

WASTE MANAGEMENT CONCEPT FOR FUTURE LARGE-SCALE NUCLEAR POWER

**I. Ganev, G. Khacheresov , A. Lopatkin, V. Naumov, V. Orlov, V. Smirnov,
L. Tochenyi - RDIPE, Russia**

ABSTRACT

The concept of radwaste management is discussed as part of the general concept of a naturally safe nuclear technology for the large-scale power industry. Deterministic exclusion of reactor accidents with fuel failure allows burning MA within the main fuel, as well as I and Tc. On attaining a certain efficiency of radiochemical separation of actinoids and FPs, it becomes possible to bury radwastes after their long-term cooling, without upsetting the natural radiation level. The paper presents the results of studies performed by RDIPE, covering the reactor proper, the radioactivity of fuel and wastes, the storage and disposal of radwastes, and the utilization of Sr and Cs.

I. INTRODUCTION

Along with evolutionary improvement of traditional nuclear technologies in the next few decades, nuclear power also needs a definite concept of its long-term development, the absence of which deprives it of a clear perspective and of major arguments in its favour. The possibility to arrest the growth and then to reduce the consumption of conventional fuels has from the outset been a major incentive to research and developments on peaceful nuclear energy. The global fuel problems came to be resolved in this century without great involvement of nuclear fuel, and forecasts up to the year 2020 show its share in the total fuel resources consumed to remain at about 5 %.

However, the alarming estimates of the available resources, such as oil and gas, and the looming environmental and international problems which are bound to be caused by the inevitable increase of the global population and fuel demands in the next century, make us go back to the concept of large-scale nuclear power, since other energy technologies still fall short of offering a realistic alternative to the traditional power engineering. But for nuclear power to be accepted as a conventional technology for mass application, it is essential to develop a nuclear technology which would fully take into account the lessons of the first stage as well as the emerging new conditions and new requirements to be met.

Of primary importance among the latter is the exclusion of reactor accidents with catastrophic radioactive releases. Their estimated probabilities are quite acceptable for predicting safe short-term development which will not go too far away from the accumulated experience of about 6×10^3 reactor-years. However, such extrapolation of experience to $\sim 10^6$ reactor-years, which may be reached in the century to come, is not convincing due to the use of the estimated event probabilities much smaller than 10^{-6} , which usually have no experimental or trustworthy theoretical basis to support them. One cannot agree with A. Weinberg in that the "Second Nuclear Era" calls for a new nuclear technology capable of deterministically excluding catastrophic accidents by virtue of natural laws which, unlike engineered systems and barriers, are entirely reliable.

But a too expensive technology, even if quite safe, has no prospects for large-scale application in the energy sector where economic criteria are of equal fundamental importance. The main reasons for the multiple increase in the cost of NPPs lie in the growing complexity of design, requirements to equipment and personnel, and licensing procedures, resulting from the mounting safety requirements. The utmost safety attained due to technical features protected from catastrophic accidents by their inherent properties and natural laws rather than through building up engineered systems and requirements, can be combined with simplification of design and requirements, and reduction of NPP costs.

Another factor of fundamental importance for economy consists in economical consumption of natural uranium which, with the present-day LWRs having operated for 10^6 reactor-years, would amount to about 200 million t - a figure 20-40 times as large as the currently expected or explored U resources of acceptable cost. In order to keep

consumption within their limits and to avoid considerable expansion of the uranium mining industry, the new technology should ensure a reduction in the specific consumption of uranium by at least an order of magnitude.

Fast reactors with liquid metal cooling, in our opinion, have the greatest potential for coping with this problem, as well as with other challenges of a large-scale power industry (safety, wastes, economy), although some types of thermal converters in the Th-U cycle are also acceptable in terms of the fuel balance. But modern fast reactors proved to be much more expensive than LWRs despite the greater fuel and energy efficiency and simplicity of design and control principles. This may be accounted for by the choice of the highly reactive Na made at an early stage, which results in a more complicated design of the reactor, cooling and refuelling systems, SG, structures, etc.

The chemical activity and insufficient boiling margin of Na do not fit in with the requirements of inherent safety as well as the low density and heat conductivity of the oxide fuel. We see the new reactor technology as a fast reactor with a chemically inert high-boiling coolant (Pb) and a high-density, heat-conducting, heat- and radiation-resistant fuel (UN-PuN) with good radionuclide retention, efficient self-regulation by feedbacks, high natural circulation of coolant, small reactivity margin, etc.

The principles of deterministic (inherent) safety should be extended to include other links of the closed fuel cycle, primarily the final disposal of radwastes, due to the questionable reliability of engineering, geological and "historical" forecasts for tens of thousands of years ahead. Fission of all actinoids in fast reactors, transmutation of the most long-lived FPs, and gradual radioactive decay of the remaining products are fundamental physical processes which reduce the hazard of the buried wastes to that of U extracted from the earth together with its α -active decay products. The technological arrangement based on the above can be convincingly proved to be safe. The main prerequisites for implementing such an arrangement include elimination of reactor accidents with fuel failure and fractionation of the burnt-out fuel composition into U, Pu and MA for fresh fuel fabrication, I and Tc for in-pile transmutation, Sr and Cs for utilization, and radwastes to be cooled and buried.

The risk of proliferation of nuclear weapons has turned into an acute problem which is barring adoption of the concept of large-scale global nuclear power with fast reactors, closed cycle and Pu. In conjunction with political safeguards, nuclear

technology has great, as yet untapped potentialities for ruling out thefts of Pu and its military uses. Development of a nuclear technology - economical, safe and protected against Pu thefts - would be an economically and strategically more effective application for funds and efforts, including efforts of nuclear weapon experts, than would be the development of special Pu burners and technologies for its vitrification and disposal.

After Chernobyl, we undertook a search for and studies on a new nuclear technology which would meet the varied requirements of a large-scale power industry, without going too far away from what has already been mastered by nuclear engineering.

While sharing A. Weinberg's opinion that creation of a deterministically safe nuclear technology is worth the efforts commensurate to those spent already, we, nevertheless, see possibilities for developing and demonstrating such a technology within a limited stretch of some 15-20 years, based on the experience amassed in peaceful and military nuclear engineering^(*).

In team with other institutes, RDIPE has carried out conceptual development work for a lead-cooled fast reactor with UN-PuN-MA fuel, designed to operate in a closed transmutation fuel cycle. Consideration has been also given to an option proposed by C. Rubbia - a slightly subcritical reactor with a cyclic proton accelerator as a neutron source - which offers convincing proof of safety with respect to reactivity accidents. The results are discussed above, with emphasis laid on the problem of radwastes. Investigations performed in many countries in the recent years have clarified the concept of the transmutation technology in terms of its physics, for which 2- or 3-fold uncertainties in the radiation characteristics have no critical implications. We shall try and describe it here in fairly simple terms, without going into design details. Its engineering development is our main objective today.

^(*) This is a normal time scale for nuclear developments. Such time, for instance, was taken to develop and bring in the BN-350, BN-600 and other reactors. Inordinately protracted creation of a commercial fast reactor must have been caused, in our opinion, by the original choice of technical features (Na, oxide fuel) which cannot give effect to the potential advantages of fast reactors in terms of either safety or economic efficiency.

It stands to reason that the start of a new stage in nuclear power development around new reactors should be tied in with solving an urgent problem of the next few decades, which consists in utilizing the Pu accumulated in storages as a result of reduction of nuclear armaments and fuel reprocessing. Several reactors of the type in question are capable of converting within reasonable time the Pu accumulated in Russia into the theft-proof form of spent fuel, thus following the CISAC recommendations, while simultaneously gaining experience for subsequent large-scale deployment.

II. LEAD-COOLED FAST REACTOR

Two-circuit 300, 600 and 1000 MWe options of lead-cooled fast reactor were studied. Main characteristics of these reactors are given in Table 1 and discussed in detail in the reports presented at the ARS-94 Conference (Pittsburgh, 1994).

300 MWe was found to be the lowest power level at which it proved possible to use nitride fuel, do without uranium blanket and ensure $CBR \sim 1$ and total reactivity margin ΔK to $t < 1$ \$.

The feature of low neutron moderation and absorption in lead allowed the following :

- increase lead inventory in the core from ~ 40 % (Na) to ~ 60 % ;
- reduce lead heating to 120°C and maximum flow rate to 1.8 m/s ;
- reduce pumping power to less than 1 % of reactor power ;
- provide high level of natural circulation more than 15 % of nominal circulation.

Large pitch in the fuel lattice makes it possible to use unducted fuel assemblies (FA) and hence avoid local loss of FA cooling.

Lead temperature at the core inlet/outlet (420°C and 540°C respectively) ensures the required margin to lead freezing (327°C) and acceptable operating conditions for fuel claddings, reactor vessel, steam generators and pumps. To avoid lead freezing under abnormal conditions, it was decided to use supercritical steam parameters with the feedwater temperature of 340°C and a jet mixer and steam pump drive in the secondary circuit. These features result in the thermal efficiency of ~ 44 %.

Besides, the use of high-density heat-conductive nitride fuel in combination with lead coolant allows :

- * - to limit average operating fuel temperature to 700°C,
- to confine most of the fission products (FP),
- to relieve stress on fuel claddings ;

- * - to minimize power reactivity coefficient while preserving the temperature coefficient ;
- to simplify control system and avoid fast runaway under all accidents in the system.

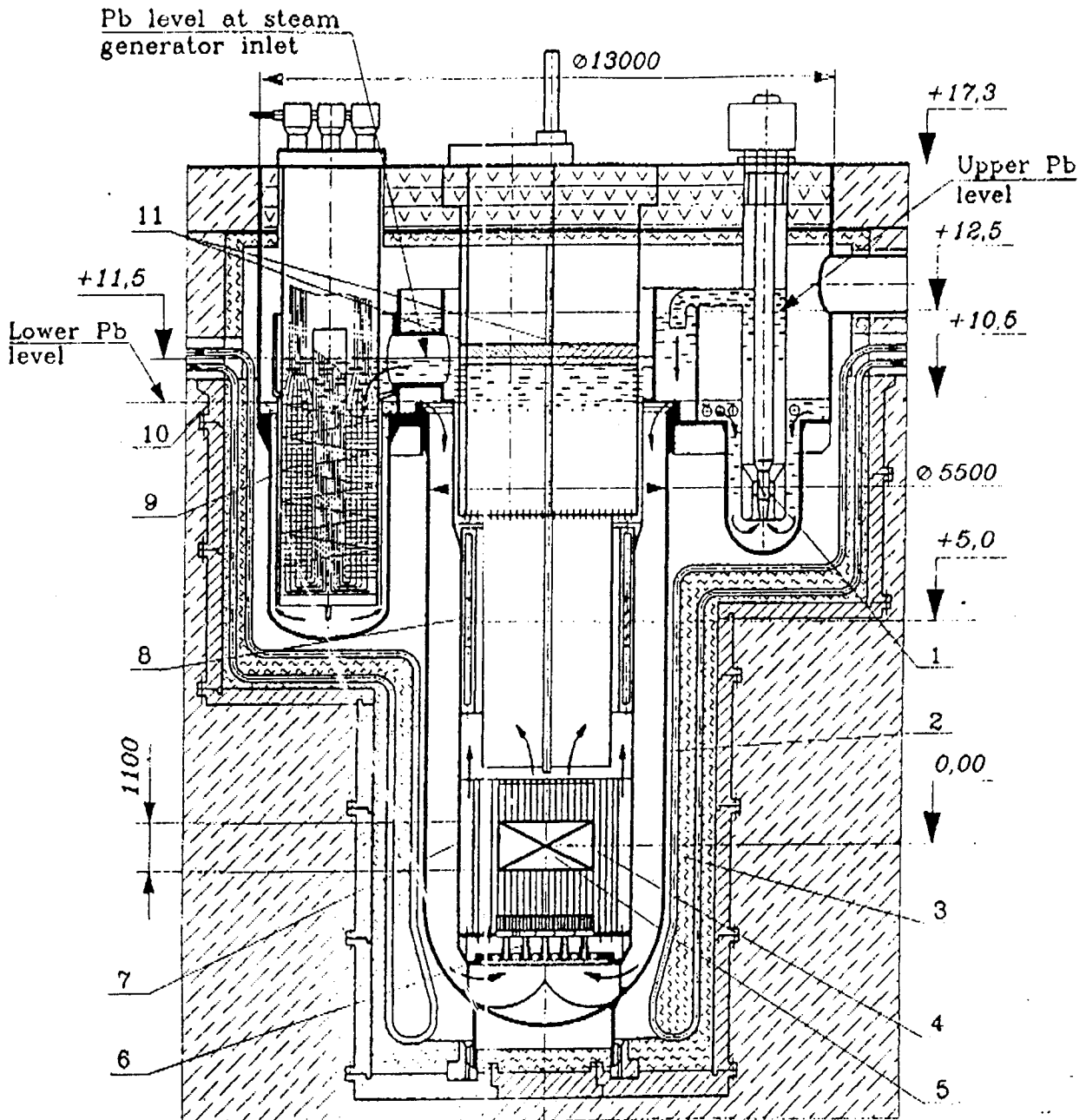
The absence of traditional uranium blanket prevents production of weapon-grade plutonium. Void reactivity effect becomes deeply negative in case of lead draining and is $< 1 \text{ \$}$ in case of steam or gas bubbles injection in the core. Taking into account high boiling temperature of lead, this prevents dangerous local positive void reactivity effect.

Control means are located outside the core. They operate based on simple reactivity control principles consisting in varying neutron leakage, which simplifies refuelling as well.

The reactor has an integral layout of the primary circuit (Fig. 1). Refuelling is performed via an in-core storage pool with the help of two rotating plugs. There is no need to wash removed FA during refuelling.

A single reactor vessel is located in a reinforced concrete vault 3 m thick. The gap between the walls of the vessel and the vault prevents loss of lead flow in case of vessel failure and partial loss of lead. Long-term emergency core cooling is based on natural air circulation in the vault.

As estimated, design simplification and natural safety features make the cost of such a plant comparable to that of LWR and even lower.



- | | |
|---------------------|---------------------|
| 1-pump | 7-separating shell |
| 2-vessel | 8-fuel storage |
| 3-thermal shielding | 9-steam generator |
| 4-CPS | 10-safeguard vessel |
| 5-core | 11-rotary plug |
| 6-support posts | |

Figure 1 - BREST-300 layout

Extreme accidents including vessel/containment failure, maximum reactivity insertion, pump trips, break in the secondary circuit etc. without scram will not result in fuel failure with catastrophic radioactive releases (over 10^4 Ci in 1 equivalent).

Fuel performance was calculated for equilibrium isotope composition obtained as a result of multiple recycling, with fission products extracted in each cycle and produced actinoids returned in the reactor for burning.

The calculations show that some 0.3 neutron in a fission is absorbed outside the core. These neutrons can be used for example for transmutation of long-lived FPs such as I-129 and Tc-99, which are released in total ~ 0.07 atoms per fission. The channels with long-lived FPs recovered during reprocessing are placed in lead reflector. Special channels with Sr and Cs are planned to be installed in the reactor as stable heat sources with the power of ~ 0.02 % N nom.

Studies are under way on the option of a subcritical reactor with a neutron source created in lead by a beam of protons injected by a cyclic accelerator.

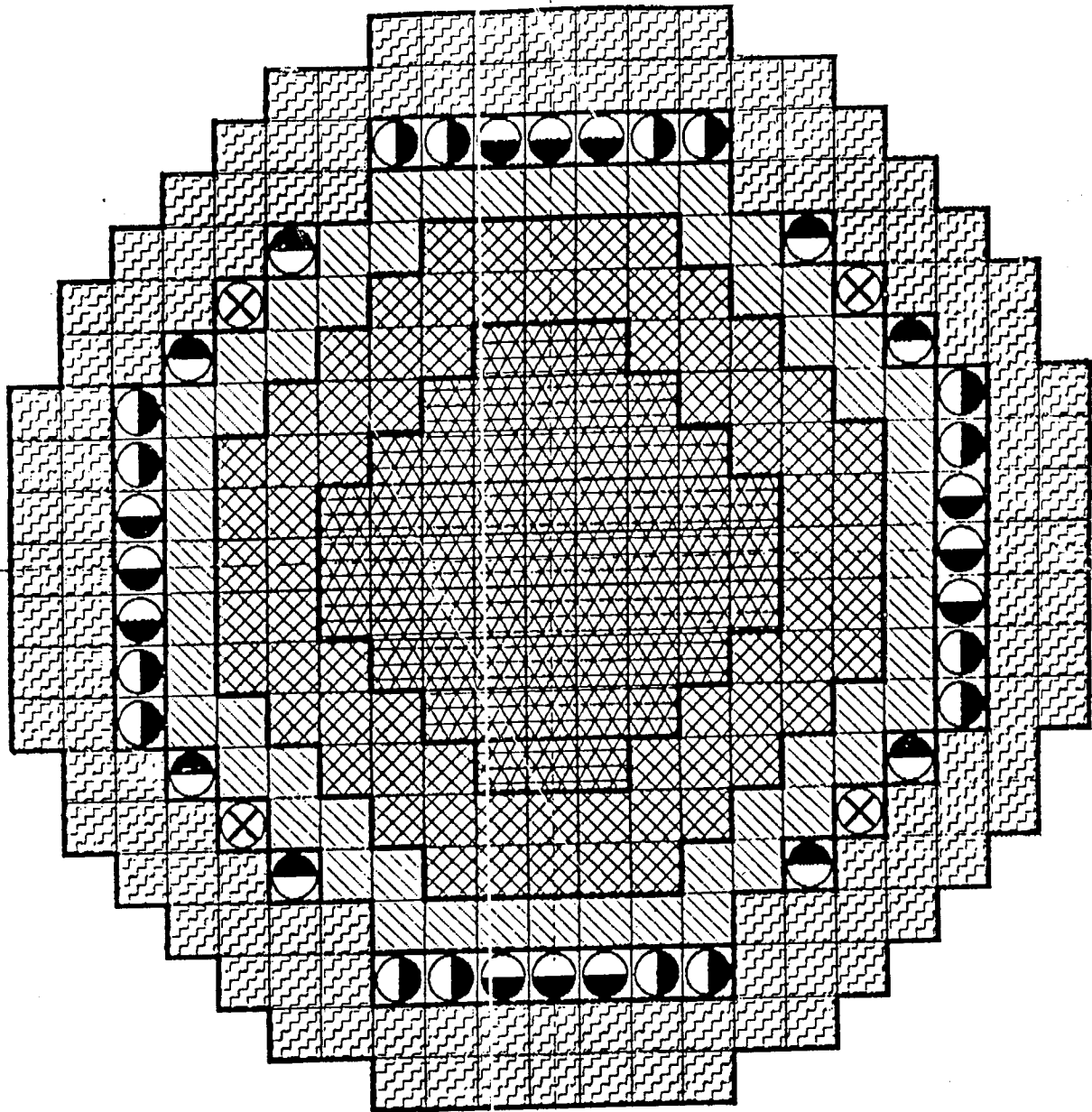
The feature of ~ 0.02 $\Delta K/K$ subcriticality with CBR ~ 1 allows to do without reactivity control means while avoiding reactor runaway both under credible and hypothetical accidents.

If accelerator adds about 10 % to the cost of the plant then such design could be economic acceptable.

The proton beam is injected vertically in an axial channel in which the lead level is regulated by gas pressure. With :

- proton energy 1 GeV,
- average current 7 mA,
- proton-neutron multiplication factor ~ 20 ,
- neutron importance in the core center ~ 1.5 ,
- subcriticality 0.02 $\Delta K/K$,

the thermal power of the reactor will be close to 700 MW, with thermal and neutronic characteristics close to these of BREST-300.











- | | | | |
|-------------------------------------------------------------------------------------|--------------------------------|-------------------------------------------------------------------------------------|-----------------------|
|  | -FA ($d_{fr}=9.1\text{mm}$) |  | -scram rod |
|  | -FA ($d_{fr}=9.6\text{mm}$) |  | -automatic controller |
|  | -FA ($d_{fr}=10.4\text{mm}$) |  | -scram rod |
|  | -lead reflector unit |  | -shim rod |

Figure 2 - Cross-section of BREST core

MAIN CHARACTERISTICS OF REACTORS

Characteristic	BREST-600	BREST-300
Electric power, MWe	600	300
Number of FA in the core, pcs.	357	185
Core diameter, mm	3190	2300
Core height, mm	1200	1100
Length of a fuel rod, mm	9.1 ; 9.6 ; 10.4	9.1 ; 9.6 ; 10.4
Lattice pitch, mm	13.6	13.6
Fuel	UN + PuN + MA	UN + PuN + MA
(U+Pu)N load, t	28	16
Pu/Pu-239 + Pu-241 load, t	3.73/2.72	2.2/1.6
Fuel life, years	5-6	5
Time between refuellings, years	1-2	1
CBR	~ 1	~ 1
Inlet/outlet lead temperature, °C	420/540	420/540
Maximum cladding temperature, °C	650	650
Maximum lead flow, m/s	1.8	1.8
Power effect, % $\Delta K/K$	0.17	0.16
Total reactivity margin, % $\Delta K/K$	0.33	0.32
β eff, %	0.35	0.35
Steam temperature at SG outlet, °C	520	520
Steam pressure at SG outlet, MPa	24.5	24.5
Thermal efficiency, %	~ 44	~ 44

III. SCENARIO OF NUCLEAR POWER DEVELOPMENT IN RUSSIA

It was assumed that after 2010 the growing demand for electricity production will be met mainly due to construction of large NPPs with the discussed reactors (up to 150 GWe by 2050 and 300 GWe by 2100) with an on-site closed transmutation fuel cycle. Besides, small nuclear plants with thermal reactors will be constructed for local electricity supply and district heating, such as LWR, HTGR, etc. (equivalent to 50 GWe). Spent fuel from these reactors will be reprocessed at central reprocessing plants, with recovered U returned to thermal reactors and Pu and MA recycled in fast reactors (together with weapon-grade Pu stockpiled as a result of arms reduction). As the growth rate slows down, thermal reactors will gradually be converted to mixed U-Pu fuel due to certain overproduction in fast reactors. Another open option is

turning to Th-U cycle. After 2100 the growth of nuclear capacity and natural Uranium consumption is halted and over T years nuclear energy remains at the level of 350 GWe, consuming only stockpiled depleted uranium (which will be sufficient for $T \sim 3$ K years). The stable period might last longer, then it may prove necessary to resume uranium production at a minor level (some 400 t per year) in case $T > 3$ K years.

About 900 K tU (Mu) will be produced up to 2100 (including already mined uranium).

The stable period T may prove much shorter if some novel energy technology, preferable in this or that respect, is developed. In this case, T might be followed by a fairly long final period of a step-by-step decommissioning of NPPs and transmutation of radioactive materials. Dedicated reactors (burners) may prove necessary to allow completing this stage in 100 years.

Assuming that 1 GWe entails 0.8 t FP per year, nuclear power of $N(t)$ GWe will produce annually $0.8 \times N(t)$ tFP/y, with total production up to T : $18(1+0.016 T)K$ t. Consumption of natural uranium per 1 tFP (currently amounting to about 200 tU/tFP) will decrease by 2100 to 50 tU/tFP and to 1 tU/tFP in ~ 3000 years.

Calculations were performed based on the VVER-1000 fuel performance as representing thermal reactors. For convenience, estimates below refer only to fast reactor fuel.

IV. FUEL COMPOSITION AND IRRADIATION CHARACTERISTICS

If MA is used in the main LCFR fuel, in 10-15 cycles the fuel composition will be close to equilibrium as summarized in Table 1 (15 cycles, stored during 1 year, % at.).

Calculations show that Cm transmutation drastically increases radiation level and heat release during fuel fabrication. Hence the Cm fraction ($\epsilon \sim 1$ %) should better be extracted and placed into storage so that Cm-242 and Cm-244 would decay.

Table 1

Th- 230	U- 232	U- 238	Np- 237	Pu- 236	Pu- 238	Pu- 239	Pu- 240	Pu- 241	Pu- 242	Am 241	Am 243	Cm 244	Cf- 249	FP
2	3.2	74	0.36	6	0.55	9.5	5	0.72	0.56	0.32	0.2	0.11	8.7	8.7
(-5)	(-5)			(-6)										(-5)

Table 2 describes irradiation characteristics of the fuel at the fabrication stage under two options : transmutation cycle with MA (without Cm) recycling and traditional cycle without MA recycling (γ and n radioactivity at 1 m distance from FA).

Table 2

	α , Ci/kg	q, W/kg	γ , $\mu\text{rem/s}$	n, $\mu\text{rem/s}$
U-Pu-MA	72	2.3	45	16
U-Pu	28	0.9	14	7

Adoption of the transmutation fuel cycle entails higher heat generation and radiation exposure which can be compensated by protection and engineered measures.

V. FUEL REPROCESSING AND FRACTIONATION

Development effort for the IFR concept (ANL, INEL) revealed that small on-site reprocessing should preferably be based on the electrochemical technology of fuel fractionation and species extraction from molten salts. Such studies are currently under way also in Russia (NIIAR) and Japan (JAERI). VNINM studied applicability of electrochemical technology for the LCFR nitride fuel, including laboratory experiments.

Calculations were performed for the following loss levels. Promising values are given in brackets ; on the right-fraction management.

P, Pu - 0.1 % (0.01 %), MA - 1 % (0.1 %). Fabrication of new fuel
 Sr - 5 %, Cs - 15 %. Utilization, sources of heat and γ radiation
 I, Tc - 10 % (1 %). Transmutation in the reactor

Radwastes are compacted and stored on site for 200 years.

Transmutation of I-129 and Tc-99 was studied for a 300 MWe reactor. Channels with I (0.4 g/cm^3) and Tc (5 g/cm^3) are located in lead reflector in the second row. Calculations taking into account neutron flux depression and resonance self-shielding showed that 90 % of Tc-99 atoms burn up in 15 years, I-129 atoms - in 23 years.

Equilibrium Sr-90 and Cs-137 amount to 205 kg ($5.6 \times 10^7 \text{ Ci}$, 187 kW) and 913 kg ($1.5 \times 10^8 \text{ Ci}$, 384 kW) per 1 GWe. They are placed in steel channels, probably together with I and Tc, to create a steady heat source in lead (about 600 kW, 0.025 % of the nominal power) to prevent its freezing during long outages.

Part of Cs can be used for setting up an on-site irradiation processing shop.

As radwastes are stockpiled, a RW storage will be constructed on site phase-by-phase over 200 years. Phase I should accommodate over the initial 50 years approximately 40 t FP (based on 1 GWe) and should be designed to provide heat removal of about 100 kW. Cm is located in special channels.

FPs in metal matrix with the concentration of $\sim 3 \times 10^3 \text{ Ci/l}$ and heat generation up to 20 W are placed in steel tubes $D * H = 0.2 * 5 \text{ m}$. 120 such tubes are located in a concrete vault (30 cu. m) cooled by natural closed circulation of dry air heated to $\sim 150^\circ\text{C}$, with heat removed in air-air and air-water heat exchangers. The storage volume is $\sim 600 \text{ m}^3$, estimated cost - less than 1 % of the plant cost.

The absence of U blanket and production of weapon-grade Pu, low reactivity margins, U-Pu-MA fuel, on-site fuel cycle prevent Pu thefts and its military application. Other security measures are planned, which together with political measures will ensure non-proliferation.

VI. RADIATION EQUIVALENT RADWASTE DISPOSAL

Uranium production is planned to be carried out with co-extraction of long-lived α active decay products (Th, Pa, Ra etc.) transmuted in reactors. This will contribute to recultivation of uranium mining areas while "giving space" for subsequent RW disposal in naturally radioactive formations.

Radioactivity of 1 t of natural uranium together with its decay products is worth 5.2 Ci. Mass of U equivalent in terms of radiation risk to the amount of radionuclide X (Λ_x Ci/tFP) accounting for 1 t of FP is :

$$m_x^0 = \frac{A_x \cdot \frac{Ci}{t_{FP}}}{5.2 \cdot \frac{Ci}{t_{uCi}} K_z^x} \left[\frac{t_u}{t_{FP}} \right],$$

where K_z^x is ratio of permissible concentrations (in Ci) in water with radionuclide X and uranium. Recently Annual Limits of Intake (ALI) were used in calculations. After reprocessing (ϵ_x is the portion of radionuclide X remaining in RW) and storage over time τ the mass of uranium equivalent to disposed RW is :

$$m_x^1 = \frac{m_x^0 \cdot \epsilon_x \cdot f_x(\tau)}{K_m^x} \left[\frac{t_u}{t_{FP}} \right],$$

where K_m^x is the factor accounting for the difference in radionuclide X and uranium migration from repository to the surface.

Short-lived radionuclides with $T_{1/2} \leq 100$ years :

Sr (29 years), Cs-137 (30 years), Sm-151 (90 years), U-235 (72 years) etc. are stored in the repository in equilibrium amounts.

$$M_x = \frac{0.8 \cdot m_x^1 \cdot N(t) \frac{t_{FP}}{\text{year}}}{\lambda_x} [t_u],$$

where λ_x is decay constant.

The time of RW storage was taken as $\tau = 200$ years since over this period the amounts of Sr-90 and Cs-137 are reduced by a factor of two orders, while the further storage is ineffective due to slower decay of Sm-151 and actinoids.

Radionuclide migration is the most uncertain factor. The data show that most radionuclides (except for I) feature much greater retention as compared with U, however, it is difficult to account for various disturbances, both natural and man-made. With short-lived radionuclides, however, stable physical and chemical forms given to RW before disposal could reasonably be expected to hold for the whole of their life and hence $K_m \sim 10$ can be taken as a minimum estimate. Maximum estimated equivalent masses of short-lived nuclides are given below for $N = 350$ GWe.

Table 3

Nuclide	Sr-90	Cs-137	Sm-151	U-232
$m_x^0 \frac{t_u}{t_{FP}}$	$2.4 \cdot 10^5$	$6.7 \cdot 10^4$	$1.1 \cdot 10^3$	$1.6 \cdot 10^3$
$\epsilon, \%$	5	15	100	0.1 100
M_x, tU	$1.6 \cdot 10^5$	$1.2 \cdot 10^5$	$7 \cdot 10^5$	$600 \cdot 6 \cdot 10^5$

Hence, the terms of radiation equivalent disposal of short-lived portion of radwastes adopted in the assumptions are fairly closely satisfied.

Long-lived I-129, Tc-99 and Cs-135 contribute rather insignificantly to the radioactivity of the disposed RW even under long-term development of nuclear power with uranium consumption at the level of 1 tU/tFP.

Table 4

Nuclide	I-129	Tc-99	Cs-135
$\epsilon, \%$	10	10	100
$m_x, tU/tFP$	1	0.1	0.08

It was adopted in the calculations that $K_I = 0.3$, $K_{Tc} = K_{Cs} = 1$. K_r was taken based on permissible concentration in water. Calculations based on ALI show significant reduction of m_I and increase of m_{Tc} .

If stored for 200 years, actinoid fraction of RW will be reduced in terms of equivalent mass of U only five-fold; the level of RW treatment (ϵ) is a major importance.

Assuming for long-lived nuclides $K_m = 1$, equivalent mass of U for actinoids amounts to : 91-180 tU/tFP with $\epsilon = 0.1\%$ and 9.1-18 tU/tFP with $\epsilon = 0.01$, as calculated based on permissible concentration in water and ALI. If $\epsilon = 0.1\%$ the results fall by a factor of 2-3 beyond the radiation equivalency, even with $T = 0$ (specific U consumption ~ 50 tU/tFP), which is within the uncertainty of the present estimates. We see that long term nuclear development will strongly depend on improvements of reprocessing technology.

VII. THE FINAL PHASE

On this stage the use of U-Pu fuel (reactors with Pu reproduction) delays the termination of nuclear power.

Accelerated termination of nuclear power is possible due to disposal of U, introduction of burners operating mainly on MA (for instance, fast reactors with liquid fuel or, probably, accelerator burners). This allows to reduce the final phase to 100 years or even shorter.

VIII. COMPARISON WITH THE Th-U CYCLE

Drastically smaller production of MA as compared with the U-Pu cycle is considered as a major argument for adopting Th-U cycle in the future. Since this cycle shows advantages over U-Pu in its neutron and fuel balance only in case of thermal reactors (with fast reactors it is much inferior to U-Pu cycle), fast reactors with a transmutation fuel cycle were compared with heavy water reactors with a closed Th-U fuel cycle with and without MA transmutation under recycling.

As follows from Fig. 3, in terms of waste radioactivity the Th-U cycle is inferior to transmutation U-Pu cycle of anyway, is not superior to it. Since in the Th-U cycle thermal reactors are inferior to fast reactors in many aspects such as neutron and fuel balance, many safety features (reactivity margin, β_{eff} etc ...), burnup and reprocessing volume, fast reactors are preferable for large NPPs provided they become cost effective.

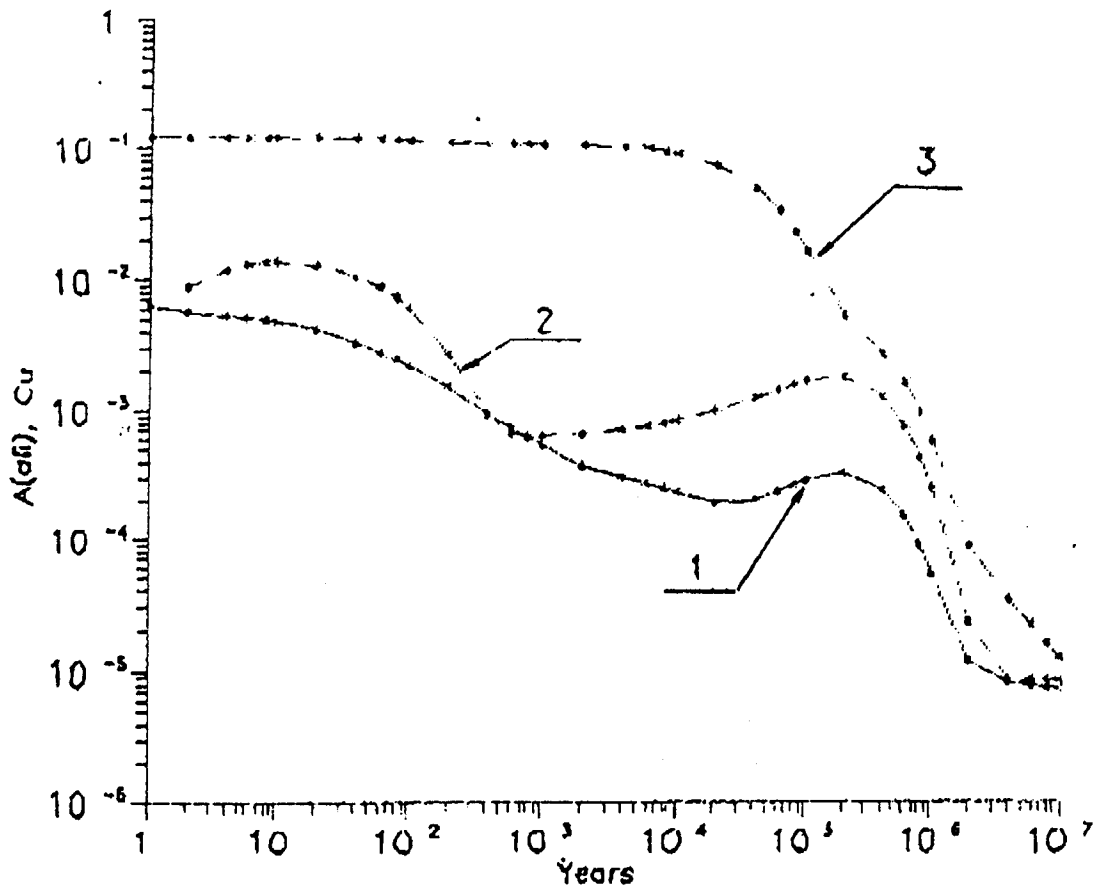


Figure 3
 Actinoide waste activity per 1 MW.day (ALI equivalent)
 from LCFR (U-Pu) [1] and CANDU (Th-U) [2] P/T fuel cycle
 and CANDU (TH-U) cycle without transmutation [3]

At the same time fast reactor loses some of its advantages in case of small power, hence thermal reactors of different types may prove preferable for local heating and electricity production. For them Th-U cycle shows certain advantages.

IX. CONCLUSION

Analysis of the concept of the U-Pu fuel cycle with transmutation in fast reactors in terms of physics points to the feasibility of radiation-equivalent radwaste disposal without upsetting the natural radiation levels, provided a number of technical problems are solved.

These primarily include creation of a reactor which would rule out accidents with fuel failure and catastrophic radioactive releases.

A major prerequisite is the development of a radiochemical technology capable of removing Pu and MA from radwastes with a residue of no more than 0.1 % and of isolating the fractions of I and Tc (< 10 %) and Sr and Cs (5-15 %). Cm should be advisably separated from the fuel for long-term cooling in a storage. If nuclear power stays in service for several centuries, still finer fractionation will be needed. Utilization of Sr and Cs, and maybe other radionuclides, will make it possible to reduce the time of radwaste cooling to an acceptable stretch of about 200 years.

Both rehabilitation of U mining sites and radiation-equivalent disposal of radwastes call for mastering the technology of U mining with co-extraction of its long-lived decay products (Th, Pa, Ra, etc ...).

There is still much time left for developing a new technology of radwaste disposal, but the possibilities of creating compositions to prevent migration of radionuclides from burial grounds need investigation and demonstration.