Lead-Bismuth-Cooled Fast Reactors

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Edited by

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PREFACE

Lead-bismuth cooled modular fast reactors are of interest to many readers. Recently, research and development (R&D) on the application of heavy liquid metal coolants (HLMC) in different nuclear technologies has been increasing widely in many parts of the world (China, Czechia, Belgium, Germany, India, Italy, Japan, Korea, Russia, USA, Sweden, Switzerland). Forty-seven experts from across the world and many Russian specialists took part in the fourth conference of "Heavy Liquid Metal Coolants in Nuclear Technologies" (HLMC-2013) held in Obninsk in 2013. Reactors with HLMC are considered as an option for Generation IV reactors.

Wide development of the R&D is conditioned by the natural properties of HLMC, namely: very high boiling point and chemical inertness in water and air, which make possible significant safety enhancements by deterministically eliminating the reasons for such accidents as the Three Mile Island accident (the USA), the Chernobyl disaster (the former USSR) and the Fukushima accident (Japan). In addition, the nuclear properties of HLMCs allow their use as a liquid metal target of proton accelerator in hybrid electronuclear systems for the transmutation of minor actinides.

The articles and papers published in the presented collection are in chronological order of their issue without any changes. So, repetitions are inevitable because almost all of the papers deal with a specific subject and cover a period of approximately twenty-five years. For that reason, there are certain differences in the results due to evolution in the design of reactor SVBR-100 and the author's opinions regarding some issues.

The first edition of the Collection, containing seventeen papers and papers, was published in 2017 by LAMBERT Academic Publishing. This second edition was intended to be an extended one, and includes nine additional articles compared to the first edition. However, Elsevier, MDPI, and SCIRP did not respond to my requests for permission to republish the ten papers I submitted, originally published in different years in their journals. Without these permissions, Cambridge Scholars Publishing cannot include these articles in the second edition. For the convenience of readers, the titles of these articles, their authors, the year of publication, the titles and numbers of the journals are listed separately in the bibliography.

Besides, in different articles and reports, the same concepts are called different names. For example, reactor installation or reactor facility, etc. Therefore, at the end of each article or report, there is a list of abbreviations.

In most cases, the compiler of the present collection is the main author of the published articles and papers.

The author would like to express thanks for the preparation of this manuscript for publication to Mr. K.G. Melnikov and Mr. I.V. Tormyshev (JSC SSC RF-Institute for Physics and Power Engineering).

CHAPTER 1

CONCEPT OF THE SELF-FUEL-PROVIDING LMFBR

G.I. TOSHINSKY

1.1 Introduction

Development of nuclear power (NP) on the basis of traditional reactors, which use less than 1% of the energy capability of natural uranium (NU), will result in the cheap resources being exhausted up to the middle of the next century and low competitivity of NP.

Transitioning to traditional fast reactors (FR) with the closed nuclear fuel cycle (NFC) can solve the problem of fuel resources for NP but has a negative effect on its competitivity (higher costs of FR and the technology of plutonium recycling). The competitiveness of NP should be improved in the distant future when the resources of cheap natural gas are exhausted.

There might be a long time interval (50–100 years) between the moment when NP loses its competitiveness and the time when the NP with FR and the closed NFC can surely compete with fossil fuel power plants. Until then NP would be less economically effective in comparison with fossil fuel power plants ("gas pause").

It is feasible to eliminate the "gas pause" by means of NP on the basis of non-traditional FRs which increase greatly the efficiency of NU energy capability without fuel chemical reprocessing and NFC closing over plutonium.

The Self-Fuel-Providing LMFBR (SFPR) is designed to operate in such a way that only fertile material (depleted uranium) is needed as a makeup fuel and, thus, it requires no Pu recycling for itself. The high efficiency of natural uranium using (EUU) in the reactor makes it possible to develop NP for hundreds of years, using available uranium resources without fuel reprocessing and with a high plutonium proliferation resistance.

The concept of long-term NP development without fuel reprocessing, based on SFPR, is considered a temporary strategy that will provide a significant time reserve for high-level development of industrial reprocessing for the closing of the nuclear fuel cycle after the mentioned period that must allow retention of the competitiveness of NP with electric power using fossil fuel.

The concept of such a reactor cooled with Pb-Bi (SVBR-600) has been developed. Use of lead-bismuth coolant (LBC) as compared to sodium results in, apart from an increase in safety, an increase in the core breeding ratio (CBR) due to a harder neutron spectrum formation. It improves greatly the EUU and also improves engineering and economic indices.

1.2 Fast Breeder-Reactor (FBR) Operation Without Fuel Reprocessing

In the reactor under consideration, the start-up loading of the core made of fuel sub-assemblies (FSAs) with a comparatively highly enriched (10–12%) uranium fuel (U-Pu fuel utilization is possible) is done only once. In the course of the reactor operation the start-up FSAs in which plutonium has already been built up under partial refuelings are gradually replaced with fresh makeup FSAs with natural or depleted uranium. There are no separate breeding zones in such a reactor. The principal condition of ensuring reactor operation in such a mode is a core breeding ratio (CBR) of more than one. This allows the criticality to be maintained due to Pu build-up.

Initially, a reactor like this was theoretically considered by Feinberg and Kunegin [1] and then by Fuchs and Hessel [2]. Later on, different investigators took some interest in this problem as well, e.g., Slesarev et al. [3].

Fast breeder-reactor (FBR) operation under such conditions is not in accord with the adopted viewpoint of FBR's role in NP, because in this case the built-up plutonium is not extracted and is not reutilized because the NFC is closed inside the reactor. However, fuel utilization is much more effective compared with the operation of NPPs with thermal reactors operated in the open NFC.

On the basis of the circumstances mentioned above, a concept was developed of FBR operating in the open NFC with slightly enriched or depleted uranium makeup [4]. Safety is of the highest priority with regard to a reactor of this type.

Fast-neutron reactor installations (RIs) cooled with LBC fall into the category of RIs whose safety is mainly provided by their inherent self-protection. This is a result of a number of features typical of this design. These intrinsic features include (1) no poisoning effects in a fast reactor, (2) a low value of a negative temperature reactivity worth coefficient, (3) compensation of fuel burn-up and fission product formation processes with plutonium build-up, (4) partial refueling of FSAs, (5) high boiling point and chemical inertness of the coolant.

Using a heavy liquid metal coolant with a low potential energy content allows a low pressure in the primary circuit at a high temperature and eliminates the possibility of a reactor thermal explosion.

1.3 FBR Safety Improvement due to Using LBC

To achieve high indices of RI safety, the choice of the primary circuit coolant is of great significance. It determines also the main technical solutions of RI design and its equipment and nuclear power plant (NPP) technical and economic characteristics to a great extent.

Among all the liquid metal coolants it is sodium that has gained the widest acceptance. This coolant has been chosen for FBR because, due to its good thermal and physical properties, sodium offers possibilities for intensive heat removal and thus for a short Pu doubling time. This was an indispensable requirement in the early stages of FBR design and development in the 1960s and 1970s. For this very reason, when considering various liquid metal coolants (LMC) for FBR in the 1950s, academician A.I. Leypunsky gave preference to sodium, although initially he had considered LBC for this purpose [5].

Yet there are some complicated problems related to realization of the reactor concept concerning a slightly enriched or depleted uranium makeup in reactors with sodium coolants. They are associated with a considerable positive sodium void reactivity effect which is characteristic of reactors with low neutron leakage. Besides, the neutrons slowing down due to sodium nuclei decreases the CBR by about 0.06 which plays a leading part in achieving a high efficiency of natural uranium utilization. For instance, as the calculations have demonstrated in the case of sodium coolant, it is difficult to provide the reactor criticality when there is a depleted uranium makeup.

At present and in the foreseeable future there is no need for such short Pu doubling times as can only be provided by FBR with sodium coolant. So, the opportunity has been afforded to turn back to studies of LBC for FBR.

In our country, long-term experience has been accumulated with respect to the eutectic LBC utilization in submarine nuclear reactors [6] which have been developed by IPPE as scientific supervisor of studies. In the process of mastering this coolant for use in Rls of submarines, several principal problems have been encountered. One of the problems is related to ensuring coolant quality in the course of operation, another to the radiation safety associated with the formation of alfa-active radionuclide polonium-210 and also some other problems have been resolved.

Making use of this coolant allows FBR safety to be improved further due to its chemically inert behavior and higher boiling temperature (about 1700 °C) that actually rules out the coolant boiling in the most stressed FSA, even in the case of the most severe accidents.

Taking into account that potential compression energy and chemical energy stores in this coolant are minimal in comparison with those of other coolants, it might result in a RI in a class with safety as high as reasonably achievable.

So, there is a chance to realize a two-circuit design scheme of steam generation and to improve technical and economic indices.

The problems limiting LBC use in large-scale future NP depend on the deficiency of bismuth resources and its relatively high cost. It might happen in this case that the bismuth cost (as part of the overall capital costs) for constructing NPP will be too large, especially when there is demand to use the ore with a low bismuth content. It may become reasonable to changeover to a purely lead coolant to be proposed for cooling FBR by RDIPE, for which there are no raw material base restrictions.

However, due to lead having a higher melting point, the lower temperature of the lead coolant should be considerably increased. This makes the problem of the coolant technology more complicated, as well as the complications of the corrosion resistance of structural materials and mass transfer. In the case of using lead-bismuth coolant, the problem took 15 years to be resolved. Besides, it also results in a more complicated operation of RI.

To reduce the cost of coolant, during the preliminary stages of concept development a lead-bismuth alloy was considered as a coolant of non-eutectic composition with the bismuth content decreased up to 10% (versus 56% in the eutectic alloy). Hence there are no problems with bismuth resources and deterioration of cost indices. At the same time (as compared to a lead coolant) its melting temperature is decreased by 77 °C (up to 250 °C) which facilitates RI operation and allows the pressure in the secondary circuit to be reduced considerably (not more than 14.0 MPa, in comparison with 24.0 MPa for a lead coolant) and reduction of maximum temperatures of the fuel element claddings up to the level which has been checked under conditions of long-term operation tests.

1.4 Efficiency of Natural Uranium Utilization

The efficiency of natural uranium utilization (EUU) in such a reactor is determined as a ratio of energy generation expressed as the equivalent mass of fission products, to the total amount of natural uranium used in the reactor during the whole operational period. EUU will increase as the period of the reactor operation increases due to the decrease of the enriched uranium start-up loading contribution to the total power generation.

The best values of EUU are achieved when forming a harder neutron spectrum due to the use of a metal fuel. In this case, depleted uranium can be used as a makeup fuel. Then EUU might achieve 5–10%. It is considerably lower than the value that could be obtained at the FBR operation in the closed NFC. But EUU is 10-20 times higher in comparison with VVER-1000 operating in the open NFC (0.5–1.0%).

When choosing certain design parameters of the reactor core, the fuel type (metal, ceramics), its density, the fraction of FSAs with refueling every cycle, then the burn-up, necessary for obtaining criticality depend only on U-235 content in the fuel makeup. When the latter decreases, EUU increases.

Such EUU in the proposed reactors makes NP development possible for extended periods without any large-scale construction of plants with the closed NFC for spent fuel reprocessing and without solving the whole complex of attendant problems.

To achieve a high EUU a long-term reactor operation is required, i.e., 100 or more years. So appropriate requirements should be met for the equipment, construction materials and operation conditions.

Because there are no reasons to expect such a long operation of RI without replacing its equipment, the whole primary circuit should be replaced in the course of NPP capital repairs.

After the reactor vessel resources are exhausted (its lifetime is taken to be equal to 50 years) the design should allow for the possibility of placing the reactor core in another reactor vessel mounted in the main NPP building. At the end of the following 50 years the first reactor vessel can be replaced with a new one after dismantling. Thus, NPP normal operation can persist up to the lifetime exhaustion of the principal building structures.

1.5 Pu Proliferation Resistance

It is sometimes considered that only weapons-grade Pu is of particular interest to terrorists who long for nuclear weapons. Yet it is known that the simplest explosive devices can be fabricated from Pu with a sufficiently high content of Pu-240 build-up in NPP reactors. Explosive devices equivalent to several thousand tons of TNT can be fabricated even from PuO_2, which may be extracted from a plundered MOX fuel, and used as an instrument of political blackmail. There is the possibility of Pu plunder in small parts in the course of fuel reprocessing despite strict controls for Pu non-proliferation.

Taking into account the above, political stability could be achieved if there was no necessity for fuel reprocessing and if long-term spent nuclear fuel storage was realized together with high radioactive fission products, that excludes unauthorized Pu proliferation ("spent fuel standard").

1.6 Results of Preliminary Studies

Results of the investigations carried out have confirmed the feasibility of FBR operation with makeup at partial refuelings with slightly enriched or depleted uranium. In this case, the EUU highest value is achieved due to the large core dimensions ($D \times H = 4.0 \times 1.4$ m), fuel volumetric fraction (not lower than 60%), utilization of fuel with high uranium density (approximately 11 g/cm^3), formation of the most hardened neutron spectrum in the core. When using metal alloyed (10% of Zr) uranium fuel with an effective density of about 75% of the theoretical one the reactor can utilize depleted uranium like makeup fuel. Thus, the highest EUU is ensured. In this case the burn-up depth achieves about 20% h.a. (that is justified at experimental assemblies of EBR-2 reactor), fast neutron damage dose on a fuel element cladding material accounts for approximately 400 dpa (it is twice the value which has been achieved in tests on ferritic and martensitic steels), the total operating time of the fuel assembly is about 30 years (that is three times more than that obtained during operation of a nuclear submarine reactor). These data have been achieved for the reactor with a thermal capacity of 1875 MWt.

Increasing the makeup fuel enrichment reduces considerably these demands. The calculations show that an increase of makeup fuel enrichment up to 4.4%, which is characteristic of VVER-1000, causes a reduction of the above parameters up to the values close to those tested experimentally. So, by the fourth lifetime EUU is three times higher than that for VVER-1000.

Accordingly natural uranium consumption becomes three times lower, i.e., the feasible duration of the open NFC operation is elongated by three times. It should also be noted that the core design and scheme of partial refuelings are far from optimum because they have been adopted in the calculations of concept preliminary developments. Further optimization should reduce requirements for operational conditions of fuel elements.

Considerable distinctions in the concentration of fissionable nuclides of FSAs with different burn-up (especially for FSAs with makeup fuel) result in non-uniform power density. It requires physical field shaping by means of choosing an optimum scheme for partial refueling of FSAs and their shuffling in the core. It is clear that such a refueling scheme is preferred where zones with a high content of plutonium are interchanged with those of lower content.

Although the concept of NP development under consideration has a lot of advantages in terms of fuel provision and safety, the NPP technical and economic indices will be of primary importance for NPP practical use. In this relation certain reasons can be cited that give us some hope that these indices will not be significantly different from those of VVER NPP of similar capacity.

They are as follows: (1) a smaller number of annually fabricated makeup fuel elements due to higher burn-up in FBR; (2) low or no expenses of uranium development and enrichment for makeup fuel elements; (3) feasibility to use an integral design RI due to a considerably low pressure in the primary circuit thus simplifying the design and decreasing the construction bulk; (4) small amounts of liquid radioactive waste (according to operation experience) requiring much lower capacities of special chemical water purification systems; (5) high reactor safety which eliminates determinatively a number of the most severe accidents, thus allowing designers to exclude a number of safety systems. All this simplifies RI design and decreases its cost.

The factors mentioned above should demonstrate the considerable benefits which are offered by this reactor design. A higher cost of the initial reactor loading and LBC (not replaced for the whole lifetime and suitable for reutilization) is more than compensated for by the factors above.

At a high EUU, it is expedient to study the problem of NP long-term development in the open NFC with the reactors under consideration.

1.7 Conclusions

The preliminary assessments performed show that it is reasonable to investigate the feasibility of FBR operation with one-through NFC in the future. It is expedient to use LBC for improving its safety. In order to operate with a depleted uranium makeup, it is necessary to meet a number of requirements providing the reactor criticality due to plutonium build-up and CBR>1.

These requirements are as follows: a large core (20-25 m^3); a high fuel volume fraction (>60%); use of dense metal fuel; a high fuel burn-up at the level of 20% h.a.

Making use of these reactors should allow the NP fuel base to be extended several times without making NFC closed. It provides improved NP safety and ensures a high Pu proliferation resistance.

List of Abbreviations

CBR	– Core breeding ratio
FR	– Fast reactor
FSA	– Fuel sub-assembly
EBR	– Experimental breeder-reactor
EUU	– Efficiency uranium utilization
HLMC	– Heavy liquid metal coolant
LBC	– Lead-bismuth coolant
LMC	– Liquid metal coolant
LMFBR	– Liquid metal fast breeder-reactor
MOX	– Mixed oxide fuel
NFC	– Nuclear fuel cycle
NP	– Nuclear power
NPP	– Nuclear power plant
NU	– Natural uranium
RI	– Reactor installation
RDIPE	– Research and development institute of power engineering
SFPR	– Self-fuel-providing reactor
SVBR	– Lead-bismuth cooled fast reactor
VVER	– Water-cooled water moderated reactor

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CHAPTER 2

THE ANALYSIS OF OPERATING EXPERIENCE OF REACTOR INSTALLATIONS USING LEAD-BISMUTH COOLANT AND ACCIDENTS THAT HAPPENED

B.F. GROMOV, O.G. GRIGORIEV,
A.V. DEDUL, G.I. TOSHINSKY,
V.S. STEPANOV, L.B. NIKITIN

2.1 Introduction

When the last NS with the RI cooled by LBC was decommissioned from the Navy, the specific phase of ship nuclear power development had been completed. The new nuclear power technology which could not be compared with any other one in the world was demonstrated to our industry.

Due to ongoing developments of the RIs using LBC for civil NPPs, the experience gained is needed for the thorough analysis and consideration in these designs to make the best use of the LBC advantages (high boiling point, chemical inertness, etc.) and to minimize the effect of its disadvantages (polonium-210 formation, melting point is ~125 °C).

The operation experience is also accompanied by the number of accidents, which are inevitable in mastering each new technology (the history of the techniques has demonstrated this), and revealed the difficulties of servicing these RIs at their base places and refueling. This experience received an ambiguous assessment from the experts who familiarized themselves with it. Some authors, who have been familiarized with this experience only by “hearsay”, allege in their memoirs that for the NSs adoption of Projects 705 and 705K (class “Alpha” in compliance with NATO terminology) of RIs using LBC was a tragic error.

Such “reminiscences” are spread by the internet network for users from other countries.

Below there are presented the key results of operating the RIs using LBC [1], causes of deaths that happened, causes of the difficulties in servicing the RIs in the base conditions and at the other phases of RIs life term, measures which have been realized in the designs of the RIs using LBC and which eliminate the causes of accidents and operational difficulties. This presentation makes an unbiased assessment of the experience gained.

2.2 The Problems of Mastering the RIs Cooled by LBC

Among the key problems which have been solved during the design and operation of this type of installation the LBC technology problem must be emphasized—i.e., the development of systems and devices for ensuring the measurement and maintenance of the LBC quality required during its long-term operation both under the normal conditions of a leak-proof circuit and in the case of partial tightness loss of the circuit in the course of repair works and reactor reload. The functioning of those systems and devices is necessary for eliminating corrosion of structure materials and slagging of the circuit by lead oxides.

Corrosion resistance of the structure materials have been ensured by using special steel alloying applying protective films in advance and maintaining the necessary concentration of corrosion inhibitor-dissolved oxygen in LBC.

The following characteristics have been obtained in the process of RI tests and operation: power and parameters of installation, the campaign lifetime, the reactivity margin, reactivity coefficients, poisoning effects, temperature distributions, dynamic parameters, coolant radioactivity, dose rates of neutron and γ-radiation behind the shield. They were in sufficiently good agreement with calculated results.

The cores and absorbing rods of the control and protection system (CPS), which have ensured total design burn-up, have demonstrated high operation serviceability.

Experience of the development and operation of RIs using LBC at the NSs and ground-based facilities enabled prototypes to be used to make certain practically important conclusions on the design of equipment for the primary circuit for reactors cooled by LBC for NPPs and ADSs.

The best engineering and economic parameters should be expected if the design of the primary circuit equipment is an integral one.

The most convenient design scheme of the SG is one in which liquid metal circulation is performed in the inter-pipe space and water or steam is circulated within the pipes. That design ensures the possibility of repairing the SG by plugging a separate pipe, which has become leaky, without dismantling the SG or losing the primary circuit tightness.

Shutdown regimes, the regimes of start-up and cooling down are realized in the easiest way if the SG operates using multiple circulation in the water-steam circuit.

Both mechanical upright pumps with a turbine or an electric driver and electromagnetic pumps may be used for coolant circulation.

LBC remaining in the liquid state under all RI operation regimes is ensured by using the SG with multiple circulation over the secondary circuit, besides that, the inlet temperature of the water supplied to the SG is higher than the LBC melting point. For initial heating up and maintaining the primary circuit under hot conditions at a low level of power in the core the system of steam or electrical heating may be used.

The substantiation of the possibility of multiple coolant "freezing-defreezing" in the RI was an important practical problem. Low shrinkage of LBC during solidifying and rather high plasticity with low strength in the solid state facilitate the elimination of RI damage when the alloy is transiting from liquid to solid form and its further cooling down to reach the ambient temperature. A special order of the temperature-time heating regime has been developed for safe RI “defreezing”.

While performing repair works and reactor reloading, it is not necessary to carry out primary circuit decontamination which is concerned with collecting, storage, transporting and reprocessing liquid radioactive waste (LRW).

Among the positive properties of the RI using LBC which have been discovered in the course of operation, one can point out the simplicity of control, high maneuverability and short time of reaching the power regime out of the sub-critical reactor state, the possibility of RI operation if there is a small leakage in the SG pipe system, high repair-fitness of the SG by plugging the depressurized pipes, the possibility of RI stable operation at any low power levels, the possibility of quickly changing the circulation regime of coolant with essential change of its flow rate, almost complete achievement of the design fuel burn-up under normal and permissible conditions of violation of the tightness of the cladding of fuel elements, maneuverability of fuel rod claddings, and almost total absence of LRW during repair works and refuelings.

2.3 The Analysis of the Accidents that Happened and the Difficulties of RIs Base Servicing

During the design and operation of RIs using LBC there were accidents at three NSs, that were the cause of the impossibility of further RIs operation. These were the RI accident at the left side of the NS of Project 645 in 1968 [2] when the core was melted partially, the OK-550 RI accident at the NS of Project 705 (Task Order 900) in 1971 when the auxiliary pipelines of the primary circuit turned leak-proofless and the BM-40/A RI accident at the NS of Project 705K (Task Order 105) in 1982 when the global corrosion damage of the steam generator (SG) pipe system of the water-steam circuit occurred and about 250 liters of radioactive coolant were spilled into the compartment [3].

One of the difficulties of RI servicing at the NS base and refueling is the necessity of a continuous steam supply into the steam heating system (SHS) of the primary circuit in order to provide the liquid form of coolant and to join up periodically the RI with the base installation to perform maintenance works on the coolant technology. Refueling should have been carried out in the “dry” dock (Gremiha settlement).

As far as the RIs of the NPPs are concerned, the problem of maintaining the coolant in a liquid state is not so urgent—the problem of maintaining the coolant in a liquid state is not so essential—because of existing outside power sources and the stationary arrangement of RIs.

Below is a description of the accidents mentioned during their progress, the analysis of their causes and technical measures for their elimination.

2.3.1 The RI Accident at the NS of Project 645

This was the only accident the cause of which was concerned with using the LBC. Because of the few studies of physical and chemical coolant operation processes, the substantiated specifications on impurity compositions in coolant, instrumentation for coolant quality control and equipment to provide the maintenance of required coolant qualities in the course of operation have been lacking. This level of LBC knowledge can be compared with that of water coolant when conveying water from the water pipeline into the steam boiler.

Because of that operation, uncontrollable accumulation of essential amounts of lead oxides in the primary circuit occurred, they could have formed when the pipelines of the primary circuit gas system, which were necessary for its repair, were depressurized, and thus air penetrated into the primary circuit. In addition, the primary circuit was contaminated by-products of oil pyrolysis which was the working medium for the seals of the rotatable shafts of the pumps that provided the gas leak-proofness of the primary circuit for gas. The oil was spilled into the primary circuit in great amounts because the oil seals had not been reliable enough.

When the rate of SG leakage increased suddenly (it had started some time before the accident), the oxides accumulated and other impurities filled the core, which was the cause of the violent decline of heat removal. A negative temperature reactivity effect was the cause of transfer of the automatic power control rod up to the upper switch terminal and spontaneous power reduction to 7% of nominal. This was the first symptom of the accident.

But the operational documentation did not include any necessary instructions for the operator on how to act when that kind of situation arose. Instead of resetting the emergency protection (EP) at the left side of the reactor, he followed the commander's directions (it occurred during Navy training) and tried to maintain the given power level by continuous removal of compensative rods (CR) out of the core. The all reactivity margin for 12 CR was released in about 30 minutes, although it was intended to provide the power margin generation for about 4000 full power hours. When the CRs were removed, the fuel in the local core area, where heat removal had deteriorated, melted and left the core together with the coolant flow. Signals of radiation hazards in the compartment that called for shutting down the RI and removing the crew into compartments distantly removed from the RI were not taken into account.

After this accident work on the problems of coolant technology was started. For many years these works have been carried out at a number of organizations under the scientific supervision of SSC RF-IPPE. As a result, the problem has been solved successfully and many years of experience following RIs operation has corroborated it.

Later the NS of Project 645 was decommissioned from the Navy and after special preservation of the RI and reactor compartment was sunk in the Kara Sea.

It is necessary to point out the main technical measures to eliminate the causes of such accidents:

– to eliminate the accumulation of oxides the maintenance of some excess inert gas pressure in the gas system of the primary circuit has been provided when repair works of equipment and fuel reloading are to be performed. To eliminate the possibility of air penetration into the primary circuit and radioactivity release into the environment the greatest possible tightness has been provided. For this purpose, special repairing and refueling equipment has been developed;

- the sensors of thermodynamic oxygen activity which enable measurement of the dissolved oxygen content of LBC and detect alloy oxidization at the very early phases have been designed and introduced;
- rejecting the use of the oil seals of the pump shafts and the adoption of water seals or gas-tight electric drivers of the primary circuit pumps. This eliminates oil penetration into the primary circuit and contamination of LBC by the products of oil pyrolysis:
- using the ejection system of high-temperature hydrogen regeneration that has been built into the RI and ensures chemical recovery of lead oxides by hydrogen (the explosion-proof compound of helium and hydrogen is used) and enables purification of even hard contaminated circuits from lead oxides if necessary;
- using the continuously operated system of coolant purification from irreducible impurities on the glass fabric filters;
- using the automatic system of coolant quality control which is equipped with sensors to continuously measure coolant quality and protective gas, ensures the preservation of oxide films on the surfaces of the primary circuit structure materials in contact with coolant and eliminates their corrosion deterioration as well as ensuring the early diagnostics deviations of normal states.

2.3.2 The RI Failure at the NS of Project 705 (Task Order 900)

Since the beginning of RI testing in 1970 and its further operation in 1971 and 1972, RI operation has been accompanied by the higher content of moisture in the air of the gas-tight compartment (GTC) where the RI was mounted. The tests have demonstrated that the causes of moisture were a lack of tightness of the seal of one of the SG covers because of a flaw in the nickel gasket, which was then changed. Besides, steam leakage through the steam heating system welds occurred, which had been made unsoundly, and there was no possibility of eliminating this leakage because of the compact assembly.

Because of cold surface sweating inside the GTC, the formation of water drops was the cause of wetting the heat insulator and “dry” shielding materials which contained chlorides.

The drops of water saturated with chlorides touched the primary circuit hot auxiliary pipelines made from austenite steel and gave rise to their corrosion cracking on the outside surface. This has been fully verified by the results of the RI inspection performed. Through corrosion damage of the primary circuit auxiliary pipelines at two of the three heat-exchanger loops and the impossibility of their repair because of compact assembly, the decision was taken to decommission this NS out of the Navy and carry out the RI inspection.

Thus, this accident was not concerned with LBC use. Similar accidents followed by the RI failure happened with pressurized water reactors when seawater touched the stainless steel pipelines of the primary circuit.

Technical measures eliminating the causes of such accidents are as follows:

In the design perspective RI for NPPs the pool type integral system of the primary circuit design has been used, which eliminates fully any primary circuit pipelines out of the monoblock unit vessel including comparatively thin-wall auxiliary pipelines of small diameter, there are no valves. Therefore, the ramified system of steam heating is eliminated. Fabricating the RI monoblock unit under the plant conditions ensures high quality and delivery of the reactor block available for operation. The integral design eliminates the possibility of coolant leakage almost completely. Besides that, a guard vessel is provided.

The integral design of RI ensures better conditions for performing the assembly works and controlling their quality.

2.3.3 The RI Accident at the NS of Project 705K (Task Order 105)

Global corrosion damage of the SG evaporation section pipes made from perlite steel occurred because the requirements for the water chemistry (WC) for feeding water for the SG were not met. This was as a result of the fact that under real operation conditions the method of reducing the oxygen content in the feedwater by electron-ion-exchange filter with copper-containing charge, which was provided by the Project, caused copper to escape into the secondary circuit. This was the cause of severe electrochemical corrosion of the piping system of the SG evaporation sections.

Because of the pipe damage, steam from the secondary circuit began to penetrate into the primary circuit where, after separation from coolant, it condensed in the emergency condenser (EC) specially provided in the gas system in case of SG leakage. As the internal volume of the EC had been filled by the condensate step-by-step, according to the signal the operator drained the EC many times by removing the accumulated condensate into a suitable tank, and thus he eliminated the essential pressure increase in the primary circuit gas system.

However, the EC drainage was stopped for unclear reasons. The heat-exchanging surface of the EC was completely flooded by water and condensation of the steam penetrating stopped. The pressure increase began at the primary circuit gas system. The strength of the gas system and primary circuit could bear the full working pressure of the secondary circuit. That is why in that case there could not be any leak-proofness break of the primary circuit.

Nevertheless, the leak-proofness break occurred, and it was caused by the following events. The gas pocket in the leakage reinjection pump (LRP) located in the pump tank below the LBC level had an adjusting manometer with an ultimate pressure of 4 kg per cm^2. In compliance with the instruction, if the RI was in operation, this manometer had to be shut off by the valve. The instruction requirement was violated and the valve was open. Due to this fact when the steam pressure in the gas pocket of the LRP tank reached ~6 kg per cm^2 and the LBC level in the internal pocket of the LRP increased with a corresponding increase of gas pressure, the sensitive manometer element was destroyed, gas escaped from the pump pocket, and under the steam pressure that was the cause of filling the gas pocket of the LRP with lead-bismuth alloy and its further leakage through the damaged manometer into the inhabited section of reactor compartment (the scheme is presented in Figure 2-1).

Radioactive air contamination by polonium-210 aerosols reached 10 maximal permissible concentration. Due to following the proper actions the crew irradiation and radioactive contamination were within the permissible limits. The analyses of crew bio-samples, which had been performed by the medical service, demonstrated that none of the crew had polonium-210 at more than 10 percent of the maximal permissible concentration.

RI examination showed that it could be restored. However, another decision was accepted. It was decided to change the whole reactor compartment of this NS with RI for the new one fabricated earlier. The motive for this decision was the following. In the course of this RI fabrication at the machine-building plant in Podolsk there was a faulty change of the steam heating system (SHS) pipes fabricated from high-nickel corrosion-resistant steel by the pipes being made from common stainless steel of the same size.

This error was found out after the RI unit was fabricated and it was impossible to change the pipes. Because the service life of stainless steel pipes was restricted by corrosion conditions, the decision was made to limit the service lifetime of the reactor unit to 25,000 hours and fabricate the reserve RI unit to use for changing the substandard one during plant repair works at the NS. In 1982 the service life of the SHS stainless steel pipes had expired, and that was the motive for changing the NS reactor compartment.

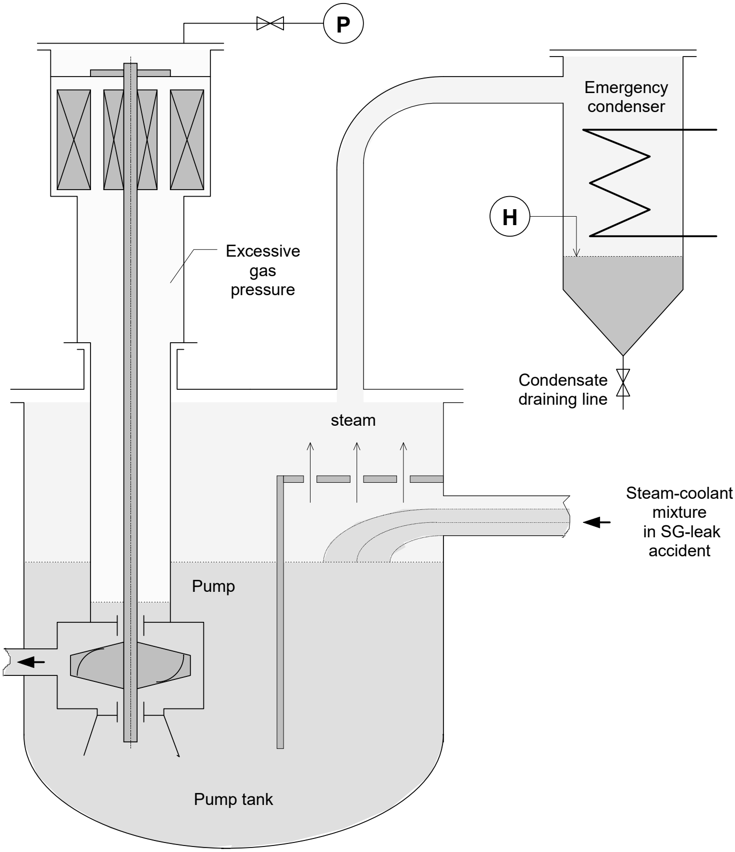

Fig. 2-1. Scheme of leakage reinjection pump.

The analysis performed has demonstrated that the cause of this accident and the accident at the NS of Task Order 900 were not concerned with the use of LBC.

The following technical measures ensure the elimination of such accidents at the RIs of the new generation:

– eliminating copper-containing materials out of the water-steam circuit;

- using more corrosion-resistant steel as a material for the SG pipes under water-steam conditions instead of low-alloyed steel of perlite class;
- providing the great extent of control-fitness and maintainability of the RI.

2.4 The Problem to Be Solved for Accelerator-Driven Systems

A number of additional problems need to be solved for ADSs:

1. Ensuring the radiation resistance of the target “window”, its strength parameters and its remote changing if the air ingress into the primary circuit and radioactivity yield into the environment occur.
2. Modifying the problem with coolant technology taking due account of slag products accumulation in coolant.
3. Ensuring the operation of target-blanket system under the conditions of iterative interruption of the accelerator proton beam or eliminating the iterative interruption of the accelerator beam.
4. Developing the technology for partial fuel sub-assemblies (FSA) reloads.
5. Designing the technology for remote fabrication of the FSA with high actinide content and heat yield.
6. Developing the questions of thermo-mechanics and thermal-hydraulics in the blanket zone adjoining the target and with high energy yield radial gradient.
7. Ensuring the radiation resistance of fuel element claddings under conditions of high-energy particles fluence effect.

2.5 Conclusions

The analysis carried out enables the following conclusions to be made.

Among the three RI accidents which happened in the early days of operating the RIs using LBC only one occurred because of LBC use (the first experimental nuclear submarine of Project 645). Its cause was the fact that at that time the comprehensive problems of coolant technology, structure materials corrosion and mass transfer in lead-bismuth circuits had not been solved. Two other accidents occurred because of manufacturing defects and operational personnel errors, which could have happened at any RI.

The causes of these accidents have been found out authentically. The causes of these accidents, which occurred at the initial stages of the development of the RIs, have been eliminated for the projected reactor installations.

List of Abbreviations

ADS	–	Accelerator-driven system
CR	–	Control rod
FSA	–	Fuel sub-assemble
EC	–	Emergency condenser
EP	–	Emergency protection (scram)
GTC	–	Gas-tight compartment
LBC	–	Lead-bismuth coolant
LRP	–	Leakage reinjection pump
LRW	–	Liquid radioactive waste
MPC	–	Maximal permissible concentration
NPP	–	Nuclear power plant
NS	–	Nuclear submarine
RI	–	Reactor installation
SG	–	Steam generator
SHS	–	Steam heating system
WC	–	Water chemistry

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