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COMPARISON OF SODIUM AND LEAD-COOLED FAST REACTORS REGARDING SEVERE SAFETY AND ECONOMICAL ISSUES

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ABSTRACT

A large number of new fast reactors may be needed earlier than foreseen in the Generation IV plans. According to the Special Report on Emission Scenarios commissioned by the Intergovernmental Panel on Climate Control nuclear power will increase by a factor of four by 2050. The drivers for this boost are the increasing energy demand in developing countries, energy security, but also climate concerns. However, if one stays with a once-through cycle the amount of high-level nuclear waste will increase substantially and there will be an upward pressure on the price of uranium in the next few decades. Therefore, it appears wise to accelerate the development of fast reactors and efficient re-processing technologies.

In this paper, two fast reactor systems are discussed – the sodium-cooled fast reactor, which has already been built and can be further improved, and the lead-cooled fast reactor that could be developed relatively soon. An accelerated development of the latter is possible due to the sizeable experience on lead-bismuth alloy coolant in Russian Alpha-class submarine reactors and the research efforts on accelerator-driven systems in the EU and other countries.

First, comparative calculations on critical masses, fissile enrichments, and burn-up swings of mid-sized SFRs and LFRs (600 MW_e) are presented. Monte Carlo transport and burn-up codes were used in the analyses. Moreover, local Doppler, coolant temperature and axial fuel expansion reactivity coefficients were also evaluated with MCNP and subsequently used in the European Accident Code-2 to calculate reactivity transients and unprotected Loss-of-Flow accidents (ULOF). Further, unprotected Loss-of-Flow as well as decay heat removal (total Loss-of-Power, TLOP) were calculated with STAR-CD CFD code for both systems.

The tight pin lattice SFRs (P/D=1.2) showed to have a better neutron economy than wide channel LFRs (P/D=1.8), resulting in larger BOL actinide inventories and lower burn-up swings for LFR. The reactivity burn-up swing of LFR self-breeder could be limited to 3$ in 3 years. The calculations revealed that LFRs have an advantage over SFRs in coping with the investigated severe accident initiators (ULOF, TLOP). The reason is better natural circulation behavior of LFR system and much higher boiling temperature of lead. An unprotected Loss-of-Flow accident in LFR leads to only a 250 K coolant outlet temperature increase whereas in SFR coolant would boil. Regarding the economics, the LFR seems to have an advantage since it does not require an intermediate coolant circuit. However, it was also proposed to avoid an intermediate coolant circuit in an SFR by using a supercritical CO_2 Brayton cycle.

1. INTRODUCTION

As predicted by the Special Report on Emission Scenarios (SRES) initiated by the Intergovernmental Panel on Climate Change (IPCC), energy demand of primary sources will increase between 1.7 and 3.7-fold until 2050 (INPRO, 2003). A mix of today's production options, however, falls short of the goal to provide a long-term, sustainable source of energy without adverse environmental/climate effects. In most of SRES reference scenarios, see Figure 1,

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the share of nuclear power is forecast to increase considerably by 2050, with a median of more than four times. However, a much larger growth of nuclear power would be needed, up to 15 times, if emissions of carbon dioxide are to be stabilized and then decreased beyond 2050-2060.

New reactor designs, which will be commissioned as a replacement and expansion of the existing nuclear reactor parks, should be competitive, safe, proliferation resistant, and meet the criteria of sustainability (GEN-IV, 2002). Next-generation systems have also to reduce the amount and radiotoxic inventory of residual wastes destined to geological repositories, hence addressing both public and scientific concerns regarding their long-term reliability. To achieve this goal, recycling of fuel and recovery of long-lived nuclides during reprocessing will be necessary. This will not only reduce the radiotoxic inventory of the waste but also allow converting long-lived actinides into energy.

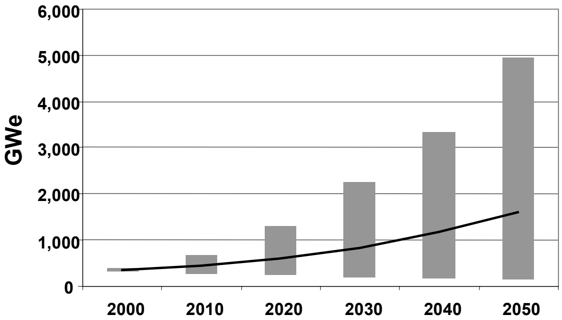

Fig. 1. *Range of nuclear power capacities as predicted by 40 SRES scenarios. Solid line represents median (INPRO, 2003).*

An efficient use of fissile fuel resources together with the ability to burn its own high-level wastes and those coming from LWRs are primary design goals of new reactor designs developed under the auspices of the Generation IV initiative. Both self-breeder (CR ~ 1) and TRU "burner" design options are hence considered. The former system is intended to operate in a pure fast reactor (FR) scheme while the latter will work in concert with LWRs or, in the double-strata scenario also together with dedicated minor actinide (MA) burners (ADS). As the purpose of self-breeder and burner is different, their fuel composition will differ and hence also their neutronic and safety characteristics. An introduction of minor actinides (americium) into the fuel significantly deteriorates the reactivity coefficients (Doppler, coolant temperature/void reactivity).

In a self-breeder configuration, the uranium content will be adjusted such that the breeding gain is kept close to zero. Minor actinides will be recycled infinitely. Their fraction in the equilibrium fuel should not exceed 2.5% in order to keep the Doppler high and coolant temperature reactivity coefficient low. In burners, which are assumed to destroy TRUs from a few (2-3) LWRs, the uranium content has to be limited in order to achieve high TRU consumption. An inert matrix is therefore applied instead of ^{238}U to dilute highly reactive TRUs. It has been shown that if MOX-recycling is not pursued in LWRs and the whole LWR TRU vector is directed into FRs, the minor actinide fraction in these systems could be kept below the 2.5% limit (OECD/NEA, 2002). Thus, there may be an incentive to re-evaluate the decision to continue with MOX-recycling in LWRs. Reactivity loss of self-breeder cores is small, allowing for extensive cycle lengths of several years. For example, in SVBR-75/100, the burn-up swing can be kept below 1$ during one year (Toshinsky, 2002). The reactivity excess available in the compensation rods is thus small excluding the possibility for an insertion of reactivity leading to a prompt criticality. On the other hand, a large reactivity reserve, usually of several tenths of dollars, is necessary in burner type of cores.

The optimal choice of core materials for GEN-IV systems is still an open question. The requirement of a fast neutron spectrum for an efficient breeding and TRU incineration implies the usage of coolants with low moderating power, such as sodium, lead, lead/bismuth eutectic, or helium. In this study, two types of liquid metals proposed as coolants for GEN-IV systems are studied - lead and sodium.

2. COOLANTS

Basic properties of the considered coolants together with lead/bismuth eutectic are summarized in Table 1:

Coolant	Na	Pb	Pb/Bi
ρ [g/cm^3]	0.847	10.48	10.45
T_m [K]	371	601	398
T_b [K]	1156	2023	1943
c_p [kJ/kg·K]	1.3	0.15	0.15
$\rho\ c_p$[J/m^3/K]	$1.1\cdot10^6$	$1.6\cdot10^6$	$1.6\cdot10^6$
k [W/m·K]	70	16	13
v [m/s]	10	2.5	2.5

Table 1. *Basic physical properties of liquid metal coolants. Densities (ρ), melting (T_m) and boiling (T_b) temperatures, specific heat (c_p), thermal conductivities (k), and maximum velocities (v) are given at 700 K.*

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Sodium has superior thermal hydraulic properties, allowing for tight pin lattices. There is a large (but not always positive) experience with operation of sodium-cooled fast reactors. While several power reactors have been shut down, BOR-60, JOYO, Phénix, and BN-600 are still operating, the latter being in quasi-commercial operation since 1982. New sodium-cooled reactors are under construction in Russia, China and India.

Sodium features a reasonably low melting temperature, but also a low boiling point (1156 K), which raises safety concerns regarding unprotected transients leading to a coolant heat-up. Sodium exhibits high chemical activity with water, water vapor and air - a limited sodium leak and fire has stopped the operation of the Japanese MONJU reactor since 1995.

The choice of lead and lead-alloys as coolants is motivated on the one hand by their high boiling temperatures, which avoids the risk of coolant boiling. On the other hand, lead and lead-alloys are compatible with air, steam, CO_2, and water, and, thus, no intermediate coolant loop is needed as in the sodium-cooled system.

Lead-bismuth eutectic provides a low melting point (398 K) limiting problems with freezing in the system and features a low chemical activity with water and air excluding the possibility for fire or explosions. A drawback connected with lead/bismuth is the accumulated radioactivity in lead/bismuth (mainly due to the α-emitter ^{210}Po, $T_{1/2}$ = 138 days), which could pose difficulties during fuel reloading or repair work on the primary circuit. However, IPPE Obninsk staff has developed methods to cope with the polonium during refueling and maintenance (Toshinsky, 2001).

Lead is considered as a more attractive coolant option than lead/bismuth mainly due to its higher availability, lower price and lower amount of induced polonium activity (by a factor of 10^4), as given in a publication about BREST-300 LFR reactor design (Adamov, 2001). Pure lead has a melting temperature of 601 K, which narrows in the reactor's operational interval to about 680-870 K. However, after more research, higher outlet temperatures will eventually be possible. Redundant electrical heaters are proposed to be introduced in order to avoid problems with freezing and blockages in fresh cores.

Lead-alloy coolant velocities are limited by erosion concerns of protective oxide layers to about 2.5-3 m/s (Novikova, 1999). Typical sodium velocities are up to 10 m/s, hence lead has, in practice, a lower heat removal capacity, which require higher pin pitch-to-diameter ratios to stay below cladding temperature limits. However, as shown later in this paper these high pitch-to-diameter ratios enhance the natural circulation capability of the coolant, and thus, the safety performance of LFRs.

Corrosion resistance of the structural material can be achieved through controlling oxygen content in lead or lead-alloy. This technology has been used in the Russian Alpha-class submarines and its effectiveness up to 820 K has been confirmed by the EU ADS research. The surface alloying by the so-called GESA method enhances corrosion resistance of the structural material further, at least up to 870 K (Wider, 2003). It should also be noted that pure lead shows to be less corrosive than lead/bismuth eutectic at the same temperature (Wider, 2003).

Fast creep of the reactor vessel during coolant heat-up transients is another important issue to be considered. It occurs significantly below the lead boiling point, ~1170 K for SS-316, 1250 K for NIMONIC alloys and possibly higher for ODS steels. These values refer to an 11 m tall vessel.

Neutronically, the lead and lead/bismuth energy loss due to the elastic scattering is significantly smaller than that for sodium. However, due to the presence of several thresholds for inelastic scattering in the energy interval from 0.57 to 2 MeV, the neutron energy loss in inelastic scattering is notably larger than for sodium. Therefore, the neutron spectrum of lead and lead/bismuth cooled reactors will be decreased for energies above 1 MeV.

On the other hand, the magnitude of the neutron flux for sodium-cooled reactor is decreased in the energy interval of 0.7-1.5 MeV, where contributions to the neutron slowing down from elastic and inelastic scattering reactions are nearly equal. Additionally, the neutron mean free path in sodium is larger than that of lead or lead/bismuth. Therefore, the leakage of neutrons and their contribution to overall neutron balance in the system is more significant for sodium.

Further, higher scattering in lead and lead/bismuth without increasing the moderation for neutrons below 0.5 MeV prevents the neutrons from escaping from the internal parts of the lead-alloy cooled cores and, at the same time, provide an excellent reflecting capability for the neutrons, which escape the core.

Hence, we can also infer that the neutron economy of the lead-alloy cooled systems would be better than for sodium-cooled counterparts having the same geometry. E.g., lead-alloy cooled, $(U,Pu)O_2$ fuelled systems require smaller plutonium enrichments than sodium counterparts to reach criticality, see Table 2.

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P/D	Na	Pb/Bi
1.2	17.0	16.0
1.6	22.0	19.0
2.0	27.5	22.5
2.4	33.0	26.0

Table 2. *Pu/(U+Pu) fraction as a function of pitch-to-diameter ratio in a model critical 1200 MW_e FR employing $(U,Pu)O_2$ fuel and cooled by sodium and lead/bismuth (Tucek, 2004).*

3. METHOD FOR NEUTRONIC AND BURN-UP CALCULATIONS

The Monte Carlo code MCNP was used for the calculation of the criticality, spatial distributions of neutron fluxes and power (Briesmeister, 2000). Reactivity coefficients were evaluated by using the perturbation model implemented in MCNP. Doppler reactivity feedback was estimated by evaluating a reactivity change upon the increase of fuel temperature from 300 K to 1800 K. MCB code was used to calculate fuel burn-up (Cetnar, 1998). Nuclear data libraries were adjusted for the temperature dependence by the NJOY code. The averaged temperatures of the core components were assumed as follows: 1500 K for fuel, 900 K for cladding, and 600 K for coolant. The composition of the actinide vector is that of spent LWR UOX fuel - see Table 3. The fuel has a burn-up of 41 GWd/tHM and it is assumed to have undergone 30 years of cooling. Depleted uranium (0.3% ^{235}U) is used in the analyses.

Isotope	Fraction
^{235}U	0.003
^{238}U	0.997
^{237}Np	1.000
^{238}Pu	0.023
^{239}Pu	0.599
^{240}Pu	0.264
^{241}Pu	0.040
^{242}Pu	0.074
^{241}Am	0.871
^{243}Am	0.129

Table 3. *Plutonium and minor actinide vector corresponding to the LWR UOX spent nuclear fuel with burn-up 41 GWd/tHM after 30 years of cooling.*

4. LFR AND SFR DESIGN MODELS

Both reactors have a power of 600 MW_e. For the lead-cooled fast reactor (LFR), the thermal efficiency is assumed to be 42 % corresponding to an improved supercritical steam cycle (Cinotti, 2004). Similarly, a supercritical Brayton CO_2 cycle could be applied in sodium-cooled fast reactor (SFR), increasing the thermal efficiency to 45% (Schulenburg, 2003).

The pellet and pin for LFR are preliminarily assumed being the same as was envisioned for CAPRA reactor (Conti, 1995), see Table 4. A pitch-to-diameter ratio of 1.8 is used, which leads to a low-pressure drop, enhancing natural circulation behavior and hence increasing margins to core damage in Loss-of-Flow accidents. The active pin height is determined from a requirement to assure the thermo-mechanical stability of the pin column (limited bending) and achieve reasonable fuel burn-up rates. In this study, an active pin length of 200 cm was used, similar to that proposed for the STAR-LM LFR design (ANL, 2005). In order to keep the axial temperature gradient in the coolant below 80 K (inlet and outlet temperatures 673 K and 753 K, respectively), the maximum coolant flow velocities of 2.3 m/s are necessary. This is well below the design limit of 2.5-3 m/s hence ensuring the erosion stability of the protective oxide layers. The height of the LFR vessel is kept at 11 m in order to ensure seismic stability of the reactor. The 600 MW_e power, the 80 K coolant temperature gradient and the 11 m vessel height are based on the planned European Lead–cooled Fast Reactor (ELFR) project (Cinotti, 2004).

Parameter	LFR	SFR
Pellet outer radius (mm)	2.4	3.0
Clad inner radius (mm)	2.5	3.1
Clad outer radius (mm)	3.0	3.45
Pitch-to-diameter ratio	1.8	1.2
S/A outer flat-to-flat (cm)	20.10	14.66
Pins per S/A	331	271
Length of upper plenum (cm)	100	100
Length of lower plenum (cm)	10	10
Active pin length (cm)	200	100
Number of S/A	289	216
Number of channels in the individual enrichment zones	4/3/3	5/3
Averaged linear power (kW/m)	7.5	24.3
Peak linear power (kW/m)	12.7	40.4

Table 4. *Design parameters of SFR and LFR core concepts considered in this study.*

The design considerations regarding SFR are based on the model of the WAC benchmark reactor (Wider, 1989). The axial and radial reflectors were removed and the active pin height is only 100 cm. As discussed above, sodium allows for higher coolant velocities than heavy

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metal coolants. Hence, tighter pin lattices with P/D ~ 1.2 can be applied for the SFR.

5. LFR AND SFR BURNERS

In order to maximize the burn-up rate of transuranics from LWRs and limit breeding, the uranium fraction in the reactor core has to be significantly reduced in comparison to standard FBR designs. The excess reactivity of highly reactive TRU fuel has to be subsequently compensated by an introduction of a diluent (inert matrix) or a neutron absorber.

In our previous study (Wider, 2005), we have shown that an MgO matrix features favorable neutronic characteristics, limiting the amount of reactivity introduced into the system during coolant heat-up.

5.1 Neutronic and Burn-up Performance

The volume fraction of MgO in $(U,Pu)O_2$-MgO fuel was kept constant at 50% ensuring fabricability and thermal stability of the fuel. The SFR and LFR cores were divided into two and three enrichment zones, respectively. Uranium/plutonium ratio in the individual zones was adjusted in order to attain criticality, while at the same time keeping radial power peaking factor below 1.3. The core designs thus resemble a configuration characteristic for a Pu-burner operating together with LWRs and dedicated MA burners in the double strata scheme.

The basic core characteristics are listed in Table 5. The duct-free sub-assembly structure was modeled explicitly in MCNP. In order to achieve criticality, average Pu fraction in the LFR core has to be *actually* slightly higher than for the SFR despite the almost double actinide mass present in the LFR. The reason is the tight pin lattice of SFR (P/D=1.2), which offers better neutron economy than the wide channel LFR design. The burn-up swing of the burner cores was roughly inversely proportional to the initial actinide mass, which means that an LFR needs half as many outages over a period of time.

Parameter	LFR	SFR
Pu fraction in the individual enrichment zones [%]	24/30/39	27.7/31.7
Average Pu fraction in the fuel [%]	32.5	30.2
m_{act} at BOL [tHM]	13.7	6.58
Burn-up swing [pcm/full power day]	15.4	30.1

Table 5. *Neutronic and burn-up performance of SFR and LFR burners.*

The spatially dependent local reactivity coefficients together with the axial distribution of the fluxes were calculated in subsequent calculations for LFR and SFR designs. Axially, cores were divided into seven equidistant meshes (five in the core and two in the lower and upper plena). Two and three radial channels were applied for SFR and LFR, respectively, corresponding to the individual enrichment zones.

Comparisons of the coolant and fuel temperature feedbacks in the individual radial channels of SFR and LFR are given in Table 6. In all channels, heat-up of the fuel and coolant during ULOF by 100 K yields a prompt negative reactivity feedback, which well compensates for the positive reactivity introduced by the coolant. In addition, the fuel heats-up faster than the coolant in all over-power conditions. Stronger Doppler reactivity feedback is provided in SFR than in lead-cooled cores in which less neutrons are scattered down to the region of pronounced resonances. The fraction of the coolant in the pin lattice of SFRs is considerably lower than in LFRs, which gives a low coolant temperature reactivity coefficient.

Radial channel	Doppler Δk [pcm]		Coolant Δk [pcm]	
	LFR	SFR	LFR	SFR
1	–30.7	–86.8	+16.7	+7.7
2	–46.0	–44.4	+30.2	–20.1
3	–23.2	–	–1.8	–

Table 6. *Doppler and coolant temperature reactivity feedback for LFR and SFR burner cores corresponding to the increase of fuel and coolant temperatures by 100 K.*

5.2 Safety Performance

Safety analyses were performed with the EAC-2 (Wider, 1990) and STAR-CD codes. The latter was also used for decay heat removal calculations. Unprotected Loss-of-Flow and total Loss-of-Power accidents were considered.

5.2.1 Unprotected Loss-of-Flow Accidents

The main parameters of the SFR under consideration are described in Table 4. This is the WAC benchmark reactor downgraded from 800 MW_e to 600 MW_e (or 1426 MW_t) in order to compare it with an LFR of the same power. The flow coast-down is described by the equation:

$$\mathbf{G/G_0 = 1 /(1+t/6s)}$$

Thus, the flow-rate will be halved in 6 seconds. When the sodium starts boiling in Figure 2, the flow is still 20% of nominal. This figure also shows that the net reactivities due to

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the Doppler and axial fuel expansion cannot prevent the boiling. It was shown that in a smaller SFR design (800 MW_t or about 335 MW_e), a strong negative radial expansion feedback of the structure and a reasonable natural circulation are needed to prevent boiling (Van Tuyle, 1989). For the Super-Phénix reactor at least an increase in grace period (of about 100 s) could be achieved using a constantly rotating flywheel that provided temporary power to the pump during a ULOF.

Figure 2 shows that sodium boiling leads to a significant positive reactivity increasing the power (see Figure 3), which leads to more boiling and voiding until fuel melts. Then, fuel pins breach and molten fuel is swept out shutting the excursion down after about half the core is molten. The fuel feedbacks could also be temporarily positive if pin failures near the mid-plane are predicted. The present calculations were performed with the European Accident Code-2 using 10 radial calculation channels.

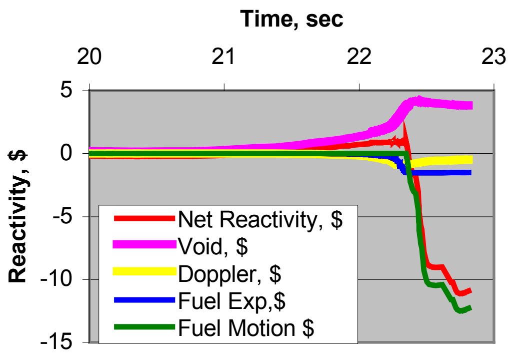

Fig. 2. *Reactivities in SFR-ULOF accident.*

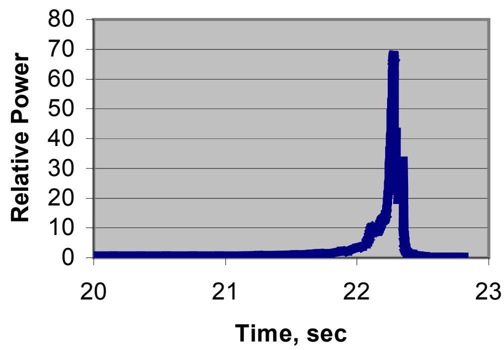

Fig. 3. *Relative power in SFR-ULOF accident.*

The next figures concern the behavior of an LFR of 600 MW_e or 1426 MW_t (see Table 4) in an unprotected Loss-of-Flow accident. No negative feedbacks are considered since the STAR-CD CFD code, which was used, has only thermal hydraulic models.

The same flow coast-down as in the previous SFR calculation is applied. As can be seen in Figure 4, LFR-ULOF leads only to an averaged outlet temperature increase of about 250 K. This is of course no significant problem since lead is still 1000 K away from boiling and in short-term perspective no corrosion problems will occur. From the multi-channel core calculations, we found that the hottest channel coolant outlet temperature will only rise to 1020 K. Nevertheless, in reality, there will also be negative feedbacks due to the Doppler and axial fuel expansion that will get the power down quickly.

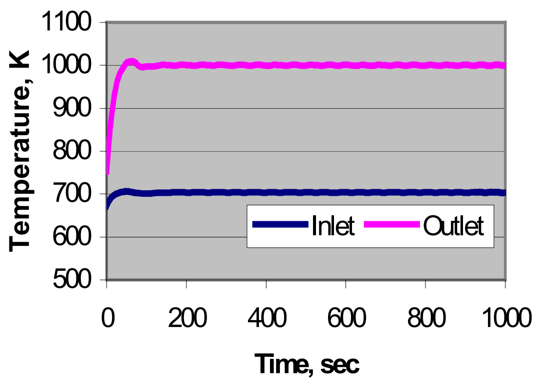

Fig. 4. *Above-core averaged and inlet coolant temperatures in LFR-ULOF accident.*

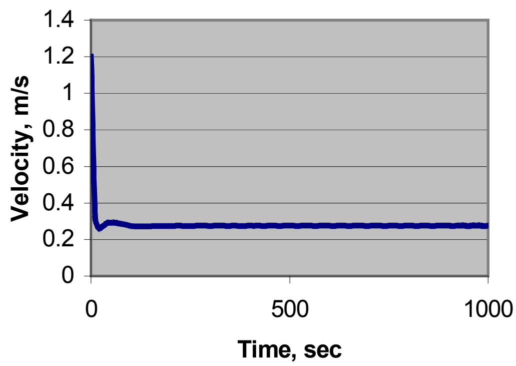

Fig. 5. *Above-core averaged coolant velocity (v_{init}=1.2m/s) in LFR-ULOF accident.*

This remarkable behavior of the LFR is due to the low-pressure drop core (P/D = 1.8) and a simple flow path design as in the Ansaldo ADS (Cinotti, 2001) and the ANL STAR-LM designs (ANL, 2005). In the latter designs, the coolant rises above the core and then continues down through a down-comer (were the heat exchangers are located) back to the core. In the ANL design,

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solely natural circulation cooling is used even for regular, steady state operation.

In the subsequent two Figures (6 & 7) it is shown how SFRs would behave during LOFs in simple flow-path designs and without negative feedbacks. Figure 6 shows that the averaged coolant outlet temperature would exceed the boiling point in about 20 sec (see also Figure 2). Thus, the simple flow-path design alone doesn't explain the difference in behavior relative to the LFR.

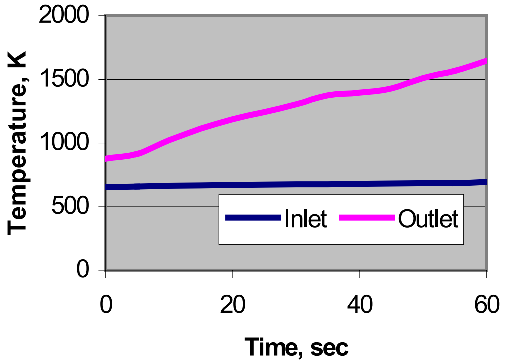

Fig. 6. Averaged outlet and inlet coolant temperatures of 600 MW$_e$ SFR with P/D of 1.2 and simple flow-path design.

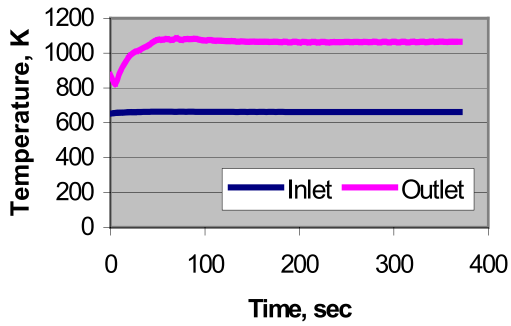

Fig. 7. Averaged outlet and inlet coolant temperatures in SFR-ULOF with large P/D= 1.8.

The main reason why the SFR did not behave similarly as the LFR must be due to high-pressure drop in the SFR core. By using the same high P/D=1.8 of the LFR in the SFR, we achieve similarly good behavior of the system in the ULOF. Figure 7 shows essentially the same temperature behavior as the LFR (Figure 4), the only difference is the 70 K higher outlet temperature of the SFR under consideration.

5.2.2 Total Loss-of-Power Accidents

In a total Loss-of-Power accident in which diesel-driven generators are unavailable, emergency decay heat removal becomes important. Here, we investigate the Reactor Vessel Air Cooling System (RVACS) for the two 600 MW$_e$ systems (Carlsson, 2000).

Figure 8 shows that the LFR gets after 2-days within 50 K of the creep limit of SS-316 for the existing stresses. Thus, a more efficient emergency cooling such as an IRACS (In Vessel Reactor Auxiliary Cooling System) may be needed. Also a water pool surrounding the guard vessel, as described in (Toshinsky, 2002), is an interesting option for the emergency cooling. For the both SFRs under consideration, the TLOP eventually led to sodium boiling.

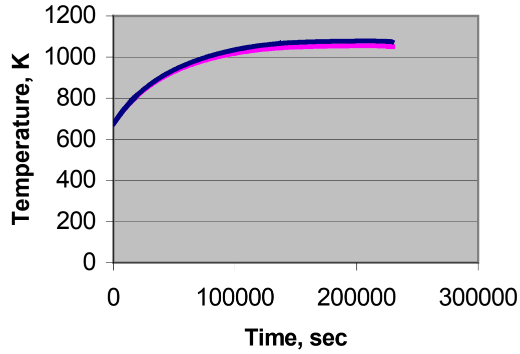

Fig. 8. Vessel temperature evolution in LFR-TLOP, heat removal is provided by RVACS. Pink line is for emissivity of 0.9, blue for 0.7.

5.2.3 Reactivity Accidents

Regarding reactivity accidents the LFR has an advantage due to lower burn-up swings – see below. In addition, if large insertions lead to pin failures, the fuel sweep-out is more likely in LFRs due to a low pressurization during fuel-coolant interaction and due to the much larger inertia of lead.

6. LFR AND SFR SELF-BREEDERS

Self-breeder reactors feature low burn-up reactivity swing. Ideally, this should be below 1$ over an extended time. The amount of reactivity that could be then accidentally inserted into the reactor is small limiting consequences of reactivity-induced accidents.

In the self-breeder reactor designs, the MgO matrix was omitted and the TRU vector consisted of plutonium and minor actinides (Np and Am) from spent LWR UOX fuel. The Pu/MA fraction, 83/17, corresponded again to a fuel decay time of 30 years and a fuel burn-up of 41 GWd/tHM. Two LFR core configurations were set-up. One corresponding to the core design with P/D = 1.8 and pin diameter of 6 mm, and the second with enlarged pin diameter, D=6.36 mm, which yielded a pitch-to-diameter ratio of 1.7. In the

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latter design, the uranium/plutonium ratio had to be increased in order to attain criticality. Other geometry and material parameters were retained the same. Actinide masses were now 27.34 and 31.54 tHM for LFR core designs with P/D=1.8 and 1.7, respectively. Actinide mass for the SFR (P/D=1.2) was 13.09 tHM.

It should be noted that the SFR core performs similarly to LFR cores during the first year of irradiation despite of much lower initial actinide mass, see Figure 9. Additional iterations on the fuel composition may show further reduction in the reactivity swing.

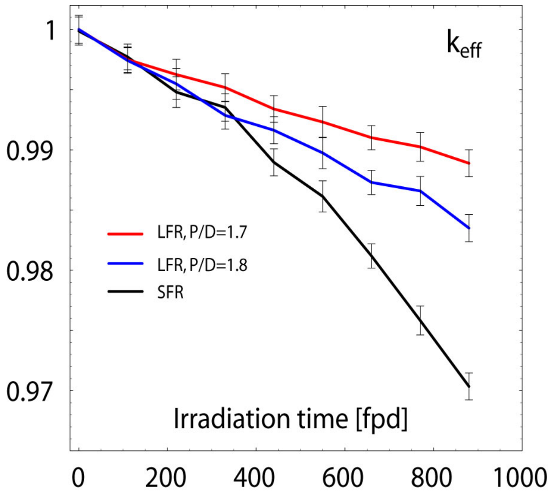

Fig. 9. *Reactivity burn-up swing for LFR and SFR cores. 2-σ standard deviation is displayed.*

As shown by Feldman et al. (2004), low power density LFRs can have very long refueling intervals of up to 15 years, which is ideal for remote locations and developing countries with no nuclear infrastructure. Modular and easily transportable LFRs would be best suited for these applications.

7. ECONOMIC ASPECTS OF SFR AND LFR

Regarding economics, fast reactors were earlier considered more expensive to build and their electricity generation cost higher than that of LWRs. However, in the last few years several Russian publications have indicated that the lead/bismuth-cooled SVBR-75/100 is cheaper to build than all other reactor types and that the electricity generation cost is even lower than that of gas-fired plants, see Table 7.

The reasons for this are that no intermediate coolant loop is needed for an LFR, and less safety-related systems have to be built. The prolonged, 8-year fuel cycle is helping to get the electricity generation cost down, too. Note that in a true LFR, lead would be used instead of the lead/bismuth. Since lead is about 10 times cheaper than lead/bismuth, the capital cost for LFR may be even lower than envisioned for SVBR-75/100.

Energy system considered	**SVBR -75/100 LFR**	**BN-800 SFR**	**Gas PGU-325**
No of plants, MWe	16 × 102	2×890	5× 325
Efficiency of the net plant, %	34.6*	46.2	44.4
Specific capital investments, $/kW (price of 1991)	661.5	783.4	600
Cost of electricity, cent/kW·h - price 91	1.46	1.56	1.75

Table 7. *Economic comparisons of LFR, SFR and gas-fired plant (Zrodnikov, 2003).*

** will be higher for supercritical steam cycle*

Note that if a supercritical CO_2 Brayton cycle is used in an SFR, the intermediate cooling loop can be omitted and the SFR could become more economically advantageous, too. However, further research is needed to confirm the viability of this approach alluding to the possibility of orifice clogging by Na-CO_2 reaction products (Weaver, 2005).

8. CONCLUSIONS

Considering the typical core lattice design configurations, the LFR showed to have advantages over SFR regarding behavior in severe accidents, ULOF and TLOP. This is due to the better natural circulation behavior of LFR design and the much higher boiling temperature of lead. Moreover, chemical inactivity of lead excludes possibility for fires or other strongly exothermic reactions with air, water, and water vapor. An LFR appears to have also an economic advantage since it doesn't need an intermediate coolant circuit and the number of reactor outages can be limited. The latter aspect is also relevant for the remote location of modular LFRs. These and larger LFRs can be used both as burners and self-breeders with some advantages over SFRs (e.g., featuring lower burn-up swing). However, SFRs could also become more economically favorable if an inert gas could be used as a secondary coolant. There is considerably more experience with sodium than lead or lead-alloys although this was not always satisfying. For LFRs, corrosion and material characteristics of steels should be further investigated, in particular under irradiation conditions.

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