beam trip) This MTBF in cyclotrons is dominated by sparking in RF and electrostatic deflection devices.

Beam interruptions due to sparking are generally short (duration <1 mn), but still too long with respect to the thermal time constant of transmuters or sub-critical multiplying assemblies. In the PSI facility the trip rate due to sparking is as high as 8500/y under good conditions [129,130]. This, however, is accounted for in the design of the 1 MW spallation target and not considered to be a problem. For cavities in general, further studies are needed, both on the mechanisms that cause a discharge and on measures for fast recovery, so that the beam can be maintained uninterrupted.

Redesigning critical components in terms of geometries that optimise field gradients, and methods of conditioning and surface treatment, are really the only measures available.

The Mean Down Time of the accelerators is generally dominated by unscheduled interruptions of longer duration lasting >1 h, usually due to component failures. To reduce their contribution one has to focus on the different types of systems separately. Preventative maintenance has to be performed, in many cases increasing operating costs. Furthermore, high voltage power devices, as used in beam deflectors (DC), or RF power amplifiers employing vacuum tubes, are more susceptible to failure than their low voltage counterparts. Important in any case are means to assist in quick fault diagnostics, ready-to-operate replacement units, fast interchangeability in all critical components and devices. Increasing the lifetime of critical components, like the electrodes of beam deflection devices, RF amplifiers (tubes) and RF couplers (windows) is an important aspect of reducing unscheduled down-time and operating costs.

Overall operating costs, including power consumption, maintainability and maintenance costs, etc., became an issue when cost effectiveness was being analysed for commercial rather than research applications of cyclotrons. Existing large facilities, however, have up to now not been optimised with these considerations in mind, so there seems to be considerable potential for improvement in this respect [131,132].

Total MTBF and MDT will always depend on the total number of critical components in an accelerator, because failure rates cannot be lowered below a certain reasonably attainable number, and total component redundancy will be precluded by costs. Ultimately, since critical components like RF cavities, high voltage power supplies, RF power amplifiers, etc will always show comparable reliability characteristics, whether used in circular or linear accelerators, a critical component count as low as possible will be an important factor in overall accelerator performance (MTBF and MDT) for ADS systems.

*Beam losses*

Concerning beam loss, the extraction of the beam from the 1 GeV cyclotron is the most critical point. A good separation between the orbits at the extraction radius is mandatory in order to achieve good extraction efficiency. It is given by the average radius and the number of turns. In the 10 MW facility these parameters have been selected so that the separation of turns is larger than in the 590 MeV cyclotron at PSI. The yearly averaged extraction efficiency achieved in routine operation of the 590 MeV cyclotron is as high as 99.98% and the same extraction efficiency can be expected in a 10 MW facility.

The truly limiting factor is the radiation dose imposed on the personnel involved in repair and maintenance. This is difficult to predict, because the dose depends not only on the beam loss in the cyclotron and beam lines, but to a larger degree on the design of the equipment, the installation of local shielding, on provision for quick and remote removal of activated components into shielded boxes and the use of manipulators [133,134]. It also depends on preventive measures like concentration of activation products in specially designed beam catchers, optimised material selection and, last but not least, the attitude of the personnel themselves in handling of activated components. The serviceability after irradiation is related rather to the design than to the amount of beam produced or lost. In the PSI facility the annual beam production has been upgraded by three orders of magnitude over the past 25 years while the dose to the personnel could be halved in the same time. The conclusion is that, with proper design strategies as mentioned above, a 10 MW facility can be handled, provided the beam transmission remains comparable to that in the PSI cyclotrons.

*Power conversion efficiency*

The power efficiency of the facility depends very much on the type of accelerator, on the size of the cyclotron and on the amount of beam loading. It is highest if the facility is operated close to the intensity limit, i.e. at the highest possible beam power for a given accelerating voltage. It can, therefore, only be known after a final design has been finished.

The PSI facility is operated at a low beam loading factor. The power efficiency of 10% is, therefore, rather low. The beam power is 1 MW, the RF power needed to produce the high acceleration voltage is about 1.7 MW and if we assume 67% conversion from mains to RF this results in another 1.35 MW lost. The pre-accelerator, injector cyclotron and beam lines need 2.6 MW and the whole infrastructure demand amounts to about 2.5 MW.

For the proposed 1 GeV cyclotron facility the power efficiency has been estimated to be about 36% [129,130]. The beam power is 10 MW, the RF power needed to produce the high acceleration voltage is about 4 MW and if we conversion from mains to RF amounts to 7 MW lost. The pre-accelerator, injector cyclotron and beam lines need 3 to 4 MW and the whole infrastructure demand has been taken as about 3 MW. The figures given are based on extrapolation from the existing facility without any consideration of power-saving technology.

#### *4.3.4.3 Multiplexed modular accelerator concepts*

A future ADS facility for transmutation or energy production must have a beam power specification of several tens of MW. If at the same time beam interruptions are restricted to the extremely low 10 to 40 trips/y, then new strategies in accelerator design will be necessary for linacs and cyclotrons. A beam availability so high can only be reached with highly redundant systems, which significantly add to the cost of a future facility. No conclusive judgement on the cost of a possible facility can therefore be formed at present in the absence of detailed projects.

A thorough investigation into the possibility of redundant subsystems has not, to our knowledge, been made. In view of the large investment involved in such a facility a redundancy in the largest subsystem, which is the accelerator itself, has to be considered.

Two scenarios can be thought of and have been proposed:

- Use of for instance three 100 MW linacs to drive four transmutation targets, as proposed by G. Bauer, with splitting of all three beams, on a pulse by pulse basis, into identical portions directed to all four target systems. If one accelerator stops operation the power to the four target systems is reduced by only 33% each [124]. For such a scenario the accelerators most probably would be linacs.
- Use of for instance three 10 MW cyclotrons driving one 30 MW target [129,130]. Again outage of one cyclotron reduces the power on the target system by only 33%. In this scenario one could even think of having a fourth cyclotron as a back-up. In this case the beam could be brought back to full power quickly if one were willing to add to the operation cost the electricity bill for keeping the fourth cyclotron running.

#### *4.3.4.4 Status of current accelerator projects*

JAERI and KEK (High Energy Accelerator Research Organisation) have been jointly proposing the multi-purpose complex facilities in the High-Intensity Proton Accelerator Project [135], and the Phase 1 Project was approved for construction. Phase 1 includes: 1) 400 MeV NC linac; 2) 3 GeV Proton Synchrotron (PS) at 1 MW; 3) 50 GeV PS at 0.75 MW; 4) the major part of the 1MW SNS facility; and

5) a portion of the 50 GeV experimental facility. The total budget of Phase 1 is 133.5 billion yen. The phase 1 will be completed within 6 years. Phase 2 will comprise construction of an ADS experimental facility including 400 MeV to 600 MeV SC linac, upgrade of SNS to 5 MW, construction of a neutrino beam line and upgrade of the 50 GeV experimental facility. R&D of super-conducting cavities for a proton linac has been performed since 1995 at JAERI. The vertical tests of 5 cell cavities of $\beta = 0.5$ and $\beta = 0.89$ have been carried out with surface electric fields of 23 MV/m and 31 MV/m, respectively, at 2 k [136,137]. Fabrication of a prototype cryo-module, which includes two 5 cell cavities of $\beta = 0.60$, is in progress and performance test will be made in 2001.

Two projects, IPHI (Injecteur de Proton Haute Intensité) in France and TRASCO (TRAsmutazione SCOrie) in Italy, are also of particular interest in view of the design and construction of an experimental ADS. A collaboration (CEA-CNRS-INFN) between the two projects has been formally established in such a way that, even though each project has its own programme, many important choices are common in order to obtain the maximum profit from the investments made by the two teams.

IPHI is a 1 MW, 10 MeV demonstrator accelerator, that could be used as front end for a high power proton linac. It consists of:

- An ECR source (SILHI, Source d'Ion Légers Haute Intensité), operated at 2.45 GHz with an ECR axial magnetic field of 875 Gauss, able to deliver a 95 keV, 100 mA proton beam.
- A normal conductive radio-frequency quadrupole (RFQ) able to provide a 500 kW, 5 MeV CW beam.
- A drift tube linac (DTL) tank that brings the proton energy up to about 11 MeV.

The SILHI source has already been built, as well as the low energy beam transport (LEBT) line. The design of the RFQ has been completed; its construction is now going on and should be completed by 2002. The construction of a short DTL tank is in progress while the definition of the high-energy beam transport (HEBT) has almost been done.

The objectives of the first part of the Italian TRASCO research programme, leaded by INFN, are:

- A conceptual design of a 1 GeV, 30 mA proton linear accelerator (linac).
- The design and construction of the TRIPS proton source and of the 5 MeV, 352 MHz CW RFQ.
- The study of possible alternatives for the linac part from 5 MeV (the output of the RFQ) up to about 100 MeV.
- The design of the high-energy section of the linac, based on super-conductive elliptical type accelerating structures, as well as the construction of some prototypical super-conducting RF cavities.

A reference conceptual design of the proton source and medium energy section – the 352.2 MHz RFQ and a DTL – has been determined, for a nominal accelerated current of more than 30 mA. The TRIPS proton source has been built and is under commissioning. A detailed design and engineering work of the 352 MHz RFQ has started and a 3 m long aluminium model of the RFQ has been built and measured for RF field stabilisation tests. Technological tests on a short copper section have been done and the first section of the RFQ is in construction. Preliminary studies of an ISCL (Independently phased Superconducting Cavity Linac) – to be used instead of the traditional DTL – have been also done. The conceptual design of the 352 MHz super-conducting LINAC, able to bring the 30 mA proton beam from 100 MeV up to 1 700 MeV, has already been worked out and is mostly based on the LEPII technology. The construction and the tests of the Nb-sputtered copper $\beta = 0.85$ single-cell and multi-cell prototypes cavities has been done at CERN, under a collaboration agreement between CERN and INFN.

#### *4.3.4.5 Concluding remarks*

As can be guessed from the preceding paragraphs, a qualitative comparison of the two accelerator concepts cannot be made at present with any reasonable degree of confidence; too many aspects still depend on further R&D in various disciplines of accelerator science and engineering.

## 4.4 Conclusions

This chapter started by asking if ADS-technology differs from the already developed FR-technology, if there may be synergy in future development and if there are significant bottlenecks that might be foreseen. The previous description has highlighted the main differences especially in the level of development of reactor technology. Synergy is possible and has to be sought in the future in fuel and materials development where particular focus is necessary on the accelerator-reactor coupling, the dynamic behaviour of ADS and the spallation target technology. In general, this chapter has indicated that:

- On the whole, the development status of accelerators is well advanced, and beam-powers of up to 10 MW for cyclotrons and 100 MW for linacs now appear to be feasible. However, further development is required with respect to the beam losses and especially the beam trips to avoid fast temperature and mechanical stress transients in the reactor.
- Various problems related to the accelerator-reactor coupling have still to be investigated. Thereby, special attention has to be given to the target, and especially the beam-window, which is subjected to highly damaging spallation particles and nuclides and corrosive environments which are not encountered in normal reactors. To this end, research programmes have been initiated in Europe and elsewhere.
- While the reactor physics of sub-critical systems is well understood, the issues regarding the dynamic response to reactivity and source transients require investigation because they are the area of greatest difference between critical and sub-critical systems.