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New Reactors Concepts and Scenarios

A. Gandini
University of Rome
Italy

NEW REACTORS CONCEPTS AND SCENARIOS

A. GANDINI
University of Rome, Nuclear Engineering Faculty
Email: gandini@dinceal1.din.uniroma1.it

In recent years an increasing interest is observed with respect to subcritical, accelerator driven systems (ADS), for their possible role in perspective future nuclear energy scenarios, as actinide (Pu and MA) incinerators, and/or claimed energy plants with potential enhanced safety characteristics. Important research programs are devoted to the various related fields of research. Extensive studies on the ADS behavior under incidental conditions are in particular made, for verifying their claimed advantage, under the safety point of view, with respect to the corresponding critical reactors. Corresponding medium and long range scenarios are being studied to cope with a number of concerns associated with the safety (power excursions, residual heat risk), as well as with the fuel flow (criticality accidents, fuel diversion, radiological risk, proliferation). In the present work we shall try to review current lines of research in this field, and comment on possible scenarios so far envisaged.

1 Introduction

The main lesson learnt so far in the four and a half decades of pacific uses of atomic energy following the first Geneva Conference for Peaceful Used of Atomic Energy in 1955, is that accidents in nuclear reactors posing risks greater than those perceived acceptable as an inevitable price to be paid for progress, are related to human factors. Such has been in fact the case with the events occurred at Three Mile Island, Chernobil, and, more recently, Tokaimura. This latter incident, in particular, has broadened the perception of major possible risks outside the reactor plant.

Since the TMI accident, extensive researches worldwide in many areas of the safety domain have been pursued, from the sophisticated PRA methodology to the concept of passive safety, based on natural laws of physics, and to the development of more friendly man/machine interfaces. But, notwithstanding these new ideas, and the excellent record in terms of safety and efficiency of power plants operated in OECD countries after the TMI event, the whole fission nuclear energy establishment, due to political, economic, and social pressures, was pushed into a sort of a limbo, with most countries freezing, or phasing out from industrial engagements. Luckily, a few nations have maintained an active commitment in industrial development and production (in particular, France and Japan), whereas a larger number engaged into R&D programs, generally oriented toward innovative concepts.

On the other end, alternative energy resources, i.e., other than gas, fossil fuels, and hydroelectricity, although significant for certain applications, do not seem adequate for satisfying the perspective bulk energy needs. In the mean time, environmental concerns, pollution and global warming, have being building up.

So, in recent years, a revived interest has being growing on the role fission nuclear energy could play in relation to a number of issues:

- responding to the energy needs in a world of increasing energy demand and in view of a medium/long term reduction/exhaustion of fossil fuels, in absence of alternative usable energy resources of comparable scale;

- reducing the large quantities of plutonium existing in military arsenals and spent fuel stocks;
- ensuring proliferation resistance;
- reducing the greenhouse effect;
- ensuring a limited long term radiological risk;
- ensuring a safe transition to an expanding nuclear phase in a period of increasing energy demand;
- ensuring the possibility of closing the fission cycle, whenever a different, important energy source is developed as a substitution, with minimal radio-toxicity risk from residual waste.

The extent of development so far gained by the nuclear industry, and still being gained along with R&D activities in many world laboratories, allows a quite large variety of concept options and to be considered answering the above demands.

In this work we shall try to give a perspective review of current lines of research in this field, and to comment on a few possible scenarios so far envisaged.

2 Reactor concepts

Among new concepts proposed in recent years, besides advanced designs based on LWR technology, entirely new lines of research have been pursued in many laboratories in relation to accelerator driven systems (ADS), i.e., consisting of subcritical systems in which the fission reaction chain is sustained by an external neutron source.

A variety of ADS concepts have been so far considered, after first proposals by Bowman[1] and Rubbia[2], the first one mostly oriented to transuranium (TRU), or minor actinides (MA) elements incineration, the second one mostly oriented to energy production. Such variety stems from possible options relevant to:

- reactor core: thermal, or fast;
- accelerator type: linac, or cyclotron;
- fuel cycle: U-Pu, or Th-U;
- coolant: Na, Pb, or gas;
- fuel form: metal, oxide, nitride, or molten salt;
- processing: aqueous, or pyro-metallurgical;
- main scope: TRU (or MA) burning, or energy production and TRU burning.

Due to the uncertainty affecting these various options, in terms of performance, costs and/or feasibility, it seems quite difficult to decide now a clear choice.

To answer these difficulties, wide internationally coordinated R&D effort is underway. Among the areas covered we may list:

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[1] 1. C. D. Bowman, et. al., "Nuclear Energy Generation and Waste Transmutation Using an Accelerator-Driven Intense Thermal Neutron Source," Nuclear Instruments and Methods A320, 336-367 (1992)

[2] C. Rubbia, et al., "A High Gain Energy Amplifier Operated with Fast Neutrons", Proceedings of Las Vegas Conference on Accelerator Driven Transmutation Technologies and Applications, 25-29 July 1994.

- accelerator technology;
- on site processing techniques (in particular, pyroprocessing);
- Pb technology (in particular, corrosion/erosion effects on the containment vessel walls);
- molten salt technology (in particular, corrosion effects on the containment vessel walls).

Above all this, a main question needs to be answered: which are the advantages of ADS systems with respect to their critical counterparts?

In principle, each role could be played, with larger or minor efficiency, by a critical, as well as by a subcritical system, although such role interchangeability may have limits due to safety considerations.

In order to make a first comparison between critical and subcritical systems, let us consider a critical reactor (energy producer and/or incinerator) with given fuel mixture formed by Pu and MA.

The fuel of an ADS counterpart, at a given subcriticality level, would then consist in a similar mixture, in which the Pu fraction could be:

- maintained the same as in the critical system, so that some absorbing material might be added, e.g., long lived fission products (LLFP) such as Te^{99} and I^{129}, consistently with the assigned subcriticality;
- just reduced, so to achieve the assigned subcriticality. Since, for achieving the same power, the neutron flux level would have to be enhanced to compensate the reduced fissile material content, a higher burning rate of MA would follow.

An intermediate option could also be chosen.

In all cases, there would be extra neutrons available for incinerating MA or/and LLFP.

Critical reactors (more likely fast ones, for their superior neutron economy properties) could as well assume the role as incinerators, although not with the same efficiency. On the other hand, critical reactors have the advantage of a better power space distribution than ADS systems. This is due to the presence, in subcritical systems, of relatively strong flux gradients in the direction of the neutron source, increasing with the degree of subcriticality. This neutron flux peaking might, however, be somewhat alleviated by adequately reducing the fuel enrichment and/or by a proper distribution of neutron poisons in core regions near the spallation neutron source (although such measures would in turn penalize the advantage of these systems as incinerators, for the relative loss of weight of the spallation neutrons and/or for the extra parasitic captures, respectively).

The use of ADS in place of critical reactors would of course introduce extra costs, in particular connected with the accelerator capital, operation, and maintenance, besides the plant efficiency penalty. Possible savings on other costs may be however envisaged.

The main claimed justification for ADS rather than critical systems appears related to safety considerations. Under this respect there are two main arguments:

- ADS's would allow, in principle, to avoid using control elements for reactor operation and reactivity compensation during burn-up, this leading to the exclusion of the reactivity accident following an inadvertent control rod extraction (there are

ADS proposals, however, in which control elements are introduced to allow an accelerator operation at constant beam current).

- the distance from criticality conditions is equivalent to an extra amount of delayed neutrons. This property, in particular, allows to consider them the best candidates as MA incinerators, in consideration of the relatively small delayed neutron fraction associated with these elements.

To the first argument we may object that for an ADS we would have to take into account the specific accident of an inadvertent insertion of all the reserve current, relatively large at beginning of life, cold condition, and to the serious problem of a loss of flow (LOF), or a loss of heat sink (LOHS) accident without current interruption. In these latter cases, in fact, it can be shown[3] that the coolant temperature reaches unacceptable levels (this problem could be however alleviated by coupling the accelerator current feed with the reactor power[4]).

The second argument seems the stronger one. An adequate subcriticality level would assure, in fact, a "safe" distance of the fundamental mode multiplication coefficient (K_{eff}) from prompt criticality condition.

For what concerns the possible introduction of an absorber control poison (in rod or dilution form) for the reactivity follow-up, we should consider that this, for assuring a comparable safety level, would imply increasing the degree of subcriticality, so to assure that an accidental control rod extraction would not bring the system reactivity beyond a prescribed level. This in turn would require a more intense proton current and then a further cost penalization.

As a conclusion, for the same safety level, it appears more efficient to use the sole current control mode for insuring the necessary reactivity adjustments during reactor operation and burn-up.

Efficiency loss

The efficiency loss of the system is directly related to the power fraction needed to run the accelerator, this fraction in turn increasing with the subcriticality. As well known[5], in a subcritical system, driven by a given external neutron source, the reactor power is proportional to $K_{eff}/(1-K_{eff})$.

For purpose of illustration, in the following a simple expression of the efficiency loss is derived, evidencing its dependence from the subcriticality level and other accelerator related parameters.

Consider the accelerator power (in MW)

$$W_{acc} = cE_p/f_b \, , \qquad (1)$$

[3] A. Gandini, M. Salvatores, I. Slessarev, "Balance of Power in ADS Operation and Safety", Annals of Nuclear Energy 27,1 (2000).

[4] A. Gandini, M. Salvatores, I. Slessarev, "The power-current feed coupling in ADS systems", ADTTA '99 Conference, Prague, June 7-11, 1999.

[5] S. Glasstone, M.C. Edlund, "The Elements of Nuclear Reactor Theory", Van Nostrand, New York, 1952, p196.

where c represent the current (in mA), E_p the proton energy (in GeV) and f_b the accelerator power efficiency[6]. We may then write the expression for the overall fission source rate as

$$\varphi\nu\Sigma_f V = n_{/mA} f_b W_{acc} K_{eff} / E_p(1-K_{eff}) \qquad (2)$$

where V is the core volume and $n_{/mA}$ (an increasing function of the proton energy E_p) represents the "effective" number of source neutrons per unit (mA) current.

A current of 1mA corresponds to 0.625×10^{16} protons/sec. We shall denote as $p_{/mA}$ this quantity. Indicating as "m" the number of neutrons produced by each proton in a lead target[7,8], and assuming a weight "g" for these neutrons with respect to the average importance of those produced by fission, it is $n_{/mA} = gmp_{/mA}$. A discussion on the weight (importance) of external source neutrons may be found in reference of note 9.

The ratio W_{acc}/W_e may be viewed as the efficiency loss fraction with respect to the plant electric power W_e (=$f_e W_t$). Replacing in above expression (2) $\varphi\Sigma_f V$ with W_t/κ, κ representing energy units (in MJ) per fission, it results

$$W_{acc}/W_e = \nu E_p(1-K_{eff})/\ K_{eff}\ \kappa f_b f_e gmp_{/mA} \qquad (3)$$

or

$$W_{acc}/W_t = \nu E_p(1-K_{eff})/\ K_{eff}\ \kappa f_b gmp_{/mA}\ . \qquad (4)$$

We may also write the expression relevant to the corresponding current intensity needed for a given power and subcriticality level:

$$c = W_t\ \nu(1-K_{eff})/\ K_{eff}\ \kappa gmp_{/mA}\ , \qquad (5)$$

and that of the beam power:

$$W_{beam} = E_p W_t\ \nu(1-K_{eff})/\ K_{eff}\ \kappa gmp_{/mA} \qquad (6)$$

It is generally assumed that the optimal proton energy E_p should be on the level of 1 GeV. This is mostly dictated by the need of limiting the material damage produced to the window by the intense proton beam[10].

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[6] i.e., the product of efficiencies AC to DC (~ 95%), DC to RF (~ 58%) and RF to X emission (~ 94%).

[7] The number of spallation neutrons (m) produced per impinging proton of energy E_p, in the energy range considered, increases along with the (empirical) expression[5]
$m = 3.717\times10^{-5} E_p^2 + 3.396\times10^{-3} E_p - 0.367$.
Recent indications give different values, about 10÷20% lower, for 1 GeV protons.

[8] S. Andriamonje, et al "Experimental determination of the energy generated in nuclear cascades by high energy beam", Physics Letters, B 348 (1995) 697.

[9] A. Gandini, "Sensitivity Analysis of Source Driven Subcrtitical Systems by the HGPT Methodology", Ann. Nucl. En., 24, 1241 (1997).

[10] This is so, in particular, due to the fact that the number of spallation neutrons per incident proton increases with its energy E_p along with the (empirical) expression shown in footnote 4, and, therefore, the proton current correspondingly decreases to produce the same neutron source.

Then, assuming k_{eff}=0.96, f_b = 0.5, f_e = 0.38, ν =2.7, m=33 and g= 1.2 (at high burnup condition)[11], and recalling that κ (energy per fission) is of the order of 200 Mev (= 3.2 x 10^{-17} MJ), we obtain

$$W_{acc}/W_e = 0.074\,, \quad W_{acc}/W_t = 0.028\,, \quad c = 0.014\ W_t.$$

For a thermal power of 840 MW (corresponding to a PRISM size reactor), at K_{eff}=0.96, assuming the parameter values above, we would obtain W_{acc}= 24 MW, corresponding to a proton current of about 10 mA. This corresponds to the value (~11 mA) indicated recently by Lawrence[12] for a new advanced linac concept and similar system parameters.

We can then say that the net efficiency of an ADS plant is so reduced, with respect to that (f_e) of its critical counterpart, and results given by the product $f_e(1-W_{acc}/W_e)$, with W_{acc}/W_e given by expression (3).

From expression (3) we can also obtain the break-even point of the subcriticality level, i.e., the value K_{break} (in the example above, corresponding to K_{eff}=0.64) at which the ratio W_{acc}/W_e equals unity. At this point all the electricity produced would be used to run the accelerator. The ADS system would be a "pure burner", with an incinerating rate more than doubled (because of the augmented flux so to maintain the same power) with respect to a similar system with K_{eff}=0.96.

A system of this type, with K_{eff} more or less far from the break-even point, would be justified in case the role of the ADSs were confined to serve as mere long lived fission products (LLFP) transmuters (as has been suggested in early ADS studies[13]), the incineration of MA being assigned to fast criticals.

3 Accelerator

The accelerators to be adopted for driving ADS systems belong to two broad categories: linacs and cyclotrons. Cyclotrons are limited to beam power intensities of the order of 10 MW (for proton energy of 1 GeV, corresponding to currents of the order of 10 mA), whereas linacs may achieve beam power intensities higher by one order of magnitude.

The high power accelerator technology has been under continuous improvement over the past decades, and have been developed into relatively efficient research and industrial tools.

Detailed studies, particularly at the Paul Scherrer Institute (PSI), where a considerable experience has been built on their 590 MeV SINQ accelerator now operating routinely at a current level around 1.5 mA, indicate that the construction of a 1 GeV

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[11] Coefficient g should account for the importance of the highly energetic spallation neutrons with respect to the average importance of the fission ones. This values changes in relation to the specific ADS system and decreases with burnup. An estimate made in a benchmark exercise (see: I. Slessarev, "IAEA-ADS Benchmark, Results and Analysis", TCM-Meeting, 17-19 Sept. 1997, Madrid) has given an averaged value of 1.1 (starting from about 1.33) for an ADS similar to the CERN's lead cooled, U-Th fueled, EA system at high burn up condition.

[12] G. Lawrence, "High power accelerator design and development for ATW systems, ADTT'99 Conference, 1999, Prague

[13] M. Salvatores, I Slessarev, M. Uematsu, "Physics Characteristics of Nuclear Power Systems with Reduced Long-Term Radioactivity Risk", Nuclear Science and Engineering, 120, 18 (1995).

isochronous cyclotron with intensity up to 10 mA is entirely feasible. This is quite adequate for a class of ADS fast systems up to powers of the order of 800 MWt (assuming a multiplication coefficient of 0.96 and a PbBi target). The choice of a cyclotron for ADS systems up to these powers seems optimal, considering that the cost of cyclotrons, due to their compactness, should results significantly lower than that of linacs.

If ADS plants with powers of the order of 1 GWt or larger, are considered, or if one single accelerator is assumed feeding a cluster of relatively small ADS modules, as in the recent proposal by Los Alamos National Laboratory (LANL)[9], linac accelerators should be adopted.

4 Coolants

Lead-Bismuth Eutectic

Nuclear reactor systems cooled by lead-bismuth eutectic (LBE) are subject to a number of serious problems, namely:

- corrosion, erosion and mass transfer;
- radioactive safety problem, particularly due to the α-active Po produced with the reaction: Bi^{209} (n,γ) $Bi^{210} \rightarrow$ Po^{210} (its half life of 138 days poses an operational, rather than a waste concern);
- freezing/defreezing problems;
- thermohydrodinamic problems;
- environmental concern, since lead and bismuth are heavy metal requiring adequate separation from the environment. When irradiated, they represent a "mixed" waste which further complicates disposal.

Major research efforts are at present underway in many laboratories to address these issues. The corrosion issue affecting the vessel integrity is a major one. A protective film consisting of oxide compounds of steel components is claimed as an adequate measure[14]. The basis of the film would be Fe_3O_4. The reliability of such a solution needs to be verified, recalling the experience in metal cooled reactors of not negligible oxide layer detachments, in presence of power/temperature transients (and, in the case of an ADS, also in presence of spallation products, if the coolant serves also as target).

On the other hand, LBE as coolant has a number of advantages, among which, relatively to sodium,

- the impossibility of explosion and fire at interaction with air, water and vapor, due to the low chemical lead and bismuth activity;
- the impossibility of the coolant boiling out, due to the high boiling point of Pb-Bi (1670°C);
- its enhanced ability (with respect to sodium) of transmuting long lived fission products, such as Tc^{99}, in case this is a strategy to be adopted. This ability stems from the relatively small neutron lethargy loss per collision associated with the PBE

[14] Yu. I. Orlov , "Stages of development of Lead-Bismuth as a coolant for nuclear reactors in Russia", MIT ATW Technical Review Meeting, 1998, Boston, Jan. 15-1998.

coolant, which allows neutrons to remain longer after many collisions within the capture resonance width ranges of those isotopes;

- superior properties for natural convection cooling;
- the possibility of increased efficiency, due to the ability of liquid Pb-Bi to support higher thermal efficiencies due to the high boiling temperature (temperature limits associated with the corrosion issue may however limit its operating value);
- transport cross section of Pb ~2 ÷3 times larger than that for Na, (although its capture cross section is ~5 ÷10 times larger than that, relatively small, of Na). Besides, the Pb atom density is 40% higher than for Na. As a consequence, the neutron economics in presence of the Pb coolant are better: lower fissile loading which in turn entails higher breeding and, consequently, smaller reactivity swing;
- retention of most actinides and fission products if released into the coolant after an accidental event.

Sodium

Sodium is the best known metal coolant for fast reactors. A main advantage is its compatibility with all the materials to be used for fabrication of fuel cladding, reactor vessel, internal components, together with its low pressure of operation.

Strong efforts have been made on the Superphenix reactor to improve the in-service inspection, which remains an issue (as for other liquid metals). Another drawback is its chemical reactivity with both air and water.

The use of sodium as coolant would require its physical separation from the spallation medium, supposed to be Pb-Bi.

Comparison of sodium with lead-bismuth has been done above, when discussing the Pb-Bi coolant.

Sodium may be considered a possible back-up solution in case Pb-Bi technology would result impracticable.

Gas

The utilization of gas (helium, or CO_2) as coolant has a number of advantages, since it:

- facilitates the in-service inspection and repair, as compared to liquid metals;
- allows the operation with fast neutron spectrum, with no adverse reactivity feedback in case of voiding as compared to liquid metals;
- has a negligible absorption of neutrons. Hence it produces no radiological concern;
- is non-corrosive.

Use of gas, however, requires operating with high internal pressure (typically 50 to 70 bars), with the related consequences on vessel loading; also, in case of loss of coolant (de-pressurization), decay heat removal is somehow more troublesome in natural circulation and would lead to very high temperatures for the fuel, and, correspondingly, for the internal structures.

Gas cooled reactors have been already developed, with thermal neutrons (built and operated) as well as with fast neutrons (conceptual design stage).

The choice of gas requires a physical separation between the spallation medium and the primary coolant, with the added constraint that this separation must withstand the loads (pressure, temperatures) related to the gas.

5 Fuels

Traditional fuels generally considered for metal cooled fast systems have been oxides and metals. There has also been a long-term interest in carbides and nitrides.

Since fuels relevant to ADS (in their role as incinerators) will not contain large amounts of fertile material, the (negative) Doppler temperature feedback would be small. This would imply to maintain the subcriticality $(1-K_{eff})$ at a large, cost penalizing level. We should remark, however, that critical systems using highly enriched uranium have been operated extensively in the past. These systems had relatively low Doppler effect but were designed to take advantage of other temperature feedbacks, such as thermal expansion (included outward bowing) and/or moderator temperature coefficients.

Other isotopes other than U^{238} could be as well considered for designing systems with enhanced Doppler, such as tungsten, hafnium or dysprosium, with resonance integrals 350, 2000 and 1500, respectively. With sufficient Doppler feedback, ADS systems could be safely run at higher k_{eff}'s.

Metal

Metal fuels have very good thermal properties and can be directly processed with pyrochemistry. Experience with the integrated fast reactor (IFR) developed at the Argonne National Laboratory demonstrated that burnups of 10 to 20 atom percent could be achieved with these fuels.

Nitride

Nitride fuels have good thermal properties and can be also used with pyrochemical processing. Traditional nitride fuels are being studied for use in burning the minor actinides while using the plutonium in fast or thermal reactors (double strata strategy)[15].

Molten Salt

Molten salt fuels allow for the fuel to be dissolved in the coolant. This provides for a homogenous mixture where fuel can be added and removed (quasi) continuously. This continuos operation transforms the concern over the reactivity swing during burnup, to be compensated by an appropriate (poison, or accelerator current) control, into that of correctly adjusting the feed (fuel inlet) and bleed (fuel outlet) operations. Correct

[15] T. Mukaiyama, "Status of Partitioning &Transmutation R&D, and research needs for ATW from JAERI perspective", ATW Meeting, USDOE/OCRWM Washington DC, 17-18 Feb. 1999.

operation of a molten salt reactor would sustain a constant amount of subcriticality, thereby avoiding burnable absorbers, control rods, or changing the accelerator power. The traditional first barrier to radionuclides release, the solid fuel form, is non-existent in molten salt fuels. However, multiple barriers can be designed to compensate for this change in safety basis. Since molten salt reactors process the fuel continuously, the quantity of fission products is reduced when compare to a solid fuel design.

Coated Particles

Fuels based on the high temperature reactor (HTR) technology with coated particles could be an option for operating at very high temperatures. There are basically two possible types of particle fuel: sol-gel, developed by General Atomics, based on graphite kernels, coated with silicon carbide, and the infiltrated kernel approach developed by BNL.

Sol-gel was developed in the HTGR program, and has been demonstrated for uranium and plutonium carbides and oxides. Other actinides should perform well, in principle.

All particle fuel should be coated to enhance fission product retention, and protected against reaction/erosion by the coolant. The choice of coating materials is based on compatibility with the kernel and coolant, and the level of burn-up desired. In general, silicon-carbide or metal-carbide are the materials of choice. A pyrolytic carbon intermediate layer is usually used between the kernel and coating for fission product retention, and to protect the outer coating for high burnups (over 700 GWd/t) with the sol-gel based particles.

The use of a bare particle geometry allows direct cooling of the particles. As a result, high power densities, hence high flux levels can be achieved. This ensures high burnup rates for the actinides and other fission products. The thermal margins in operating and accident scenarios are also high.

The coated particle fuel considered above was developed in relation to thermal reactors. There is also an interest to develop a similar fuel form for fast, or intermediate, systems (possibly, with some ceramic outer layer, as TiN, and with particles of ~10-fold larger radius). Little experience exists so far in this field and a fairly large R&D effort would be required.

6 Fission product burning

ADS incinerators are considered also for transmuting Tc^{99} and I^{129} by neutron absorption. The design of the target material is under development but several issues can be addressed. Tc^{99} plus a neutron becomes in the order of tens of seconds stable Ru^{100} which has similar physical properties to Tc^{99}. However, I^{129} plus a neutron becomes Xe^{130} in the order of tens of hours and Xe^{130} is a noble gas. In the case of Tc^{99} a target can be developed that will allow a very long residence time. In the case of I^{129} pressure buildup is a concern which must be accounted for in the target design and could limit the target life. Another difference between the two fission products is their cross section. Tc^{99} has a thermal absorption cross section of 20 barns and a resonance integral of 300 barns. I^{129} has a thermal absorption cross section of 30 barns but a resonance integral of only 50 barns.

As previously mentioned, the use of Pb-Bi as coolant would help transmuting these fission products, due to the larger number of scattering collisions, with respect to lighter coolants, required by a neutron for escaping a capture resonance width.

In order to help the transmutation of Tc^{99} and I^{129}, some moderation of the flux is useful. Unfortunately due to high thermal fission cross section of the MA, thermalization of the neutrons can result in local power peaking problems.

7 Different ADS Concepts

In principle, any reactor design, reconfigured for source-driven subcritical operation, is a viable option for consideration as the sub-critical blanket/burner for an ADS system.

In previous sections our main reference have been an IFR-like (Pb-Bi or Na cooled) system. Below, we shortly describe two other concepts of interest as possible ADS candidates.

Molten Salt Systems

In the molten salt option, fuel and coolant are obviously the same medium. A reactor using a molten salt mixture of uranium, thorium, beryllium and lithium fluorides has worked with success for several years in Oak Ridge.

A major advantage of molten salts are the great flexibility for the insertion of MA and LLFP, together with the capability of accommodating very high burn-ups. As molten salts, one can envisage chlorides or fluorides. In the first case, operating temperatures are reasonably low (600 to 700°C), but chlorides have some adverse properties, such as corrosion (particularly with respect to steel) and a relatively high neutron absorption. In the second case, operating temperatures would be higher, but fluorides seem more stable. In any case, use of molten salts would need a very lengthy R&D effort, due to larger unknowns associated with material selection, operation and general engineering approach to design, construction and licensing.

At JAERI a molten- salt fast ADS concept with chloride salt as target, fuel and coolant, and with on-line processing, is studied as an alternative[15] within their development program. The material compatibility problem is the major concern.

In US a thermal molten salt ATW transmuter has been also analyzed[16]. The system design utilizes NaF-ZrF carrier salt. The advantages claimed for this design are:

1. No backend chemistry. Once the fuel is fluorinated it is maintained in the core until discharge.
2. Immediate denaturing of the plutonium once mixed with the molten salt of the core. This would meet antiproliferation requirements.
3. Due to the homogenous nature of the fuel the reactivity change with burnup can be minimized by continuous feed/bleed operations. A multiplication coefficient (K_{eff}) closer to unity might then be adopted.
4. No need for fuel fabrication.

[16] Annu. Rev. Nucl. Part. Sci., 48:505-556.

The disadvantages are:

1. Liquid fuel. This implies the reduction of one containment barrier (but there is no reason to assume that equal safety cannot be achieved).
2. Corrosive fuel. Although this was a problem with the original molten salt reactor, corrosion resistant materials have been developed for molten salt operation.
3. Mixing of fuel implies that 20% of the actinides remain at discharge. Another type of reactors should be developed to further reduce the actinides.
4. Less favourable neutron economy with a thermal spectrum. The neutrons per fission are about 7% less at thermal energies. The parasitic losses to fertile actinides are about a third greater for thermal systems. This can be compensated by an appropriate accelerator power and/or fuel enrichment increase.

High Temperature Gas Cooled Systems

High temperature gas-cooled systems use helium (or CO_2) as coolant, whereas the fuel elements contain coated particles embedded in a graphite matrix. In the German HTGR (critical, modular) design[17] , fuel spheres, each six centimeters in diameter, form a "pebble bed", whereas prismatic blocks are favored in the USA (General Atomics[18]).

Reactivity control can be enhanced by adding burnable poisons or fertile material to the fuel element. The addition of these materials would reduce the anticipated rapid drop in reactivity, and lengthen the cycle time.

The particle form would offer a high-integrity final disposal option requiring no additional processing. In this case, the particles would be encapsulated in a graphite matrix in a container which can then be placed in a suitable repository.

The particle nature of the fuel offers the potential for on-line re-fueling, similar to that employed in the German PBR, which provides additional flexibility for reactivity control. The significant improvement of large gas turbine generators developed over the last decade is quite reviving now the interest on these systems.

8 Scenarios

A variety of possible scenarios exist, depending on the type of cycle (open, or closed), fuel type (solid, molten salt, coated particles), strategy, etc.

Double strata

We may recall first the so called "double strata" option (closed cycle) consisting in:

[17] W. Gröger, "Safety Aspects of New HTGR Designs in the Federal Republic of Germany", Int. Workshop on Safety of Nuclear Installations of the Next Generation and Beyond, Aug. 28-31, 1989 Chicago.

[18] M.P. La Bar and W.A. Simon, "The Modular Helium Reactor for the Twenty-First Century", The Uranium Institute Twenty-Second Annual Symposium, 3-5 September 1997, London.

- a first layer comprehending LWR (and possibly) fast reactors, burning U and Pu,
- a second layer comprehending MA (and possibly Pu) ADS, and/or critical reactor burners.

For what concerns the first layer, the fissioning of plutonium in MOX form in LWR, as is presently done with a single recycle strategy, and might in the future be incremented with multirecycling, would reduce, if not eliminate, the use of fast reactors for Pu burning. The degree of reduction would of course depend on how many passes are made through MOX-fueled reactors and what portion of the total fuel loading includes Pu.

Ultimately, the use of MOX fuel recycling would lead to a larger discharge of minor actinides per unit burnup. If enriched uranium is blended in with MOX, a transition would be possible (after some relevant safety concerns will be solved) to an equilibrium condition in which only minor actinides would reach the ADS plants (in the second layer). In this case, the production of MA would more than triple that relevant to the once-through fuel cycle. The support ratio, i.e., the number of LWRs associated with each ADS TRU-burner, would also triple, increasing from 6÷7 LWRs (of nominal 1000 MWe) per ADS (Pu+MA) burner (of nominal 1000 MWt) to about 20 per ADS (MA) burner.

Burning all the plutonium in fast reactors rather than in LWRs within the first layer would greatly improve the support ratio. In this case it could possibly be in fact between 40 and 80 fast reactors (of nominal 1000 Mwe) per ADS plant. In turn, no less than 6÷7 LWRs would be associated with each fast reactor.

For what concerns the second layer, although a large variety of ADS concepts could be claimed as viable options, we think that first candidates for demonstration should be chosen starting from proven concepts, using fuel in solid form. This in consideration of the large experience so far built with them during many decades in the nuclear energy field, and of the strong selection occurred for what concerned system operation, burnup performance, reliability, safety.

Consistent R&D developments on promising alternative, potentially attractive, options, such as molten salt, or (within an open cycle logic) advanced high temperature gas cooled systems for actinide burning, should be of course pursued in a longer range perspective.

So, taking into account the demands for a final MA long term solution within the double strata logic, we envisage, as second stratum (ADS, or critical) systems, IFR-like reactors, integrated on site with pyrometallurgical fuel reprocessing and fabrication plants. This would also have the merit of eliminating the hazard associated with the transportation of highly active materials.

IFR-like systems would benefit from the long experience with fast reactors (in particular, with EBR II). It still needs to be established which coolant, Pb-Bi or Na, should be preferred.

Pb-Bi has a number of advantages, as we have already seen, but the corrosion/erosion effects, and the Po related radiation issue are serious problems. Fast reactors cooled by Pb-based eutectics have been studied in US[19,20] and designed and operated in Russia[21].

Na has known problems which should however be well coped with by existing technology. The smaller efficiency in LLFP transmutation doesn't seem to be per se a decisive discriminant with respect to Pb-Bi coolant. Its major drawback appears rather that of not having the capability of lead to dissolve within its bulk the fuel in case of an accident leading to fuel melting.

As far as the choice of ADS vs. critical systems is concerned, the main argument concerns safety: in particular, the possibility of a criticality accident leading to a prompt critical configuration. This would be a crucial issue in case of a pure MA burner.

Fast ADS systems can be made safer at the expense of system efficiency and at the not insignificant extra capital and maintenance costs associated with the accelerator. The probability of accidents leading to prompt criticality can be made negligible, assuming we may exclude the possibility of a current cut-off failure leading to fuel melt-down (and consequent possible re-compaction into a supercritical mass).

The safety of critical systems can be as well enhanced, by proper selection of (modular) size, fuel composition and form, coolant, and design.

As far as the β_{eff} issue relevant to MA burners is concerned, it could be answered by the addition in the fuel of an appropriate fraction of plutonium.

The U-Th cycle

Although the interest in uranium-thorium cycle is currently very limited, there are some interesting advantages associated with it. It can in fact significantly reduce the production of actinides, thus alleviating the long term radiological risk. In addition, this cycle can enhance proliferation resistance by mixing the thorium with uranium, so that U233 is diluted as it is produced. This resulting mixture is called DTU (denatured-thorium-uranium). In one recent study in a double strata logic, the support ratio for a DTU-LWR fuel cycle, with recycle of thorium and uranium, was determined to be 14 DTU-fueled LWR reactors per ATW transmuter plant.[22] Mixing uranium with thorium, if on one side reduces the U^{233} problem, on the other one creates that of Pu production. Its degradation at equilibrium conditions might reduce the issue, but the concern remains. There seems however that the production of Pu^{238} (a strong source of spontaneous neutrons and decay

[19] J.R. Liaw, E.K. Fujita, D.C. Wade, "Comparative Neutronic Analysis of Pb- versus Na-Cooled LMR Cores", Proceedings of the 1992 Topical Meeting on Advances in Reactor Physics, March 8-11, 1992, Charleston, SC.

[20] S. Yiftah and D. Okrent, "Some Physics Calculations on the Performance of Large Fast Breeder Power Reactors", USAEC Report ANL-6212, Dec. 1960.

[21] Yu. I. Orlov, "Stages of development of Lead-Bismuth as a coolant for nuclear reactors in Russia", MIT ATW Technical Review, 1998, Jan. 15-16.

[22] Beller, D. E., Sailor, W. C., Venneri, F., and Herring, J. S., "A Closed, Proliferation-Resistant Fuel Cycle with Th-U-O_2-Fueled LWRs, Th, U, and Np Recycle, and Accelerator-Driven Transmutation of Waste (ATW)", Global '99: International Conference on Future Nuclear Systems, Nuclear Technology - Bridging the Millennia, Jackson, WY, Aug 29-Sep 3, 1999.

heat), enhanced with a mixed urania-thoria fuel cycle[23], might alleviate this problem. A significant fraction of this isotope in the plutonium mixture is in fact considered as making it proliferation resistant, i.e., unusable for military applications.

Single stratum scenario

If the second stratum would be constituted only by critical fast reactors, mainly devoted to MA incineration, a single stratum strategy might be envisaged in which these systems might play the role of burners of MA together with the plutonium produced in LWR, along with the same old IFR logic.

Long term scenario

As already mentioned, although a quite positive experience with LWR reactors in the last decades cannot be forgotten, it is widely perceived that a broader, world-wide nuclear energy expansion would pose a number of concerns related to the safety (in particular, power excursions, residual heat risk), as well as to those associated with the fuel flow (criticality accidents, fuel diversion, radiological risk, proliferation).

Moreover, the double strata scenario described above, although answering many crucial issues, leaves unanswered the last one mentioned in the Introduction. With such a scenario, in fact, the present nuclear energy production structure (reactor park and reprocessing plants) is conserved as it is. A long term, adequate equilibrium state, prerequisite to a third reduction phase, appears controversial and rather undefined.

Speculations can be made in relation to a long term (Single Stratum) scenario responding to these issues. The basic logic would consist in a gradual shifting at a certain point from the uranium-plutoniom cycle into the thorium-uranium one. Two main options may be envisaged:

1. The first one would consist of advanced (in terms of safety and performance) fast reactor plants, with fixed fuel elements, viewed as an evolution of the IFR (Na, or lead cooled) concept. This option may encounter opposition for to alleged possibility of criticality accidents (even if these occurrence can be reduced to a quite remote probability event). An ADS fast reactor version might be envisaged, if all the issues related to the accelerator performance (in particular, the current interruption issue) will be reduced. Concerns related with proliferation issues would need also to be specifically addressed.

2. The second one would consist of the so called mobile fuel reactors (MFR). In this case two reactor concepts may to be considered as possible candidates:

- The molten salt reactor (MSR);
- The pebble bed reactor (PBR).

Adopting mobile fuel, a quasi-constant multiplication factor is maintained. With this second option, safety and proliferation issues appear better addressed.

[23] K.D. Weaver et al., "Analysis of Thorium-Uranium Fuel Using MOCUP and SCALE", Trans. ANS, 81, 61 (1999).

An evolutionary strategy may be considered in above long term scenario, in the sense that it can be adapted to each of the three phases (expansion, steady state, reduction) considered and adequately responding to the objectives outlined above.

Within such strategy, to allow a transition to the third, reduction phase, the use of the thorium cycle would be required, for the advantages associated with it.

MFR

As mentioned before, one of the main advantages with the (ADS, or critical) MFR systems is the possibility of drastically reducing the need of a reactivity (or current, in case of an ADS-like concept) reserve during operation and burnup. This removes, in particular, the risk of a reactivity accident consequent to an inadvertent control rod extraction.

Critical MSR and PBR designs are generally well known, although there are different versions.

As far as MSR systems are concerned, we may recall in particular, the TASSE concept[24] (a modification of the MSR thermal design proposed by Bowman[25]). With this concept it would be possible to avoid, in an energy expanding phase (doubling time of the order of 20÷30 years), the (economically penalizing) fuel reprocessing step, the same fuel being recycled in a feed and bleed mode. The bleed fuel at quasi-equilibrium composition (in the expanding phase) would be used for feeding, when the doubling of the necessary mass is achieved, a new system.

As far as the PBR is concerned, we remind the old German (thermal) reactor project.

As belonging to the MFR family, the PBR has the desirable property, mentioned above, of maintaining the multiplication factor quasi-constant during the reactor life. Small operation changes could be obtained by proper power adjustments of the coolant pump, exploiting the large (negative) temperature coefficients which characterize these systems. An important feature of the PBR design is its operation at very high temperatures (~ 850°C), which allows:

- on one side, to reach very high thermal efficiencies (of the order of 50%);
- on the other side, to consider these systems also for the production of process heat (for instance, for hydrogen production).

The same feed an bleed logic of the TASSE concept might be applied also to the PBR.

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[24] I., Slessarev, M. Salvatores, V. Berthou, "Concept of Thorium Fueled Subcritical System for Energy Production and TRU Incineration without Wastes (TASSE)", 3rd International Conference on Accelerator - Driven TransmutationTechnologies and Applications, Praha (Pruhonice), 7-11 June 1999.

[25] 1. C. D. Bowman, et. al., "Nuclear Energy Generation and Waste Transmutation Using an Accelerator-Driven Intense Thermal Neutron Source," Nuclear Instruments and Methods A320, 336-367 (1992)

The potentiality of the PBR systems, also with respect to their possible role as ADS, may be also inferred from the results of a recent work by X. Raepsaet et al.[26]. In their paper it is demonstrated in fact that in a HTR of the GA type with fixed fuel elements, after 500 GWd/t, a K_∞ value is reached of the order of 1.01 with 66% PU+MA fissile (from PWR). In a corresponding PBR at equilibrium conditions with similar average burnup (and then similar K_∞) the oldest pebbles (ready for been definitely discharged) would have a burnup of the order of 80%. For a large, and then low leakage, reactor, possibly with an external neutron source, it would be possible, in principle, to adopt this concept for an effective actinide burner with a relatively reduced content of MA.

MFR reactors may be viewed as critical, as well as ADS systems. The choice of ADS, rather than a critical concept, would be finally dictated merely by safety considerations, or, for the MFR concepts, for operating, at equilibrium, with less enriched fuel feed.

Transition, equilibrium and phasing out

In above long term scenario, a transition lasting several decades would be foreseen during which a gradual substitution of present reactors plants are replaced by these new concepts (IFR-like advanced fast reactors with fixed fuel elements, or MFR's). During this (presumably, expanding) phase, these reactors should be (at least partially) fed by existing stocks of spent fuel from old generation reactors, and/or by military plutonium. At the end, an equilibrium condition would be eventually reached in which the (average) feed fuel isotopic distribution would be quasi-constant in time.

At this equilibrium phase, the transition into, or the inception of a U-Th cycle would take place. The two cycles (U-Pu and U-Th) could coexist for an indefinite time. This gradual transition to the U-Th cycle would allow, on one side, to optimize the exploitation of natural resources and, on the other side, to minimize the long term radiological risk.

When the time arrives to start the phasing out of the fissile cycle (presumably, for the advent of a new, more advantageous source of bulk energy), a definite transition from the U-Pu cycle into the U-Th one should be established. During this phasing-out period, major quantities of TRU would be destroyed, until, at equilibrium, the fresh feed would consist only of thorium plus, possibly, some fissile (U^{235}), to upgrade the fuel. The amount of added fissile would depend on the system concept. In case of ADS, such fission content could be significantly reduced by an appropriate increase of the subcriticality, this in turn implying an increasing fraction of the produced energy for feeding the accelerator.

At the end of the fission era, a negligible quantity of fissile material would remain, in terms of change of the overall radioactivity preexistent the fission energy exploitation.

[26] X. Raepsaet et al. "Fuel Cycle Performances in High Temperature Reactor", Global '99: International Conference on Future Nuclear Systems, Nuclear Technology - Bridging the Millennia, Jackson, WY, Aug 29-Sep 3, 1999.