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Power reactors and sub-critical blanket systems with lead and lead–bismuth as coolant and/or target material

Utilization and transmutation of actinides and long lived fission products

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FOREWORD

High level radioactive waste disposal is an issue of great importance in the discussion of the sustainability of nuclear power generation. The main contributors to the high radioactivity are the fission products and the minor actinides. The long lived fission products and minor actinides set severe demands on the arrangements for safe waste disposal.

Fast reactors and accelerator driven systems (ADS) are under development in Member States to reduce the long term hazard of spent fuel and radioactive waste, taking advantage of their incineration and transmutation capability. Important R&D programmes are being undertaken in many Member States to substantiate this option and advance the basic knowledge in this innovative area of nuclear energy development.

The conceptual design of the lead cooled fast reactor concept BREST-OD-300, as well as various other conceptual designs of lead/lead–bismuth cooled fast reactors have been developed to meet enhanced safety and non-proliferation requirements, aiming at both energy production and transmutation of nuclear waste. Some R&D studies indicate that the use of lead and lead–bismuth coolant has some advantages in comparison with existing sodium cooled fast reactor systems, e.g.: simplified design of fast reactor core and BOP, enhanced inherent safety, and easier radwaste management in related fuel cycles. Moreover, various ADS conceptual designs with lead and lead–bismuth as target material and coolant also have been pursued. The results to date are encouraging, indicating that the ADS has the potential to offer an option for meeting the challenges of the backend fuel cycle.

During the last decade, there have been substantial advances in several countries with their own R&D programme in the fields of lead/lead–bismuth cooled critical and sub-critical concepts, coolant technology, and experimental validation. In this context, international exchange of information and experience, as well as international collaborative R&D programmes are becoming of increasing importance. It is with this focus that the IAEA convened the Advisory Group Meeting on Design and Performance of Reactor and Sub-critical Blanket Systems with Lead and Lead–Bismuth as Coolant and/or Target Material, in co-operation with the Research and Development Institute of Power Engineering (RDIPE), the Institute of Physics and Power Engineering (IPPE) and Minatom in the Russian Federation.

This meeting, which assembled sixteen participants from eight countries, drew upon the vast experience of a group of international experts in order to review and discuss the recent R&D developments in critical and sub-critical concepts, coolant properties, and experimental and analytical validation work, as well as to exchange information on the experience accumulated, and to discuss the issues requiring further R&D. A total of twenty-four presentations and/or statements were made by the participants.

The IAEA expresses its appreciation to all the participants of the Advisory Group Meeting for their valuable contributions and also to the Member States that have made available experts to assist and participate in this meeting. The IAEA officer responsible for this publication was Young-In Kim of the Division of Nuclear Power.

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SUMMARY

1. INTRODUCTION

The Advisory Group Meeting (AGM) on Design and Performance of Reactor and Sub-critical Blanket Systems with Lead and Lead–Bismuth as Coolant and/or Target Material was held in Moscow, Russian Federation, 23–27 October 2000. The AGM was convened by the IAEA and was hosted by the Research and Development Institute of Power Engineering (RDIPE) in Moscow, in co-operation with the Institute of Physics and Power Engineering (IPPE) in Obninsk, and Minatom. Its purpose was to provide a forum for international information exchange on all the topics relevant to lead (Pb) and lead–bismuth (Pb–Bi) cooled critical and sub-critical reactors. In addition, the AGM aimed at: (1) finding ways and means to improve international co-operation efforts in this area; (2) obtaining advice from the Member States with regard to the activities to be implemented in this area by the IAEA, in order to best meet their needs; and (3) laying out the plans for an effective co-ordination and support of the R&D activities in this area.

Attendance at the AGM included sixteen participants from eight countries (Belgium, China, India, Italy, Japan, Kazakhstan, the Republic of Korea, the Russian Federation) and one international organization (IAEA). Presentations and/or statements with regard to the status of the respective R&D programmes were made by all participants.

An important result achieved at the AGM was extended information exchange on the recent R&D developments in the following fields: critical and sub-critical concepts, coolant properties, experimental and analytical validation work. The fast reactor concept BREST-OD-300 (Russia), as well as various conceptual designs of heavy liquid metal cooled fast reactors pursued in Japan were presented. Research and development work on hybrid (accelerator driven) sub-critical systems were presented by the Republic of Korea (HYPER), India (fast-thermal hybrid accelerator driven sub-critical system ADS), Italy (European “energy amplifier demonstration project”), Belgium (development of an ADS R&D experimental facility), China (study on ADS Pb/Pb–Bi spallation target physics), Sweden (investigations of void coefficients in sub-critical lead–bismuth cooled blankets), and Kazakhstan (basic nuclear data for ADS).

Important results were further achieved during the visit to the IPPE in Obninsk on 25 October 2000. An extensive scientific and technical exchange covering the following topics was held during this visit: spallation targets, sub-critical target/blanket systems, multi-purpose small fast power reactors with lead–bismuth cooling (modular design), lead–bismuth thermal-hydraulics, and steel corrosion resistance in lead–bismuth. Of particular interest were the visits in two IPPE laboratories: the experimental facilities for heavy liquid metal coolant corrosion tests, and the experimental facilities for heavy liquid metal coolant (HLMC) technology.

2. STATUS OF DESIGN DEVELOPMENT AND PERFORMANCE

2.1. Heavy metal cooled fast reactors

2.1.1. Japan

The conceptual design study on various types of HLMC reactor plant concepts was conducted in JNC. Through preliminary design study, JNC identified some technical features on each concept and roughly evaluated economical competitiveness based on the total weight of

NSSS. Finally, Pb–Bi cooled medium tank type reactor was selected as a most promising concept. JNC insisted that lower melting point of Pb–Bi coolant makes it more attractive than Pb coolant and the amount of Bi resources seems not to be a significant issue based on roughly but conservative estimation. JNC is now performing design study on a medium sized and complete natural circulation cooled reactor with Pb–Bi coolant. This concept takes advantage of superior natural circulation characteristics of Pb–Bi and seeks simplification of NSSS and BOP designs by employing passive features.

2.1.2. Russia Federation

The conceptual design of a fast reactor BREST with (U, Pu)N fuel and lead coolant has been developed such as to consistently implement the principles of natural safety without a sufficient deviation from materials and technology which were proven in defense and civil nuclear power facilities. Designs for reactors with the power in the range of 300 MWe (BREST-300) to 1200 MWe (BREST-1200) were optimised and proved to meet safety and non-proliferation requirements. The use of lead coolant in the BREST reactor designs allows the following design features: single vessel or pool type arrangement without a metal vessel, two circuits in the main and emergency cooling systems; no special system to wash off coolant from fuel assemblies during refuelling; improved passive control and protection features with threshold response; simpler design of steam generators; and simpler rooms in the cooling circuits and other nuclear power plant constructions.

Neutronic experiments were performed at U-Pu-Pb critical assemblies to validate reactor physics and revise the nuclear data. Experimental work on lead coolant technology and in-pile testing of nitride fuel to study the corrosion of nitride fuel and structural materials in lead have been conducted and are being conducted. The BREST-300 engineering design is now in progress. Specific projects on R&D of closed nuclear fuel cycle technology, and development and construction of a demonstration power plant as well, have been suggested at the international collaboration level.

Mononitride (U, Pu)N fuel was chosen for BREST-300 due to favorable properties such as its high actinide density and high thermal conductivity. The use of lead bonding allows operating under so called cold fuel conditions, i.e., with maximum temperatures below one third of the melting point. For the BREST design, a moderate average burnup of 6% average is foreseen. Two fabrication processes — carbothermic reduction of oxides and direct nitriding of metals — were developed at the Bochvar Institute (VNIINM). Oxygen and carbon impurities resulting from the carbothermic reduction method could be surpressed down to 0.2% weight fraction. The metal nitriding method is still preferred due to the possibility of obtaining pellets with densities above 90% after sintering. The lead bonding concept was validated by heating lead filled fuel pins up to 800°C inner cladding temperatures for 2000 hours. No chemical reaction between lead and cladding was observed.

The irradiation experience of nitride fuels in Russia encompasses 15 years of operation of a UN core in BR-10. For (U, Pu)N, the highest burnup reached was 6%. A test of the high temperature performance was made by irradiating He bonded (U, Pu)N at 1045 W/cm up to sign of liquid metal or nitrogen gas formation was seen in subsequent destructive examinations. (Pu, Zr)N fuel was also fabricated and irradiated successfully up to 8% burnup in BOR-60. On-site pyroprecessing of spent BREST fuel is foreseen. KCl and LiCl were used for testing the performance on UN, 99% recovery efficiency was achieved, even though no recovery from the electrolyte was performed. One still has to decide whether to directly recover U and Pu metal from the cathode or to use a liquid process for the purpose. The issue of PuN decomposition was raised in the discussion, but it was assured that no observation of metal or gas formation in Russian nitride irradiation had been made. It appears that the moderate burnup projected for the BREST fuel justifies the high pellet density (as compared to the west/European recommendation of 85% TD), but laboratory tests of the high temperature stability of the BREST fuel would be required to ensure that the fuel performs as expected under accident conditions.

The sodium cooled loop type reactor BOR-60 has been designed to accommodate one experimental channel in the fifth row of its fuel core for testing BREST fuel. The experimental loop consists of a mockup of the fuel assembly containing 4 complete fuel rods. The fuel assembly is cooled independently from the reactor core primary system by a flow of liquid lead. The lead is pumped, and is cooled by the sodium that flows inside the reactor through a double wall arrangement. The oxygen content is constantly monitored and controlled, as well as the eventual release of fission gases (Xe and Kr) revealing a pin failure. The auxiliary systems have been already mounted and fully tested while the loop will be installed in the reactor in January 2001 after delivering of the structural materials to be tested in it. The experiment is planned to be run for 2 years reaching 4% burnup in the uranium–plutonium nitrate fuel with an average spectrum of 234 keV and flux of 10^{15} n/cm^2 s. For comparison, the maximum and average discharge fuel burnup foreseen in the BREST-OD-300 reactor will be 10% and 6%, respectively. After this two-year experiment no other plans have been made for the BOR-60 reactor experimental channel and it could be in principle available to the international community for a new experiment. Many laboratories have already showed a strong interest in this loop. The lead loop will not be available because it will be completely dismantled to be analyzed after the irradiation.

The small fast reactor SVBR-75/100 design was developed on the basis of reactor operation experience obtained from Pb–Bi cooled nuclear submarines. It was shown that such a reactor concept that meets safety and nonproliferation requirements, could have multi-purpose uses, e.g., as nuclear power desalination plants in developing countries, and as part of large modular nuclear power plants in developed countries.

2.2. Accelerator driven sub-critical system

2.2.1. Belgium

The ADS facility was proposed in the MYRRHA project by SCK•CEN. The nuclear waste, and particularly the minor actinides and long-lived fission products, are one of the issues affecting the acceptance by the public of nuclear energy. Because it is controlled by an external source, the ADS is seen as a promising tool towards the transmutation of those materials. MYRRHA is intended to be a first step towards an ADS-DEMO facility. The MYRRHA concept aims at providing an irradiation facility for ADS related research: structural materials, fuel, liquid metals, sub-critical reactor physics, and subsequently on applications such as the transmutation of waste. The main originality of the MYRRHA concept resides in its windowless spallation source. The objective of a fast neutron flux ($\sim 10^{15}$ n/cm^2 s) on a low power machine (30 MW) leads to a very compact core and spallation source. The project is presently in a pre-design phase. The purpose in this phase is to identify the main technical issues, to present a machine configuration and provide realistic cost estimates for the construction. International cooperation has already been set up on particular topics and it is the will of SCK•CEN to further broaden this cooperation in the future. The pre-design phase is expected to be completed in spring 2001. The detail design, engineering, safety studies, construction and licensing will start in 2008 as final objective for the first operation.

2.2.2. India

India considers the Pb–Bi coolant as an essential part of the ADS design. The fluid dynamics and heat transfer characteristics of Pb–Bi are being investigated now. Two types of cooling methods are considered — natural convection and other by gas injection. India also has interest in benchmarking computational studies in heavy metal coolant systems which have been done so far. For this objective, some experiments have been planned with heavy metal coolant loops with mercury in the near future and with Pb–Bi in the years to come. In this programme, the international collaboration is invited. The area of priority will be to gain experience and data on properties of structural steels which are compatible with heavy metal coolants. This is required for the construction of system equipment and the running of heavy metal coolant process systems. India is inclined to join such activities on partnership basis.

2.2.3. Italy

The Italian ADS programme is divided into two subprogrammes, which strongly interact one with each other and are based on the C. Rubbia design of an Energy Amplifier. The first is a fundamental R&D programme named TRASCO (from the Italian acronym of waste transmutation) and the second is an industrial programme related to system design. The TRASCO programme is further divided in two parts: the first part, lead by INFN, is devoted to the development of a 1 GeV, 30 MeV proton linac accelerator based on RFQ, DTL, and super-conducting cavities technology. The second part is devoted to the sub-critical system. In this framework the CHEOPE and LECOR facilities were built at ENEA's Brasimone site. They will be used for testing the basic technology related to Pb–Bi coolant (loading/unloading operations, oxygen control, instrumentation, and filtering) and the compatibility of structural materials. Experiments are also planned for the MEGAPIE and the MYRRHA projects. The TRASCO programme has been already extended, and ENEA is building the first large Pb–Bi pool facility in the world, CIRCE, that is currently being installed in the old PEC reactor facility of Brasimone and will be put into operation in the second half of 2001. It consists of a 9 m high and 1.2 m large pool filled with 100 t of Pb–Bi where a series of tests will be done devoted not only to test basic phenomenology (such as gas injection for natural circulation enhancement, oxygen monitoring and control in pool conditions), but also component mock-ups of the demonstration facility proposed by the Italian community. The use of the PEC facility will be also available for international collaborations. The system design studies which are performed by Ansaldo, CRS4, ENEA and INFN, aim at completing a pre-engineering design of a 80 MW Pb–Bi ADS demonstration facility which will be the Italian proposal for a European demonstrator to be built in the framework of a European collaboration.

2.2.4. Kazakhstan

The Institute of Nuclear Physics (INP) in Republic of Kazakhstan has been performing some research for the multi-purpose ADS as well as for fast reactors in terms of nuclear data and material science. Some new material data on physical and mechanical properties of irradiated structural materials of BN-350 have been obtained. New approaches and models applicable to the calculation of spallation residues and fission fragments have been developed.

2.2.5. Japan

The ADS core design performed by JAERI was analyzed in terms of void worths as function of pin pitch. It was shown that Pb–Bi coolant always yields lower void worths, with a transmutation from positive to negative worths for pin pitches exceeding 1.5 times pin diameters. It was emphasized that Monte Carlo methods need to be used in order to obtain correct void worth values. In the Sing Sing core design performed at RIT, boron carbide is introduced into the core in order to minimize neutron capture rates on the minor actinides. Hence helium production in the fuel can be surpressed, and reactivity losses are mitigated. The price paid is an increase in void worths that become positive even using Pb–Bi coolant and large pin pitches. The advantage of Pb–Bi over sodium is still clear though, allowing operating the core sub-criticality with a k-eigenvalue 3000 pcm higher for the Pb–Bi cooled version. There is a wellknown uncertainty in the inelastic cross sections evaluations of Pb void worths, and is as such very significant. Uncertainties concerning the minor actinides cross sections could also affect these evaluations. It was pointed out that the scenario of having local voids in the center of the core, induced by, e.g., the release of fission gas from fuel pins should be investigated.

2.2.6. Republic of Korea

A conceptual design of HYPER system and related R&Ds are being performed in KAERI, aiming at the transmutation of nuclear waste and the energy production. Some major design features of the Pb–Bi cooled HYPER system have been developed through the investigation of various transmutation system concepts in terms of the transmutation capability, safety and the proliferation resistance of related fuel cycles.

The core is designed to produce 1000 MW_{th} with reflector assemblies filled with liquid lead being located at the core perimeter. Burnable absorbers are designed to surpress a large reactivity swing resulting from a relatively small amount of fertile in the HYPER core. Either TRU-Zr alloy or (TRU-Zr)-Zr dispersion metallic fuel is under consideration, based on the pyroprocessing process. From the trade-off studies for fuel fabrication, a dispersion metallic fuel is being considered more favorable for the HYPER system in terms of high discharge burnup. TRU and fission products (FP) will be loaded separately for incineration. TRU is loaded as a fuel to drive the system and FPs are loaded as a target (FP assembly) to be irradiated in a special region of the reactor core. Long-lived FPs such as ^{99}Tc- and ^{129}I will be incinerated using a localized thermal neutron flux obtained by inserting some moderators in the FP assembly. FP assembly (target) configurations are different for Tc and I, respectively.

A three loops system with Pb–Bi coolant is chosen for the preliminary design of HYPER cooling system. The Pb–Bi core coolant is also used as a spallation target. The beam target design is optimized for stable and safe operation of the target and for reasonable lifetime of the beam window. Beam diameter and window thickness are determined based on the simulation results.

There are few experimental data available for justifying the TRU-Zr metallic fuel for the HYPER system. In addition, the development of Pb–Bi coolant/target technologies is very challengeable in the HYPER system design. Experiments to solve the key technical issues related with TRU-Zr fuel and Pb–Bi coolant/target will be performed from 2001 to 2003 and the conceptual design of the HYPER system will be completed by 2006.

2.2.7. Russian Federation

The test loop TS-1, the first target complex in the world, has been designed by IPPE and will be operated for testing a liquid metal Pb–Bi cooled target in the proton beam of the linear accelerator in the Los Alamos National Laboratory (LANL).

3. CONCLUSIONS

The AGM stressed that nuclear energy is a realistic solution to satisfy the energy demand, considering the limited resources of fossil fuel, its uneven distribution in the world and the impact of its use on the planet, and the expected doubling of the world population in the 21st century and tripling of the electricity demand (especially in the developing countries). However, the AGM concluded that the development of innovative nuclear technologies must be pursued meeting the following requirements: (a) deterministic exclusion of any severe accident; (b) proliferation resistance; (c) cost competitiveness with alternative energy sources; (d) sustainable fuel supply; and (e) solution of the radioactive waste management problem.

4. RECOMMENDATIONS

On the basis of the technical presentations and discussions, both at RDIPE and during the visit at IPPE, the AGM has identified the following areas of mutual interest:

- Corrosion resistance, embrittlement and radiation ageing of structural materials in lead, as well as in lead–bismuth coolants;
- Technology of advanced fuels, such as nitride plutonium–uranium fuel, including their irradiation testing in fast research reactors;
- Fuel reprocessing techniques that would support the non-proliferation regime and would be unsuitable for the production of weapons-grade materials;
- Radioactive waste treatment techniques, including transmutation of minor actinides and long-lived fission products in fast reactors and/or ADS;
- Nuclear data libraries, codes, and experimental validation studies.

Considering the importance of development work in the area of innovative nuclear technologies, the participants in the AGM suggested that wide bilateral and multilateral collaborations on the above mentioned issues would greatly benefit national efforts.

More specifically, the following R&D topics that would benefit from an international collaboration were identified:

1. Updating of the heavy metal nuclear data
2. Fast reactor and ADS fuels and material development;
3. CRP on computational fluid dynamics methods and computer code validation;
4. Nitride fuel experiments in BOR-60;
5. Information exchange on the operational experiences of heavy liquid metal coolant (HLMC) experimental facilities as well as reactors, and discuss the creation of a reliability database for HLMC components.

NUCLEAR POWER STRATEGY: REQUIREMENTS FOR TECHNOLOGY

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Abstract

Need and feasibility of large-scale nuclear power technology in the future nuclear power development have been considered from the viewpoints of contemporary requirements for sustainable fuel resources, small environmental impact, enhanced safety, improved economics competitive to fossil fuel technologies, political neutrality and geopolitical resistance. To challenge and resolve these global requirements, a large-scale fast reactor concept with a closed fuel cycle which can implement the principle of natural safety and can embrace non-proliferation regime as well, has been proposed. In this context, the development of large-scale nuclear power technology has been offered for its gradual transition to large-scale nuclear power. Joint development efforts at an international level are substantial to achieve success in this nuclear power technology development.

1. THE POSSIBLE ROLE OF NUCLEAR POWER

Speaking about the possible role of nuclear power in sustainable world development, there are at least three questions to be answered:

1. Is large-scale nuclear power essential to future development?
2. Is it feasible to have modem nuclear power transformed for large-scale deployment?
3. When will large-scale nuclear power be practically needed?

These questions are placed in the order of their increasing complexity. The first among them is indeed the simplest. To have it answered, let us formulate the requirements for a large–scale power industry and see which of the existing and emerging energy technologies can meet them.

Such requirements are fairly obvious:

- Sustainable (unlimited) fuel resources;
- Sustainable environmental performance (small waste quantities);
- Sustainable accident resistance (exclusion of severe accidents at power units and fuel cycle facilities, presenting a threat to public safety);
- Political neutrality (minimized political restrictions on materials involved in the fuel cycle);
- Sustainable economics (competitiveness against "unsustainable" fossil fuel technologies);
- Sustained geopolitical resistance (availability to developing countries).

The energy technologies complying with all these requirements will be referred to as sustainable large–scale technologies.

Let us have a look now at present–day energy technologies in the context of the above requirements. Their characteristics are following:

- Fossil fuel power and nuclear power using thermal reactors are basically unsustainable energy technologies;

– Alternative energy technologies (solar, wind, geothermal, tidal, biomass, etc.) are basically unfit for large-scale deployment;
– Thermonuclear fusion is not an energy technology of the 21st century;
– Nuclear power based on fast reactors has the greatest potential for large–scale deployment and sustained operation.

Thus, we have actually answered the first question concerning the basic need for large–scale nuclear power, as there appears to be no reasonable alternative to it – at least in the 21st century. But large-scale nuclear power will not be produced by a mere increase in the scale of contemporary nuclear power based on thermal reactors.

Let us address the second question of whether it is feasible to transform modern nuclear power for large–scale deployment. It will not be an exaggeration to say that all the problems of modern nuclear power are in one way or another related either to its military descent or to radwaste. Let us attempt to analyze the origins of these problems.

The military origin and inertia of technical thought have shaped up the perception of nuclear power as a technology of harnessing he nuclear explosion energy and as a technology for production of nuclear weapons. Due to concerns about radwaste and the "coal–oil" thinking, closing of the nuclear fuel cycle is viewed as a technology involving burial of large quantities of long-lived high-level waste.

2. THE REQUIREMENTS FOR FUTURE NUCLEAR POWER

The future of nuclear power depends on the answers to two cardinal questions. First, is nuclear power able to get rid of its birthmarks? This question can be expanded as:

– Can we deal with the feared possibility of a nuclear reactor turning into an explosive device by making it technically impossible? and
– Can we rule out in practice the possibility of illicit diversion of nuclear fuel cycle materials for military or terrorist purposes?

Another cardinal question: Is it feasible to have a closed nuclear fuel cycle with negligible impacts on the biosphere? In an expanded form this question can read as:

– Is it feasible to have low-waste spent fuel reprocessing? and
– Is it possible to make radwaste sufficiently safe for burial in geological formations?

If even one of the answers is negative, mankind is bound to give up such a technology sooner or later. We are interested, of course, in the affirmative answers. But are they possible? Yes, they are!

Let us consider what has to be done to exclude severe accidents involving a radiation threat to population.

A change-over from the concept of "defence against dangerous power plants" to "rejection of dangerous plants" implies development of a nuclear reactor with conclusively demonstrated ("deterministic") exclusion of severe accidents such as prompt criticality excursion, steam and hydrogen explosions.

According to Russian studies the greatest potential for "deterministic" exclusion of severe reactor accidents is offered by fast reactors without U–blanket, having a high-boiling liquid metal coolant.

There is a way towards making nuclear power politically neutral: switch–over from “political control over nuclear power” to “politically neutral nuclear power” calls for technological support of non-proliferation regime, including:

- Elimination of U-blanket in fast reactors with core breeding ration (CBR) ~1;
- Separation of pure Pu to be eliminated from spent fuel reprocessing;
- Phaseout of enrichment technologies.

Let us present requirements for an environmentally acceptable nuclear fuel cycle. Change-over from the concept of "clean fuel–dirty waste” to the concept of “dirty fuel–clean waste” implies:

- Spent fuel reprocessing with separation of fuel (U, Pu), acitinides (Am, Cm), long lived fission products;
- Transmutation of Am, ^{99}Tc, ^{129}I (the low activity of the latter two allows their transmutation till future times);
- Storage of short–lived fission products for ~200 years;
- Radiation–equivalent disposal.

Let us identify some of the stepping–stones for radiation equivalent radwaste management. Investigations have shown:

- Small waste quantities in “dry” (high–temperature) reprocessing;
- Efficient transmutation in fast reactors without U-blankets;
- Physical possibility of a radiation balance between buried waste and mined uranium.

Its technological feasibility and economic expediency are yet to be demonstrated.

All the safety requirements placed upon a new nuclear energy technology are embraced by the notion of “natural safety”. The concept of natural safety is an extension of the inherent safety principles, including:

- Deterministic elimination of severe reactor accidents;
- Technological support of non-proliferation regime;
- Low waste reprocessing of spent fuel with radiation–equivalent waste disposal.

3. REQUIREMENTS FOR NEW NUCLEAR POWER TECHNOLOGY

Let us group internal and external requirements to the new nuclear energy technology.

- Internal requirements:
  - Natural safety;
  - Fuel breeding, i.e., fast reactors with a closed fuel cycle.
- External requirements:
  - Competitiveness;
  - Availability to developing countries.

In respect of competitiveness, the requirements to the new nuclear technology are as follows:

- It should be made competitive in the next few decades, rather than in some indefinite future when cheap fossil resources are exhausted and quotas or penalties for greenhouse. Gas emissions are established;
- NPPs with fast reactors of a new generation should cost less than modern LWR plants.

Studies made in Russia have shown that the cost of NPPs can be drastically reduced only through implementing the concept of natural safety.

Another external requirement to the new nuclear technology is a new nuclear technology should be made available to developing countries. It is an impotent factor of geopolitical stability. The decision to give up developing fast reactors of a new generation with a closed fuel cycle:

– Adds to the risk of international conflicts over the diminishing rich fossil fuel sources;
– Confines the developing nations to economic backwardness, by overburdening their economy with fossil fuel production and transportation;
– Compels the developing countries to create their national nuclear technologies along traditional lines, which may lead to loss of control over non-proliferation regime.

Thus, we have an answer to the second question posed at the beginning of this presentation: it is feasible to transform today's nuclear power for large–scale deployment based on fast reactors with a closed fuel cycle by consistently implementing the principle of natural safety.

4. LARGE-SCALE FAST REACTOR DEPLOYMENT

We are yet to address the third, most difficult question: when will large–scale nuclear power be in demand? On the other hand, warning predictions exist which apparently should have caused efforts to create a technological foundation for large–scale nuclear power in the near future. Such predictions are following:

– Demographic:
  - Population doubling by mid–century (mostly in developing countries).
– Energy:
  - Doubling of primary energy requirements and trebling (to 6000 GWe) of electric energy requirements by the year 2050.
– Environmental:
  - Industrial pollution of biosphere;
  - Greenhouse effect (fossil energy sector).
– Political:
  - Proliferation of nuclear weapons (Pu of weapon and energy origin).
– Geopolitical:
  - International conflicts over dwindling rich oil and gas sources;
  - Backwardness of many national economies saddled with fuel production and transportation.

On the other hand, out of the two options for nuclear power development in the 21st century:

– No-growth option: with nuclear power frozen at the present–day level;
– Development option: with large–scale deployment of nuclear energy technology.

The former, consisting in limited development involving reactors of present–day types, is gaining in popularity.

This is the more surprising as none of the burning problems faced by nuclear power can be resolved within the framework of the no–growth option.

Some of these problems are following:

– Safety;
– Non-proliferation (including Pu disposition);
– Competitiveness;
– Spent fuel and radwaste management;
– Fuel availability.

All of them are facets of one fundamental challenge: the future of nuclear energy. The adequacy and effectiveness of today's actions taken to resolve the above specific problems, depend on our vision of the nuclear power in the 21st century.

In the no growth case, the safety problem is dealt with essentially by building up engineered safety features. In the development case, the same problem is resolved by implementing the concept of natural safety.

In the no-growth case, the problem of non-proliferation is reduced to political consolidation of the international non-proliferation regime. And the problem of Pu disposition is treated as technical and economic problems of Pu storage and as a political problem of reducing its stockpiles.

In the nuclear power development case, the non-proliferation problem is addressed as follows:

– By transferring spent fuel from storages to the reactors and fuel cycles facilities affording best protection against Pu theft or illicit diversion;
– By integrating Pu from reduced warheads and spent fuel into the closed cycle of fast reactors, balanced in terms of Pu, with technological support of non–proliferation; in such a case, primary Pu separation from spent fuel of thermal reactors and fabrication of first cores for fast reactors of the new generation are assumed to take place either at reprocessing facilities of nuclear countries or at special nuclear technology centres under international jurisdiction;
– By transferring thermal reactors into a Th–U cycle balanced out for ^{233}U (technological support of non-proliferation regime, with spent fuel reprocessed without separation of ^{233}U);
– By abandoning the technologies of U enrichment, Pu and ^{233}U separation (technological foothold for non-proliferation of nuclear weapons).

In this case, plutonium is utilized as part of mixed uranium–plutonium fuel in:

– Operating fast reactors,
– Fast reactors of a new generation (the basis of future nuclear power).

In the no-growth case, the problem of spent fuel management amounts to:

– Long–term storage:
  - National;
  - International.
– Pu recycling in thermal reactors is inexpedient:
  - Strategically (loss of capability for nuclear power development);
  - Economically (with cheap U);
  - Environmentally (additional exposure during MOX fuel fabrication);
  - Technologically (additional demonstration of nuclear safety).

With the nuclear power development option, spent fuel management includes the following aspects:

- Spent fuel reprocessing with separation of U and Pu, minor actinides (Am, Cm), long–lived fission products (^{137}Cs, ^{90}Sr, ^{99}Tc, ^{129}I);
- Transmutation in fast reactors;
- Actinides (Am), long–lived fission products unsuitable for industrial or medical uses (^{99}Tc, ^{129}I: due to their low activity this operation can be postponed till distant future);
- Storage of short-lived fission products for ~200 years;
- Radiation-equivalent disposal of radwaste;
- Radioactivity balance between buried waste and U ore extracted from the earth (assuming that buried waste will contain no more than 0.1% of Pu, Am, Cm found in spent fuel, up to 1% of ^{137}Cs, ^{90}Sr, ^{99}Tc, ^{129}I).

And finally, the fuel problem in the no growth case is viewed as the problem of closing the fuel cycle of LWRs based on the use of MOX fuel. It should be borne in mind here that:

- Pu recycling in thermal reactors as a means to increase their resources is inexpedient technologically, economically and environmentally;
- Uranium saving to be gained from closing the fuel cycle of LWRs is small (~20%) and cannot serve as a decisive point in favour of such conversion;
- Closing of the LWR fuel cycle with once-through use of MOX fuel would halve the chances for fast reactors to be brought in later.

Moreover, multiple Pu recycling in LWRs eliminates all possibilities for large–scale nuclear power development based on fast reactors.

In the nuclear power development case, the fuel problem is tackled by closing the fuel cycle of fast reactors. In such a case:

- The techniques of choice for reprocessing of fast reactor fuel are nonaqueous processes: electrolysis of molten salts and gas–phase fluorination, in which the USA (ANL, Idaho) and Russia (NIIAR) have amassed substantial experience.
- By going from the oxide option to fuel of higher density and heat conductivity with equilibrium composition of fuel (× ~10% for Pu) in rods of d ~10 mm, by getting rid of U blankets (BR = CBR ~1), and by resorting to “dry” reprocessing of spent fuel without U and Pu separation, we can count on a reduction in the costs of closed nuclear fuel cycle technologies.

Thus, it has been shown that nuclear power can successfully tackle its problems by taking the development course. Gradual large–scale development of nuclear power offers solutions to global challenges of:

- Energy production increase;
- Prevention of anthropogenic buildup of greenhouse gases in the atmosphere;
- Restoration of nuclear power competitiveness and simultaneously to its internal problems which are:
  - Consolidation of non–proliferation regime;
  - Environmentally acceptable radwaste management;
  - Elimination of accidents presenting a radiation threat to population.

5. CONCLUDING REMARKS

It should be specially emphasized that development of nuclear technology for gradual transition to large–scale nuclear power is impossible without wide international cooperation. Joint efforts to meet both national and global energy requirements are called for.

Russia is ready to suggest specific projects, already started at the national level:

- Research and development of closed nuclear fuel cycle technology with radiation-equivalent disposal of radioactive waste and technological support of non–proliferation;
- Development and construction of a demonstration power plant with a fast reactor and pilot fuel cycle facilities in conformity to the concept of natural safety.

Proponents of this approach in Russia have been working along these lines for a number of years already and the progress made to date is briefly described below:

- Work on the lead coolant technology – at an advanced stage;
- Research on corrosion of nitride fuel and structural materials lead – well advanced;
- BREST-300 engineering design – in progress;
- Site for construction of the demonstration plant of natural safety – chosen;
- Feasibility study for the commercial BREST-1200 unit – in progress.

NUCLEAR POWER OF THE COMING CENTURY AND REQUIREMENTS TO THE NUCLEAR TECHNOLOGY

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Abstract

Current state of nuclear power in the world has been considered and the reasons for its falling short of the great expectations relating to its vigorous development in the outgoing century are considered. Anticipated energy demand of the mankind in the next century is evaluated, suggesting that with exhausted resources of cheap fossil fuel and ecological restrictions it can be satisfied by means of a new nuclear technology meeting the requirements of large–scale power generation in terms of safety and economic indices, moreover, the technology can be elaborated in the context of achievements made in civil and military nuclear engineering. Since the developing countries are the most interested parties, it is just their initiative in the development of nuclear technology at the next stage that could provide an impetus for its actual advance. It is shown that large–scale development of nuclear power, being adequate to increase in energy demand, is possible even if solely large NPPs equipped with breeders providing BR ≥1 are constructed. Requirements for the reactor and fuel cycle technologies are made, their major aspects being: efficient utilization of Pu accumulated and reduction of U specific consumption by an order of magnitude at least, natural inherent safety and deterministic elimination of accidents involving high radioactive releases, assurance of a balance between radiation hazard posed by radioactive wastes disposed and uranium extracted from the ground, nuclear weapons due to fuel reprocessing ruling out potentiality of Pu diversion, reduction of the new generation reactor costs below the costs of today's LWR.

1. INTRODUCTION

The doubling of the global population expected by the mid–century, mostly due to the developing countries, and the ever–increasing number of nations taking the course of industrial development are bound to cause at least doubling of the world's demand for primary energy and trebling of the demand for electricity.

The growth of energy production will be in all probability accompanied by gradual depletion of cheap hydrocarbon reserves and by a rise of their prices. The world fuel market is likely to be increasingly affected by the endeavors of various countries to preserve their national hydrocarbon resources as a chief item of apart, for many of them, and as fuel for transport and raw materials for chemical synthesis.

The looming hazard of international conflicts around oil and gas sources is another factor to be reckoned with.

Use of fossil fuels, including huge coal resources, can prove to be closer to its end than is currently expected, on account of the emissions of combustion products and global climatic changes. Besides the rise of fuel prices, measures taken to minimize harmful releases are bound to add to the capital costs of the energy sector.

The main provision of the long time reliable energy supply would be the development of new energy technologies capable of large–scale and economical replacement of fossil fuels.

With half a century of practical experience behind them, fission reactors might seem to be an eligible and realistic alternative to the conventional energy sources. But deployment of thermal reactors fueled with ^{235}U is limited by the resources of cheap uranium reactors are assessed at somewhat more than 10^7 t, which in terms of the energy equivalent is less than the estimated resources of oil and gas, let alone coal. This means that ^{235}U–fueled reactors are incapable of having a greater effect on the global consumption of conventional fuels. With its current share in electricity production, nuclear power based on traditional reactors, mostly LWRs, can go ahead for another ~40 years to supply the demands of fuel–deficient countries and regions. Nuclear power can be deployed on a much larger scale, using fast reactors – which only some 20 or 30 years ago was believed attainable within this century. However, contrary to expectations, the first generation of fast reactors proved to be much more expensive than LWRs. It is only to be regretted now that the root causes thereof were never brought to light with the result of an ingrained prejudice against fast reactors stamped as inevitably costly machines. The hazard of proliferation of nuclear weapons is also associated with fast reactors and the closed fuel cycle.

For these reasons as well as the grave accidents at TMI and Chernobyl, as energy saving measures and others the expected large–scale deployment of nuclear power was never brought into effect, deferred till some not entirely definite time in the future.

2. ENERGY TECHNOLOGY OF THE NEXT CENTURY

Meanwhile, studies show that it is possible to create a nuclear technology which will meet the safety and economy requirements of large–scale power production without going too far from the achievements of civil and military nuclear engineering. If in the next few years, the States concerned recognize the vital need for resolving the problem, and doing so in good time, and if they succeed in adopting a definite concept, the engineering development and demonstration of the latter can be carried out within a reasonable period of about 20 years. Thus the stage would be set for a new nuclear power which in the next century could take on a major part – say, half – of the increase in the global demands for fuel and energy. This means that nuclear power should grow by an order of magnitude from its present–day level of ~340 GWe by the mid–century and then double or treble its capacity before the end of the century.

Nuclear technology development has long since acquired the traits of an international effort and in the century to come will be guided by the global energy requirements, with countries concerned joining forces to carry it on. In so far as such a need is now greatest in the developing countries, it is their initiative in developing a nuclear technology for the next stage that can really a practical turn to work.

Such an initiative would undoubtedly find support with Russian nuclear experts whose vast experience and capabilities are currently in poor demand in the country, as well as with specialists from other countries seeking applications for their expertise.

Creation of a new nuclear technology would also answer the fundamental needs of industrialized nations and ought to be supported by their governments with the understanding that this technology would not add to the risk of a sprawl of nuclear weapons.

The criteria for adopting a nuclear technology for the next stage stem from the fairly general view of the future nuclear power discussed below.

3. LONG-TERM SCENARIO OF NUCLEAR POWER DEVELOPMENT

A long-term scenario of nuclear power development, which is of course a now tentative job, is envisaged for three options. The first option (Option 1), assumes the continued advancement of nuclear power on traditional thermal reactors with ^{235}U fuel – mostly LWRs.

Consuming ~200 t of natural uranium a year per 1 GWe and given ~10^7 t of cheap uranium, LWRs will have a total output of ~5 × 10^4 Gwe·yr, with their operation in reactor–years making roughly the same figure, and will produce ~10^4 t of fissile Pu (~200 kg/y per GWe). Reuse of Pu, recovered now at the facilities built in France, UK, and Russia, could increase the fuel resources of thermal reactors by 20 to 25%. But the low cost–effectiveness of using MOX fuel in these reactors offers no incentive to expansion of these facilities, while the spread of this technology in the world would add to the hazard of proliferation of nuclear weapons. Plutonium burning in thermal reactors, which has a low efficiency, would constrain or even bar the way to large–scale deployment of breeders at the next stage. That is why this scenario assumes that the thermal reactors of the first stage will continue operating mostly in the open fuel cycle.

Thermal reactors of different types are most likely to find application in a longer term as well, owing to their advantages in some areas of energy production: small and medium nuclear plants (tens and hundreds of MW_{th}) are well suited to meet local heat and electricity needs of remote regions where construction of transmission lines and fuel delivery are difficult and costly, or to provide high–grade heat for some processes. But to do so, thermal reactors will subsequently have to shift to the Th–^{233}U fuel cycle and a BR of ~0.8 to 1 (Option 2), with the ^{233}U deficit covered by breeders.

But centralized electricity production at large NPPs (of GWe capacity), with power transmission over hundreds of kilometers to regions with a million–size population, will in all probability remain the main sphere of nuclear energy application. Electricity is still the most universal and convenient form of energy, well suited for transmission and final uses; its generation grows at the quickest rate and will account for the predominant part of fuel consumption in the next century (its current share being roughly 1/3).

The experience with high–voltage transmission lines amassed in Russia among other countries and the anticipated advent of economical superconducting lines in the next century, open up possibilities for transmitting electricity from large NPPs over thousands of kilometers and for expanding its export. The trend toward miniaturization observed in other industries is opposed here by the indivisibility of the technological process and by the increase in specific costs with reduction of power, especially for NPPs.

Electricity being a standardized and universal product, changes in the market demand do not entail reorganization of the production process, which together with the ease of long–distance transportation is another point in favor of large power plants and reactor units. This does not preclude the use of nuclear energy for heat supply, while utilization of waste heat from NPPs remains an important problem yet to be solved.

On these grounds, large–scale nuclear power deployment, envisaged in Option 3 with a conventional start in 2020, is assumed to involve construction of mostly large NPPs. Such a scale can be provided only by breeders with BR ≥ 1.

An essential objective of this stage is the cost–effective and safe utilization of Pu produced both by the reactors of the first stage (10^4 t) and as a result of nuclear arms reduction (possibly over 200 t). With the use of Pu, BR≥1 is attainable only with fast reactors, which predetermines their principal role at this stage.

One of the main reasons for using light–weight and heat–conducting sodium as a coolant in the first fast reactors, was its capability to remove high heat fluxes from the fuel, with a resultant decrease in the fuel inventory and in the Pu doubling time, T_2. In the post–war decades, the annual rate of energy production growth reached 6 to 7% (up to 12% in the USSR) and short doubling time, T_2, was regarded as an important criterion in fast reactor development. Along with the high power density, another associated requirement was a high breeding gain (BR ~1), for which purpose a uranium blanket was provided. Studied as long-term options were high–density and heat–conducting fuels, such as metal alloys, monocarbides and mononitrides, which afforded a simultaneous rise in the BR and the power density.

The situation is considerably different now. The growth rates have dropped (a threefold increase of electricity production over slightly more than 50 years corresponds to an average rate of ~2% a year), Pu is building up in large quantities, so short doubling time, T_2 is no longer needed. The scenario can be fulfilled by fast reactors with BR ~1 and moderate power density. The 10^4 t of Pu and 1.5×10^4 t of ^{235}U in the spent fuel of the first–stage reactors allow bringing in fast reactors ~4000 GWe in capacity, using Pu mixed with slightly enriched (1 to 4%) uranium (additionally enriched regenerated fuel of thermal reactors). As nuclear power settles on an even keel, these reactors will move into the ordinary U-Pu cycle. With an optimum CBR of 1.05 (minimum reactivity variations), nuclear generating capacities can reach ~8000 GWe due to Pu breeding in the early 22nd century. Therefore, their development should be governed exclusively by the safety and economy criteria. These goals can be met by replacing sodium with a chemically passive high–boiling coolant, eliminating the uranium blanket with assured in–core breeding CBR = BR ~1, and using high–density, heat-conducting fuel instead of oxide fuel (for the purpose of attaining CBR ~1 and reducing reactivity margins, rather than increasing power density). It will be shown below that these and other measures result in an economically efficient high–power fast reactor with an essentially higher level of safety.

Excess neutrons in a fast reactor without a U blanket in the U–Pu cycle and a high flux of fast neutrons endow fast reactors with the advantage of transmutation of long–lived radionuclides to resolve the problem of radwaste without creating special burners. The equilibrium fuel composition (CBR ~1) opens the way for the use of a reprocessing technology which consists basically in rather limited removal of fission products and rules out Pu extraction in this process. Use of such a technology in "non–nuclear" countries would afford a certain degree of their independence from nuclear nations without violating the international nonproliferation regime.

This discussion leads us to the conclusion that the choice of fast reactors in the U–Pu cycle as a basis for large–scale nuclear power, made by its founders back in the 1940s–1950s, remains valid in the new conditions as well. But these conditions and the experience amassed call for new approaches to the creation of such reactors.

It was already mentioned above that thermal reactors also have some scope for long-term development in certain fields of power production with a switch–over to the Th–U cycle in the future. With their contribution to the future nuclear power assessed at 10 to 20%, it can be shown that the ^{233}U deficit in these reactors may be covered without too much trouble by providing fast reactors with a small thorium blanket to utilize part of the leakage neutrons. With breeding well–established and the problems of radwaste settled – mainly through transmutation of long–lived actinides – there seem to be no constraint on the duration of nuclear power operation from the viewpoint of cheap fuel resources and radwaste accumulation. But a complete concept of nuclear development should incorporate, among other things, the final stage with phaseout of NPPs and elimination of large quantities of radioactive material from the reactor inventories. This suggests the need for effective burners without nuclear fuel reproduction, which makes the ongoing quest and studies in this area meaningful. However, even if our hopes for the reasonably early advent of economical and safe breeders do come true, the engineering development of such burners will not be started until some more remote time in the future.

4. REQUIREMENTS TO REACTOR AND TECHNOLOGY, CHOICE OF REACTOR TYPE

4.1. Uranium consumption

Efficient utilization of stockpiled Pu, reduction of specific uranium consumption by no less than an order of magnitude with no need to provide short doubling time.

High–power fast reactor in the U–Pu cycle, moderate power density, CB = CBR ~1, no uranium blanket. Reactor with CBR ~1 should have the power of no less than ~300 MWe. CBR ~1 also dictates the use of high–density fuel. For many reasons, UN–PuN fuel appears to be an optimum choice.

4.2. NPP safety

Exclusion of severe accidents which may result in fuel failure and large radioactive releases (fast runaway, loss of coolant, fire, steam and hydrogen explosions).

If the operation period of nuclear power in the next stage of its development exceeds 10^6 reactor-years, the probability of the above accidents should be kept well below 10^{-6} per reactor-year. Probabilities of such a level obtained by PSA methods have neither operational experience (the existing nuclear power has operated for about 10^4 reactor–years) nor convincing theoretical data to support them. PSA techniques are useful for planning safety improvements at NPPs and allow quite dependable predictions related to the near-term nuclear power development, but they are unsuitable for preparing a really strong safety case for large–scale nuclear power.

Therefore, reactors of the next stage should present no risk of such accidents under whatever human errors, failures or damages to equipment and safety barriers, i.e. these accidents should be deterministically excluded owing to the intrinsic physical and chemical properties and behaviour of the fuel, coolant and other reactor components (natural safety).

Needless to say, there is no way to avoid radioactive releases in case of total reactor and plant destruction as a result of a nuclear attack or a fall of a large asteroid, and these events should be mentioned in the design documentation as exceptions. All potential accidents in a naturally safe reactor, except for those mentioned above, are treated as design–basis events.

Fast reactors with high-density fuel, operating in the U–Pu cycle, can be designed to have optimum CBR ~1, no Xe and Sm poisoning, small power reactivity effect due to the use of fuel with high heat conductivity, and small effect of delayed Np decay, so as to keep the total reactivity margin at the level of $\Delta k_{tot} < \beta_{eff}$ and hence exclude fast runaway under any erroneous actions or accidents in the reactivity control system. Without uranium blanket, a fast reactor has a deeply negative integral void effect. Passive control and cooling elements, feedbacks including large negative temperature coefficient dk/dT, a high level of natural circulation of coolant, prevent dangerous temperature growth under off–normal conditions.

Sodium interaction with air and water, which may cause hydrogen generation and lead to a loss of coolant, and the possibility of a local positive void effect showing up during boiling of this coolant, suggest that it should be traded for another, chemically inert, coolant which boils at a much higher temperature. With no necessity to provide high power density in the core and short doubling time, T_2, it becomes possible to use, for instance, the heavy coolant which has been successfully employed in Russian naval reactors, namely Pb–Bi eutectic or Pb which is close to the former in all physical and chemical characteristics but for the melting point. Use of Pb settles the problems of the high cost and small resources of Bi, and of volatile ^{210}Po with its high alpha activity, produced from Bi. The problems caused by high melting temperature of Pb (327°C) can be resolved through the use of proper temperature conditions and reactor cooling so as to stay within acceptable steel temperatures and to exclude blockage of lead paths under off-normal conditions.

The deterministic safety requirement implies that the ultimate design–basis accident (UDBA), i.e., the accident which covers any event resulting from human errors or multiple failures of equipment, including loss of forced cooling, failure of the scram function, insertion of full reactivity margin, damage to outer barriers such as containment and reactor vessel should not cause fuel failure and radioactive releases such that would require evacuation of people from the territory around the plant.

Analysis of hypothetical (non–credible) accidents including large rapid reactivity addition, fuel failure and collapse with resultant secondary criticality, has been performed optionally, with a view to obtaining ultimate estimates.

Extreme external impacts leading to destruction of the plant, reactor and its vault, will be mentioned in the design documentation, but their analyses are also optional.

4.3. Radwaste

Any predictions concerning safe disposal of large amounts of radwaste for tens of thousands of years give rise to doubts about the validity of geological and especially “historical” forecasts for such remote future.

These doubts can be removed if the radiation hazard from buried radwaste is brought into balance with that of uranium extracted from the earth (radiation–equivalent radwaste disposal), and this is adopted as a requirement for nuclear technology.

The requirement can be satisfied in the following way:

– Long-lived products of U decay (Th, Ra) can be co–extracted with uranium and then handled together with actinides. This step will also facilitate rehabilitation of U mining areas on completion of the work there;

– U, Pu, and other actinides produced during reactor operation, first of all, Am can be returned to reactor to be transmuted by fast neutrons into fission products;
– Radwaste can be subjected to treatment with a view to removing actinides so that it contain only ~10^{-3} of Pu;
– After cooling, radwaste can be brought into a mineral–like state or some other physical and chemical form not prone to migration in soil;
– Radwaste can be buried in naturally radioactive geological formations remaining after U mining, in such amounts that they will be equivalent to extracted U in terms of their radiation hazard.

It should be pointed out that long-lived Np produced in reactors has low activity, and can be dumped untransmuted without disturbing the radiation equivalence. Moreover, if returned to reactors Np adds to the fuel activity since it produces highly active ^{238}Pu and ^{236}Pu which decays to ^{232}U.

Cm, whose main isotopes have a relatively short half-life and a high activity, especially neutronics, can also significantly increase fuel activity if returned to reactor for transmutation, thus impeding fuel refabrication. Therefore, it would be better if Cm were separated from fuel during reprocessing, cooled for some 50–70 years and then returned to reactor in the form of decay products, i.e. Pu isotopes.

Attainment of radiation equivalence between radwaste and mined uranium would also benefit from separation of Sr and Cs so that only 1–10% of them will remain in radwaste. Extracted Sr and Cs could then be utilized as radiation or heat sources. Long–lived I and Tc (with 1–10% of them going to wastes), if extracted, can be returned to reactors for transmutation. The remaining radwaste (with the activity of about 10^4 Ci/L) can be stored in casks cooled by dry air under natural circulation. The activity of radwaste stored in this way would fall by three to four orders of magnitude in 200 years, which simplifies the technology of final disposal of radwaste and enhances its safety analysis shows that such storages can be designed to be simple and not very expensive.

There would be no problems or risk associated with long–distance transport of radwaste if fuel cycle facilities and radwaste storages were set up on NPP sites.

4.4. Nonproliferation

A fast reactor with CBR ~1 and no uranium blanket operates with fuel of equilibrium composition and has no need for Pu separation or addition during fuel fabrication. To adjust the fuel composition, it is sufficient to add ^{238}U to compensate for its burnup.

This fact allows putting forth the following requirement: reprocessing technology should be such as to rule out Pu separation. In this case, reprocessing will boil down essentially to removing fission products from the fuel. In the context of their influence on reactivity, it is acceptable to remove fission products (FPs) so that ~1–10% of them remain in the fuel, which would simplify the above technology and facilitate its choice, though increasing the fuel activity, in particular during refabrication. This is not a major complication, however, since the process is remote anyway. Besides, a high activity of fuel is another warranty against its theft.

The main point here is that such a technology will not add to the risk of proliferation and hence may find worldwide application.

Needless to say, there is no way in which any new fuel cycle technology can rule out illegal application of existing techniques of Pu separation, in particular from LWR fuel, or uranium enrichment for the purpose of obtaining weapons–grade materials. This problem can be successfully dealt with only by political steps meant to enhance the nonproliferation regime and improve the safeguards. Moreover, it has to be resolved irrespective of the further route of nuclear power and nuclear technology.

Promotion of the breeding technology appears to be the most cost–effective way of utilizing Pu accumulated in spent fuel from modern reactors. With this option, Pu would be taken from cooling ponds and put into reactors and fuel facilities, which affords maximum safeguarding and reduces the risk of illegal plutonium separation and utilization.

Fast reactors without uranium blanket, with CBR ~1 and moderate power density, have many traits and possibilities essential for attaining this goal:

- There is no U blanket to produce weapons-grade Pu and with a small reactivity margin in these reactors, it is no longer possible to put U assemblies in the core for Pu accumulation;
- Small reactivity variations during refueling, moderate power density and on–site fuel cycle allow quasicontinuous on–load refueling. Spent fuel can be cooled during 3 to 12 months in an in–vessel storage facility and then sent directly for reprocessing and refabrication. Hence there will be no need for out–of–pile storages for spent and fresh fuel. Such fuel handling can largely simplify supervision and practically excludes fuel thefts.

Initial reprocessing of spent fuel from thermal reactors and fabrication of first cores for fast reactors will have to be done at facilities available in the nuclear countries, but this dependence will not be so strong as in case of regular supply of enriched uranium. Consideration may be given to setting up nuclear technology centers on the basis of these facilities under international jurisdiction.

The aqueous technology widely used now and other options being studied at the moment are tailored to existing reactors which require Pu separation. To meet the above requirement, it is necessary to alter the existing reprocessing technologies or to develop a new one, and this is one of the major challenges along with the development of new reactors. Physical methods of fuel treatment, in which FP removal relies on a factor of two difference in atomic weights, may prove to be the most effective and simple solution.

No concept of a closed fuel cycle satisfying the requirements of large–scale nuclear power, has been suggested yet. It will not fail to appear, however, once the objective is properly defined, and then the requisite technology can be developed and demonstrated within the period of the reactor development.

4.5. Economics

With cheap fuel, it is the NPP cost, which has grown considerably during the last years on account of safety improvements, that is largely responsible for the cost of nuclear energy generation. NPPs of the next stage should be cheaper than modern LWRs, to be economically competitive in many countries and regions.

As is the case with most of the sophisticated technological systems, NPP cost is determined by many things. No separate improvement in one area (for instance, use of natural water circulation instead of pumps in BWRs) can reduce this cost by more than a few per cent.

NPPs with fast reactors should be made at least twice cheaper, which calls for some basic solution extending to the main equipment, systems and structures, whose high costs stem from safety requirements. The answer in this case comes from the natural safety philosophy.

High safety level of new plants, achieved mostly owing to elimination of potentially dangerous design solutions and due to the use of the laws of nature, will make it possible to simplify plant design, lower requirements for basic and auxiliary systems, structures and personnel, and will obviate the need for additional safety systems. These potentialities can be translated into plant design based on consistent application of the natural safety philosophy.

5. AN EXAMPLE OF A NATURALLY SAFE FAST REACTOR

The conceptual design of a fast reactor with UN–PuN fuel and lead coolant (BREST), developed not so long ago, proves that it is possible to meet all the above requirements keeping to a proven technology.

Design and analytical studies were performed and then optimized for reactors with the power in the range from 300 to 1200 MWe. Experiments carried out at U–Pu–Pb critical assemblies sought to validate reactor physics and revise the nuclear data. Steels were subjected to long–term corrosion testing in Pb circulation loops. Experiments were performed to study Pb interaction with air and water of high parameters, interaction of nitride fuel with Pb and steel claddings, and other things.

Calculations on the ultimate design–basis accident, as it was defined above, showed that this reactor can survive it without fuel failure and with moderate radioactive releases. Investigations into hypothetical accidents confirmed that the reactivity addition of up to several β_{eff} at a rate of up to 50 β/s does not cause lead boiling and large release of mechanical energy. According to these studies, lead density, which is close to that of fuel, and convective flows prevent fuel collapse which may otherwise result in the formation of a secondary critical mass.

A lead-cooled fast reactor has a simpler design than sodium–cooled fast reactor:

- Single vessel or pool–type arrangement without a metal vessel (reactor is placed directly in a concrete vault with thermal insulation between concrete and lead);
- Two circuits in the main and emergency cooling systems; decay heat removed by natural circulation of air in tubes located in the lead coolant of the primary circuit;
- No special system to wash off coolant of FAs during refueling;
- Reactivity control provided mostly by lead in tubes located in the side blanket; lead level in tubes is regulated by gas pressure;
- Passive control and protection features with threshold response; high level of natural circulation of coolant; less stringent requirements to the speed of operation with simplification of control and protection systems;
- Simpler design of steam generators, with no need for fast–acting leak detection systems and quick–response valves;
- Less sophisticated fire protection, ventilation and other support systems and components; simpler rooms in the cooling circuits and other NPP constructions.

Cost estimations and comparisons confirm that it is possible to reduce capital costs of such NPPs and the cost of their electricity, as compared to those at VVER plants.

Operating experience of reactors with a heavy coolant, extensive in–pile testing of nitride fuel, calculations and experiments performed in the course of the conceptual design, made the basic aspects of the concept clear enough to embark on engineering development. Considering that the latter will require additional studies and tests, it will take some 10 to 15 years to develop and build an experimental reactor or a demonstration unit which can be put into demonstration operation in about 20 years.

EVOLUTION OF THE TECHNICAL CONCEPT OF FAST REACTORS: THE CONCEPT OF BREST

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Abstract

A new reactor concept BREST, which will adequately meet the variety of cost-efficiencies and safety requirements set for power industry demand within contemporary understandings, has been developed based on the lessons learned from a 50-year experience in the fast reactor development in Russia. The BREST reactor concept adopts most features of the IFR concept. However, several advances in consistent implementation of natural safety principle and in increase of unit power without sacrifice due to its reduction have been achieved in the BREST reactor design. Applications of the high dense and heat-conductive mono-nitride fuel, and the lead–bismuth coolant having high degree of natural circulation are incorporated to facilitate the long-lived high-level radwaste treatment and the natural safety principles in the BREST reactor design. The 40-year experience gained in the lead–bismuth cooled submarine development and the results obtained from core physics experiments and lead coolant experimental works allowed to begin detailed design of a demonstration power unit consisting of BREST-300 reactor and on-site fuel cycle facilities. The construction of the BREST-300 power unit is scheduled for its completion in the first decade of the 21st century. A feasibility study for two BREST-1200 power units demonstrated its superior economics to LWRs of the same power output. Several calculational and experimental works, and design efforts for clarifying the applicability of developed lead–bismuth coolant technology to the lead coolant are under way. An international cooperation on those efforts would substantially contribute to the BREST reactor development.

1. TECHNICAL CONCEPTS OF FAST REACTORS

In the course of nuclear weapons development in the USA and the USSR in 1940s E. Fermi, A. Leipunsky and other nuclear physicists had evaluated a unique surplus of neutrons in the fast U-Pu reactors. If calculated per a burned Pu nucleus excluding 1 neutron for continuation of the chain reaction, about 2 neutrons are left available as compared with the thermal reactors, viz. 1 neutron in case of ^{235}U or Pu and ~1.2 for the Th-^{233}U cycle.

It allows with a high degree of confidence to reach breeding ratio BR ≥1, enhance by two orders of magnitude the degree of uranium utilization and involve uranium from low-grade ore in the fuel balance. As a result, nuclear fuel resources will increase dozens of thousands or, perhaps, millions times, so we can justifiably consider them drainless.[1]

At the same time, the surplus of neutrons is the most important prerequisite for resolution of other global challenges of nuclear power formulated by E. Fermi as back as 1944: safety and dependent cost-efficiency of the reactors, radwaste, non-proliferation.

Having understood that conventional power was limited by available fuel resources (as well as the environmental concern, as we should add at present) and willing to use the advantages of defense nuclear power achievements, the outstanding physicists initiated development of civil nuclear power which would be capable to replace “chemical” power industry.

In 1952 the first pilot fast reactor EBR-I was commissioned in the United States of America to generate electricity by means of a nuclear source.

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[1] Water in the world oceans contains ~ 4×10^9 t of dissolved uranium salts which can be extracted at the cost greater by only one order of magnitude than the current costs, as estimated by Japanese specialists. For the fast reactors it is quite acceptable. Total uranium resources in the earth crust amount to ~10^{14} t, or average concentration of ~ 10^{-6}. The appreciable portion of the resources can be accepted as fuel for the fast reactors.

The successful experience with the BR–5 reactor with oxide fuel and Na (1959, USSR) paved the way for development of the first nuclear power plants (NPPs) with similar reactors in the USSR, France and the UK in 1970s–1980s.

The experience of thermal reactors on the basis of ^{235}U fuel designed in 1940s in the USA and the USSR for production of weapons–grade Pu and U (graphite, heavy–water reactors) and in 1950s for nuclear submarines (light water reactors) can also be used in nuclear power. As long as inexpensive uranium with its concentration in the ore $\geq 10^{-3}$ is still available (potential resources are estimated as 10^7 t), the cost of fuel for thermal reactors will not be high, and such NPPs can successfully compete with conventional power plants (fossil, oil and gas), provided that their construction is rather inexpensive. These arguments set the basis for the first stage of nuclear power development in 1950s with the use of above–mentioned reactors (the first NPP was commissioned in the USSR in 1954).

However, since less than one percent of uranium can be utilized in thermal reactors, the cheap resources of uranium proved to be in a shorter supply than oil and gas resources, to say nothing about coal resources. For this reason a large contribution of nuclear fuel to the world nuclear power cannot be expected. Indeed, after the peak 16% (6% in the fuel balance) and a much higher share in some countries where fuel is scarce, the uranium contribution will be decreasing in the future decades according to the estimations.

Scarce supply of cheap uranium had been a matter of concern from the very beginning of nuclear power development, but Pu produced in the thermal reactors ($\sim 10^4$ t Pu per 10^7 t of consumed U) was supposed to be used as fuel for the fast reactors which would not be limited by fuel resources. After the World War II nuclear power was growing at the record pace and many scientists believed that civil nuclear power riding along the success in defense area could constitute thousands GWs by the end of this century.

To attain this goal, the fast reactors should be designed to provide high rate of Pu breeding, i.e. high breeding ratio (BR) and high fuel power density. Therefore, the following design solutions were selected for the first generation: light heat–conductive Na, U blanket that could make up for a decrease in BR when oxide fuel was applied because of its well–known characteristics to facilitate designing. Dense and heat–conductive metal U–Pu alloys, monocarbide and mononitride fuel were put under R&D activities for the future applications, as these fuels were preferable for fast reactors along with "dry" fuel reprocessing technology, as opposed to water–chemistry techniques for production of weapons–grade Pu.

The events that took place in 1970s and 1980s had changed drastically the situation in power industry, and particularly in nuclear power. Anticipated rates and scales of nuclear power development turned to be unprepared and unclaimed due to the following:

– Oil crisis in 1970s had been mitigated by means of energy–saving measures and stabilization of the fuel market. Rate of power industry progress in developed countries was slowing down;
– The increased rate of the natural gas production. Implementation of gas turbines for the purposes of power industry and resulting rise of its cost–efficiency;
– Accidents at the nuclear power plants, concern over radwaste disposal, anti–nuke movement. NPP safety had been improved but at the expense of higher cost of electricity generated at NPP;
– Fast reactors proved to be the most expensive. Higher risk of nuclear weapons proliferation in case of closed nuclear fuel cycle.

A swell in nuclear power development in 1960s and 1970s was followed by stagnation. Construction and even designing of fast reactors have been phased down at first in the USA and now in Europe. To dispose accumulated spent fuel and Pu, the closed LWR fuel cycle and various burners were developed, though being uneconomic measures. As a result, the large–scale development of nuclear power on the basis of fast reactors could be significantly reduced or even eliminated at all.

The interest in nuclear power and fast reactors is kept up by Asian countries striving for industrial development and lower dependence on fuel suppliers. However, as yet these countries just follow the way went by the USA, Russia and Europe in this century.

Meanwhile, according to the estimations, population will double and the world demand for energy will rise three times, mainly owing to the needs of developing countries, in the coming century. These trends would probably involve the global risk of depleting the cheap resources of organic fuel, destabilization of the fuel markets and getting to hazardous limits on combustion products release, especially if more coal would be burned.

Stabilization of chemical fuel consumption will require advanced power technologies capable to provide large–scale replacement of fuel. Fast reactors are believed to be the most realistic and efficient solution, provided that some specific concerns associated with fast reactors will be resolved.

The designs of EFR and BN–800 (the latter's NPP construction on the site of Beloyarsk has already started) demonstrate the capabilities of evolutionary development of a traditional concept of fast reactor design, cost reduction being one of them.

Nevertheless, the lessons learned from a 50-year experience and new conditions set for power industry demand for development of a new concept for fast reactor which will adequately meet the variety of cost–efficiency and safety requirements in their present–day understanding.

2. NEW EVOLUTIONARY CONCEPTS

The TMI and Chernobyl accidents revealed not only defective design and operation procedures applied at the nuclear power plants, but also inadequacy in the philosophy of safety as it was developed on the basis of overall positive, though limited, experience obtained at the defense nuclear reactors and first civil NPPs. The safety concept allowed large margins for reactivity effects and chemically active coolants that boil at low temperature and might lead to reactor runaway accidents, loss of cooling capability, fire, steam and hydrogen explosions. These hazards can be mitigated at the expense of heavy–duty engineered systems and barriers, tightening of the requirements to equipment and personnel. Though severe accidents could not be completely eliminated, these safety measures reduce probability of such accidents along with higher cost of NPP. PSA methods allow to predict at a certain degree of confidence the safety of nuclear power plants, as long as the evaluations are kept close to the accumulated experience of 10^4 reactor–years. However, similar predictions for large–scale nuclear power which would operate during $\sim 10^6$ reactor–years cannot be based on neither experimental data, nor convincing theoretical justification.

The term "inherent safety" coined by A. Veinberg and introduced in common use after the TMI accident could be regarded as a key principle of a new nuclear power philosophy that includes the economic concerns as well, since the main expenses for nuclear energy go for provisions of NPP safety. It seems reasonable to apply this philosophy to the whole fuel cycle, including radwaste and non–proliferation problems which are not clearly categorized as yet.

The Russian synonym “natural safety” is believed for us to be a proper term for this extended conception. A natural safety approach to the nuclear reactors means, first of all, the rejection of potentially dangerous design solutions for attaining safety and their replacement with rather reliable, as opposed to engineered measures, physical and chemical properties and relationships inherent for fuel, coolant and other components. So when A. Veinberg claimed for deterministic safety rather than scholastic absolutes, he meant reduction of the occurrence probability of severe accidents associated with a catastrophic release of radioactivity to evidently negligible values. Low reactivity margins and effects, the use of chemically inert coolant that boils at high temperature and other measures make it possible to exclude fuel damage under any human errors, failures of components and even external impacts, except for extreme nuclear or other impacts leading to a complete NPP destruction. Other beneficial factors are as follows: efficient feedback, high degree of natural circulation of coolant and air for the purposes of decay heat removal.

For the reason of neutron surplus, fast reactors can use fuel with a balanced composition (Pu: ^{238}U ~0.1, CBR ≅1). Only dense fuels conform to this requirement which is the most important factor of inherent safety, as it allows to reduce a total reactivity margin under high heat conductivity of fuel to $\Delta k_{tot} \leq \beta_{eff}$ as compared to dozens of β_{eff} for modern reactors. Many other advantages can also be obtained.

The balanced composition of fuel also permits to refuse the extraction of Pu during fuel reprocessing. In combination with rejection of enriched uranium, the transfer to fast reactors will eliminate the major factors of risk associated with nuclear weapons proliferation as nuclear power spreads all over the world. Realization of the international non–proliferation regime can be facilitated and reduced to supervision over possible set-up of illegal Pu extraction and U enrichment facilities. Fuel reprocessing technology can also be facilitated.

Neutron surplus in fast reactors is beneficial in other respects as well. In particular, radwaste management problem can be resolved in line with natural safety principles by means of transmutation of all actinides into fission products (FPs) and long–lived FPs (^{129}I, ^{99}Tc, etc.) into short-lived and stable nuclides. Radwaste decontamination from actinides to the factor of ~10^{-3} would provide the balance between radioactivity of disposed radwaste and uranium taken from mines.

3. BREST REACTOR DESIGN DEVELOPMENT

Development of a new concept of fast reactor design had been started in Russia in 1970s after the commissioning of the BN–350 reactor and the completion of BN–600 design with a view to resolve revealed challenges. The Chernobyl accident and other events in nuclear power gave a stronger impulse to this work. From the beginning of 1990s, thanks to the support provided by Ministry for Atomic Energy, Ministry of Science and Russian Academy of Sciences, these activities have been performed in a more consistent and planned way.

The work was much influenced by the concept of IFR developed in ANL, USA. Many provisions of this concept were applied to our design, including the use of dense fuel, on–site fuel cycle facilities with dry fuel reprocessing and some others. At the same time, we believe that Na cooling and metal fuel could not properly meet the natural safety principles, whereas the advantages of a modular design (PRISM) do not compensate the economic losses because of decreased unit power and make it more difficult to use fuel with balanced composition.

According to preset goals, BREST reactors should be designed such as to implement consistently the principles of natural safety without a sufficient deviation from the materials and technology which had been proved in defense and civil nuclear power facilities. On the other hand, advanced, though not yet completely justified design solutions should not be sacrificed for the sake of quick construction, as happened with the previous designs of fast reactors, which is why their huge potentialities could not be implemented fully.

Slowdown of the power growth rate in the world (estimated for the next century as just a little over 2% per year, i.e. triplication of installed capacity for 50 years) and accumulation of up to $\sim 10^4$ t of fissile Pu produced in thermal nuclear reactors allow to neglect the need for a short doubling period and a basic requirement for excessive Pu production. As a result, the highest level of safety and cost–efficiency became the top priority.

Lower fuel power density makes it possible to transfer from Na to heavy, low–activated, chemically inert lead (Pb) coolant that is characterized by low absorption and neutron moderation capabilities. In terms of its physical and chemical properties, except for the melting temperature, Pb is similar to lead–bismuth (Pb–Bi) coolant which has been used for 40 years in the nuclear reactors for Russian submarines.

There are some concerns regarding applicability of the corrosion–resistant technology developed for Pb–Bi coolant to Pb coolant. Another problem is avoidance of cold water ingress to steam generators that may lead to Pb freezing. Design efforts, calculations and experimental works are underway to clarify and eliminate the problems mentioned above.

The high dense and high heat–conductive mononitride fuel with balanced composition, tested in the BR-5 reactor and the loop experiments, has been selected. This year it is planned to install a loop channel with Pb and BREST fuel into the BOR-60 reactor.

A series of physical experiments with Pu–U–Pb core have been conducted and some of them are underway at the BFS critical assembly.

The obtained results allow to begin detailed design of a demonstration power unit with the BREST-300 reactor and on–site fuel cycle facilities with dry fuel reprocessing technology. The first stage of detailed NPP design as applicable to the site of Beloyarsk has been completed. Designing and construction of the power unit are scheduled for completion within the first decade of the 21th century.

A feasibility study has been performed for the NPP with two BREST-1200 units. The study demonstrated that the cost of NPP with an inherently safe fast reactor would be significantly lower than the cost of NPP with LWR of the same power output.

The investigations and developments performed so far let us hope that demonstration of the fast reactor and fuel cycle of natural safety will open up the optimistic prospects for a new stage of nuclear power capable to find the answers to the urgent fuel and energy problems. Participation of other countries interested in nuclear power production can substantially contribute to the success of this project.

DESIGN FEATURES OF BREST REACTORS AND EXPERIMENTAL WORK TO ADVANCE THE CONCEPT OF BREST REACTORS

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Abstract

Principle design features of BREST-300 (300 MWe) and BREST-1200 (1200 MWe) lead–cooled fast reactors are presented in this paper. Several experimental works have been performed or under way in order to justify lead-cooled reactor design concepts. BREST reactor designs of different outputs have been developed using the same principles. In conjunction with the increased output and the implement of inherent safety concept, a number of new solutions, which may be applied to the BREST-300 reactor design too, have been considered in the BREST-1200 reactor design. The new design features adopted in the BREST-1200 reactor design include: pool-type reactor design not requiring metal vessel, hence, not limiting reactor power; new handling system allowing to reduce central hall and building dimensions as a whole; emergency cooling system using field pipes, immersed directly in lead, which may be used to cool down reactor under normal conditions; by--pass line incorporated in coolant loop allowing to refuse the actively actuating valve initiated in pumps shut down.

1. DESIGN FEATURES OF BREST REACTORS

We guess, large-scale power will demand reactors of various sizes, but centralized power generation using large-scale nuclear power plants, will be the thoroughfare of the development. This has resulted in necessity to consider large–scale reactor concepts satisfying the criteria of new technology, e.g., the BREST-1200 reactor. The same principles as ones used in the BREST-300 reactor were applied in developing the BREST-1200 reactor, while BREST-300 may be considered both as mid-size series power reactor, and test and experimental reactor, designated to gain operating experience, finally develop and check new engineering solutions, specifying safety and economy of lead-cooled fast reactors. The reactor power (1200 MWe) was chosen due to the fact that there are turbines of supercritical parameters, such as K-200-240LMZ, being manufactured in Russia.

The BREST-1200 reactor facility (as well as BREST-300) represents dual-cycle cooling, steam generating power unit, comprising reactor itself with steam generators, pumps, refuelling equipment, control and protection system, concrete well with thermal insulation, steam turbine plant, heat removal system being used in reactor cooldown, reactor heatup system, reactor overpressure protection system, primary coolant cleanup system, gas cleanup system and other subsidiary systems.

High density (14.3 g/cm^3) and highly conductive (20 W/m-K) mononitride mixed fuel (UN–PuN) is discussed as the fuel of high compatibility with the lead and fuel cladding material. Chromium ferrite–martensite steel will be used as fuel cladding.

Lead, ensuring good thermal fuel–coolant interaction, is filled between the fuel and cladding in a gap provided by fuel element design to reduce fuel temperature resulting in relatively low fission product release from the fuel inwardly the cladding.

For the purpose of providing large cross section for coolant flow passage, increasing the power removed by lead natural convection, reducing coolant heatup value, and mainly avoiding loss of cooling the affected fuel assemblies caused by local flow blockage of the fuel assemblies, all the in-core fuel assemblies are can-free. Such fuel assembly design allows coolant to flow across the core preventing the affected fuel assembly from burnout. Fuel assemblies installed in reactor reflector have leak–tight cans. The first line of these fuel assemblies is used as control member channels, and fuel assemblies in the lines from 2 to 4 may content I and Tc for transmutation, as well as Sr and Cs as stable heat generator.

Three–zone flattening of lead incremental heating and fuel cladding temperatures by shaping fuel assembly power density and lead flow rate by the use of fuel elements with different diameters but similar plutonium content in the fuel to be loaded is applied in the core instead of usual flattening of radial power density distribution by fuel enrichment.

Uranium screens, used conventionally in fast reactors, have been replaced for efficient lead reflector, albedo characteristics of which are better than those of uranium dioxide, to reduce neutron leakage, improve neutron flux flattening, provide operating conditions without reactivity change during core cycle, and achieve complete in–core fuel breeding.

The use of chemically inert, high–boiling lead as the primary coolant, enabled to refuse three–circuit heat removal system, and adopt simpler dual-cycle cooling system with steam superheating by steam and feed water preheating up to 340^{o}C by live steam at supercritical parameters.

Heat removal from reactor core is carried out by lead coolant forcedly convected by using pumps. Lead coolant is pumped to 2 m height about suction chamber lead level and fed to free level of the ring discharge chamber. Then the lead comes down to core support grid, through fuel assemblies from the bottom upwards, to be heated up to 540^{o}C, and is fed to common dump chamber of "hot" coolant, further the coolant comes up and through distributing header pipes flows to steam generator inlet cavities and shell space. While coming down the shell space, the lead coolant gives its heat to secondary coolant, which flows through steam generator tubes. The lead coolant cooled to ~420^{o}C flows up through the ring gap and pours out to pump suction chamber, from which it is again pumped to the discharge chamber (Figs 1 and 2).

Lead circulation through reactor core and steam generator is carried out by the difference between the levels of "cold" and "hot" coolant developed by the pumps, rather than the head developed by the pumps. Nonuniformity of lead flow through the steam generators with one or few pumps shut down is excluded, because flow inertia in fast pump shutdown is provided by equalizing coolant levels in discharge and suction chambers (for about 20 s).

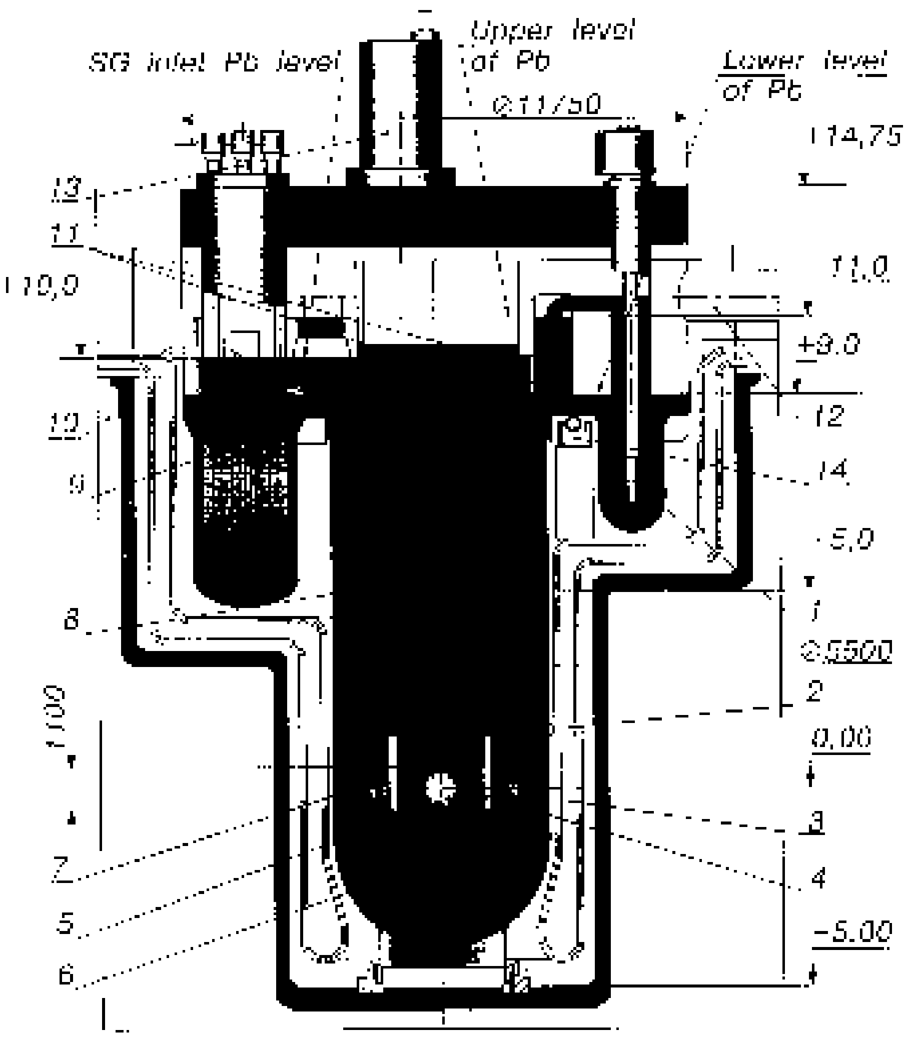
As for proposed primary coolant flow system, while flowing the coolant reaches free level twice, that results in surfacing and escaping the majority of lead vapour bubbles, being generated in accidents caused by steam generator tube depressurization.

Integral−loop primary circuit layout is applied to reduce consequences of potential accident caused by steam generator tube ruptures, in which the steam generators and main circulating pumps are placed outside reactor main vessel. Such layout together with the chosen lead flow circulation system and steam relief units from the reactor vessel to the pressure−suppression pools will prevent critical quantity of steam from intake by reactor core and reactor vessel from overpressure. In comparison with integral layout conventional for fast reactors, the BREST reactor design enables to reduce reactor dimensions and lead loop volume.

In−vessel spent fuel storage, spaced from the core and protected from being irradiated, enables to accelerate and simplify spent fuel unloading from the reactor by precooling the spent fuel to the level of decay heat release, allowing its handling without being cooled by force.

The lack of high pressure in the lead circuit and relatively high lead freezing point contribute to crack self-healing that prevents loss of coolant, fuel element melting, and radioactive lead melt blowdown to reactor room.

In conjunction with increased power and to completely implement the concept of inherent safety in the BREST-1200 reactor design, a number of new solutions have been taken as compared to the BREST-300 reactor.



1 — pump
2 — reactor vessel
3 — thermal protection
4 — CPS
5 — core
6 — support pillars
7 — dividing barrel with a support grid
8 — in-pile FA storage
9 — steam generator
10 — concrete vault
11 — rotating plugs
12 — emergency throw of steam
13 — transfer machines
14 — vessel support

FIG. 1. General view of BREST-300.

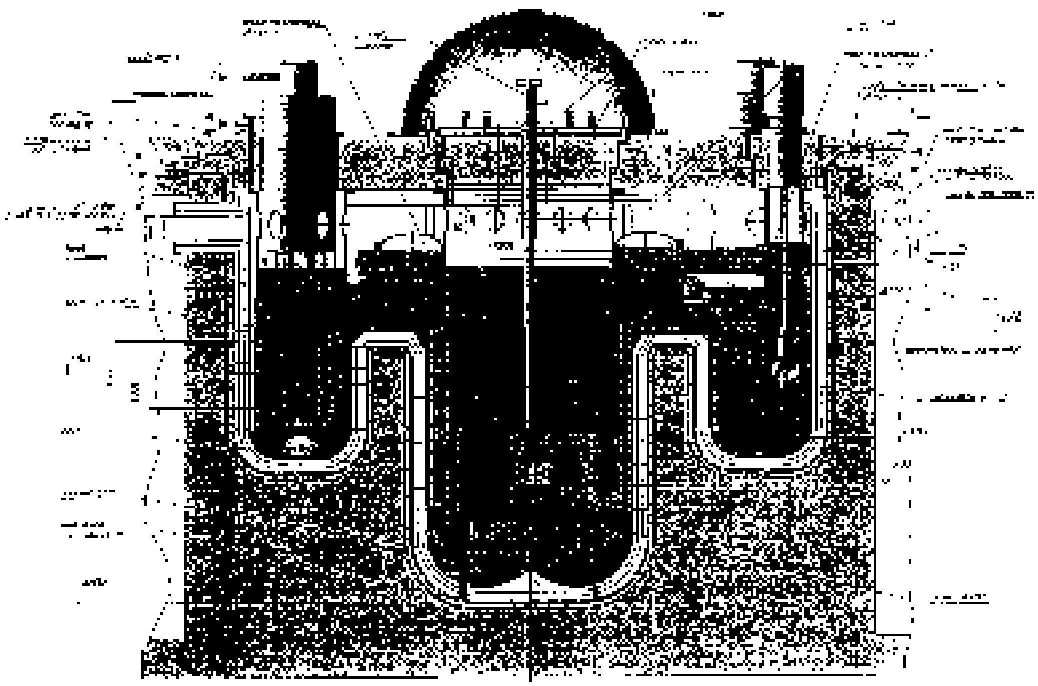


FIG. 2. BREST-1200 reactor.

Increased reactor dimensions and weight is a problem for vessel manufacturing, shipment and installation, seismic stability. Pool-type layout of the reactor and steam generators is adopted in the BREST-1200 design, where those are installed inwardly concrete well with thermal insulation without metal vessel. Natural convection of air, circulating through pipes (120 mm pipe size), the downcomer and upcomer legs of which are placed inwardly the load bearing concrete, is used to maintain the concrete temperature within the admissible limits. Reinforced concrete body is lined inwardly with 8–10 mm thick anchored steel. Air pipes are secured to the lining from outside, and thermal insulation is anchored from inside, being made in the form of stainless steel clad thermal insulating units of 3 × 200 = 600 mm total thickness. “Keramvol” plate is chosen as the thermal insulation. The reduced design conductivity of the thermal insulating units was taken as 0.2 W/m^{o}C.

Regarding the BREST-300 reactor, the lack of screens and relatively small size of reactor core allowed one to place control rods outside the core, around its side surface, implement the potential to control the reactor by influence on neutron leakage through varying lead column level in the reflector; it enables to consider the potential to refuse control and protection system having conventional mechanical drives.

Large dimensions of BREST-1200 core in radial direction has reduced the importance of side neutron leakage significantly. In comparison with the BREST−300 reactor the efficiency of lead columns in the lateral reflector has reduced by a factor of three, that is why making the BREST−1200 reactor subcritical in outages ($\Delta k = 2\%$) is carried out by absorber rods, placed inwardly the tubes of the fuel assembly structures and held above the core in upper position by lead flow. As forced convection ceases, heavy rods come down the core. As circulation starts, rod groups (each having its own lead inlet) subsequently reset to upper position. The main functions in reactivity control (Δk ~0.3%) are carried out by the members placed in the lateral blanket, mainly by lead columns, the level of which in the channels is regulated by gas pressure.

To simplify installation and dismantling of the gas driven control members due to increased number of gas driven regulators in the BREST-1200 reactor (from 28 in BREST-300 up to 64 in BREST-1200) it has resulted in necessity to route the feeding mains from the top, that is a feature of BREST-300, supply the drive gas from core bottom. A flexible pipe was used to supply the gas, with the pipe inserted through the guide channel on the side of the coolant downcomer path in the guide pipe of the control member, installed in the center of reflector unit.

As for the BREST-1200 reactor, heat is transferred to the emergency cooling system by air circulating due to the natural draft in field pipes, immersed directly in liquid lead in steam generator wells. Atmospheric air enters internal field pipe being a downcomer portion, and comes upward through the gap between the internal and external pipes, which is the upcomer portion. Heated air enters exhaust stack through the collecting headers and is released into the atmosphere. An emergency cooling system incorporates 264 field pipes with internal and external pipe diameter of 140 and 210 mm, respectively. Interior side of the external pipe has 16 fins of 200 mm width with each extending along the heated portion. With reactor operated at normal conditions the emergency cooling system is in hot standby and the system capacity in this mode of operation is as low as possible from the viewpoint of maintaining the temperatures of outlet circulation circuit at the level allowing one to start the system at full capacity immediately. The emergency cooling system may be initiated by either opening a shut valve by active or passive signal, or passive unit actuation by air temperature growth at Field pipe outlet. At 420^{o}C the capacity of emergency cooling and reactor well cooling system (this system may be used as a train of emergency cooling) is ~11 MW with 70 m exhaust stack height. Field pipes of the emergency cooling system may be also used in standard reactor cooling with forced air convection. In this case fans are used instead of blowers. Power output taken by this cooling system is ~1% of full power.

As radial size of reactor core increases, negative component of positive reactivity effect at reflector ends decreases. Neutron absorbers are proposed to be placed in fuel assemblies at 200 300 mm distance from the ends of fuel column to increase this component, that enables to reduce core sensitivity to lead density variation in essence to the value characteristic for the BREST-300 reactor core.

New fuel assemblies and core components handling pattern is proposed for BREST-1200 (Fig. 3) as compared with the BREST-300 reactor. The handling operation at these reactors is carried out in two steps: moving components in and out of reactor by in–reactor handling. A conveyor, having two ramp conveying pipes and movable tungsten–loaded container located on the side of the reactor to install and fix from surfacing core components (spent fuel assemblies, fresh fuel assemblies, lead reflector unit, control member) in it, is used in BREST-1200 to move the spent fuel assemblies, fresh fuel assemblies and other core components in and out of reactor. Floor–mounted handling machine is used in the BREST-300 reactor to carry out this operations. By moving the bridge and trolley the handling machine installs fuel assembly enclosing booth coaxial with discharging channel–lock and joins the booth with lock chamber by lowering the booth, the chamber being installed on the rotable plug. In–reactor handling includes operations to load fresh fuel assemblies, reflector units and control members from the in-reactor storage to the core, as well as unloading the spent fuel assemblies and other components from the core to the in-reactor storage; regarding the BREST-1200 reactor these operations also cover insertion of the spent fuel assemblies into the conveyor container and withdrawal of the fresh fuel assemblies out of the conveyor container. In–reactor handling is carried out using rotable plugs and in-reactor handling machine, installed onto the small rotable plug. The rotable plugs are designed for guiding the in-reactor handling machine to the co-ordinate of the specific cell of the core and in–reactor storage; regarding the BREST-1200 reactor the plugs are used also for guiding the in–reactor handling machine to conveyor container co-ordinate. Handling pattern used in the BREST-1200 reactor allows one to reduce significantly the central hall dimensions in comparison with those of the BREST-300 reactor, where upper handling pattern with handling through rotable plug is used.

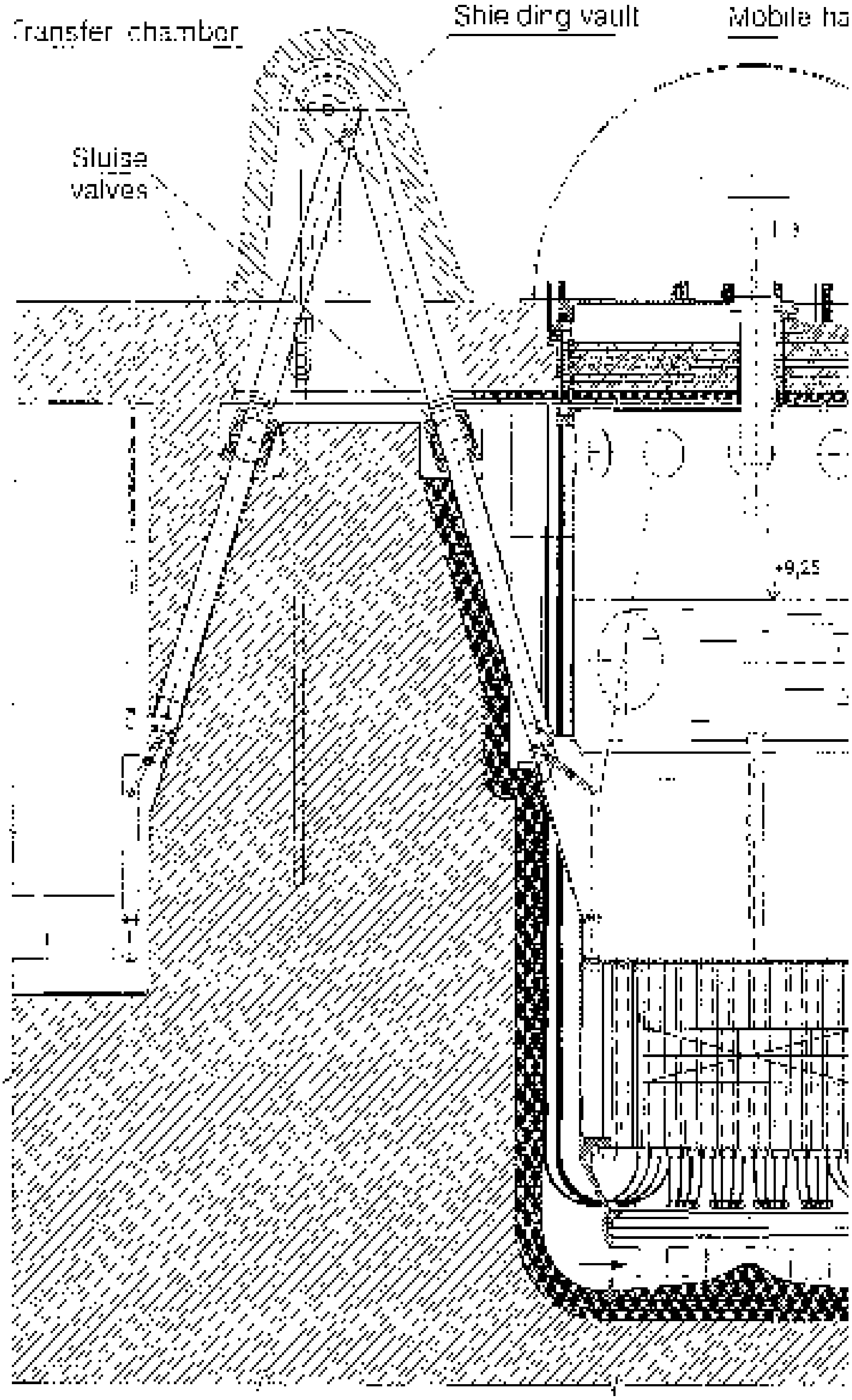


FIG. 3. Schematic of refueling BREST-1200 reactor.

Check cargo valves interconnecting pump suction chamber with reactor discharge chamber are used in the BREST-300 reactor to provide coolant natural convection. All check valves open in pump operation at rated capacity and the lead is pumped to the reactor discharge chamber through these valves and through upper overflow pump pipes, too. The check valves become closed by hydrodynamic forces arising in moving the coolant from the reactor discharge chamber to the discharge chambers of shutdown pumps with one or two pumps shut down. Thus shutdown of one or two pumps will not result in lead coolant back flow through the pumps.

As clear from the BREST–type reactor incident design study including severe accidents, the reactor is stable to the incidents and large–scale fission product releases are impossible.

Comparative economic estimates allow one to hope the expenses not to be higher than those for light water reactors.

Conceptual design development has confirmed the potential to build BREST–type reactors of various power with inherent safety for large–scale nuclear power in future.

Table I outlines the specifications of 300 and 1200 MW lead-cooled fast reactors within the framework of conceptual design.

TABLE I. TECHNICAL CHARACTERISTICS REACTORS BREST–300 AND BREST–1200

Characteristic	BREST-300	BREST-1200
Thermal power, MW	700	2800
Net electric power, MW	300	1200
Number of FA in the core	185	332
Core diameter, mm	2300	4755
Core height, mm	1100	1100
Fuel element spacing, mm	13.6	13.6
Fuel element diameter, mm	9.1; 9.6; 10.4	9.1; 9.6; 10.4
Core fuel	UN + PuN	UN + PuN
Core fuel load, t	16	63.9
Load of Pu/(^{239}Pu + ^{241}Pu), t	2.1/1.5	8.56/6.06
Characteristic	BREST-300	BREST-1200
Fuel lifetime, yrs.	5	5-6
Refueling interval, yrs.	1	1
Inlet/Outlet lead temp., ^{o}C	420/540	420/540
In. (water)/Out. (steam) temp., ^{o}C	340/520	340/520
Maximum cladding temp., ^{o}C	650	650
Maximum lead velocity, m/s	1.8	1.7
Lead flow rate, t/s	40	158.4
Efficiency, %	43	43
Core breeding ratio (CBR)	~1	~1
Effects of reactivity, % $\Delta k/k$		
– Power	0.16	0.15
– Total (max) reactivity margin	0.35	0.31
β_{eff}	0.0036	0.0034
Lifetime, y	60	60

2. EXPERIMENTAL WORK TO ADVANCE THE CONCEPT OF BREST REACTORS

The experimental works conducted during 1990–1994 at Institute of Physics and Power Engineering (Obninsk, Russia), Central Research Institute of Structural Materials (St.–Petersburg, Russia), All–Russian Research Institute for Technical Physics (Russia), Institute of Steel and Alloys (Moscow, Russia), Polytechnical Institute (Nizhny Novgorod, Russia), and IGR reactor (Semipalatinsk, Russia), are discussed.

2.1. Neutronic experiments to justify neutronic analysis

Different BREST–300 reactor core characteristics have been studied in experiments at a series of three critical assemblies (BFS-61, BFS-61–1, BFS-61-2) differing from each other with side reflector configuration at the Institute of Physics and Power Engineering (IPPE) (Obninsk, Russia). The critical assembly core height was 86.7 cm, and the core radius changed with the variation of the side reflector configuration and ranged from 44.6 cm to 49.8 cm. Such characteristics as critical parameters, mean cross section ratio, reactivity coefficients for different materials, Doppler reactivity effect in sample heating, control member prototype worth, reaction rate distribution, void effect in Pb, reactivity effect in hydrogen insertion, reactivity effect in fuel melting simulation, distribution of fission neutron weights, and delayed neutron effective fraction were measured. Different codes with different constant systems were used by the experts of IPPE and the Kurchatov Institute (Moscow, Russia) to compare the experimental and analytical values.

For the purpose of verifying and supporting BREST–300 reactor analyses the following critical experiments were analyzed by Monte Carlo method at the Kurchatov Institute:

- Three configurations at BFS–2 rig (IPPE);
- Critical experiments at PF assembly (IPPE) with Pb blocks and highly enriched uranium;
- Six compositions with cylindrical core and highly enriched uranium and Pb disks in the core and reflector at critical assembly ROMB (All–Russian Research Institute for Technical Physics);
- Six compositions with spherical core of semi–nuclear grade plutonium and reflector of highly enriched uranium at assembly ROMB (All-Russian Research Institute for Technical Physics).

The spectrum of neutron leakage from the Pb sphere was measured using C_f-source in the range from 0.1 to 10 MW, and inelastic scattering cross sections for Pb were measured by time-of-flight method at IPPE to define more accurately the differential nuclear data. From the above experiments, nuclear data for Pb have been defined more accurately.

2.2. Hydrothermal experiments to justify core hydrothermal analysis

The following studies have been performed at the sodium-potassium (22% Na + 78% K) eutectic rig, IPPE [2]:

- Heat transfer coefficient in square fuel element grids with relative pitch S/d = 1.28–1.46 has been defined;
- The effect of fuel element square spacer grids on heat transfer has been defined;
- Heat transfer at initial portion of fuel element square grids has been defined;
- The effect of power density varying with height on heat transfer coefficient has been defined.

The experiments for defining the effect of coolant technology and corrosion films on heat transfer coefficient have been conducted at the Pb rig, IPPE. Necessary heat transfer coefficients and empirical dependencies for the core and loop hydrothermal analysis have been obtained as a result of experiments.

2.3. Structural steel corrosion tests for service life at lead-cooled nonisothermal rigs

Structural steels for reactor cores were tested for service life during 8500 hours at two rigs with test section temperatures 620 and 650^{o}C, and lead flow velocity 1.6 m/s, IPPE. The effect of temperature, coatings, alloying elements, steel structure, coolant technology and other parameters on steel corrosion resistance in Pb was studied at the rigs. After the in–Pub tests, oxide films and mechanical properties of the steels have been studied.

Structural steels for the vessel and steam generators were tested for service life during 12 500 hours at the rig with test section temperature 550^{o}C, and lead flow velocity 1.7 m/s, Central Research Institute of Structural Materials. The effects of temperatures, coatings, alloying elements, steel structure, coolant technology stress and other parameters in Pb were studied at the rig. Welded joints and steels in zones of stagnation were tested. After the benchmark in–Pb tests, steel corrosion resistance and mechanical properties were studied.

Fuel element prototypes simulating the fuel by molybdenum and uranium nitride with Pb sublayer and different additions to it were tested for service life in furnaces during 5000 hours, at temperature drop through the fuel element from 540 to 700^{o}C, and constant temperatures 650 and 700^{o}C, IPPE. The effects of temperature, coatings, additions to the Pb sublayer, alloying elements, steel structure and other parameters on steel corrosion resistance were studied at fuel element prototypes. After the in-furnace tests of the fuel element prototypes, the effects of Pb–sublayer on steel corrosion. resistance were studied.

As evident from the results of the conducted tests and studies, there is a possibility to use the Pb–Bi coolant technology for the Pb coolant. To achieve steel corrosion resistance as high as possible, the parameters of oxygen treatment have been chosen preliminarily. Structural steels for the core, vessel and steam generator have been chosen. Steel corrosion rate for service life of the core, steam generator and reactor have been predicted preliminarily.

2.4. Lead inflammation

When discussing lead–cooled reactors at seminars and scientific and technical councils, any time an issue on its combustion arises. As referred to [3], which was written on the base of GOST 12.1.044–88 “Fire and explosion hazard of substances and materials. List of parameters and methods of their definition” lead is classified as combustible substance. With dust of size 74 µm self ignition point of aerogel is 270^{o}C, and aerosuspension is 580^{o}C. Regarding iron parameters, it is also the combustible substance with dust of size 74 µm, the self ignition temperature of aerogel is 170^{o}C, and aerosuspension is 320^{o}C. Therefore studies and experiments were carried out on liquid lead ignition.

Lead ignition processes with slow air leak in and prompt air breaking through, as well as oxygen supply to the gas space over the lead were simulated at liquid lead temperature 1200^{o}C in Institute of Steel and Alloys. Ignition, flash and combustion both with and without additional ignition source in the form of powerful spark discharge were not observed.

In lead metallurgical fabrication and refining using open kettles or reverberatory furnace at 1200^{o}C with oxygen–enriched air blow through, no ignition and combustion of lead and vapour were observed at ~1100^{o}C of the exhaust gases and vapours.

Thermodynamic analysis and modeling of liquid lead ignition possibility at 900–1200°C have shown that there is no liquid lead ignition, even if all the initial agents are taken at high temperature equal to 1200°C, since lead and lead oxide evaporation reaction requires more heat than oxidation reaction. The cycle of lead oxidation reactions may provide evaporation processes that will be endothermic and the main thermodynamic ignition condition will not be fulfilled. As resulted from the experiments and thermodynamic analysis, liquid lead is not combustible.

2.5. The experiments on the rupture of steam generator tubes

Experiments on bubbling water and steam at pressure up to 24.0 MPa and temperature 140–350°C through liquid lead at temperature 550–600°C and 50–3000 mm thickness were conducted in Polytechnical Institute, N. Novgorod. The possibility to abolish "vapour hammer" in tube rupture was studied.

Experiments on rupture of the tube containing water of supercritical parameters were conducted at All–Russian Research Institute for Technical Physics (Russia), and the effect of the rupture on the adjacent tubes was studied.

Simple engineering solutions allowing to avoid "vapour hammer" have been found. It has been shown that the rupture of one steam generator tube will not cause the rupture of adjacent tubes. A possibility to use double-cycle cooling system in the reactor has been demonstrated.

3. DEVELOPMENT OF LEAD-COOLED FAST PILOT REACTOR OF 300 MW POWER WITH ON–SITE FUEL CYCLE AT RDIPE SVERDLOVSK BRANCH SITE

Technical documentation for NPP with lead–cooled fast pilot reactor of 300 MW power with on–site fuel cycle at RDIPE Sverdlovsk branch site is planned to develop in 1998 – 1999. The reactor is aimed at:

- New coolant management;
- Neutronic and hydrothermal study;
- Tests for service life;
- Demonstration of stability to accidents with and without scram:
    - Insertion of total reactivity margin;
    - Primary and secondary pumps off;
    - Steam generator tubes broken;
    - Freezing and unfreezing;
    - Imposition of simultaneous accidents;
    - Limiting accidents;
- "Devised" accidents.

REFERENCES

[1] ADAMOV, E.O., et al., Conceptual design of BREST–300 lead–cooled fast reactor, Proceedings, International Topical Meeting on Advanced Reactor Safety, ARS'94, Pittsburgh, USA, 509–516.

[2] TSIKUNOV, V.S., The next generation of fast reactors, Nuclear Engineering and Design, 173 (1997) 143–150.

[3] Edited by BARATOV, A.N. and KOROLCHENKO, A.Ya., Handbook "Fire and explosion hazard of substances and materials and means to extinguish this", Khimija Publishing House (1990) (book one, 329-330, book two, 161).

CURRENT STATUS AND PLANS FOR DEVELOPMENT OF NPP WITH BREST REACTORS

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Abstract

Current status and plans for the development of nuclear power plant with a pilot lead-cooled 300 MW(e) rated fast reactor (BREST-OD-300) are presented in this paper. The development project was commenced by the order of Ministry for Atomic Energy in 1998 and includes detailed design documentations for BREST-OD-300 and on-site fuel cycle facilities, and R&D activities for design verification.

1. ORDER OF MINISTER FOR ATOMIC ENERGY, DATED 17. 07. 1998

On development of detailed design documentation for the nuclear power plant with a pilot demonstration fast reactor with lead coolant of 300 MW(e) output (BREST–OD–300) and on-site fuel cycle at Beloyarsk NPP site, and commencement of R&D activities to justify the design. The purpose of the work is to demonstrate feasibility of designing NPP such as to meet the following main requirements:

- Avoidance of severe accidents at the reactor facility and nuclear fuel cycle facilities associated with radiological hazard for public;
- Radiation–equivalent disposal of radwaste and their temporary storage in the region;
- Technological support of non–proliferation regime;
- Competitiveness.

The following enterprises and organizations are involved in the project:

- Research and Development Institute of Power Engineering (NIKIET);
- Institute of Physics and Power Engineering (FEI);
- All-Russian Research Institute for Inorganic Materials (VNIINM);
- St. Petersburg “Atomenergoproekt” (AEP);
- State Construction and Design Institute (GSPI);
- Research Institute of Atomic Reactors (NIIAR);
- Sverdlovsk branch of Research and Development Institute of Power Engineering (SF NIKIET);
- “Hydropress” Pilot Design Bureau;
- Central Design Machine Building Bureau (TsKBM);
- “Hydromash” Design Bureau;
- “Atomenergoproekt” State Planning and Design, Research and Survey Institute;
- All-Russian Planning and Design Technological Head Institute (VNIPIET);
- Central Institute for Power Engineering (TsVTT);
- Central Design Institute (ITsP);
- Research and Production Association State Enterprise (NIKIMT);
- All-Russian Institute for Thermal Engineering (VTI);
- “PROMETEY” Central Research and Design Institute;
- Department of Integrated Automated Systems (OKSAT).

2. MATERIALS ALREADY DEVELOPED

- Justification of capital investments in NPP construction;
- Justification of capital investments in construction of on–site fuel cycle facilities:
  - Master plan;
  - Technological solutions;
  - Main building;
  - Turbine hall and secondary circuit;
  - Civil engineering design;
  - Construction arrangement draft;
  - Preliminary safety analysis report;
  - Assessment of the ecological impact.

The following basic technologies were developed:

- FA shearing;
- Fuel regeneration;
- Fuel element fabrication;
- FA fabrication.

Detailed design documentation for the following components and equipment of BREST–OD–300:

- Reactor facility;
- Steam generator;
- Pump;
- Floors;
- Reactor vault;
- Refueling machine;
- Reactor facility systems as follows:
  - Coolant technologies;
  - Gas system;
  - Coolant purification of impurities;
  - Heatup of lead and filling the reactor facility with lead coolant;
  - Reactor overpressure protection.

3. FEASIBILITY STUDY FOR BREST–1200

- Examinations;
- Experimental activities;
- Main stages of the activities:
  - RF Government decree on construction, 2000;
  - Experimental activities, 1998–2006;
  - Designing activities, 1998–2001;
  - Examination performing, licensing and beginning of construction, 2002;
  - Commissioning, 2007.

EXPERIMENTAL STUDIES OF BREST-OD-300 REACTOR CHARACTERISTICS ON BFS FACILITIES

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Abstract

The BFS−77−1 critical experiments were performed to validate and verify the software and cross section data base aiming at confirming the neutronics analysis of lead cooled BREST−OD−300 fast reactor core. For the validation and verification, other experimental works have been also completed to justify software and cross section data base, and real accuracy of calculation of characteristics of the BREST−OD−300 fast reactor core. It was drawn from the analysis of current status of evaluated neutron data for lead that additional experiments are required in order to refine lead cross section values. Calculated characteristics of control rods and reactivity balance still require justification.

1. CODES AND CROSS SECTION DATA USED IN THE REACTOR ANALYSIS FOR EXPERIMENTAL JUSTIFICATION

1.1. Computer codes

Current core neutronics analysis made within the framework of lead cooled BREST-OD-300 fast reactor design is based on the ABBN-93 data base library [1] using neutronics codes, namely: SYNTES [2], JARFR [3] and 3D [4]. Precise calculations are made using Monte Carlo method in the MCNP code and the ENDF/B−VI file library.

In the course of the analytical treatment of the experimental results obtained on the BFS and other facilities, neutronics characteristics of the critical assemblies are evaluated by the basic code TRIGEX [5] used for fast reactor analysis: with ABBN−93 cross section data. Heterogeneous structure of the BFS assemblies is taken into account, and all necessary corrections to the TRIGEX code. Homogeneous diffusion calculation results are evaluated using the FFCP [6] and TWODANT codes. Monte Carlo precise calculations are made on the basis of the MCNP and MMK-KENO codes [7].

1.2. Cross sections data base

The set of group cross section data ABBN-93 [4], developed at SSC RF IPPE is taken as a basis. The set of cross section data ABBN-93 is the new Russian system of the group cross section data with traditional (28 groups) and multi−group (299 groups) division. This system has been designed for the analysis of neutron and photon patterns and their functionals in the core, blanket and shielding of different types of nuclear reactors. The ABBN-93 system is based on the files of evaluated nuclear cross sections data recommended by experienced specialists. These files extracted from the libraries of evaluated nuclear data, namely ENDF/B-VI, JENDL-3, BROND-2 and some other sources, are included into the Russian library of evaluated data FOND-2 (SSC RF IPPE) [8].

The cross sections and codes system CONSYST/ABBN (SSC RF IPPE) provides different applications of the ABBN-93 cross section data. The key codes of CONSYST/ABBN system are CONSYST, PRECONI and PRECONS Fortran codes. The PRECONI and PRECONS codes couple the group cross section data of ABBN-93 with such Russian computer codes as SYNTES (VNILAES), JAR5R (SSC KI), 3D (NIKIET) and TRIGEX (SSC RF IPPE). CONSYST code provides cross section data for the kinetics codes such as ANISN and TWODANT, as well as for codes intended for precise calculations of the reactor characteristics using Monte Carlo method, for instance NIMK-KENO and MCNP. The scheme of the data base system interaction with the computer codes is shown in Fig. 1.

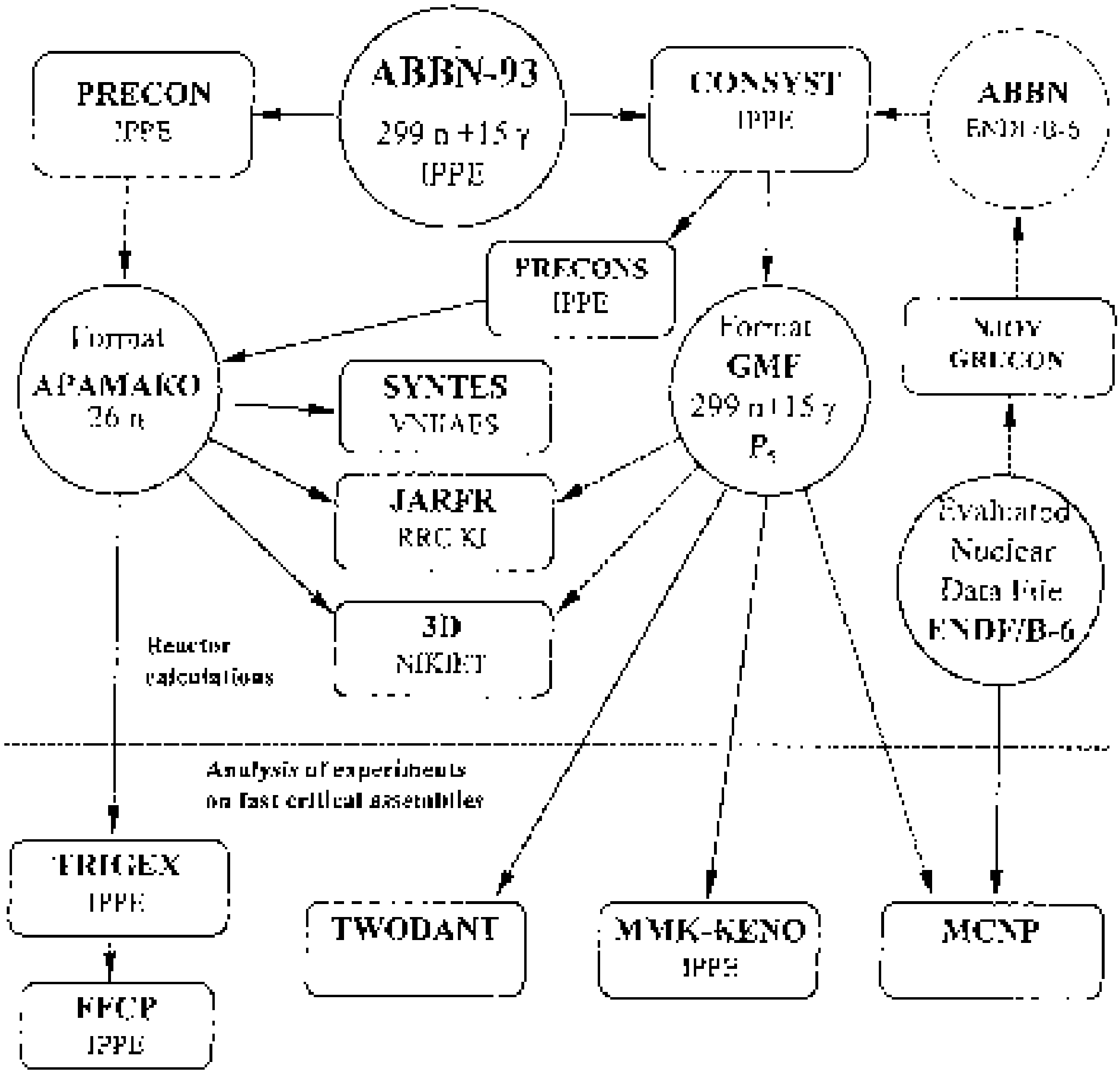

FIG. 1. Scheme of interactions between different calculational codes.

1.3. Experimental studies performed

Accuracy achieved by now in the evaluation of physical characteristics of lead cooled BREST–OD–300 reactor core is based on the analytical treatment of the following experimental studies:

- Measurement of escape cross sections under U238(n,f) reaction threshold, thus determining the number of fissions in the reactor in the fast neutron range, this being the basis for the conclusion on the reliability of description of inelastic scattering within 0.8–10.5 MeV energy range;
- Series of spherical critical experiments with uranium and plutonium fuel using lead reflector of variable thickness [9];
- Series of experiments conducted on the ROMB facility (VNIITF, Chelyabinsk) with the “pancakes” made of high enrichment uranium and lead;
- BFS–61 series of experiments conducted on the BFS-1 facility [10];
- BFS–77 series of experiments conducted on the BFS-1 facility;
- BFS–64 series of experiments conducted on the BFS-2 facility (not completed).

1.4. Planned experiments

The following additional experimental studies are planned now in order to justify existing accuracy of calculation of physical characteristics of lead cooled BREST-OD-300 reactor:

- Several critical assemblies having 30 and 60% fuel enrichment to be tested on the BFS-2 facility in order to study and refine physical properties and cross sections of lead and bismuth;
- Full scale modeling of the BREST-OD-300 reactor on the BFS–2 critical facility.

1.5. Analytical treatment of experimental results

By now, the analysis of BFS-61 critical assembly experiments has been completed, and the analysis of BFS-77 critical assembly experiments has been started (preliminary results of this analysis are given below).

Calculations were made by the basic code used for fast reactor analysis, namely TRIGEX (SSC RF IPPE) with the ABBN-93 cross section database. All required corrections to the homogeneous diffusion calculation results made by the TRIGEX code have been taken into account, using the FFCP and TWODANT codes. Precise calculations were made by Monte Carlo method using the MCNP and MMK-KENO codes.

1.6. Estimation of components of calculational uncertainties

For the BREST-OD-300 reactor model, calculations were made to estimate sensitivity of such parameters as k_{eff}, breeding gain (BGA), Doppler temperature reactivity effect (DTRE) and void reactivity effect (VRE) with respect to the cross sections. The cross sections component of the calculation uncertainty was also estimated. This analysis was made using the INDECS system [11]. All the results of experiments and analytical treatment were stored into the LEMEX and LSENS database of the INDECS system.

BREST-OD-300 SSC RF IPPE
NPP Department

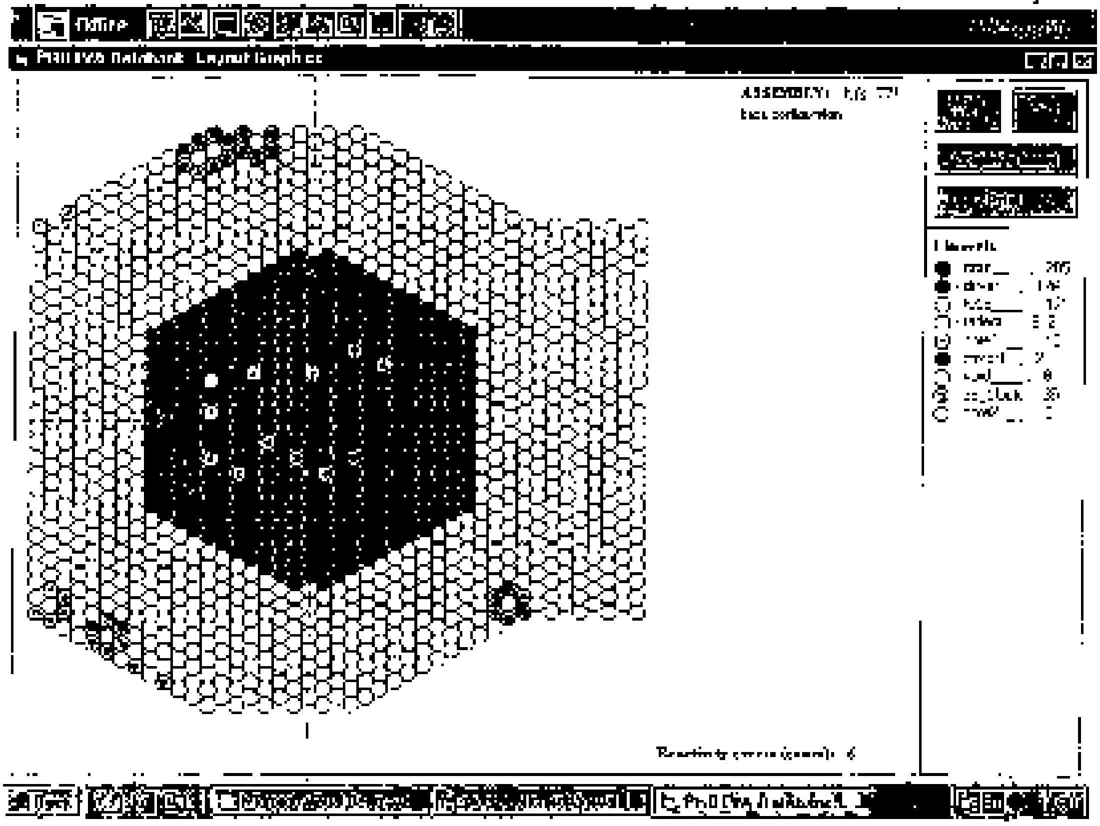

FIG. 2. Loading layout for BFS–7–1 assembly.

BREST-OD-300 SSC RF IPPE
NPP Department

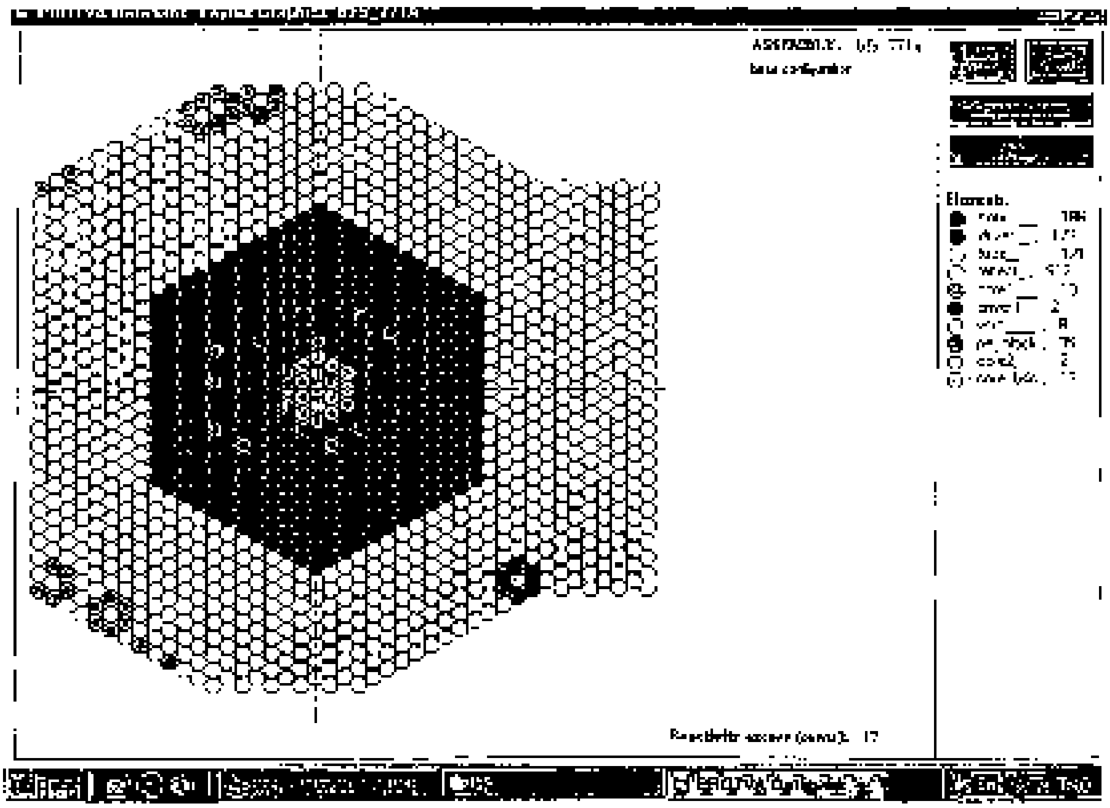

FIG. 3. Loading layout for BFS–7–1a assembly.

2. RESULTS OF EXPERIMENTAL STUDIES CARRIED OUT ON BFS-77 ASSEMBLIES

The BFS-77 critical assembly was created on the BFS-1 critical facility, and experimental studies were performed on core 1 modelling of the BREST-OD-300 reactor with lead coolant. Critical loading of BFS-77-1 included:

- 217 rods of test core having composition of BREST–OD–300 reactor core 1 type;
- 178 rods of driver zone;
- 812 rods of radial reflector made of depleted uranium dioxide.

Two types of the BFS-77-1 assembly were tested. The BFS-77-1a assembly differed from BFS-77-1 by that in 19 central rods the lower cell of the upper lead cavity was replaced with 5 discs of natural boron carbide. In order to keep criticality, two rods were added on the periphery of the driver zone. Layouts of loading of the BFS-77-1 and BFS-77-la critical assemblies are presented in Figs 2 and 3.

2.1. Criticality parameter

Calculations were made using the TRIGEX code for the BFS-77-1 critical assembly in the homogeneous diffusion approximation: k_{eff} (TRIGEX) = 0.9757. Correction for heterogeneous structure of the assembly evaluated using the MMK-KENO code was Δk_{eff} (heter.) = +0.0145. Kinetic correction evaluated using the TWODANT code for the BFS-77-1 assembly was Δk_{eff} (kinetic) = +0.0073. Thus, calculated k_{eff} value for the TRIGEX code is k_{eff} (TRIGEX) = 0.9975 0.0030. Calculation made by the Monte Carlo method using the MMK-KENO code with the ABBN-93 data base in 299 groups in P_5 order of scattering with detailed description of the whole geometrical structure of the critical assembly in exact compliance with the loading layout, of each tube and pellet of the BFS-77-1 assembly gave k_{eff} = 1.0015±0.0006. Calculation made by the MCNP code using evaluated data files ENDF/B-VI [5] with detailed description of the whole geometrical structure of critical assemblies resulted in: a) k_{eff} = 1.00025±0.00013 for the BFS–77–1 assembly and b) k_{eff} = 1.00074±0.00013 for the BFS-77-la assembly.

2.2. Spectral indexes and fission number distribution

Table I gives results of calculations made on spectral indexes for the BFS-77-1 assembly. The results were obtained using the MCNP code and the TRIGEX diffusion code (the FFCP code was used for evaluation of corrections for heterogeneity). Figure 4 shows axial distribution of Pu^{239} fission numbers in the BFS-77-1 assembly evaluated by the TRIGEX code.

2.2. Spectral indexes and fission number distribution

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TABLE I. k_{eff} AND SPECTRAL INDEXES FOR BFS−77−1 ASSEMBLY

Indexes	MCNP calculation	TRIGEX calculation	
	C/E	Hete./Homo.	C/E
k_{eff}	0.9999±0.0030	1.0223	0.9975±0.0030
C238/F235	1.013±0.050	1.13	1.056±0.050
F238/F235	0.985±0.030	0.93	1.011±0.030
F239/F235	0.995±0.014	0.99	0.996±0.014
F240/F239	1.103±0.033	0.95	1.043±0.033
FNp^{237}/F239	1.039±0.042	0.95	1.079±0.042
FPu^{238}/F239	1.063±0.022	0.98	1.056±0.022
FAm^{241}/F239	0.908±0.039	0.95	0.985±0.039
FAm^{243}/F239	1.018±0.080	0.94	1.15±0.08

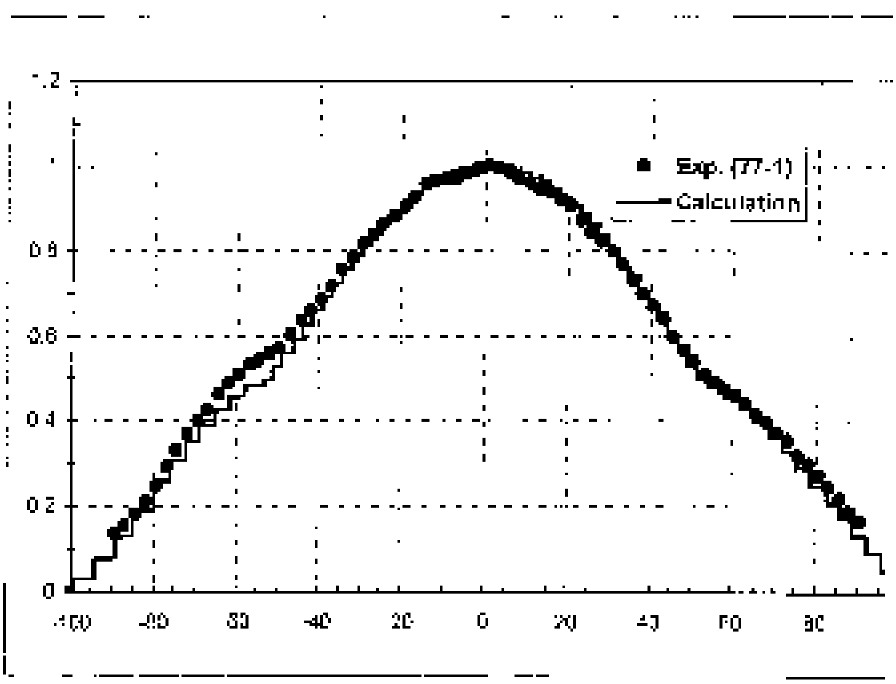

FIG. 4. Pu^{239} fission rate distribution via height of BFS−77−1 assembly.

2.3. Measurements of lead void reactivity effect

Void reactivity effect (VRE) is determined as the effect of lead disks replacement with the empty cans in either core (70 items) or the upper axial blanket (36 items). All the presented results have been obtained by direct variation calculations. Comparison of VRE calculated and experimentally determined values for the upper axial blanket and the core after replacement of lead discs with the empty cans is given in Table II. Corrections for heterogeneity and kinetics have been introduced in the calculated values. Table III presents similar comparison referred to one tube (experimental error value is ±0.7 cents).

TABLE II. VOID REACTIVITY EFFECT (ACCUMULATED, CENTS)

Voiding zone	Number of tubes	TRIGEX homogeneous	Δ kinetic	Δ heterogeneous	C-E
T	19	-26.0	+9.4	+0.8	+0.4
O	34	-42.9	+12.9	+1.3	-1.0
P	56	-65.5	+21.2	+1.8	+2.1
	85	-92.0	+25.6	+2.6	+5.9
C	19	-3.6	+6.1	-12.1	-11.0
O	34	-4.8	+9.5	-20.8	-19.4
R	56	-6.7	+16.3	-33.1	-26.5
E	85	-6.5	+21.1	-48.3	-35.5

Table III. VOID REACTIVITY EFFECT (PORTION, CENTS)

Total number of Tubes	Number of tubes in portion	Top		Core	
		Experiment portion (per tube)	Calculation portion (per tube)	Experiment portion (per tube)	Calculation portion (per tube)
19	19	-16.2 (-0.85)	-15.8 (-0.83)	+1.4 (+0.074)	-9.6 (-0.505)
34	15	-11.5 (-0.77)	-12.9 (-0.86)	+1.9 (+0.127)	-6.5 (-0.433)
56	22	-16.9 (-0.77)	-13.8 (-0.63)	-0.3 (-0.014)	-7.4 (-0.336)
85	29	-25.1 (-0.89)	-21.3 (-0.73)	-1.2 (-0.41)	-10.2 (-0.352)

Effect of replacement of 4 rods with 4 sleeves: –40.0±1.0 cents

Calculated value –38.0 cents.

2.4. Measurement of reactivity effect caused by the fuel density change

Experiments were made with the fuel rods containing 5 cells of the following core composition: 4Pb + UO_2 + Pu + U + 4Pb + C + 4Pb + U + Pu + UO_2 + 4Pb. Both upper and lower axial blankets referred to the standard fuel rods.

– Effect of replacement of 7 standard rods located around assembly axis (5–2–2 channel) with experimental rods –20.0±0.5 cents

– Calculated value –19.2 cents.

2.5. Measurement of reactivity effect caused by replacement of 4 standard fuel rods with the lead sleeve mock-ups

Lead sleeve mock-ups differed from the fuel rods of the test zone fuel rods by that the core composition was substituted for the lead disks of 109 mm height. The mock-ups were installed in the following channels: 5–2–2 (20^{th} row from the left, 19^{th} channel from the top), 4–1–1 (18^{th} row, 18^{th} channel), 5–4–2 (22^{nd} row, 18^{th} channel) and 6–3–2 (20^{th} row, 21^{st} channel).

2.6. Measurement of worth of mock–ups with absorber

Mock–ups containing absorbers differed from the test zone fuel rods by that the natural B_4C disks were used instead of the core composition. Effect of replacement of lead sleeve mock–up with that containing absorber was studied.

Results of measurements made on single mock-ups are as follows:

- mock–up–1 (5–2–2) –116.0±1.0 cents; calculated value -116cents;
- mock–up–2 (4–1–1) -114.0±1.0 cents; calculated value -114 cents;
- mock–up–3 (5–4–2) -115.0±1.0 cents; calculated value -114 cents;
- mock–up–4 (6–3–2) -114, 0± 1. 0 cents; calculated value -114 cents.
- Effect of replacement of three mock–ups: –322.0±2.0 cents;
  Calculated value –318 cents.
- Effect of replacement of four mock–ups: –401.0±3.0 cents;
  Calculated value –397 cents.

2.7. Measurement of rod worth

Rod worth calculation results are presented in Table IV. These were obtained using the TRIGEX code with the first order perturbation theory formulas.

TABLE IV. WORTH OF SOME RODS

Rod type	Coordinate on map layout	Worth, β_{eff}	TRIGEX calculation	MCNP calculation
Reflector	4–12–3	0.07	0.042	
Reflector	5–14–8	0.08	0.038	
Test zone	5–6–4	0.11	0.082	0.091±0.05
Test zone	5–10–6	0.10	0.080	0.13±0.05
Driver zone	5–11–7	0.31	0.307	0.36±0.05
Driver zone	5–13–8	0.30	0.276	0.32±0.05

3. RESULTS OF ANALYTICAL TREATMENT OF THE OTHER EXPERIMENTS AND ACCURACY OF CALCULATIONS

3.1. Experiments performed on BFS-61 assemblies

In 1990–1991, experimental study on modeling fast reactor with lead coolant and plutonium nitride based fuel was carried out on the BFS–61 critical assemblies at SSC RF IPPE to order of SSC KI [9].

Calculated values of criticality parameter for the TRIGEX code with the ABBN-93 cross sections data are presented in Table V. Uncertainty values attached to the results of calculation have been obtained as 1/4 of evaluated correction for heterogeneity, which in this case is predominant. These results were confirmed by the precise analysis using Monte Carlo method by the MMK-KENO code (see Table VI). In these calculations, all geometrical features (including BFS basket and its supporting plate) of each assembly were presented. Calculations on the MMK-KENO code were made using the ABBN-93 data base in 299 groups, P_5 approximation for order of scattering, statistical accuracy of the calculation was ±0.0005.

TABLE V. CALCULATED VALUES OF CRITICALITY PARAMETER

BFS Assembly	k_{eff} calculation by TRIGEX				
	Reference calculation	Heterogeneity Correction	Kinetics correction	Other corrections	Calculated values
61–0	0.9624	+2.54%	+1.0%	+0.1%	0.998±0.006
61–1	0.9653	+236%	+0.9%	+0.1%	0.998±0.006
61–2	0.9677	+2.26%	+0.74%	+0.1%	0.998±0.006

TABLE VI. RESULTS OF PRECISE EVALUATION OF CRITICALITY PARAMETER

BFS Assembly	k_{eff} calculation by MMK–KENO	
	Heterogeneous analysis using ABBN–93	Heterogeneous analysis using JENDL–3.2
61–0	0.9995	0.9959 (–0.4%)
61–1	0.9971	0.9935 (–0.4%)
61–2	0.9977	0.9926 (–0.5%)

For the sake of comparison, Table VI also contains the results of calculation made on the basis of cross section data for lead taken from the JENDL-3.2 evaluated data library. These calculations are under ABBN-93, so the criticality parameter of the BFS-1 assemblies is underestimated by 0.4–0.5%. It should be noted that in the ABBN-3 cross section data system, value taken from the previous version of the JENDL-3 evaluated data library was used for lead.

Measured spectral indexes were also analytically treated within the framework of the TRIGEX code taking into account space and resonance heterogeneous effects, evaluated using the FFCP code.

Table VII gives calculated values of spectral indexes compared with the results of measurements. These measurement results are presented along with the errors evaluated by the experimenters taking into account the error of corrections introduced into the calculations.

TABLE VII. COMPARISON OF SPECTRAL INDEXES CALCULATION DATA AND EXPERIMENTAL RESULTS

Functional	Introduced corrections	Calculated values	C/E
C8/F5 (1)	1.000	0.128	1.001±0.024
C8/F5 (2)	1.000	0.137	1.025±0.023
C8/F5 (3)	1.000	0.144	1.001±0.021
FS/F5 (1)	1.000	0.0305	0.968±0.030
F239/F235	0.999	1.059	1.002±0.015
F240/F239	0.992	0.271	1.050±0.020
F241/F239	1.000	1.261	1.002±0.015

3.2. Spherical critical assemblies given in Handbook [10]

Series of experiments: HMF–027 and PMF–035 from the International data bank of evaluated experiments on critical safety Handbook [10] were treated analytically. These are spherical critical systems using uranium and plutonium fuel with lead reflector of variable thickness. Calculations were made by the MMK–KENO code in 299 groups, P_5 order of the scattering.

Results of this analysis based on the ABBN–93 cross sections are presented in Table VIII (statistical accuracy of the calculation is indicated in brackets). Results of calculation made on the critical assemblies with thicker reflector and not included in the collected book [10], are also presented. As it can be seen from the table, evaluation of these critical assemblies has been made at lower level as compared to that of the two previously indicated and included into the Handbook [10].

Table VIII also gives for comparison the results of calculations made using lead cross section data from JENDL-3.2 evaluated data library. As it follows from the table, results obtained using ABBN-93 cross section data and different evaluations of lead cross sections (ABBN-93 and JENDL-3.2) are in a good agreement with the experimental results (k_{eff} = 1), the discrepancy being within ±0.3% value, while the disagreement between two evaluations is as high as 1%.

TABLE VIII. CRITICALITY CALCULATION RESULTS FOR SPHERICAL ASSEMBLIES

Assembly	Fuel	Reflector thickness, cm	k_{eff} ABBN–93	k_{eff} JENDL-3.2
HMF–027 [10]	U^{235}	3	1.0003(6)	1.0021 (+0.2%)
HMF–027a	U^{235}	14	1.0028(6)	1.0118 (+0.9%)
PMF-035 [10]	Pu^{239}	3	0.9977(6)	1.0012 (+0.4%)
PMF-035a	Pu^{239}	28	1.0106(6)	1.0211 (+1.0%)

3.3. ROMB cylindrical assemblies

Evaluation of a series of experimental results obtained on ROMB facility (VNIITF) with disks made of high enrichment uranium and lead (6 compositions) was made using MMK-KENO code in 299 groups, D5 order of scattering. Results of this evaluation with ABBN-93 database are presented in Table IX with statistical accuracy of calculations indicated in brackets. Table IX also gives for comparison the results of calculations made using lead cross section data from JENDL-3.2 evaluated data library. As it follows from the table, results obtained using ABBN-93 cross section data are in a good agreement with the experimental results ($k_{eff} = 1$), the discrepancy being within ±0.5% value, while the disagreement resulted from calculations using JENDL-3.2 lead cross sections is up to +1.6%.

3.4. Measurement of escape cross sections under U^{238} fission threshold

Comparative analysis of cross sections of total and inelastic scattering was made for lead taken from the ABBN–93 group cross section library, which was obtained by reprocessing of the FOND–2 home library of evaluated neutron data files [8] and values taken from different libraries of evaluated neutron data, such as BROND-2, ENDF/B-VI, JEF–2 and JENDL-3.

TABLE IX. CRITICALITY EVALUATION FOR THE SERIES OF ROMB CYLINDRICAL ASSEMBLIES

Assembly	Lead fraction	Reflector	k_{eff} ABBN-93	k_{eff} JENDL-3.2
KC-1	Pb/U = 0	None	0.9948(5)	0.9948
KC-2	Pb/U = 0	Provided	0.9946(5)	0.9967 (+0.2%)
KC-3	Pb/U = 0.5	Provided	0.9944(5)	0.9997 (+0.5%)
KC-4	Pb/U = 0.8	Provided	0.9973(6)	1.0087 (+1.1%)
KC-5	Pb/U = 1.1	Provided	0.9993(5)	1.0156 (+1.6%)
KC-6	Pb/U = 1.5	Provided	0.9986(5)	1.0130 (+1.4%)

It has been revealed that the largest discrepancies between the data taken from different libraries are observed within 0.1–1.5 MeV energy range. The most considerable discrepancies exist as far as the ENDF/B-VI neutron data library is concerned, that is the total cross section is significantly overestimated in this energy range. The total cross section value of natural lead taken from the JENDL-3 library is in a good agreement with the experimental data set. Inelastic scattering cross section values of natural lead taken from different libraries were compared and this revealed that neutron inelastic scattering cross section value in the BROND-2 library is groundlessly overestimated.

Behaviour of inelastic scattering cross section in the vicinity of threshold (energy over 300 keV) has significant influence on the escape cross section value under the threshold of U^{238}(n,f) reaction, determining the number of fissions in the fast neutron energy range in the core. Based on the experimental measurement of the escape cross section value under the threshold of U^{238}(n,f) reaction, conclusion can be made on the reliability of description of inelastic scattering in lead within 0.8–10.5 MeV neutron energy range.

Escape cross section under U^{238} fission threshold is determined as follows:

$$\sigma_{escape} = \frac{\int_0^\infty dE\, f(E)\, \sigma_f^{238}(E) \left\{ \sigma_a^{Pb}(E) + \sigma_s^{Pb}(E)\, f_s^{Pb}(E \rightarrow E') \left[1 - \sigma_f^{238}(E') / \sigma_f^{238}(E) \right] \right\}}{\int_0^\infty dE\, f(E) \sigma_f^{238}(E)} \quad .$$

Table X presents ratio of this value for the libraries under consideration to the experimental data obtained by N. Bethe, et al (1957) that was later confirmed by the results obtained by I.I. Bondarenko, et al (1961). Similar data are given in Table X on escape cross section under the threshold of Np^{137} fission and Al^{27}(n,p) reaction.

3.5. Cross sections component of calculation uncertainty

Evaluation was made of the uncertainty of BREST–OD–300 reactor analysis associated with using the ABBN–93 cross section data base, and the INDECS, code and archives system [11] was used for that. Accuracy was evaluated for the criticality parameter k_{eff}, breeding gain (BGA), void reactivity effect (VRE) and Doppler temperature reactivity effect (DTRE).

TABLE X. CALCULATION TO EXPERIMENT DATA RATIO FOR ESCAPE CROSS SECTION UNDER REACTION THRESHOLD

Data	C/E		
	U^{238}(n,f)	Np^{237}(n,f)	Al^{27}(n,p)
BROND–2	1.70	1.90	1.09
ENDF/B–VI	1.07	1.19	0.99
ENDF/B–VI.2	1.15	1.30	1.13
JENDL–3	0.97	1.00	0.99
JENDL–3.2	1.51	2.00	1.08
Experiment	0.712±0.043	0.208±0.050	2.21±0.15

The INDECS system includes data computer archives: LEMEX, LTASK, LSENS, and LUND, as well as specially developed CORE code package for making statistical analysis of the observed discrepancies between calculated and experimental results. The LUND database involves W matrix of uncertainties and covariances of the ABBN–93 cross sections without taking into account integral experiments carried out on the critical assemblies and reactors. For such materials as Fe, Cr, Ni, Mo, Pb and U^{238}, for which escape cross section under U^{238} fission threshold was measured — experiments performed by N. Bethe, et al and I.I. Bondarenko, et al. The evaluation of covariance matrices of uncertainties for (n,n') reaction was made taking into account these experiments.

Sensitivity of k_{eff}, BGA, VRE and DTRE with respect to the cross sections was evaluated for the BREST-OD-300 reactor model. This evaluation was made using the TRIGEX code.

Uncertainties of considered physical characteristics of BREST-OD-300 caused by the neutron data, namely the ABBN-93 group cross sections are presented in Table XI. Void reactivity effect uncertainty is given in %Δk/k units, while the absolute values of BGA uncertainties are presented, and DTRE uncertainty is given in percent of the effect. DTRE contribution to the total uncertainty is mainly determined by the uncertainties in the knowledge of Doppler increments of the resonance self–shielding factors. As follows from Table XI, the largest concern on “the micro data level”, i.e., without taking into account macro experiments, is caused by k_{eff} accuracy. It will be recalled that for the traditional core design target accuracy of critical parameter evaluation is 0.5%. However, k_{eff} error can be significantly decreased by taking into account the results of experiments carried out on fast critical assemblies. Table XI also gives evaluated total cross section components of calculational uncertainties of BREST-OD-300 physical characteristics, obtained by taking into account macro experiments carried out on critical assemblies.

As it can be concluded from Table XI, the accuracy of calculation prediction of k_{eff} value has decreased from 2.2% down to 1.0% taking into account macro experiments. Slight reduction of VRE prediction accuracy has occurred, while for DTRE it has been kept the same. This is quite natural, since the experimental data available were sufficient for refining only spectral component of DTRE uncertainty whose contribution to the total uncertainty is small, while Doppler increments of the resonance self–shielding factors are predominant in making such contribution.

TABLE XI. EVALUATED CROSS SECTION COMPONENTS OF CALCULATION UNCERTAINTIES OF BREST–OD–300 PHYSICAL CHARACTERISTICS

Parameter	Cross section component of uncertainty	
	Without experiments (micro data level)	With critical assembly Experiments
k_{eff}	2.2%	1.0%
BGA (+0.11)	53.1% (±0.06)	26.2% (±0.03)
VRE (+0.9%Δk/k)	41.296 (±0.4% Δk/k)	38.5% (±0.4 Δk/k)
DTRE (–0.49%Δk/k)	12%	12%

4. CONCLUSION

Validation and verification of the software and cross section database were carried out conformably for neutronics analysis of lead cooled BREST–OD–300 fast reactor core. For the validation and verification, experimental work has been completed to justify software and cross section database, and real accuracy of calculation of characteristics of lead cooled BREST–OD–300 fast reactor core. Analysis of current status of evaluated neutron data for lead has been made. It was drawn from the analysis that additional experiments are required in order to refine lead cross section values. Calculated characteristics of control rods and reactivity balance still require justification.

REFERENCES

[1] MANTUROV, G.N., NIKOLAYEV, M.N., TSYBOULYA, A.M., System of group cross section data ABBN–93, Part 1: Nuclear data for calculation of neutron and photon radiation patterns, Issues of Nuclear Science and Technology, Series: Nuclear Data Base, **1** M., (1996) 59 (in Russian).

[2] SELEZNYOV, E.F., SYNTES code, Abstracts of paper presentations, Issues of Nuclear Science and Technology, Series: Physics and Technology of Nuclear Reactors, **6** (43) (1984) 56–58 (in Russian).

[3] JARFR–l code package, Abstracts of paper presentations, Issues of Nuclear Science and Technology, Series: Physics and Technology of Nuclear Reactors, **9** M. (1985) 66 (in Russian).

[4] BARINOV, S.V., 3D code, Algorithms and codes for nuclear reactor neutronics analysis, Proceedings, Seminar Neutronics–99, 26–28 October 1999, Obninsk, Russian Federation (in Russian).

[5] SERYOGIN, A.S., Abstracts of paper presentations, TRIGEX code for low–group reactor analysis in three–dimensional hexagonal geometry, Issues of Nuclear Science and Technology, Series: Physics and Technology of Nuclear Reactors, **4** (33) M., (1983) 59–60 (in Russian).

[6] BEZBORODOV, A.A., RYAZANOV, B.G., SAVOS'KIN, M.M., Analysis of heterogeneity effects using FFP method in the fast neutron critical assemblies, Issues of Nuclear Science and Technology, Series: Physics and Technology of Nuclear Reactors, **2** (1986) 8 (in Russian).

[7] BLYSKAVKA, A.A., MANTUROV, G.N., NIKOLAYEV, M.N., TSYBOULIA, A.M., CONSYST/MMK–KENO code package for analysis of nuclear reactor characteristics using Monte Carlo method in multi-group approximation with scattering indicators in P_N approximation, Algorithms and codes for nuclear reactor neutronics analysis, Proceedings, Seminar Neutronics–99, 26–28 October 1999, Obninsk, Russian Federation (in Russian).

[8] BLOKHIN, A.I., FURSOV, B.I., IGNATYUK, A.V., Current status of Russian evaluated neutron data libraries, Proceedings, Nuclear Data for Science and Technology, Gatlinburg, Tennessee, USA, ORNL, ANS, **2** (1994) 695.

[9] International Handbook of Evaluated Criticality Safety Benchmark Experiments, NEA/NSC/DOC(95)03, NEA OECD, Paris (1999).

[10] ALEXEEV, P.N., EFIMENKO, V.F., MATVEENKO, I.P., et al., Justification of characteristics of lead cooled fast reactors on the critical assembly BFS–61, Abstracts of paper presentations, Seventh All Soviet Union seminar on the issues of reactor physics (1991) 68–69 (in Russian).

[11] MANTUROV, G.N., INDECS code and archives system, Issues of Nuclear Science and Technology, Series: Nuclear Data Base, **5** (89) (1984) 20 (in Russian).

MONONITRIDE U–Pu MIXED FUEL AND ITS ELECTROCHEMICAL REPROCESSING IN MOLTEN SALTS

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Abstract

A synthesis process of mixed U–Pu mononitride fuel from initial oxides and alloys has demonstrated its promising nature and feasibility of achieving low contents of oxygen and carbon. The UPuN cores with 92–94% TD was fabricated by compacting and sintering at 1600–1700°C, using the mononitride produced by a carbothermal method from oxides. It has shown that mononitride fuel is compatible with ferritic-martensitic steels at 1200–1300°C for 5 hours. It has also demonstrated that mononitride fuel has a high radiation resistance and a good compatibility with structure materials under irradiation in BR-10 and BOR-60 fast reactors. A compact electrochemical process for nitride fuel reprocessing with a significant removal of fission products and a small amount of high-level wastes has been developed and demonstrated its feasibility to develop a new nitrides synthesis process for establishing a safe closed fuel cycle.

ANNOTATION

The processes of the production mixed mononitride fuels from the parent U–Pu alloys and their oxides are being developed in VNIINM. This is the result of their final products of pyroelectrochemical, water–chemical processes that use mononitride fuel reprocessing and the parent materials for mixed nitrides production from the oxides using carbothermal process at ~1750°C (Fig. 1) [1–7].

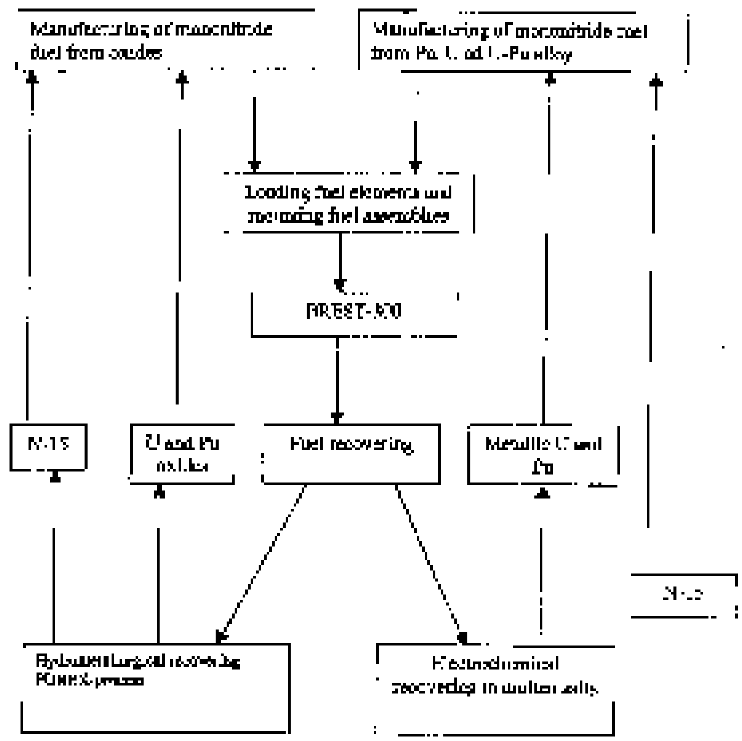

FIG. 1. The diagram of closed fuel cycle.

Moreover it is intended to use a mixed nitride produced from the oxides of the power plutonium and uranium for the first loading of the BREST–OD–300 core. It has been demonstrated that the synthesis process of the nitride fuel is simpler, has less operations does not need complex equipment, provides an opportunity to develop a continuous process and to produce the powder for the direct manufacturing of fuel cores.

As the result of the research of carbothermal process of the mixed nitride fuel production the main process parameters and the future research directions for the fuel quality improvement have been determined.

Radiation tests of the nitride fuel in the reactors SM-2, MIR, BR-10 and BOR-650 under the heat density 350–1000 W/cm demonstrated its high performance, the compatibility with structural steels and metal coolants (Na, Na–K, Pb, Pb–Bi, etc.).

The nitride U–Pu fuel properties and experience available on electrochemical refining and metals production were the basis for the research and development of the electrochemical process for this fuel reprocessing in the molten haloid salts. The first stage of the research has been finished and it demonstrated the feasibility of this process.

1. INTRODUCTION

New criteria to the fuel production technology, its reprocessing, waste minimization and fission products transmutation are imposed by the nuclear power development [8, 9].

On site fuel reprocessing including fissile and radioactive materials transportation provides safety improvement and is a non–proliferation measure. The same role the joint circulation of U, Pu and high level activity minor actinides plays within the fuel cycle. In this connection the development of safe, compact dry reprocessing process without fine removal of fission products with small amount of wastes is an important task. We believe that the electrochemical reprocessing is the most promising among other similar processes.

The improvement of the core construction by the use of a mononitride fuel and lead as a coolant provided the possibility to develop the conceptual design of fast reactor BREST-OD-300 with improved safety and economic characteristics in the Research and Development Institute of Power Engineering (RDIPE).

The development of self–protected production and reprocessing processes for nitride fuels provide a basis for the creation of a safe closed fuel cycle. Taking into consideration all the requirements to the mononitride fuel, this kind of fuel (as well as carbonitride fuel) has an unique set of properties; the high concentration of fissile element, thermal conduction (growing with the growth of the temperature), melting point, good compatibility with structural materials and coolants (Na, Pb, Pb–Bi, etc.), and is considered as the most promising fuel for fast reactors (see Table I).

Graphite, chemically pure argon, nitrogen and argon–hydrogen mixture (Ar + 8% H_2) were used as the parent materials.

Uranium dioxide oxygen factor was within 2.01–2.05. The charge containing oxiBy now in Russia and abroad the number of technological researches and reactor tests of mononitride fuel has grown.

TABLE I. COMPARISON OF FUEL PROPERTIES

Properties	$UPuO_2$	UPuN	UPuC
Density, g/cm^3	11.05	14.32	13.62
Content of fissionable element in 1 cm^3 of fuel, g/cm^3	9.74	13.53	12.96
Thermal conductivity, W/m ^{o}C within the range 500–100^{o}C	2.2–2.0	20 – 22	24 – 26
Melting point, K	2950	3050	2700
Interaction with coolant Na, Pb, Pb–Bi	Interaction with Na, producing Na_3UPuO^4 + Q	No interaction with Na, Pb, Pb–Bi	No interaction with Na, Pb, Pb–Bi
Creepage, Mpa at			
1150^{o}C and 1 kg/mm^2	5×10^{-4}	3×10^{-3}	1×10^{-4}
1150^{o}C and 6 kg/mm^2	8×10^{-3}	6×10^{-2}	5×10^{-3}
1450^{o}C and 1 kg/mm^2	9×10^{-2}	2×10^{-3}	9×10^{-3}
1450^{o}C and 6 kg/mm^2	2×10^{-2}	3×10^{-1}	9×10^{-2}

Reactor tests of the mononitride fuel successfully finished or are under progress in EBR-II, Phoenix, BR-10, BOR-60, etc. at different density and burnup. However the scale of technological research and studying of the mononitride fuel properties is tremendously less than similar work with oxide fuel.

Actually by now the starting basis for the gradual studying of the mononitride fuel has been created. Two technological plans were proposed in the field of the synthesis of the mixed mononitride:

A. The production from parent oxides in nitrogen atmosphere by carbotermal method (using oxide of energetic plutonium);
B. The production from parent metals U, Pu using their hydrogenation by hydrogen and nitration by nitrogen.

It is possible to use both methods to manufacture cores with different density, though the process of manufacturing cores with density exceeding 90% using the first method is more difficult and as the result more expensive. Electrochemical reprocessing of a spent nuclear fuel including mononitride fuel in molten haloid salts mostly meets common requirements to the reprocessing of fuel.

It is predetermined by physical and chemical properties of mononitrides of uranium, plutonium and fission products and based on the experimental experience in the field of metals and oxides refining. The main arguments for electrochemical reprocessing of mononitride fuel are following:

- Irradiated mononitride fuel consists of conductive nitrides the formation energy of which is less than the formation energy of their haloid salts. It should make anodic solution of nitrides easier;
- The feature of the electrochemical reprocessing in melts is the compactness of the equipment and the possibility of multiple use of the electrolyte after periodical removal of fission products. The wastes are in the hard form;
- It is difficult to separate U, Pu and minor actinides because of the close values of release potentials;
- Ensuring of nuclear, radiation and explosion safety;

– No need in long holding of irradiated fuel;
– Reduction of fuel amount in use;
– Possibility of fractional release of FP elements with close values of release potentials;
– It is expected that the final product of reprocessing will be the alloy U, Pu, MA and partly with RE, that is suitable parent material for the low temperature (up to 600^{o}C) nitrides synthesis process [4, 10, 11].

It is possible to carry out the process of electrochemical reprocessing in electrolyzers with hard and liquid cathodes.

2. THE STUDY OF SYNTHESIS AND MANUFACTURING TECHNIQUES FOR FUEL COLUMNS OF MIXED URANIUM NITRIDE AND PLUTONIUM

Now there is significant amount of power (reactor) plutonium dioxide. The alloy U–Pu should be used as parent material after development of the electrochemical reprocessing UPuN fuel in molten haloid salts. The use of these parent materials for the mixed nitride fuel production is an important component of the development of the BREST reactor.

That is why the development of methods for producing of mixed mononitride fuel using these parent materials began.

2.1. The study of carbothermal manufacturing technique for mixed mononitride [5, 6]

The study was held in argon atmosphere using multi–chamber facility and adjoint process equipment.

Mixed mononitride, containing ~13% PuN, is proposed to be used in the BREST-OD-300 reactor design. Therefore the charge was calculated using the equation as follows:

$$0.87\ UO_2 + 0.13PuO_2 + 2.04\ C + 0.5\ N_2 + 0.16\ H_2 = U_{0.87}Pu_{0.13}N + 2CO + 0.04\ CH_4.$$

Up to 2% excessive carbon was added to bond oxygen from the poststoichiometric oxide and oxygen and water vapor absorbed by carbon [5]. Uranium and plutonium oxides, as well as carbon in the form of the black, flades and carbon was mixed by ball mill during 4–5 hours. The obtained charge was compacted under 1.5–2.0 t/cm^2 pressure to produce charge pellets of 6–15 mm diameter and 4.0–6.0 mm height. The density of pressed–powder compact with different type of carbon used not varied significantly and was about 40–42% TD. The briquetted charge was loaded into the vacuum–compression furnace with graphite heater and was heated gradually. Two versions of thermal treatment were tested:

– Up to 1400^{o}C heating in vacuum followed by temperature rise in nitrogen atmosphere and soaking at 1600–1700oC during 1–5 hours and cooling in argon atmosphere;
– Heating up to 1700^{o}C in chemically pure nitrogen flow, soaking at the same temperature during 5–7 hours and cooling in argon.

Hydrogen was added in several tests at the final third of the soaking at 1600–1700^{o}C temperature to reduce carbon content.

No carbidizing agent was found to give important advantages. Oxygen and carbon content was within 0.2–0.3% each.

The effect of carbonitriting length was conducted at 1650°C. It was found that at this temperature oxygen and carbon contents reduce from 0.5 and 0.63% to ~0.2 and ~0.3% respectively as thermal treatment duration increases from 1 to 7 hours.

First of all the addition of hydrogen to nitrogen at the final stage assists to reduce a little the content of carbon to 0.15% by weight. The parameter of the crystal lattice $U_{0.87}Pu_{0.13}N$ was 4.891±0.001 Å. In future it is expected to study the optimization of the synthesis UPuN regime.

The started development of the technique to produce mixed mononitride using charge preheating in vacuum at 1400°C (the first option) enabled to reach better results for shorter (by −2 hours) treatment length as compared with the second option. However it seems that the second method will allow to develop uninterruptible process and manufacture the mixed mononitride in large amount.

2.2. The study of mixed nitride synthesis technique using parent metals

Parent metallic U and Pu smelted in vacuum at ~1200°C during 30 minutes. In accordance with the other option after electrochemical reprocessing and refining smelt the alloy U−Pu will go to the synthesis. The mixed nitride synthesis is based on the reactions as follows:

$2Me(alloy) + 3H_2 = 2MeH_3$

$MeH_3 + x\ (1 + x)N_2 = MeN_{1+x} + 1.5H_2$.

The synthesis of the mixed nitride fuel was I the horizontal stainless steel apparatus that was heated by pulled-on furnace. The source U and Pu ingots were placed to the steel tray, which was put into the apparatus. The obtained alloy was hydrogenated with purified hydrogen at 180–220°C and then nitrated by chemically pure nitrogen at 220–550°C. Plutonium content in parent alloy and the change of nitration duration will reduce the amount of the produced sesqui−uranium nitride. This mode of operation resulted in nitrogen content in mixed nitride 6.4–6.7%. The produced mixed nitride looks like a powder with up to 30−40 μm particle size and is suitable for further producing of fuel columns.

It should be noted that hydrogenation and nitration operations are succeeded in one apparatus without overcharging intermediate products.

2.3. Study of fuel core fabrication from mixed mononitride

Mononitride produced by carbothermal method from oxides required additional grinding, fractionation to grain size less than 40 μm and classification. Mixed nitride powder synthesized from initial metals (alloys) proved fit for direct fabrication of the cores. Granulated and original UPuN powders were used for choosing the initial powder form. It was ascertained that compaction in monocavity moulds did not necessarily require preliminary granulation of the powder. Initial nitride powders were compacted under a pressure of 1-8 t/cm^2. The density of the core blanks produced was 45–55% of theoretical density (TD). The blanks compacting pressure effect was mainly pronounced up to 3–4 t/cm^2 [1–6].

The cores were sintered in vacuum and in nitrogen atmosphere ($Ar + N_2$) at a temperature of 1550–1700°C, depending on the nitride powder initial grain size and required density. Figure 2 presents microstructure of ~94% TD mixed mononitride fuel. The grain microhardness amounted to 650–700 kg/mm^2 [1–6]. The UPuN cores fabricated from oxides and metals as initial materials featured a high degree of homogeneity. Schematic flow diagrams of UPuN production from initial oxides and alloys, as well as of core fabrication, are provided in Figs 3 and 4.

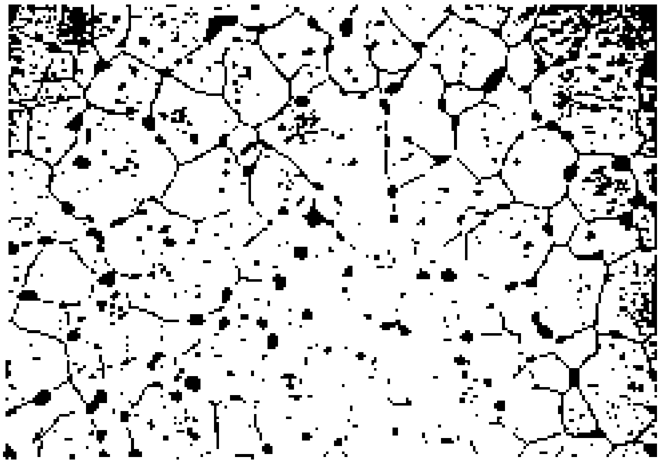

FIG. 2. Microstructure of UPuN fuel core.

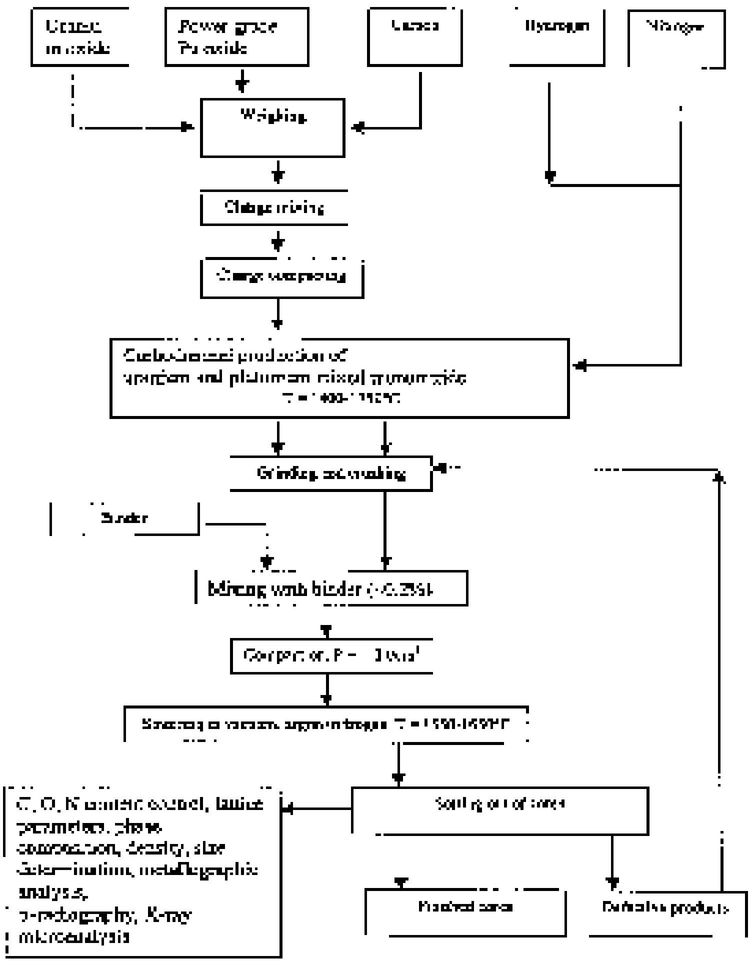

FIG. 3. Flow diagram of mixed mononitride fabrication from U and Pu oxides for first loading of the BREST-300 reactor.

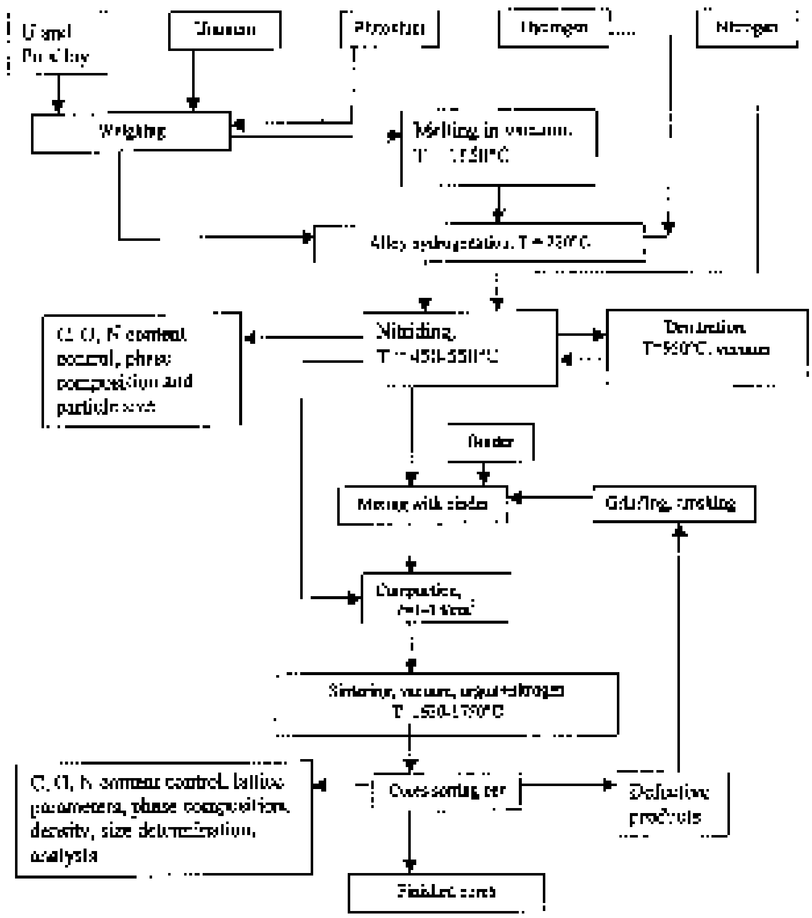

FIG. 4. Flow diagram of mixed mononitride fabrication from metals and alloys.

3. STUDY OF MONONITRIDE FUEL AND LEAD COMPATIBILITY WITH FUEL CLADDING MATERIALS

Compatibility under isothermal conditions of certain compositions was studied, specifically:

- Mononitride fuel (U0.9-0.85PU0.1-0.15N) – steel EP-823, EP-450;
- Mononitride fuel–lead;
- Steels EP-823 and EP-450-lead; steel, lead, mononitride fuel.

Fuel and steel compatibility was studied using lead-filled assemblies/containers with fuel pellets and steel disks (Fig. 5).

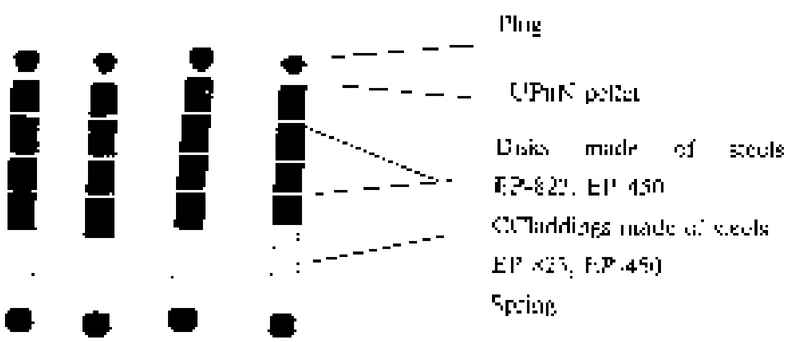

FIG. 5. X-ray patterns of loaded containers for compatibility studies.

The studies permitted ascertaining that mononitride fuel of the above-mentioned composition does not interact either with lead or with cladding steels EP-823, EP-450, EA-172 at temperatures of 650 and 800°C for up to 2000 hours and at temperatures of 1200 and 1300°C (chosen for simulation of emergency situations) for 5 hours. Moreover, no change in the fuel surface layers washed by lead was detected and no changes in near the surface area of steel adjacent to fuel were revealed. Judging by the structure of diffusion samples, steel components penetration into mononitride fuel was also absent, which was confirmed by the results of X-ray spectral microanalysis.

4. RADIATION STUDIES OF MONONITRIDE FUEL

Cores fabricated of initial metals and oxides, the content of both oxygen and carbon in them varying from <0.1% to 0.4–0.5%, were used to study radiation-induced properties of mononitride fuel.

Radiation tests of fuel elements with the cores made of uranium mononitride and uranium and plutonium mixed mononitride, their density being 83–92% TD, fabricated from oxides and metals, were carried out in the SM-2, MIR, BR-10, BOR-60 reactors with linear rating from 350 to 1045 W/cm and burnup up to ~9% of heavy atoms (h.a.). Austenitic steel was mainlyused as cladding material. All the fuel elements retained their leak tightness. It is worth noting that the lifetime of the BR-10 first loading amounted to approximately 5.5 years.

Fuel swelling and release of gaseous fission products (FP) depend on the temperature in the fuel core center and O–C content in the fuel. Reduction of both oxygen and carbon content from 0.4–0.5% to less than 0.15% involves a decrease in swelling from 2.0–2.5% to 1.1–1.5% and gas release from 50–55%, to 20–25%, in case of 9.0% h.a. burnup. Increase in oxygen and carbon concentration in fuel gives rise to increase of interaction and etching zone and decrease in ductility. In the interaction zone a high content of carbon is observed, which is nearly twofold higher compared with the initial one in fuel elements with the cores containing oxygen and carbon in excess of 0.3%.

In the cores fabricated of mixed uranium and plutonium mononitride after irradiation in the BOR-60 reactor at linear rating of 1000–1045 W/cm no change in plutonium concentration from the center to periphery was observed, iodine and caesium corrosion of the cladding being absent.

5. RESULTS OF NITRIDE FUEL ELECTROCHEMICAL REPROCESSING STUDY

Research on non-irradiated mixed mononitride fuel reprocessing by the PUREX-process, conducted at VNIIM, showed that it can provide quantitative recovery of uranium and plutonium actually under conditions of oxide fuel reprocessing. The process of mixed mononitride fuel dissolution occurs steadily and at a higher rate than that of oxide one. The behaviour is similar to the one observed during uranium mononitride dissolution. Uranium and plutonium oxides are final products of the reprocessing. The studies have shown feasibility of making use of the production line for oxide fuel reprocessing available at the RT plant (PA "MAYAK").

The UN cores, their density being 85–92% TD, were used in the studies. Experimentally chosen equimolar mixture of potassium and lithium chlorides with 9–10% UCl_3 was used as electrolyte. To avoid a special operation of UCl_3 production, it is introduced into the electrolyte by means of metal uranium anodic dissolution prior to refining. Argon is used as gas medium over the electrolyte. Electrolysis is conducted at a temperature of 600–650°C [4, 10–12].

TABLE II. RESULTS OF UN ANODIC DISSOLUTION [4]

Temperature (^{o}C)	Current density (A/cm^2)	Anode potential (V)	Current efficiency (g/A–h)	Yield of trivalent ions (%)
500	0.1	-0.32	2.873	89.2
	0.25	-0.11	2.679	62.0
	0.5	+0.15	2.451	31.2
600	0.1	-0.41	2.917	94.3
	0.25	-0.35	2.89	83.8
	0.5	-0.22	2.644	57.3

The data obtained show that uranium mononitride decomposition occurs with U tri– and tetravalent ions dissolution in the electrolyte. Temperature growth promotes increase in the share of the lower valency ions, whereas increase in anodic dissolution rate results in its decrease. The studies performed permitted recommending the following electrolysis conditions [10, 12]:

– Temperature, ^{o}C	500–620
– Current density, A/cm^2	
• Anodic	0.15–0.2
• Cathodic	0.4–0.6
– Gas medium	Argon
– Reprocessing rate, g/cm^2 h	0.45–0.6

Schematic flow diagram of mononitride fuel electrochemical reprocessing is suggested (Fig. 6) on the basis of the research performed. It makes allowance for the results of studies on similar reprocessing techniques of metal, nitride and oxide fuel in Russia, USA and Japan.

Spent mixed mononitride fuel, its cladding being removed, is loaded into a perforated graphite crucible or molybdenum metal basket, serving as anode. Anodic decomposition (dissolution) of nitride fuel is carried out at a temperature of 550–650^{o}C. The available experimental data permit claiming that anodic dissolution of irradiated fuel is complete. Noble metals (~99.9%), molybdenum (~98%), technetium (~95%) and zirconium (~98%), will precipitate in subanodic space as slime.

Rare earth elements (REE), Sr, Cs, I and other fission products will pass into the electrolyte. Minor actinides (MA) in the form of nitrides, in a way similar to uranium and plutonium nitrides, will decompose (dissolve) in the electrolyte. Strontium (Sr), caesium (Cs) and, partially, iodine (I) will pass to the electrolyte.

A portion of elementary iodine can be released to the electrolyzer atmosphere. It has to be bound by a getter. Acceptable physicochernical properties of Ce, Mn, Y, Na and their good compatibility with structural steels permit their use as getters.

Nitrogen evolved in the course of anodic dissolution as a gaseous phase can be absorbed by the getter. At the stage of fuel dissolution, U, Pu and MA nearly completely pass to the electrolyte.

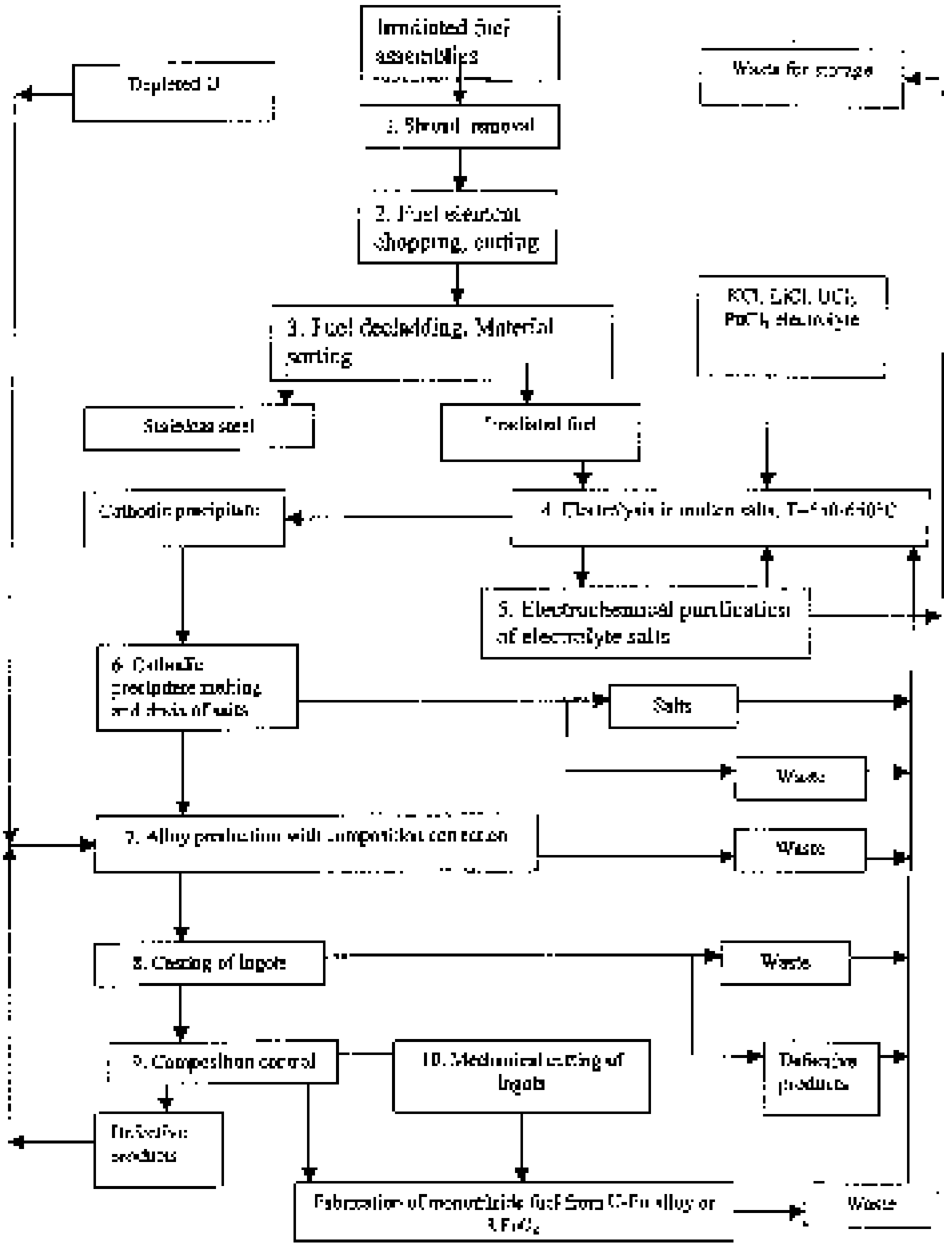

FIG. 6. Flow diagram of mixed mononitride fuel electrochemical reprocessing in melts [10, 11].

Cathodic separation of uranium, plutonium from the electrolyte is performed at a temperature of 550–650°C. There are two variants of U, Pu and MA separation, i.e. on solid and liquid cathodes. From technological viewpoint, it proved appropriate to separate the metals on solid cathode. The use of liquid Cd cathode will necessitate its subsequent sublimation, although the use of liquid cathode permits obtaining U, Pu, MA in a more compact form. In the process of electrochemical refining REE (5–10% of the amount formed) are separated on the cathode simultaneously with U, Pu and MA. Total amount of U and Pu recovered from spent fuel and separated on the cathode, will make up ~99.95% from the amount received for reprocessing.

As a result of electrochemical reprocessing, U, Pu, most of MA and 2–10% REE fission products are separated on the cathode. The resulting cathode precipitate containing U, Pu, MA, REE and 25–35% of electrolyte is fed for separating melting. The metal alloy produced, i.e. U–Pu–MA–REE after its composition correction (^{238}U is added in the amount approximately corresponding to burnt out Pu) is fed for low–temperature refabrication of nitride fuel. Electrolyte separated during melting is returned to the electrolyzer. The electrolyte is reused for fuel reprocessing. After accumulation of 10–15% FP the electrolyte is fed for purification and composition correction.

The FP separated as metals, oxides, chlorides in solid form are compacted and shipped for disposal or long–term controlled storage. It is assumed that from 0.01 to 0.05% of uranium and plutonium remain in solid waste resulting from mononitride fuel electrochemical reprocessing. Fission products in the waste are present partially in elementary state and partially in the form of chlorides (in spent electrolyte).

In the process of electrolysis, uranium and plutonium separation on the cathode takes place. Simultaneously most of MA is separated, their isolation potentials being similar to the ones of uranium and plutonium. MA low concentration in the electrolyte means that the process of their reduction on the cathode is longer than that of U and Pu.

A portion of MA remains in the electrolyte making up 2–10% of the initial amount in spent fuel. After removal from the electrolyzer of the cathodic precipitate containing U, Pu, MA and REE, the electrolyte is purified electrochemically from MA and REE, separated on the cathode. By varying electrical characteristics of the process one can implement fractional purification of the electrolyte, separating MA and REE. The known thermochemical characteristics of most elements classified as FP show promise of successful solution of the problem.

MA, REE and dissolved technetium (Tc) can be extracted from the electrolyte by means of its 2–3-fold electrochemical purification with isolation on the cathode. Caesium (Ce) and strontium (Sr) can be extracted from the electrolyte electrochemically using the liquid cadmium cathode. Iodine evolved can be bound by getters made of Y, Na, Mn, Ce into thermally stable compounds YI_3, NaI, MnI_2, CeI_3.

Thus, it proved possible to separate most of FP on the cathode by means of electrochemical reprocessing. Insignificant amount of FP, Pu (<0.01%) and U (~0.05%) will be forwarded to a long–term storage of high–level waste. A process for electrolyte purification from Pu, Am and FP by zone melting with a high efficiency has been developed at NIIAR.

During electrochemical reprocessing, U, Pu, MA and, partially, REE are separated. The produced U–Pu alloy is similar in its isotopic composition to the initial one and obviates the necessity of pure plutonium introduction in the course of fuel refabrication. It proved more preferable to use the liquid cathode. Joint separation of U, Pu, MA, and partially REE, featuring a powerful radiation, increases self–protection of the process and permits developing new processes of nitrides synthesis.

6. CONCLUSIONS

1. The studies on synthesis of mixed U–Pu mononitride fuel from initial oxides and metals have demonstrated its promising nature and feasibility of achieving low contents of oxygen and carbon (less than 0.2% for fuel made of oxides and 0.1% of metals).
2. It is shown that cores with 92–94% TD can be fabricated by compacting and sintering at a temperature of 1600–1700°C [1–6].
3. Mononiride fuel is compatible with ferritic-martensitic steels EP–823, EP–450 and lead up to 800°C for 2000 hours (period of taking measurements) and at 1200–1300°C for 5 hours.
4. Mononitride UN, UPuN fuel has high radiation resistant properties and compatibility with austenitic and ferritic-martensitic steels under irradiation in the BR–10, BOR–60 reactors at heat rating of 350–1045 W/cm up to burnup from 4 to 9% h.a. [5].
5. Feasibility of setting up a compact pyroelectro chemical process of nitride fuel reprocessing in molten halide salts is shown.

REFERENCES

[1] BLANK, H., et al., Dense Fuels in Europe, J. Nuclear Mater., No. 166 (1989), 95 - 104.

[2] ARAI, Y., FUKUCHIMO, S., Fabrication of UPuN Fuel Pellets, J. Nucl. Mater., **168** (1989) 280.

[3] ROGOZKIN, B.D., RESHETNIKOV, F.G., BIBILASHVILY, Yu.K., Production and Studies of U and Pu Nitrides as Nuclear Fuel and Forms of Weapon's Grade Plutonium Storage, Global–1995, **2** (1995) 1359.

[4] ROGOZKIN, B.D., ORLOV, V.V., et al., Properties and Synthesis and Reprocessing Technology of Mononitride Fuel for Inherently Safe Reactors, Proceedings, International Topical Meeting on Advanced Reactor Safety, ARS'94, 17–21 April 1994, Pittsburgh, USA.

[5] ROGOZKIN, B.D., STEPENNOVA, N.M., FEDOROV, Yu.E., et.al., Mononitride Mixed Fuel for Fast Reactors, Paper presentation, Technical Committee Meeting on Unconventional Options for Pu Disposition, 7–11 November 1994, Obninsk, Russian Federation.

[6] RESHETNIKOV, F.G., ROGOZKIN, B.D., et. al., Study of Methods for Uranium Monocarbide and Mononitride Core Fabrication for Fuel Elements Used in Fast Reactors, Atomnaya Energiya, **35** (6) (1973) (in Russian).

[7] VOLSKY, A.N., STERLIN, Ya.M., Plutonium Metallurgy, Nauka Publishing House, Moscow (1967) 36 (in Russian).

[8] ORLOV, V.V., SILA-NOVISTSKI, A.G., et al., Mononitride Fuel and Large Scale Nuclear Power Industry, Studies on Fuels with Low Fission Gas Release, IAEA-TECDOC-970 (1996), Vienna, 155.

[9] ORLOV, V.V., GROMOV, B.F., SOLONIN, M.I., ADAMOV, E.O., Nonconventional Inherently Safe NPP Concepts, Atomnaya Energiya, **72** (4)(1992) 317 (in Russian).

[10] The White Book of Nuclear Power, edited by ADAMOV, E.O., Moscow, (1998).

[11] KOBAYACHI, F., OGAWA, T., AKABORI, M., KATO, Y., Anodic Dissolution of Uranium Mononitride in Lithium Chloride Eutectic Melt, J. Am. Ceram. Soc., **78** (8) (1994) 2279–2281.

[12] SHENTYAKOV, V.V., et al., Patent No. 2079909 of May 20, 1997, Priority as of September 27, 1994 (in Russian).

THE BOR-60 LOOP-CHANNEL DESIGN FOR TESTING THE BREST REACTOR FUEL

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Abstract

The paper is devoted to development and calculated substantiation of the design of the autonomous lead-cooled loop (AILCL) intended for testing the fuel pins prototypes in the BOR-60 reactor for the BREST-OD-300 reactor. The design features of the loop, its characteristics, instrumentation are considered. The auxiliary systems required to provide the loop operation are described. The main neutron-physical and thermohydraulic characteristics of the loop are presented. The basic operating conditions of the loop are shown. Great consideration is given to analysis of faults and failures affecting the loop serviceability and reactor safety. It is shown that the BOR-60 reactor safety is provided under all normal modes and postulated failures.

1. INTRODUCTION

During the BOR–60 operation a great experience was accumulated on testing fuel, structural and absorbing materials (Table I). A modern complex of material science laboratories at RIAR allows all necessary investigations of irradiated materials. The tested materials include 311-823 ferrito-martensitic steel and UPuN nitride fuel – analogues of materials used in the BREST-OD-300 prospective fast reactor. High irradiation parameters were obtained.

A specific feature of the operating conditions of the BREST reactor fuel and steel is their contact with the aggressive coolant-lead. Therefore, to substantiate serviceability of these materials there must be, if possible, full modeling of the required parameters and conditions. The only facility, at which these investigations can be performed, is the BOR-60 experimental fast reactor. A cell available in the 5th row of the core with direct access to the reactor allows instrumented investigations of different materials under any modes up to emergency ones.

A great experience of designing and testing different experimental devices was accumulated. The most complicated device is an autonomous loop channel made by a 2-circuit scheme. The channel is made as a capsule and is entirely in the reactor with the required communications brought outside. The channel was used in the flow rate blocking experiment up to destruction of fuel pins. The channel was designed so that all fragments of fuel pins remained inside, nothing got into the reactor, no discharge problems occurred.

The gained experience allows us to hope for the possibility of creating the analogous lead-cooled channel.

TABLE I. MAIN TRENDS OF RADIATION INVESTIGATIONS

Material		Type
Fuel	ceramics	UO_2, UO_2-PuO_2, UC, UN, UPuN, UPuCN
	metal	U, UPu, UPuZrNb
	cermet	U-PuO_2, UO_2-U, UN-U
Absorbing	samples	Ta, H_f, Dy, Sm, Gd, AlB_6, AlB_{12}, EuO_3
	CPS rods	CrB_2, B_4C, $Eu2O_3$, Eu_2O_3+H_2Zr
Structural	stainless steels	0Kh18N9, Kh18N10T, EP-450, EP-823, 03Khl6N9M2, EP-912, EI-847, EP-172, ChS-68, VKh-24
	high-nickel alloys	E-16, Kh20N45M4B, VtsU
	refractory materials	V, W, Mo, Nb
	zirconium alloys	E-110, E-635, E-125
	graphites	GRP-2-125, M6-6, GR-280, ARV, IG-11, PGI
Electro-technical	insulation	$A1_2O_3$, SiO_2, Si, mica
	cables	KTM, KNMS(N)
	magnets	UNDK
Others	special ceramics	GB-7, IF-46, TsTS, $LiNbO_3$
	biological protection materials	concretes

The design studies confirmed this possibility. Table II presents some characteristics and parameters of the BREST reactor as compared to the design parameters of the lead loop. It is obvious that they are close enough, which allows modeling of the operating conditions of the upper part of the BREST reactor fuel pin. The autonomous lead-cooled channel (ALCL) is designed to substantiate the design solutions of the created fuel elements of the lead-cooled BREST-OD-300 fast reactor, to perform in-pile tests of the fuel pin models.

TABLE II. SOME COMPARATIVE PARAMETERS OF THE BREST AND BOR-60 REACTORS

Parameter	BREST			BOR-60
	Central area	Interim area	Periphery area	Core 5th row, lead loop
Fuel height in fuel pin, m	1.1			0.45
Fuel pin diameter, 10^{-3} m	9.1	9.6	10.4	9.4
Relative step of fuel pins	1.3–1.5			1.25
Fuel	(88%U–12%Pu)N			
Linear power, W/cm	427	413	353	334
Coolant velocity, m/s	1.8	1.66	1.52	1.1
Inlet temperature, ^oC	423	423	423	470
Outlet temperature, ^oC	575	579	604	544
Cladding temperature (internal surface), ^oC	658	666	668	640
Cladding temperature (external surface), ^oC	629	642	649	615
Maximum fuel temperature, ^oC		974	971	950

The autonomous lead-cooled channel (ALCL) is designed to substantiate the design solutions of the created fuel elements of the lead-cooled BREST-OD-300 fast reactor, to perform in-pile tests of the fuel pin models. The loop is installed into the standard cell D23 of the 5th row of the core having direct access from above through the rotary plugs of the reactor.

To provide the serviceability under different operating modes the autonomous lead-cooled loop is equipped with the following auxiliary systems:

– System of the loop heating up and lead filling and maintaining the oxygen potential;
– Gas system of failure detection of the loop fuel pin claddings;
– Information system to measure the main loop parameters and to create an archive.

The following principles make a basis of the loop design:

– Preserving the loop integrity under normal operation and design basis accidents;
– Exclusion of the possible violation of the operational limits and/or safe operating conditions during installation into the reactor, operation and withdrawal from the reactor;
– Loop operation must not give rise to misalignments of energy release, which can cause damage of the reactor fuel pins;
– Exclusion of the possible unforeseen movement of the loop causing the change of reactivity.

2. LOOP DESIGN

Table III presents the main loop characteristics.

The loop gas cavities are blown with inert gas–argon. The working overpressure of inert gas in the loop – 0.005 ~0.01 MPa.

TABLE III. MAIN CHARACTERISTICS AND PARAMETERS OF LEAD LOOP

Parameters	Value
Power, kW	53
Linear power, kW/m	33.4
Core height, m	0.45
Power non-uniformity factor over fuel pin height	1.16
Lead coolant	OOGOST-22861-77*
Pump	Axial centrifugal, immersion type
Pump head, m	2.5
Lead velocity, kg/s	375
Lead rate, m/s	1.1
Sodium flow rate, kg/s	0.5
Lead temperature at the core inlet/outlet, ^{o}C	470/560
Electric heater power, kW	10
Oxygen content in lead, 10^{-6} %mass	≤ 5
Heating up-cooldown rate, ^{o}C/min	< 1
Sodium temperature at the loop inlet/outlet, ^{o}C	320/410

* index of the State Standard.

During the BOR-60 shutdowns and transportation-production operations there must be provision of the lead coolant melting. For this purpose in the annular channel below the loop pump there are heating pins with the total power of 9 kW.

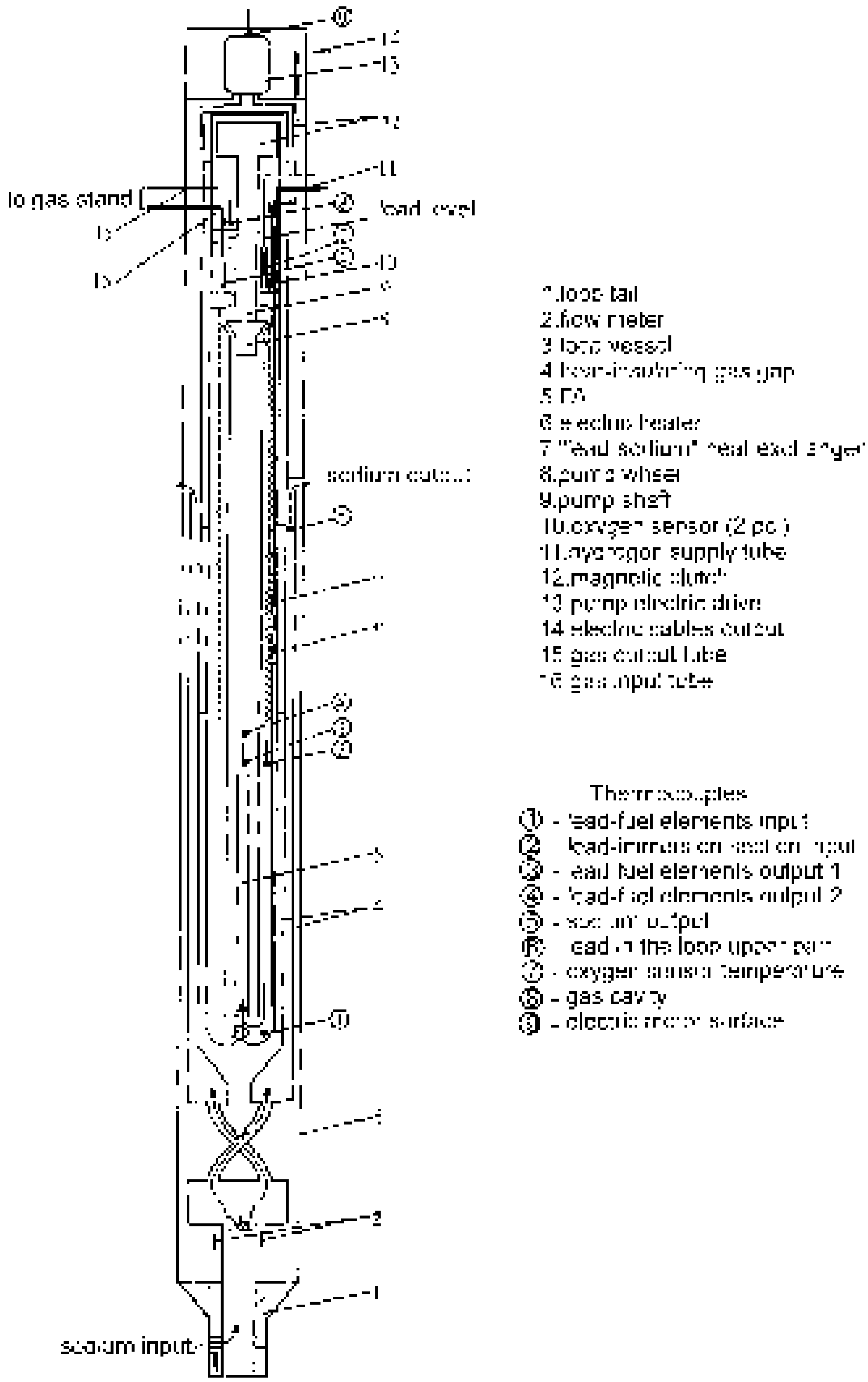

FIG. 1. Scheme of the lead capsule-loop.

The loop being developed incorporates the following pins (Fig. 1):

– Fuel assembly (FA) with fuel pins under test;
– Centrifugal pump for the lead coolant circulation; electromagnetic clutch;
– Lead chamber, sodium chamber, upper wrapper, lower wrapper with a tail;
– The pump electrically driven tight box;
– Tight cylindrical wrapper for the electromagnetic clutch and cables connectors;
– Gas circulation loop with measurement systems (volumetric activity, pressure, flow rate).

Inside the channel there is the following:

– Thermocouples;
– Electromagnetic eddy transducer of sodium flow rate;
– Activity solions;
– Section electric heaters.

On the outside of the channel there are feeder cables and tight connectors. The loop FA consists of 4 fuel pins placed in a square lattice (Fig. 2). Main FA characteristics are presented in Table IV.

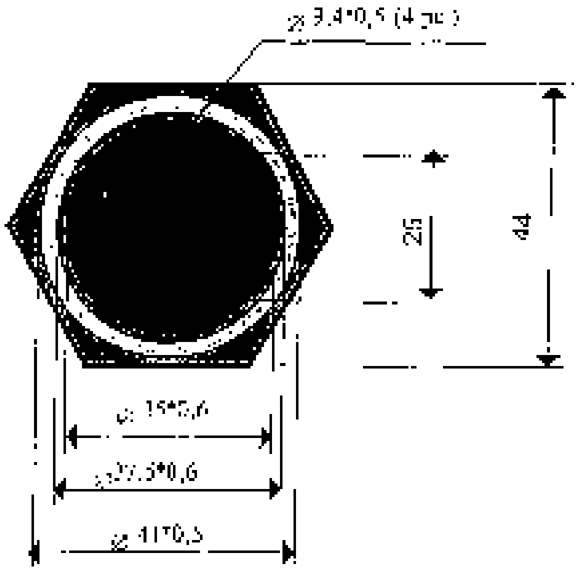

FIG. 2. Loop cross-section in the core central plane.

TABLE IV. MAIN FA CHARACTERISTICS

Parameter	Value
Fuel pins quantity	4
Fuel pins diameter, mm	9.4
Cladding thickness, mm	0.5
Contact sublayer	lead
Relative step of fuel pins	1.25–1.30
Fuel burnup, %h.a.	~4
Core height (fuel part), m	0.45
Power operation time, h	11 000
Fuel	(88% U–12% Pu)N

Figure 3 presents a scheme of the fuel pin. In its lower part there is a tail, a molybdenum bush, on which fuel rods are placed. In the upper part of the fuel pin there is a compensation gas space and upper end of the fuel pin. Circulation of the lead coolant is provided by the axial centrifugal pump of immersion type built in the loop. The moving moment is transferred from the electric motor to the pump part by the electromagnetic clutch. To place the inlet and outlet connections of the blowing gas, to install the connectors for feeder cables from the parameter control sensors and section electric heaters, there was developed a tight cylindrical wrapper installed on the loop upper flange placed in the small rotating plug of the BOR-60 reactor.

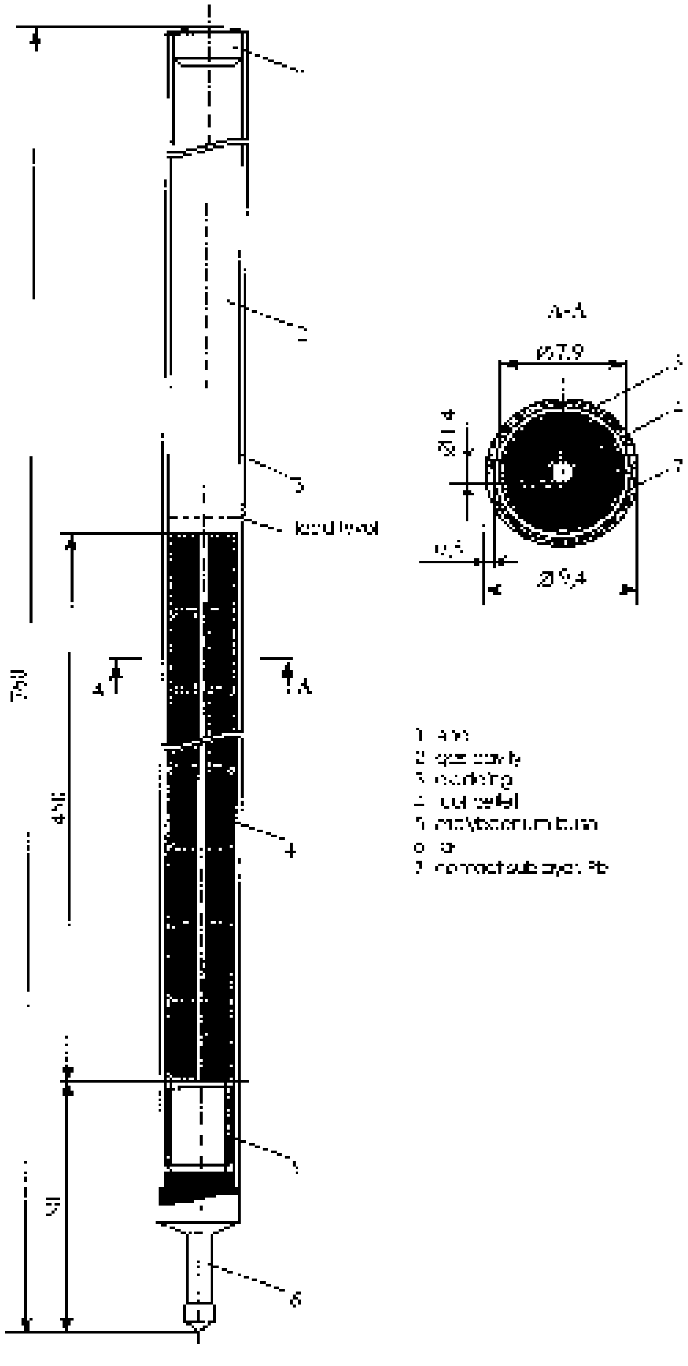

FIG. 3. Loop experimental fuel pin.

The loop is cooled with the BOR-60 reactor sodium, which is supplied from the high-pressure chamber of the reactor collector to the annular gap of the heat-exchange area placed in the loop periphery.

The lead circuit is separated from the sodium one by a double wall with a heat-insulating gap and heat-exchange section that provides the required temperature difference between lead and sodium, lead-to-sodium heat transfer. The double wall also decreases the probability of the inter-circuit seal failure, for which two walls must be destroyed simultaneously. In case of seal failure of one wall the gap will be filled with the coolant and the lead temperature will considerably decrease. It will allow diagnostics of the accident progression at an early stage.

Power supply of the loop is provided from two sources: a pump from the highly reliable uninterrupted power source, and the internal heaters — from the system or switched autonomous power supply.

3. EXTERIOR AND AUXILLARY SYSTEMS

3.1. System of lead filling, coolant preparation and structural materials passivation

The system includes a section electric furnace with power of 9 kW for the loop heating up to 550°C, a lead melting tank with the coolant capacity of 7 L, gas lines for gas mixtures preparation and supply to the autonomous channel, a vacuum system for the channel gas removal, control devices of temperatures, coolant level, gas flow rate and composition (Fig. 4).

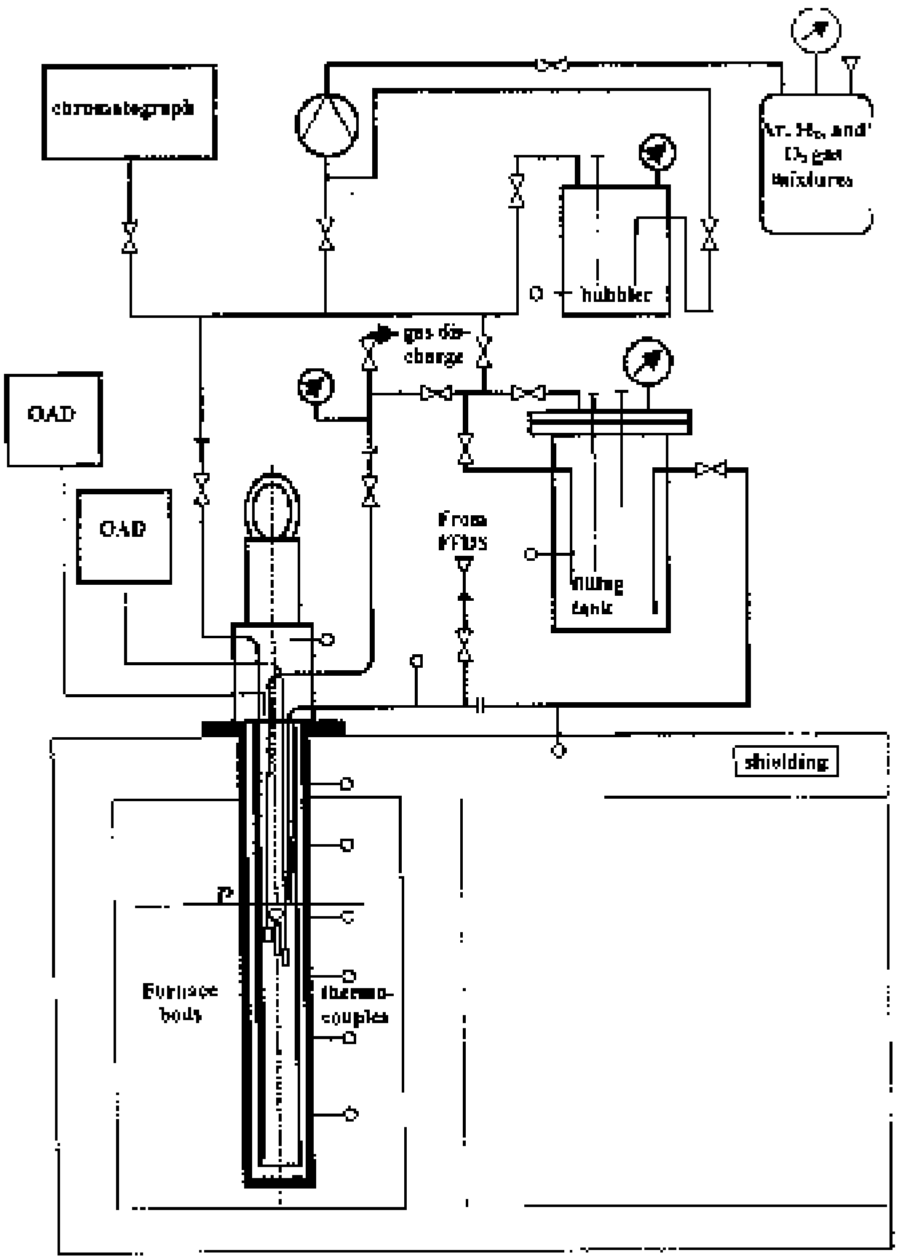

FIG. 4. Flow chart of load coolant loop filling OD – oxygen activity detector.

The section electric furnace is placed in one of the dry keeping channels (DKC) of the BOR-60 irradiated devices. The loop is immersed into the electric furnace, heated up to 500°C. The loop lead circuit is vacuumized and filled with C00 lead from the melting tank. The lead filling level is controlled by indications of the in-loop thermocouples and indications of the mobile level indicator in the melting tank. The required level is set by the pressure difference in the loop and tank gas cavities. After lead filling, the coolant circulation is organized using a centrifugal pump to mix it in the channel and to control the thermo-dynamic oxygen activity using the DAK sensors. If the thermo-dynamic activity (TDA) is less than 0.5×10^{-5} rel. un. at 500°C, the coolant oxygen saturation is needed to oxidize the materials surfaces.

The lead circuit materials are passivated to create oxide coatings on the internal surfaces and to repair defects in the pre-applied oxide coatings. The oxygen activity under the passivation mode is maintained in the range of $(0.5–1) \times 10^{-3}$ by supply of the argon–oxygen mixture (up to 20 vol.%) to the coolant via the gas tube above the coolant level. At the same time the in-loop coolant circulation is maintained. Passivation is completed at TDA increase over 1×10^{-3} rel. un.

Regeneration is carried out at oxygen excess in the lead circuit: TDA >5×10^{-3} at 500°C. Regeneration is performed by supply of the recovery gas mixture (argon–hydrogen–water). A gas mixture is prepared comprising: Base-technically pure argon; hydrogen – up to 30 vol.%. The ratio of water–hydrogen partial pressure is $1 < P_{H_2O}/P_H < 10$. Gas mixture is fed to the coolant by a tube below fuel pins at 500°C. The oxygen thermodynamic activity TDA of coolant is monitored with an oxygen activity detector. The coolant regeneration is completed when hydrogen consumption of mixture is stopped or TDA is decreased to 5×10^{-4} per unit value.

The heated and lead-filled loop is installed in the reactor when a centrifugal pump is switched on and loop heaters are under operation. The reactor sodium temperature must be no less than 200°C; lead is maintained in melting state due to its circulation and energy release of heaters. When next reactor refueling shutdown the loop is removed and the temperature is kept above lead melting at the expense of decay heat and lead circulation. A temporary storage of channel is provided in the electric furnace heated to 400~500°C. With decreasing the residual energy release the pump can be switched off and the temperature is kept using electric furnace regulation.

3.2. In-loop faded fuel detection system (FFDS)

According to safety regulations the FFD system is a necessary loop component and based on measurements of radionuclide volumetric activity in the gas channel cavity. The system must provide gas transport from the cavity over the loop lead coolant, gas volumetric activity measurement and cladding damage identification of tested fuel pins during the wide irradiation period under all reactor operation modes (from irradiation of tight fuel pins at a low burnup to cladding and fuel failure, that corresponds to Xe-133 volumetric activity ranging from 30 kBq/L to 100 TBq/L). Defects are identified using an experience gained in defect identification of the BOR-60 standard fuel pins and their correction by calculated values of radionuclide FP yield with parameters of the BREST-OD-300 fuel pins tests. The FFDS incorporates lines of gas transport from the loop, gas stand consisted of tight blower, measuring volumes, radiometric and spectrometric radiation detectors with collimators, lines of gas supply, blower and sampling, locking fittings, gas pressure and flow rate instrumentation, lead shielded stand as well as information collection and processing system

(Fig. 5). According to measurement conditions the gas stand must be outside the reactor pit and can be located in the BOR-60 central hall. Gamma-spectrometer, secondary radiometric and dosimetric instruments as well as PC providing information collection, processing and storage are located outside the central hall.

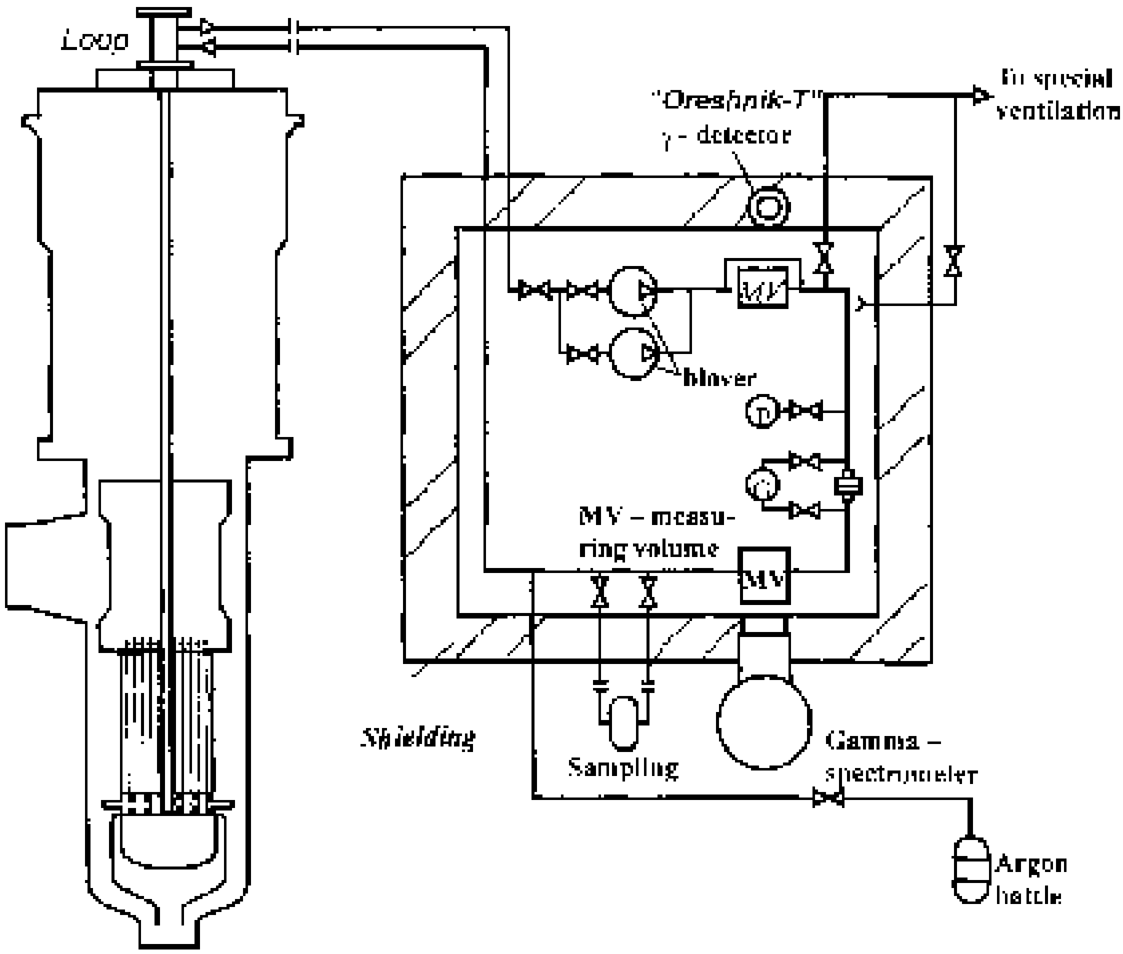

FIG. 5. Flow chart of measuring loop FFDS.

Under the blower action gas is fed below the lead level, bubbles and degases its upper layer, mixed in the gas cavity, sampled in the upper part for measurement and conveyed in the gas transport line from the reactor pit to the gas stand. The radionuclide volumetric activity is measured in the measuring volume with a germanium detector through a collimator in the lead-shielded gas stand. The information treatment, analysis and accumulation are performed with on-line mode PC. To measure the volumetric radionuclide activity the system is calibrated by measurement of gas samples under standard conditions. The failed fuel pin identification and activity change interpretation of the loop gas cavity are carried out by a special technique developed and validated with physical calculations.

The radiometric detector (gamma-counter) provides a continuous indication of the gas activity level. The dosimetric sensors are used for the radiation situation monitoring in the region of gas stand. The locking fitting services to shut off the gas lines when the loop is removed from reactor and to shut off supply, blower and sampling lines through which the whole loop gas volume is filled with the prepared gas mixture. These lines provide gas supply or disposal to special ventilation (if necessary) as well as gas sampling for measurement system calibration. The gas parameters are monitored with pressure and flow rate detectors equipped with corresponding instrumentations.

3.3. Parameter control and information system

The detectors monitoring the coolant and fuel pin state and providing the operative information about the sodium flow rate and temperature in the different loop points are installed along the length of the channel in the upward and downward fluxes of the lead coolant (Fig. 1).

A temperature is measured with thermocouples having 1.0 mm diameter. The position of the upper lead coolant level is monitored with thermocouples Nos 6 and 8. The reactor sodium flow rate of the loop input is monitored with an eddy electromagnetic detector. Flow rate, pressure and temperature detectors are used for monitoring of the lead filling loop and FFDS.

The signals of coolant temperature of the fuel pin output come to the reactor emergency shutdown system. The signal of temperature reduction comes to the alan-n signal system for intercircuit leak monitoring.

The oxygen activity of lead is measured with two oxygen activity detectors. The signals, which are not connected with the BOR-60 control system, arrive to the independent computer information system.
To determine a neutron fluence and spectrum over height the channel is loaded with a tube contained activation monitors. Five monitor sets are arranged on the welded stainless steel tube at the level of 0, ±100 and ±200 mm. from the central core plane. The sets incorporate the following materials: Nb, Fe, Ti, Cu.

The detectors are fabricated from the metal foil of 0.1 mm thickness, from which the disks of 1.0 mm in diameter are cut out. The detector sets are located in the marked quartz ampoules spaced with stainless steel inserts over the tube height.

4. NEUTRON-PHYSICAL AND THERMAL-HYDRAULIC CHARACTERISTICS

All the calculated values are obtained for automated design complex (ADC) of the BOR-0 neutron-physical characteristics based on TRIGEX-CONSYST2-BNAB-90 programme realizing small group (to six groups) diffusion approximation in three-dimensional hexagonal geometry. The isotopic kinetics was estimated by the CARE programme. These programmes are used for the BOR-60 design for a long time. The high accuracy of obtained values was proved with numerous calculated and experimental investigations.

The calculated values of the main loop neutron-physical characteristics (reactor power of 55 MW):

- Maximum linear (volumetric) power: 334 W/cm (76.3 kW/L);
- Channel power of D-23 cell: 53 kW;
- Maximum neutron flux density: 2.55×1015 /cm^2·s;
- Maximum rate of steel damage dose accumulation: 9.6×10^{-7} dpa/s;
- Axial non-uniformity coefficient of fuel column (0.45 m) for the following parameters:
  - Energy release: 1.16;
  - Neutron flux density: 1.15;
  - Rate of steel damage dose accumulation: 1.18;
  - (Pb(n,γ)) neutron-capture reaction rate: 1.14.

- Radial non-uniformity coefficient of D26 cell of core central plane for the following parameters:
  - Neutron flux density: 1.10;
  - Rate of steel damage dose accumulation: 1.14.
- Loop loading reactivity: +0.17%Δk/k;
- Fraction of neutron flux density (above 0. 1 MeV) of loop fuel part makes up 77.5%;
- Average neutron energy of D23 fuel cell: 234 keV.

The burnup of fuel pins was estimated. The following irradiation mode (seven irradiation microcompaigns (MC)) was considered: reactor thermal power of 50 MW; power irradiation time ~100 days. Thus, the reactor energy of one MC is 120 000 MWh and total energy (seven conventional MC) makes up 840 000 MWh. The loop is irradiated during 700 days (16 800 h). During this period the average fuel burnup is 3.3%h.a. and the maximum –3.8%h.a. The rate of fuel burnup changes from 9.7×10^{-7}%h.a./MWD at the beginning to 9.2×10^{-7} %h.a./MWD at the end of irradiation.

During irradiation the isotope mass of U^{235}, U^{238}, Pu^{239} and Pu^{241} is decreased by 20.4% (0.6 g), 3.2% (30.6 g), 6.2% (7.4 g) and 12.8% (0.3 g), respectively. A negligible Pu^{239} burnup is related to its accumulation as a result of U^{238} neutron capture. The isotope mass of U^{236}, Np^{237}, Pu^{240}, Pu^{242}, Am^{241} and FP was increased by (0.1 g), (0.14 g), 37.7% (2.9 g), 6.5% (0.03 g), (0.17 g), and (35.5 g) respectively (* means that isotope was absent in the initial fuel composition).

The hydraulic calculation of lead and sodium circuit was performed by conventional techniques. As a result of the performed calculation a square pressure dependence of flow rate can be obtained:

$\Delta P = k\, G^2$,

where k = 134 for sodium circuit and k = 0.2 for lead circuit;
ΔP = pressure loss (m);
G = mass coolant flow rate (kg/s).

The stationary thermal–physical operation mode of lead loop was estimated by the OSETR programme designed for calculation of temperature field of axis symmetrical heat exchanger. The heat exchanger incorporates M axis symmetrical tube walls forming the (M – 1) annular channel system. The one–phase coolants flow through the channels. The wall thickness, material and thermal–physical parameters can be different. The heat sources (sinks) are provided for both in walls and in coolants. One of three boundary conditions should be given at the internal and external radial boundaries of heat exchanger. The coolant velocity and temperature of the channel input should be known. The heat exchanger is divided into calculated areas over height. The walls are divided into calculated rings on a radius. The system of differential equations defining the temperature field is solved by finite difference method. In this case the following assumptions are made:

- Axial heat flowing along walls is very small in comparison with radial one;
- Uniform field of coolant temperature and velocity;
- Thermal conductivity heat transfer of coolant is negligibly small in comparison with convection heat transfer;
- Absence of dissipation energy;
- Absence of ray heat exchange;
- Incompressible one–phase coolants.

The channel operation was estimated for two modes: “cold” and “hot”. In “cold” mode the channel is loaded in the reactor which is shutdown or operates at a low power. For this mode the inlet sodium temperature is no less than 200^{o}C. As for the lead temperature it is no less than 350^{o}C. In order to achieve this mode an electrical heater should be installed in the lead coolant. The heater power will be dependent on the sodium and lead flow rate, heat exchange height and sodium temperature of loop input. “Hot” operation mode means reactor power of 55 MW. In this case the power released in the loop fuel is 53 kW. The electric heater is switched off. Optimization of thermal–hydraulic characteristics when changing the lead and sodium flow rate, height of heat–exchange area and thickness of gas gap in heat–isolating wall gave rise to some results presented in Table V. The selected structural parameters were used as a basis of a project.

TABLE V. LOOP HEAT-HYDRAULIC CHARACTERISTICS

	“cold” mode	“hot” mode
Lead flow rate, kg/s	3.5	3.5
Sodium flow rate, kg/s	0.11	0.45
Lead temperature of core input, ^{o}C	398	518
Maximum temperature on internal fuel pin surface, ^{o}C	398	620
Maximum temperature on external fuel pin surface, ^{o}C	398	596
Lead temperature of core output, ^{o}C	398	562
Lead temperature of heat exchange input, ^{o}C	497	404
Lead minimum temperature, ^{o}C	398	394
Sodium input/output temperature, ^{o}C	200/470	330/385
Fuel pin power, kW_t	-	53.0
Heater power, kW_t	10	-

5. LOOP OPERATION MODE AND SAFETY ANALYSIS

The manufactured loop is placed in the furnace of lead filling stand and connected to the service lines. The melt lead is pressed to the channel heated to 450–500^{o}C. The level is monitored with two thermocouples located in the upper channel part. In addition the amount of lead pressed to the loop from the tank is monitored using a tank level indicator.

After filling the circulation pump is switched on. The lead is mixed. This loop mode can be maintained for some days. In this case the monitoring temperature distribution through the channel, pump and drives parameters, (temperature, rotation rate, engine current) and oxygen contained in lead are performed. The loop is loaded in reactor at sodium temperature of 250^{o}C. In this case the loop temperature is steadily monitored and the pump is under operation. When delaying loading procedures the internal electric heaters (if necessary) can be switched on. They must be switched on when the channel is inserted in reactor D-23 cell. At zero reactor power and loaded channel the sodium flow rate should not exceed 100 m^3/h.

With rise of reactor power an unspecified procedure is required for step-by-step flow rate increase from about 100 m^3/h to nominal value when the lead maximum (560^{o}C and minimum (390^{o}C) temperatures are monitored. The mode of power rise must also provide non-exceeded ultimate parameters of standard and experimental assemblies.

After nominal power output the process of loop fuel pin irradiation is started up to achievement of 3–4% planned burnup (~1.5 year).

Under irradiation the main parameters are monitored and corrected, if necessary. During long-term operation at low power the lead flow rate is decreased so that maximum operating temperature of 560°C will be approached. The failed fuel pin claddings are checked by steady activity measurement of gas pumping through the stand. The excess of maximum temperature is avoided by set-up of audible signal and blocking signal for reactor shut down (t_l = 560°C, t_2 = 580°C). When the loop pump switching off the melting of channel fuel pins can be prevented by of scram system since time of pump run-out does not exceed 2 s.

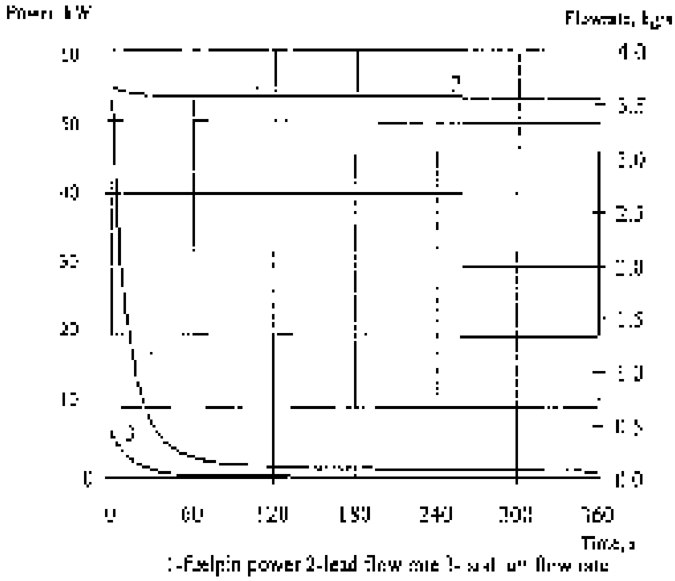

FIG. 6. Change of loop parameters when scram system action.

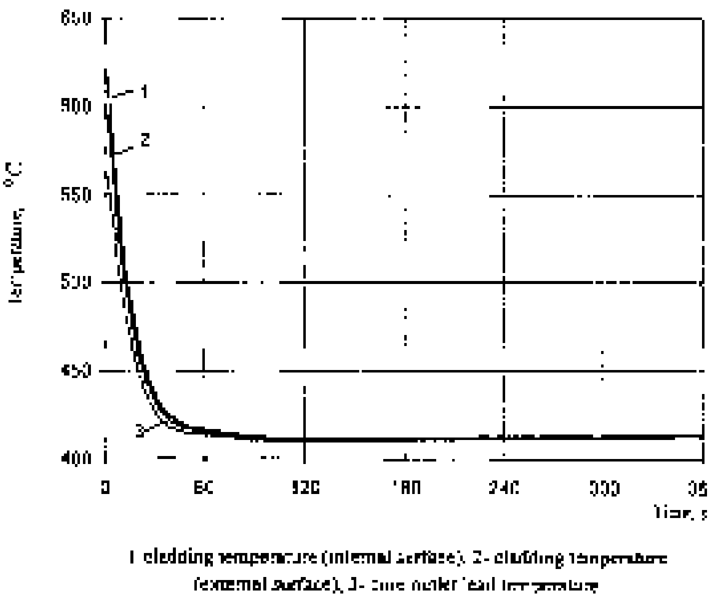

FIG. 7. Change of loop pump temperature when scram system action.

The power is decreased in reverse order using step-by-step decrease of coolant flow rate to ~100 m^3/h and temperature monitoring mentioned above. The reactor emergency shutdown is also designed mode for the channel. In this case the pump of channel continues operating and lead temperature is decreased in accordance with a decrease in the reactor power (Figs 6 and 7). After shutdown the loop is cooled in the reactor for 4 days. In this case if necessary (with decrease of reactor inlet temperature) the internal electric heaters are switched on to prevent lead freezing. The circulating pump continues operating. Within 4 days the loop is unloaded, cleaned from sodium using the standard technology and placed in the furnace of filling stand. During transportation and switched-off internal heater, the loop is cooled with air at the expense of natural convection. Lead temperature is decreased at the rate of $0.02^{o}C/s$. At initial minimum temperature of $390^{o}C$ the lead freezing temperature of $327^{o}C$ will be achieved through about 40 min. In case of electric heater failure or long delay occurred in reloading lead can freeze in the loop. In this case it is necessary to switch off the circulating pump beforehand in order to prevent its failure. In future lead melting should be carried out from top to bottom in sections when the lead-filled loop (LFL) is placed into the furnace. Lead melting is followed by temperature control.

The following failures may occur in the course of channel operation:

- Pump stopping as a result of its de-energizing or seizing;
- Blocking of flow rate in (through) fuel pins;
- Leakage of the channel walls;
- Fuel pin leakage;
- Gas-non-tightness of the loop;
- Channel leakage in reloading.

The above-mentioned faults have been used for making the list of initial events for different accidents. Safety analysis was carried out with the help of the DINBOR verified calculation programme complex and the OSETR programme which was used in analysis of transient and emergency conditions in the loop with natural circulation and was in good agreement with experimental data.

5.1. Pump stopping

An instantaneous seizing of the pump (stoppage of impeller) is considered as an alternative. It is possible in deformation and failure of the pump components and the loop itself. In order to prevent fuel pin melting in quick decrease of flow rate the reactor shut-down should be performed with the help of scram system, which is actuated when the increase of the pump engine current or lead temperature in the loop according to two thermocouples is alarmed. Figures 8 and 9 show the transient processes in the loop for this accident. The scram system is actuated if the temperature is increased from 560 to $580^{o}C$. Due to disagreement of laws on flow rate and power decrease and low level of natural circulation of lead in the loop as well the short-term increase of fuel pins temperature is occurred in comparison with the limiting operating temperatures (fuel pin cladding-up to $740^{o}C$, fuel-up to $1000^{o}C$). Melting of fuel and fuel pin cladding does not take place. As a rule such short-term increases of temperature don't cause fuel pin leakage. Such emergency conditions don't have influence on reactor safety. Blocking of flow rate through fuel pins. This type of accident may occur as a result of blocking of open flow areas in fuel assembly in particular with lead oxides or small size particles, which might be left after installation accidentally. According to the operation experience of fast reactors the probability of this accident is very low and considerable blockage causing fuel pin failure in BN-reactors is placed among beyond-designed accidents. The open flow area of cell in the LFL fuel assembly is by an order of magnitude greater than in BN fuel assembly that is why the probability of this accident in the lead-filled loop is lower.

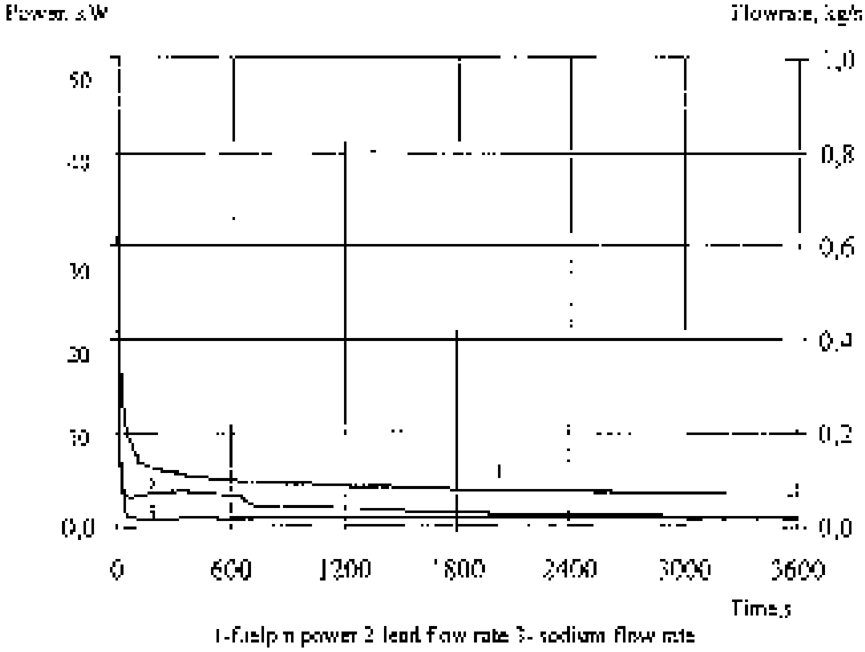

FIG. 8. Change of loop power and coolant flow rate when loop pump switching off and scram system action.

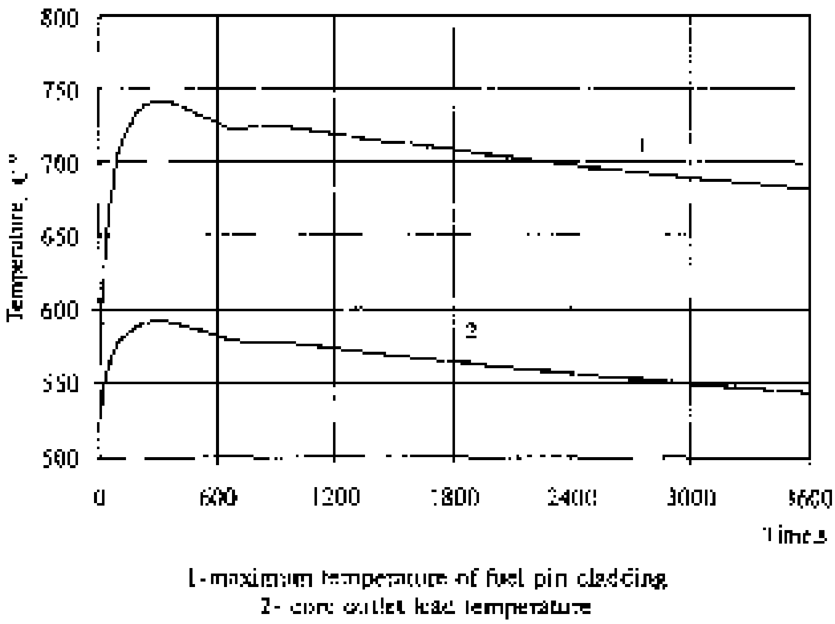

FIG. 9. Change of temperature when loop pump stopping and scram system action.

If the flow rate is blocked the lead temperature will increase and it will be recorded by available thermocouples. The scram system provides the automatic reactor shut-down within ~1 s according to the lead outlet temperature increase in fuel assembly. If the emergency protection system doesn't come into action (beyond-designed accident) fuel pins melt and fuel runs down and penetrates into one of the intercircuit walls. Energy release in the circuit decreases sharply and molten fragments solidify due to good sodium cooling. Some lead find its way into sodium but as far as generally sodium temperature does not exceed 320°C in nominal rating conditions lead freeze and the process is stopped.

5.2. Fuel pin leakage

This event is not the emergency situation for standard fuel pins because the reactor can operate with several leaky fuel pins and the limits of safe reactor operation are not exceeded. However more detailed analysis is required due to the fact that the lead-filled loop has the outer gas circuit. Leakage of one fuel pin out of four is considered to be the most probable situation for two cases: when the gas circuit is tight and when its additional non-tightness is occurred. The probability of the second case is very low because it assumes the simultaneous occurrence of two events. In case of leakage all fission products release in coolant and gas cavity and in case of gas circuit non-tightness they release into the environment. Failed fuel detection system will record this event according to gas activity and a decision about further actions right up to the reactor shut-down will be taken.

5.3. Intercircuit leakage

This situation may be caused by fabrication defects, corrosion processes in lead, thermal stresses. As this takes place, lead (or sodium, in case of outer wall leaking) fills up the gap, heat-transfer efficiency increases sharply and lead temperature decreases by 50 ~100°C along all sections. This event is alarmed and the reactor operator has time for the reactor shut-down. The probability that two walls leak is low and this event may be placed among beyond-designed accidents. However we study it.

If two walls leak (we postulate the conservative leakage in the lower part of the lead-filled loop), lead finds its way into the gas cavity that is formed between walls and then into the sodium circuit. As a consequence of rather low pressure in the circuit, great break of walls is practically impossible (equivalent diameter of 2–2.5 mm for a hole is taken conventionally). That is why lead flow rate through the leakage place will be low and lead will begin running down by sodium circuit under the influence of gravity up to the moment when lead and sodium pressures are equalized at the leakage place. It will take place within 25–30 s and approximately 2.31 L of lead out of 5 L will discharge into the loop. Lead freezing (327°C) in the lower part of the loop nearby the reactor collector will be caused by rather high sodium flow rate and its low temperature (as a rule no more than 320°C). Small portion of lead can come into the collector but the probability is very low, because the solidified lead blocks the open flow areas with respect to sodium.

Even if all displaced lead (2.3 L) finds its way into the collector, its sodium concentration will be 0.1 weight %. According to the published data sodium solubility of lead in the range of 100–260°C is evaluated by means of the following expression:

$$\mathrm{Lg}S = 6.06 – 2465/T, \text{ weight\%}.$$

If we extrapolate this function up to 320°C (outlet temperature) we receive S = 80 weight%. When the temperature is 250°C (minimum sodium temperature) S is 22 weight%, that is, lead is dissolved in sodium completely and with great margin. It will be present in it in the form of Na_4Pb(Na_5Pb) or $Na_{15}Pb_4$ chemical compounds. The great solubility margin excludes lead precipitation or suspension formation as well as the influence on hydraulic and heat exchanging properties at all operating temperatures. Lead content in fast reactor sodium is not normalized. It should be noted that lead shows high corrosivity with respect to generally used reactor materials (1X18H9). The corrosion rate of 1X18H9 steel in pure lead is 0.5 mm per year at 600°C and it is 0.1 mm per year at 400°C. As lead thermodynamic activity is proportional to its small portion, it will be nearly four orders of magnitude lower than for pure lead. Accordingly the corrosion rate will be substantially lower. In such a way the considerable influence of dissolved lead on corrosion of structural materials is not expected for such rare accident. In case of sodium circuit blocking owing to sodium discharging and freezing cooling of LFL fuel pins will become worse sharply and in case of the strong blocking it will be carried out owing to sodium leakage between fuel sub-assemblies. Possible consequences of blocking are studied above.

5.4. Gas-non-tightness of the loop

This situation may occur owing to leakage of loop walls nearby the upper gas cavity or outgoing gas tubes. As this takes place, the gas from the loop may release into the reactor sump or in the reactor hall nearby FFDS gas stand. The consequences will have a radiation character and they are studied below. Non-tightness is recorded by radiation monitoring system and then the reactor shut-down is performed.

The detailed analysis of radiation conditions during ordinary experiment on fuel pin irradiation in the lead-filled loop and as well as during different emergency conditions has been carried out. Well-known methods that analyze the activity of fission products released under the fuel pin cladding and from under it into the coolant have been examined. The acceptable conditions for fuel pin irradiation in the LFL model and constants have been selected. Radionuclide activity in fuel pins, coolant and the LFL gas cavity have been calculated. Possible radionuclide leaks in case of emergency situations have been estimated, and radiation consequences of such events have been analyzed. Results of lead activity calculation are given in Table VI.

TABLE VI. SPECIFIC ACTIVITY OF LEAD, IMPURITY AND CORROSIVE PINS IN THE LEAD-FILLED LOOP

Radionuclide	$T_{½}$	Equilibrium activity (GBk/L)
Pb-203	52.1 h	0.37
Pb-204m	66.9 min	37.0
Pb-207m	0.8 s	1850
Pb-209	3.31 h	7.4
Zn-65	245 days	0.26
As-76	26.3 h	4.1
Sb-124	60.2 days	3.7
Po-210	138 days	0.037
Mn-54	280 days	0.185
Fe-55	2.7 yrs.	1.85
Co-60	5.3 yrs.	1.85×10^{-2}

Fission product activity in fuel is the main source of possible radiation consequences in different emergency conditions. Accumulation of activity with regard to BOR-60 neutron-physical parameters in D-23 cell (where the loop is located), fuel composition, temporary irradiation conditions and decay were calculated in accordance with the AFPA and CARE programmes. The reliability of calculations performed in accordance with the programmes is confirmed by coincidence of the calculated and experimental results on fuel composition after long-term irradiation of fuel assemblies of different compositions. The results are within the limits of experimental errors and (Table VII).

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TABLE VII. ACTIVITY OF MAIN FISSION PRODUCT GROUPS IN FUEL PIN, Bq/FUEL PIN

Nuclide	$T_{1/2}$	Irradiation time		
		1 year	1.5 year	2 years
Gases				
Xe-135	9.08 h	214E+11*	265E+11	264E+11
Xe-133	5.24 d	202E+11	251E+11	249E+11
Xe-135m	15.6 min	467E+10	579E+10	576E+10
Kr-88	171.6 min	460E+10	571E+10	567E+10
Kr-87	76.4 min	359E+10	446E+10	444E+11
Volatile (T < 1000 K)				
I-134	52.6 min	212E+11	264E+11	262E+11
I-133	20.8 h	200E+11	248E+11	247E+11
I-135	6.61 h	186E+11	231E+11	229E+11
Cs-138	33.4 min	172E+11	214E+11	212E+11
I-132	2.28 h	153E+11	190E+11	189E+11
I-131	8.04 d	108E+11	135E+11	134E+11
Nonvolatile:				
Mo-99	66.02 h	178E+11	221E+11	220E+11
Ru-103	39.35 d	176E+11	210E+11	228E+11
Rh-103m	56.1 min	176E+11	209E+11	228E+11
La-140	40.2 h	161E+11	200E+11	200E+11
Ba-140	12.8 d	161E+11	200E+11	199E+11
Zr-97	17.0 h	155E+11	193E+11	192E+11
Te-132	78.2 h	149E+11	185E+11	184E+11
Ru-105	4.4 h	148E+11	184E+11	183E+11
Rh-105	35.4 h	148E+11	184E+11	183E+11
Ce-141	32.5 d	144E+11	174E+11	185E+11
Zr-95	64.0 d	122E+11	137E+11	155E+11
Nb-95	35.1 d	114E+11	116E+11	141E+11
Sum of nuclides		3.41E+14	4.17E+14	4.24E+14
Nuclide sum total		1.66E+15	2.06E+15	2.07E+15

* read as 214E+11 = $214 \times 10^{+11}$.

Gas release will not take place in case of cladding failure. In the steady state the gaseous fission product release through the defect in the fuel pin cladding is determined from their delay in convective mixing. According to the published data $\mu = 2 \times 10^{-4}$ L/s is taken for gases and volatile nuclides. Fission product release into the loop gas cavity was calculated with the use of conversion rate factor $\alpha = 10^{-3}$ L/s. Fission product activity in the lead coolant and gas space of the loop was calculated using the estimated values of fission product activity in fuel pin after irradiation during 2 years. The calculations were made for the above-mentioned emergency situations (Table VIII).

Analysis of emergency situations and calculations on radionuclide release and distribution demonstrate high safety and protection. Radiation conditions remain normal during ordinary experiment on the BREST-300 prototype fuel pin irradiation and as well as during the designed accidents connected with leaking in the loop gas cavity. In case of accidents

TABLE VIII. RESULTS OF ANALYSIS FOR RADIATION CONSEQUENCES OCCURRED IN BREST-300 FUEL PIN TESTING IN BOR-60 LEAD-FILLED LOOP DURING ORDINARY IRRADIATION PROCESS AND DIFFERENT ACCIDENTS

Analysed situation	Determining nuclides and their activity	Critical parameter	Estimated value	Allowed value	Consequences
Normal conditions for fuel pin Irradiation	In stand: Ar^{41} – 40 MBq/L H^{3}– 1.4 GBq/L Po^{210} – 53 Bq/L	Dose rate nearby FFDS stand	0.3	7.8	Radiation conditions in BOR-60 working rooms are not changed. Releases are absent.
Gas-non-tightness of the loop in the reactor gas cavity	Po^{210} – 0.008 Bq/L	Change of activity in the reactor gas cavity	Ar^{41} – 40 MBq/L	Sum of- inert radioactive gases - 12 GBq/L	Any radiation. Consequences are not expected.
Gas-non-tightness of the loop in the reactor sump or FFDS stand	In release according to B-4: Ar^{41} – 30 MBq H^{3} – 1 GBq	Discharge via RIAR ventilation duct	Ar^{41} – 30 MBq Po^{210} – 40 Bq/days	Sum of inert radioactive gases 1295 GBq/days Po^{210}–7.4 MBq/days	Release of gas and alpha-active aerosols is less than 0.0001 against governing level and it does not make radiation Condition worse.
Gas leakage in the reactor hall	In the reactor hall: H^{3}– 2×10^{4} Bq/m^{3}	Activity nearby FFDS stand	Ar^{41} - 6×10^{4} Bq/m^{3} H – 2×10^{6} Bq/m^{3} Po^{210} – 3 Bq/m^{3}	Ar^{41} – 7×10^{5} Bq/m^{3} H – 4.4×10^{5} Bq/m^{3} Po^{210} – 13 Bq/m^{3}	Short-term increase of H^{3} concentration, excess of the main Irradiation limit for personnel is impossible.
Fuel pin leakage in the lead-filled loop	In FFDS stand: Xe^{135} – 3100 MBq, Cs^{138} – 780 MBq, Kr^{88} – 530 MBq	Dose rate nearby FFDS stand	26	29	Radiation conditions in the service area will not be made worse.
Fuel pin leakage in the lead-filled loop in case of its leakage in the reactor cavity	Addition of Xe^{135} – 0.9 GBq Xe^{133} – l GBq to the reactor gas cavity	Change of activity in the reactor gas cavity	Sum of inert radioactive gases – 4.7 GBq/L	Sum of inert radioactive gases - 12 GBq/L	Any radiation. consequences are not expected.
Fuel pin leakage in the lead-filled loop in case of its leakage in the reactor hall	Leakage in the reactor hall: Xe^{133} – 4900 MBq, I^{131} – 270 MBq, Cs^{137} – 30 MBq and other gases and aerosols	Volumetric activity of gases and aerosols in the reactor hall	Xe^{135} – 2.2 I^{131} – 0.13 Cs^{137} – 0.2 MBq/m^{3} and other gases and aerosols	Permissible volumetric activity: Xe^{135} – 0.16 I^{131} – 0.001 Cs^{137} – 0.0001 MBq/m^{3} and other gases and aerosols	Permissible volumetric activity nearby FFDS stand might be increased by 200 – 300 times by gas and by 2000 times by aerosols. Concentration of gases and aerosols in the reactor hall in case of quick gas leakage is 14–150 times higher than permissible volumetric activity. In case of slow gas leakage the increase does not take place (with the exception of I^{131} and Cs^{137} by 20 – 50%). Main Irradiation limits will not be increased.

involving the imposition of independent events (fuel pin leakage in the loop and simultaneous break of the gas line) the limited additional exposure of the personnel is possible but it does not exceed the main irradiation limit. As this takes place, I^{131} daily discharge via the RIAR *ventilation duct* might be exceeded but it will not cause the increased population exposure.

Studied emergency situations and their consequences demonstrate that the reactor *safety does not decrease over the time of lead-filled loop operation*. The requirements of safety regulatory documents are maintained.

LEAD COOLANT AS A NATURAL SAFETY COMPONENT

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Abstract

This paper validates the selection of lead coolant for fast reactors which allows ensuring reactor facility safety thanks to its inherent qualities and properties. It also addresses how natural safety principles are realized in lead-cooled BREST reactors.

1. INTRODUCTION

Creation of large-scale nuclear power is possible only with a radical improvement of safety of nuclear power plants (NPPs) and associated industries, preservation of economic competitiveness of nuclear power and with a considerable increase in the efficiency of using uranium while transferring to fuel breeding. Only the preclusion of severe accidents may conclusively justify the right of future nuclear power for existence and its public acceptability.

Modern thermal water-cooled reactors as well as fast sodium–cooled reactors do not fully preclude accidents involving fuel damage and radioactive releases during reactor runaway, loss of cooling, boil-up and ignition of the coolant and the moderator. Their further development seeks to reduce the probability of such accidents through creation of additional engineered barriers, meeting of highest requirements for the quality of equipment, construction, control systems and for operating personnel. And though all these measures make NPPs more complicated and costly, they, however, do not eliminate the hazard of accidents.

The analysis of major accidents has lead to a new philosophy which is based on the principle of natural, or “intrinsic” safety that is achieved not by build–up of engineered systems and making high requirements for equipment and personnel, but also through using fundamental physical and chemical properties and regularities inherent in nuclear fuel, coolant and other reactor components. In this case, safety is achieved not by a reduction of the probability but by preclusion of dangerous accidents which allows relying on the efficiency of nuclear power, on simplification of constructions and structures.

Hence, the correct and validated selection of the coolant as one of the basic components of a nuclear system, will allow solving the tasks and fulfill the requirements that the new generation reactors are facing.

The coolants used in nuclear reactors should:

– Ensure an intensive and stable heat exchange at as low power consumption rate for pumping as possible;
– Have a sufficient heat resistance;
– Have a higher boiling temperature and a lower melting temperature (for liquid phase operation in a wide temperature and pressure range);
– Have a low chemical activity for reducing the danger during handling and improving the corrosion resistance of structural materials;
– Be accessible and convenient during storage and transportation;

– Be stable during in–pile radiation exposure;
– Have a small neutron capture and scattering cross sections (to ensure as low losses of neutrons during nuclear reactions as possible);
– Be low-activated during exposure to radiation (to reduce the activity of the plant's primary circuit).

It is difficult to fully meet all these requirements, so basic requirements, are imposed in each separate case. It was so when sodium was chosen as the coolant for fast reactors. Besides its cheapness and chemical compatibility with fuel and steels, the advantage of sodium as compared to other liquid metals that previously determined the choice of it, is its capability of removing heat from high rated fuel. It results in a reduction in specific fuel load of the reactor and Pu doubling time which were accepted then as the strategic criteria in development of fast reactors.

However, because of its chemical activity relative to water and air, sodium coolant does not have qualities fully ensuring natural safety of the reactor. For all their improbability, there cannot be excluded situations leading to an intense sodium–air contact which leads to the self–maintaining burning of sodium with release of airborne radioactivity of millions Ci, and then to the loss of cooling with fuel damage and much more releases. A limited temperature margin to sodium boil–up within the limits of its two average heating values in the core (~200^{o}C) do not preclude the danger of boil-up, termination of fuel cooling and realization of void reactivity effect. An increase in fuel radioactivity during return to the reactor and burning of all actinides result in more risk connected with the emergency fuel damage and radioactive releases. Since such accidents are not precluded, also of doubt is the expediency of implementing this technology, but radiation-equivalent burial of radioactive waste is impossible without it.

Other liquid–metal coolants such as Bi, Sn, Ga, Pb, Hg, Pb–Bi, have their own pronounced drawbacks in terms of cost, radioactivity (Bi), corrosion activity (Sn, Ga), toxicity (Hg, Ga), activation during neutron radiation (Ga, Sn). So two coolants — lead (Pb) and lead–bismuth (Pb–Bi) alloy — may be chosen for further consideration.

The Pb–Bi alloy was considered as a coolant as long ago as in the 1950s but the preference then was given to sodium which was discussed above. Russia has a unique experience of using Pb–Bi as a coolant in ship reactor facilities. This experience may be also used in fast reactors. The conducted research has shown that at a moderate power rating of the core (as no short doubling times are needed that is only favorable for reactor safety), heavy liquid–metal coolants such as lead and lead–bismuth are most suitable coolants for cooling fast reactors in terms of meeting safety criteria.

Lead and the eutectic alloy of lead–bismuth (55.5% Bi) have much in common as coolants in terms of their nucleonic properties (similar cross sections of elastic and inelastic neutron scattering and small cross sections of neutron capture, reactivity void effect values, etc.). These coolants have slightly different maximum permissible rates (determined by erosion processes and hydraulic resistance), temperatures in the core (determined by corrosion resistance and strength of core components). They are slightly different in thermal properties: density (4%), thermal conductivity (5%), dynamic viscosity (33%), Prandtl number (30%), boiling temperature (175^{o}C — Pb, 1670^{o}C — Pb–Bi). At the same time, there are considerable differences in allowable lower temperatures of the coolant (~200^{o}C) due to lower lead–bismuth freezing temperature of 123.6^{o}C.

High lead temperature is not an obstacle for its use since low neutron moderation and absorption allow raising its volume fraction in the reactor due to an increase in the relative fuel element spacing, reduce the pumping rate and power consumption and reduce heat–up, as the maximum temperature remains within the limits allowing using ordinary class steels and there is a margin to the lower (freezing) temperature.

Problems connected with high melting temperature may be solved by selection of temperature modes and the reactor cooling pattern without an excess of the temperature permitted for steels and the closing of the coolant circulation paths during accidents.

Lead and lead–bismuth also differ in the coolant radioactivity level thanks to highly radiotoxic and volatile α–emitters - polonium isotopes, ^{210}Po first of all, generated under neutron exposure. The ^{210}Po generation is possible via two trains:

on ^{209}Bi nuclei:

$$^{209}Bi(n,\gamma)\ ^{210}Bi\frac{\beta^-}{5.0d}\ ^{210}Po\ \frac{\alpha}{138.3d}\ ^{206}Pb_{st}\ ,$$

on nuclei of ^{208}Pb whose contribution to the polonium activity is required to be taken into account at the Bi content lower than 10^{-2} %:

$$^{208}Pb(n,\gamma)\ ^{209}Pb\ \frac{\beta^-}{0.14d}\ ^{209}Bi(n,\gamma)\ \frac{\beta^-}{5.0d}\ ^{210}Po\ \frac{\alpha}{138.3d}\ ^{206}Pb_{st}$$

$$^{208}Pb(n,\gamma)\ ^{209}Pb(n,\gamma)\ ^{210}Pb\ \frac{\beta^-}{22.3y}\ ^{210}Bi\frac{\beta^-}{5.0d}\ ^{210}Po\frac{\alpha}{138.3d}\ ^{206}Pb_{st}\ .$$

So the polonium activity magnitude is determined by reaction on ^{209}Bi nuclei and the equilibrium activity is ~10 Ci/kg, as in the lead coolant (5×10^{-4}% Bi) it is determined by reaction both on ^{209}Bi and to ^{208}Pb and may reach of 5×10^{-4} Ci/kg at the end of the reactor service life. Such lead–bismuth coolant activity creates difficulties even at the normal operating mode. At the cover gas leakage rate of 0.01% of its volume a day, the release of ^{210}Po to the central hall may reach (if the gas circuit is not cleaned of polonium) 200 fold maximum permissible concentration (MPC) (1% MPC for lead). To ensure that MPC (9.3×10^{14} Ci/L) is not exceeded for personnel in the central hall, it is necessary to comply with very high requirements for the cover gas circuit leak-tightness which will require additional capital investments.

During accidents involving the reactor vessel (cover) and building destruction at extreme effects resulted in the reactor shutdown with a temporary rise in the coolant temperature in the reactor cavity to over 1000 K, fuel elements remain intact and the radioactivity leakage from the fuel remains at the design level, the daily release of polonium from the reactor with lead coolant will be ~3 Ci, the radiation hazard of which is approximately equivalent to that of the total release of all other radionuclides during the accident. Such release approximately corresponds to level 5 on the International nuclear events scale (accidents with a risk for the environment). The measures for cleaning lead of bismuth and other radionuclides would allow mitigating the consequences of the accident to level 4 or even to level 3.

The polonium daily release from the reactor with lead–bismuth coolant during such the accident will be ~6×10^4 Ci. Such release corresponds to level 7 on the INES (global accident, long-term impact on the environment, possibility of acute ray decease and effect on the health of the population living in large territories, Chernobyl accident level).

Speaking of large-scale nuclear power, it is necessary to evaluate the coolant metal reserves available in the world. The explored reserves of Bi (1972) in the world were ~160 thousand t and its worldwide production was ~3000 t at a cost of $7–14 per kg. The explored and prospective lead reserves are ~100 million t at the annual production of ~3 million t and the cost of high pure lead of ~$1 per kg. With the 1 GW reactor's coolant demand of 15 thousand t and the use of all of the explored Bi for nuclear power needs, some 20 plants may be provided with lead–bismuth coolant. Therefore, speaking of the prospects of Pb–Bi coolant, it should be understood that it may be used for a limited number of power plants, but it is not enough to build a large-scale nuclear power of the future.

During the development of the conceptual design of the fast lead-cooled reactor BREST, the lead coolant properties have allowed obtaining new properties of the reactor based on the natural safety principles.

2. HIGH DENSITY OF LEAD

The conducted calculation research has shown that no critical mass is formed during core destruction due to similar fuel and lead densities and convective flows that scatter fuel during drawing together of fuel components and formation of mass close to critical.

It provides for the possibility of circulating the coolant without increasing reactor dimensions thanks to a difference in the free levels in the discharge and suction chambers that creates steam separation conditions during depressurization of the steam generator, and excludes its ingress to the core in hazardous amounts.

It excludes the fall of fuel assemblies (FAs) to the core and damage during refueling.

3. LOW MODERATION AND ABSORPTION OF NEUTRONS

It allows to use wide lattices which results in:

- Low power consumption for pumping and high level of natural circulation;
- The possibility of using sheathless FAs and exclusion of their overheating during local flow blockage.

It allowed using the equalization of the lead heat-up and fuel element temperatures in different radial areas of fuel elements with different diameters, but with the same Pu content. Such the method has allowed producing the required distributions both of power density and coolant flow rates as well as ensuring the adequate equalization of lead temperatures at the outlet from different FAs and the peak temperatures of fuel claddings which are the main factor determining fuel element efficiency.

4. GOOD REFLECTING PROPERTIES

The substitution of the uranium reflector with a lead reflector has allowed reducing neutron leakage and improving neutron fields equalization, and provided for a high negative reactivity void effect during lead drain from the reactor.

There is a possibility of regulating the reactor through lead substitution with gas.

The absence of screens and relatively small dimensions of the core have allowed accommodating core protection system (CPS) controls beyond the core boundaries which has enabled FA reloading without disengagement of CPS rods with their drives.

5. HIGH MELTING AND BOILING TEMPERATURE

The absence of high pressure in the lead circuit and a relatively high freezing temperature preclude accidents involving the loss of coolant and core cooling, melting of fuel elements, leakage of radioactive lead to the reactor rooms due to lead freezing and crack remedy.

There precluded accidents connected with the local void effect during boiling (T_{boil} ~2300^{o}C at the core pressure of ~10 atm).

6. LEAD DOES NOT BURN AND POORLY INTERACTS WITH WATER AND AIR

It allows using a two-circuit reactor cooling pattern, and air passive cooldown systems.

7. LEAD IS POORLY ACTIVATED

Maintenance and repair of the equipment are facilitated. Lead may be repeatedly used in other plants if NPP is decommissioned which is easier thanks to the simplicity of its handling in freezing.

Hence, the conducted analysis and development of conceptual designs of the BREST reactor have shown that lead coolant is a liquid metal coolant most fully meeting the requirements of natural safety and having sufficient margins for the development of large-scale nuclear power.

REFERENCES

[1] ANDREEV, P.A., GREMILOV, D.I., FEDOROVICH, Ye.D., Heat Exchangers of Nuclear Power Plants, Leningrad, Sudostroenie (1969).

[2] ADAMOV, E., ORLOV, V.V., et al., The Next Generation of Fast Reactors, Nuclear Engineering and Design, 173 (1997) 143–150.

[3] ADAMOV, E., ORLOV, V.V., et al., Conceptual Design of BREST-300 Lead-Cooled Reactor, Proceedings, International Topical Meeting on Advanced Reactors Safety, ARS'94, Pittsburgh, USA (509–516).

ISSUES OF LEAD COOLANT TECHNOLOGY

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Abstract

Starting from 1989 with the BREST-OD-300 reactor design development, some experimental studies related to lead coolant have been performed to provide feasibility of using liquid lead as a coolant in the closed circuit. Further comprehensive analytical and experimental studies are required to develop lead coolant technology for the BREST-OD-300 reactor design. General work program for justification of lead coolant technology are planned aiming at not only getting the information on the components required for the technology realization and their design features but also revealing the possibility of use of data obtained in the course of lead–bismuth circuit operation for the lead circuits. The main results performed so far for justification of BREST-OD-300 reactor coolant technology are presented in this paper. The results confirmed the possibility of using the experience gained on lead–bismuth coolant justification on the stage of the BREST-OD-300 reactor design development. In addition, the results provided directions for further justification of lead coolant technology based on experience gained on lead–bismuth coolant.

1. INTRODUCTION

Requirement of improvement of nuclear reactor safety and reliability impels the designers to look for the advanced coolants as compared to traditional cooling media (water, sodium, etc.). Among such coolants liquid lead is considered. Below are described lead coolant properties making it attractive for use in the reactor heat removal system.

Lead has appropriate neutronic characteristics. Its capture cross section is low, so it can be used as a coolant for either intermediate or fast neutron reactors. Activation of lead in reactor is low, and it is explosion–proof media. It has low vapor pressure and high boiling point (~1750^{o}C), thus assuring low pressure in the primary circuit of the reactor. Lead is comparatively cheap, so it can be produced on a large scale.

Besides, physical and chemical properties of lead are close to those of lead–bismuth eutectic alloy, which for many years has been successfully used as a coolant of propulsion reactors. Large scope of information has been gained on physical, chemical, thermophysical and other properties of the alloy. Technology of this coolant has been developed as well as devices for its implementation. Methodology and experimental base is available which can be used for justification of lead as a coolant of power reactors.

2. DEVELOPMENT OF LEAD COOLANT TECHNOLOGY

On the basis of the considerations mentioned above, development of BREST-OD-300 reactor design was initiated under ENTEK management with the broad circle of specialists from different research centers being involved.

On the other hand, use of lead as a coolant for the BREST-OD-300 reactor requires development of lead coolant technology conformably to specific design and parameters of nuclear reactor. This is because of the following reasons:

– Physical and chemical properties of lead are quite different from those of coolants used on the wide scale in the nuclear power (water and alkali metals), for which technologies were mainly developed;

– There is no experience gained anywhere in the world on the use of lead and its alloys in the facilities of size comparable to that of BREST-OD-300 reactor or having similar design life; lead–bismuth coolant was used in comparatively small size facilities with heat removal system design quite different from that of BREST-OD-300 reactor.

It should be noted that starting from 1989, in the course of work performed under the assignment by ENTEK within the framework of BREST reactor design development, some processes related to lead coolant (such as hydrogen regeneration, oxygen enrichment, etc.) were used on test facilities of the Institute for Physics and Power Engineering (IPPE) and some other organizations, which will probably be the integral parts of lead coolant technology of the BREST-OD-300 reactor. However, the main efforts in the process of this work were aimed at proving feasibility of liquid lead use as a coolant in the closed circuit, development of filling, start-up and operation modes of test facilities and studies of mass transfer processes (mainly corrosion) in the circuit. Temperature conditions and design characteristics of test facility components were far from those of the BREST-OD-300 reactor.

It follows from the above considerations that comprehensive analytical and experimental studies are required in order to develop technology of BREST-OD-300 reactor lead coolant. These studies should be carried out with maximum effectiveness in order to get information on the components required for this technology realization and their design features as soon as possible.

It has been already determined on the basis of the work performed that lead coolant technology should for sure include the following procedures:

– Hydrogen reduction of lead from its oxides;
– Control of the coolant quality using Ar–H2O–H2 ternary mixtures and mass exchangers with solid phase oxide filling agent;
– Coolant purification by filtering from solid phase mixtures which cannot be reduced.

Results of studies (both already completed and planned) are required primarily for experimental confirmation of feasibility of use of the above mentioned systems and equipment as well as their effective operation.

Another important goal of studies is the analysis and comparison of mass transfer processes in the circuits with lead and lead–bismuth coolants considered for two different conditions:

– Coolant technology systems are in operation;
– No coolant technology systems are used.

Results of these studies are needed for revealing the possibility of use of data obtained in the course of lead–bismuth circuit operation, for the lead circuits.

Below presented are the main stages of the general work program of justification of lead coolant technology, planned for the nearest three years:

– Determination of optimum thermodynamic conditions of liquid metal circuit in order to minimize intensity of mass transfer processes and assurance of corrosion resistance of structural materials;

– Studies on the process of slag formation in different operating modes of the primary circuit of the BRIEST-OD-300 reactor, such as circuit filling, repair and maintenance, core refueling, leaks in the circuit and in the steam generator;
– Tests and improvement of technology of the circuit surface cleaning by providing coolant–gas two-component flow;
– Development of methods and devices for coolant filtering;
– Development of methods and devices for gas purification and BREST-OD-300 reactor gas communications protection against material corrosion products and lead vapors;
– Development of methods and means required for the primary circuit surface protection against corrosion;
– Development of methods and devices for control of condition of the coolant and internal surfaces of the primary circuit;
– Experimental and methodological support (as far as coolant technology is concerned) of tests of the fuel elements for the BREST-OD-300 reactor, using lead channel of the BOR–60 reactor;
– Studies on the environmental issues related to heavy metal coolant in case of design basis and beyond design accidents of the BREST-OD-300 reactor;
– Development and justification of input data for designing coolant technology system of the BREST-OD-300 reactor and its operating rules.

3. THE MAIN RESULTS OF WORK PERFORMED IN 1999 ON JUSTIFICATION OF BREST-OD-300 REACTOR LEAD COOLANT TECHNOLOGY

– Followings are the works that were performed in 1999 to justify the lead coolant technology for the BREST-OD-300 reactor:
– Justification of the option of heavy metal coolant technology system and the main requirements to this system made on the basis of experience gained on operation of lead and lead–bismuth circuits;
– Experimental confirmation of serviceability of proposed technology system of heavy metal coolant under the temperature conditions of the BREST-OD-300 reactor;
– Revealing the increase of oxide dissolving contribution to the general hydrogen regeneration process of the coolant (as compared to that of lead–bismuth circuits);
– Revealing the possibility of decreasing hydrogen content down to the lower limit of explosion safety and fire safety (CH_2 <4–5 vol.%) in the process of regeneration;
– Development of immersed mass exchanger design with solid phase filling agent for dissolved oxygen supply to the melt in the course of passivation protection of structural materials;
– Development of technology of production of solid phase oxide filling agent of mass exchanger;
– Experimental confirmation of mass exchanger serviceability in the course of passivation protection of materials under conditions of intensive provocation of corrosion;
– Development of concept of gas route protection against structural material corrosion products and lead vapors;
– Justification of filtering materials (MFVE-3 glass paper, etc.) for purification of cover gas of the BREST-OD-300 reactor at the temperature up to 150^{o}C;
– Confirmation of serviceability and thermal resistance of MKTT-2.2A filtering material at T = 420^{o}C (for 1000 hours as a reference time period);
– Confirmation of possibility of non-isoconcentration distribution of dissolved oxygen under BREST-OD-300 reactor temperature conditions.

The main results of the above works are:

- Closeness of qualitative and quantitative characteristics of the main physical and chemical processes in lead and lead–bismuth coolant circuits was experimentally confirmed. This confirms the possibility of using experience gained on lead–bismuth coolant justification on the stage of BREST-OD-300 reactor design development;
- Directions of further justification of lead coolant technology were finally determined on the basis of experience gained on lead–bismuth coolant bringing to use.

On the other hand, further realization of the program of justification of lead coolant use in the BREST-OD-300 reactor would require considerable material expenses and strenuous experimental and analytical studies.

EXPERIMENTAL AND COMPUTATIONAL STUDY ON CORE THERMOHYDRAULICS OF BREST-TYPE REACTORS (LEAD COOLING)

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Abstract

Experimental studies of heat-transfer coefficients and temperature fields of fuel rods for the BREST-type lead-cooled reactors have been performed, using thermo-hydraulic models which consist of fuel subassemblies with square rod arrangement and use an eutectic alloy sodium-kalium as a coolant. In the results obtained from the experimental models with three different pitches, thermohydraulic nonuniformity appears along perimeters of rods located in the energy release area due to different coolant heating around the rods, and it leads to the reduction of heat transfer coefficients. In addition, a computational code has been developed to evaluate thermohydraulic performances of the BREST-300 reactor core. Code verification using the experimental data showed a satisfactory agreement. All these experimental and computational studies will contribute to thermohydraulic substantiation of BREST-type reactor cores.

1. INTRODUCTION

The results of thermohydraulic studies for the experimental demonstration lead–cooled fast reactor BREST–OD–300 are considered. The conceptual studies of lead–cooled fast reactors have shown that this direction is perspective from a point of view of design of reactors which have inherent advanced safety features [1, 2]. Taking into account a low level of heat transfer coefficients of the lead coolant in contrast to sodium (BN–type reactors) and also practically unexplored square rod arrangement used in these reactors, it is necessary to investigate how heat transfer coefficients depend on the Peclet number (Pe), the rod pitch (s/d), initial thermal sections which are variable along energy release length and core radius, spacer grids and other factors which are characteristic for the BREST–type reactor.

2. ORGANIZATION OF RESEARCH AND SIMULATION

Experimental studies of heat transfer coefficients and temperature fields of fuel rods for the BREST–type reactors have been carried out using three thermohydraulic models which have identical structures (Figs 1.a and 1.b) and are only distinguished by pitches: s/d = 1.46, 1.28, and 1.25. The models consist of a subassembly of 25 model fuel rods with square arrangement and located into the rectangular cover. Along the central model fuel rod (Fig. 2) which was rotary (gasket obturating), the surface temperature measurements were conducted along perimeter and length of the model fuel rod by microthermocouples fixed in the surface or moved along energy release length. The coolant temperature was measured in all cells at the model bundle outlet and also at the model inlet and outlet in the headers.

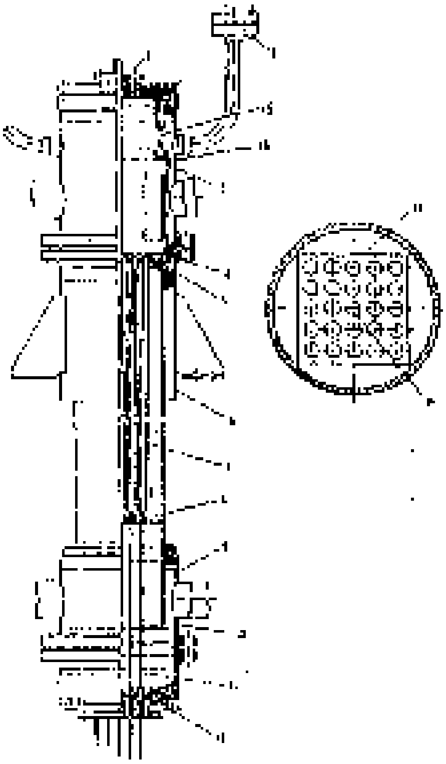


FIG. 1.a. Test facility.

1 – gasket obturating, 2 – thermocouples outlet, 3,10 – upper and lower header, 4 – thermocouples lattice, 5, 8 – upper and lower centering lattices, 6 – model vessel, 7 – model rods, 9 – guiding vessel, 11 – power supplier, 12 – power supplier obturating, 13 – square cover, 14 – rotary (measuring) model rod, 15 – support bolt, 16 – vessel.

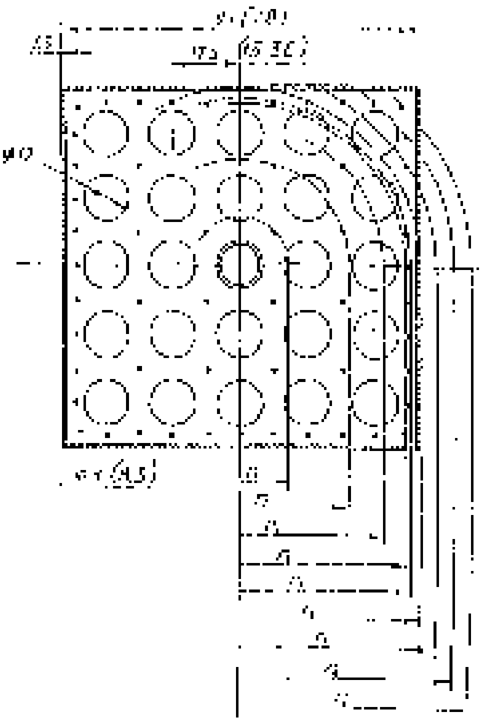


FIG. 1.b. Cross-section of model assembly and arrangement scheme of thermocouples in the coolant: r_1, - r_9, - radius where thermocouples equidistant from assembly center are located

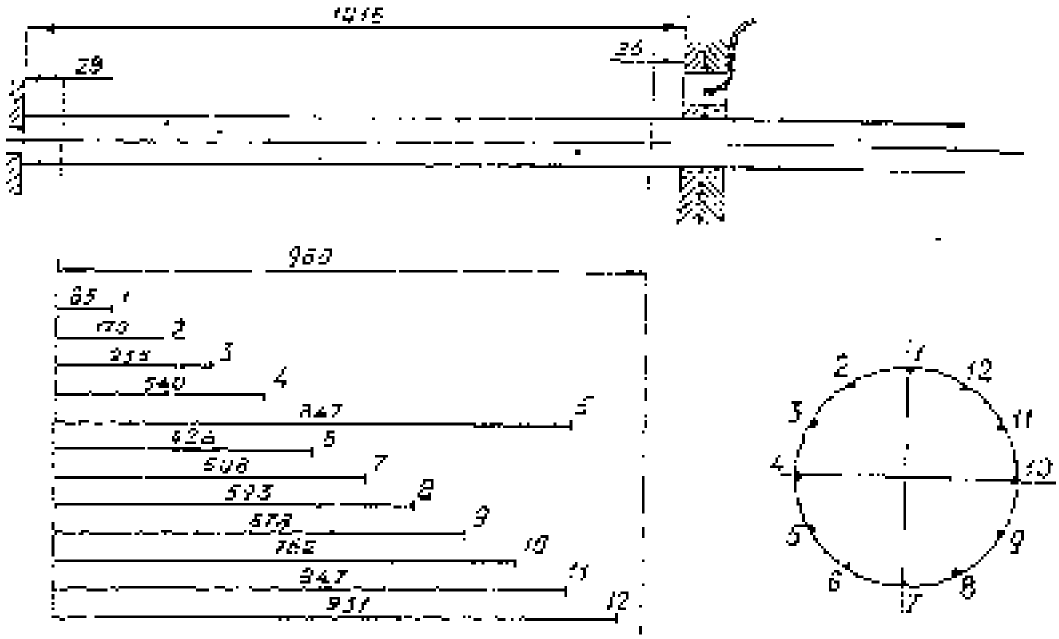


FIG. 2. Scheme of thermocouples arrangement m the surface of the model fuel rod and at the rod bundle outlet.

As a simulating coolant a eutectic alloy sodium–kalium (22% Na + 78% K) was used because it has the Prandtl number close to the numerical value of the lead Prandtl number. It ensured identity of heat exchange processes occurring at the contact "fuel rod–coolant" interface in the case when coolants considered were "clear" and when there were not thermal–chemical phenomena at the heat exchange surface.

Thermal simulation of fuel rods of the BREST-OD-300 reactor (fissionable fuel is uranium or plutonium mononitride, the cover is stainless steel, the interlayer is lead) was rather strict (accuracy – 5%) for the fourth harmonics of temperature field Fourier series expansion ($\kappa_0 = 4$) being the main harmonics for the regular square rod arrangement (Figs 3 and 4).

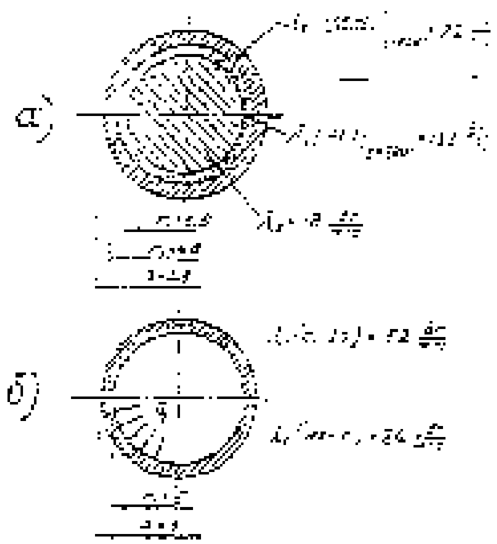


FIG. 3. For thermal simulation of the BREST-300 reactor rods (a) by model rods (b).

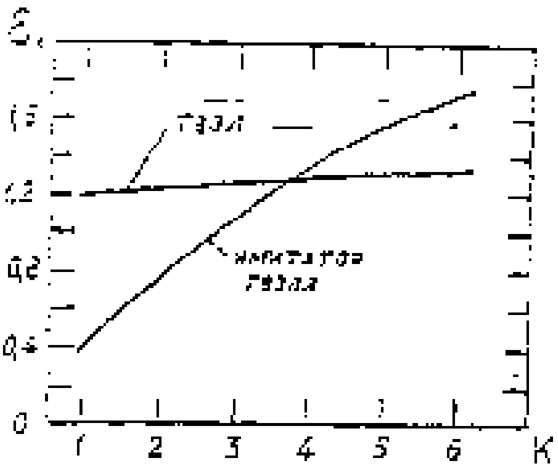


FIG. 4. Behavior of equivalent thermal conduction parameters of the BREST-300 reactor rods and their Model rods as a function of harmonics number (K) for temperature field Fourier series expansion.

The model fuel rods with lateral spacer grids (Fig. 5) with overlapping of passage cross-section for the coolant ε_p = 10 and 20% were used in addition to the smooth model fuel rods

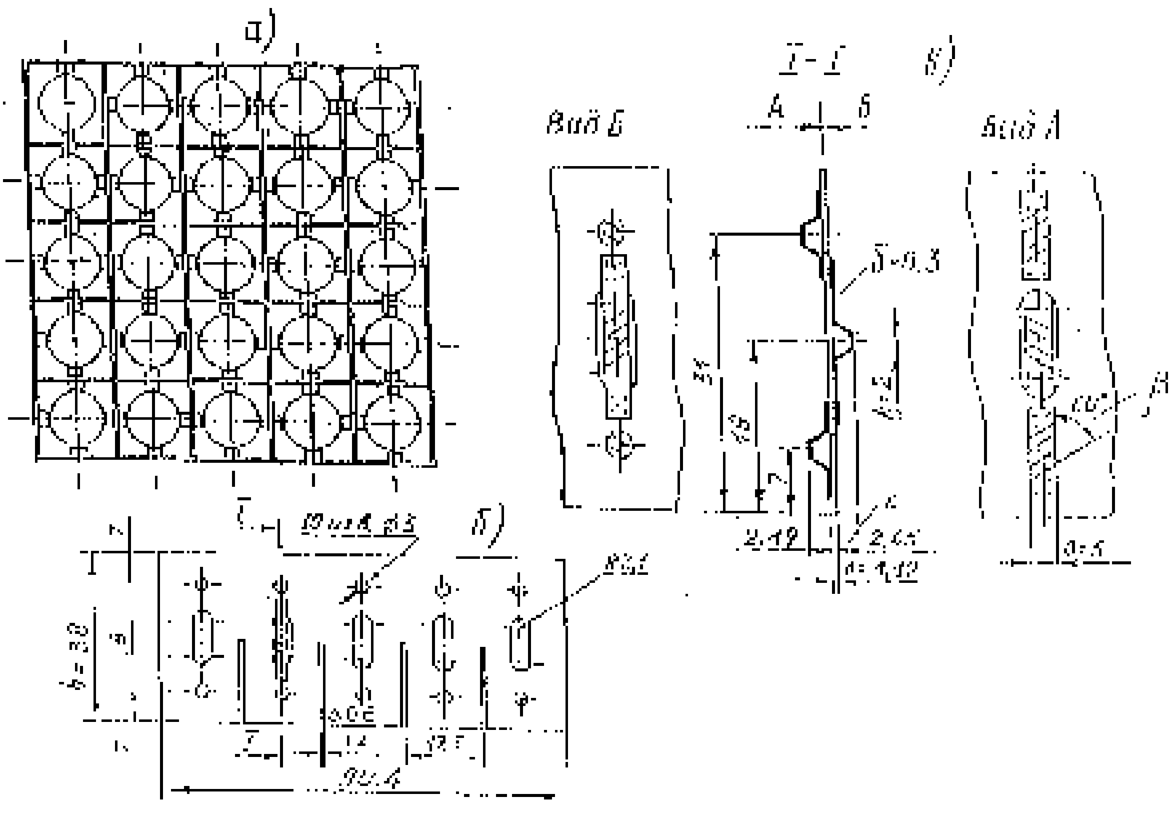


FIG. 5. Spacer grid assembled (a), spacer plate (b), and setting of spacers (c).

3. OBTAINED EXPERIMENTAL RESULTS

Length of initial thermal sections (Z_H) (Figs 6 and 7) in calibration of hydraulic diameter of a regular fuel rod cell (d_r) is described by the formula:

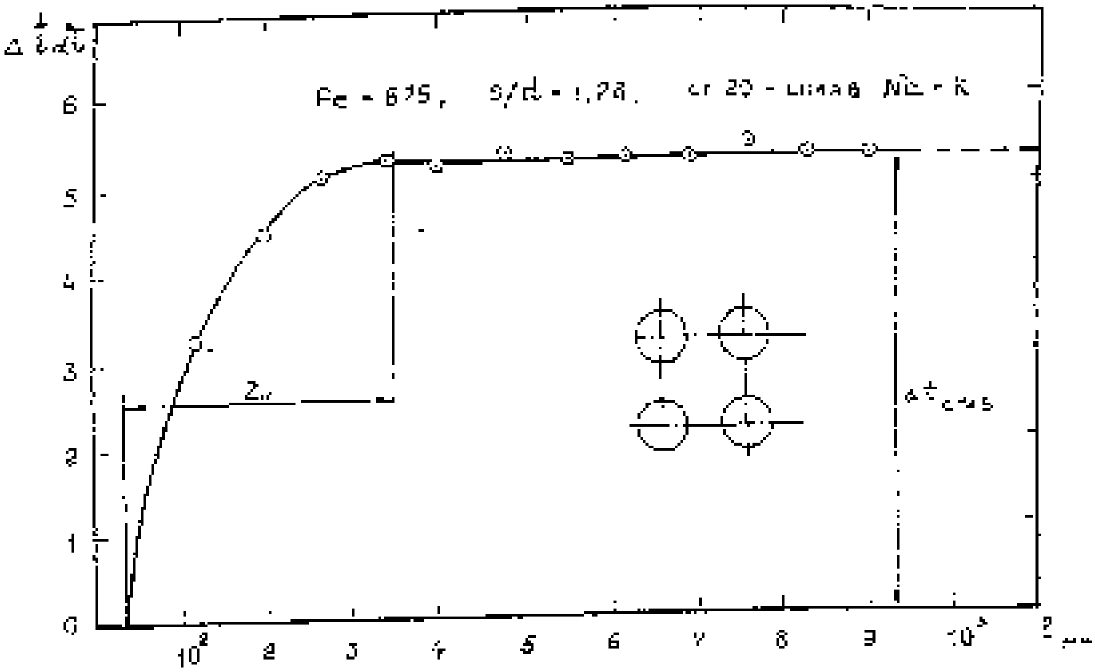


FIG. 6. Change of thermal pressures "wall–liquid" along the energy release.

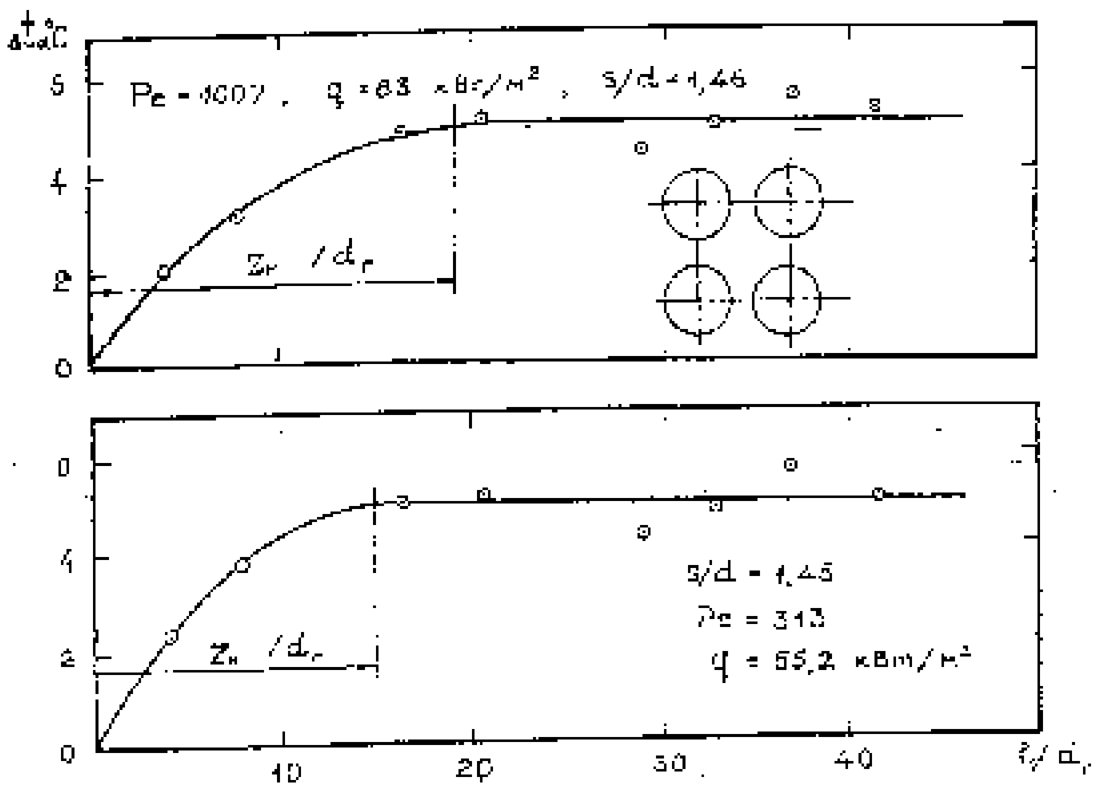


FIG. 7. Change of thermal pressures "wall-liquid" along the energy release.

$$Z_H/d_r = A - B/(255 + P_e)\,, \qquad (1)$$

where: $A = 42.3\,(s/d)^{-1.66}$, $B = 10680\,(s/d)^{-2.24}$,

$130 \le P_e = 1260$ for $s/d = 1.25, 1.28$ and $300 \le P_e \le 2000$ for $s/d = 1.46$.

Transition function (Fig. 8) for s/d = 1.25, 1.28, and 1.46 is described by uniform relation in a range 300 < Pe < 1000 with an error ±0.04:

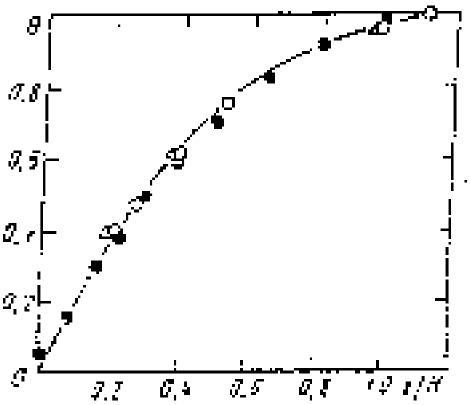


FIG. 8. Experimental transition function θ for various values of the Peclet number and pitch:
● - Pe = 887, s/d = 1.28;
○ - 313 and 1.46,
Δ - 1007 and 1.46.

$$\theta = \left[\mathrm{th}(2.3\,z/z_H)\right]^{1+z/z_H}\,. \qquad (2)$$

The relations for Z_H and θ(z) were used to evaluate the influence of variable energy release on the heat transfer coefficient with reference to BREST-300 parameters (Figs 9–11). The decrease of the heat transfer coefficient at the end of the energy release zone as compared with the stabilized value was 14–24% for normal reactor operation (Pe = 1600–2800).

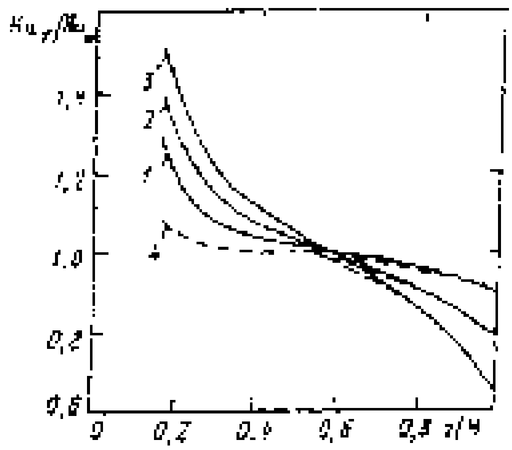


FIG. 9. Distribution of Nu_γ / Nu_8. along the core length for hydrodynamically stabilized flow for k_z = 1.1 (1), 1.2 (2), 1.3 (3), q = const (4), —–, ---- calculation using formula (2), and (3) for relaxation length $Z_H/5$, respectively.

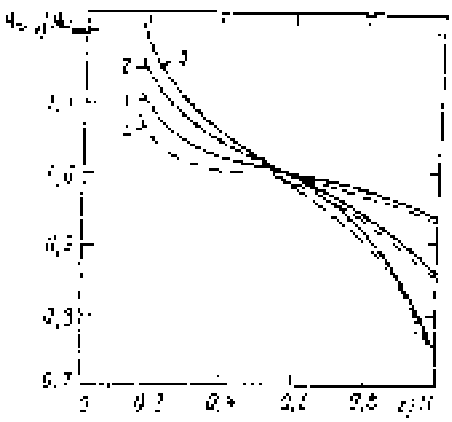


FIG. 10. The same things which are in FIG. 8 but for hydrodynamically instabilized flow.

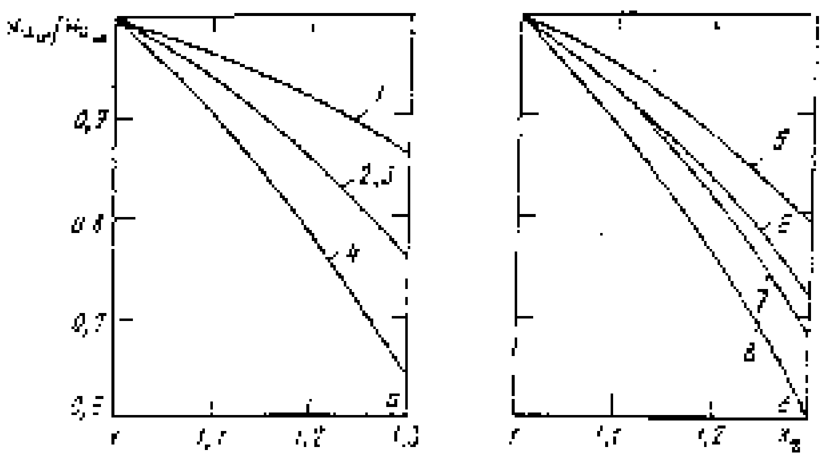


FIG. 11. Dependence Nu_γ / Nu_8 on k_z at assembly outlet for s/d = 1.3 (a) and 1.5 (b), Peclet number 160 (1), 1600 (2), 280 (5), 2800 (6) for hydrodynamically stabilized flow and 160 (3), 1600 (4), 280 (7), 2800 (8) hydrodynamically instabilized flow.

The Nusselt numbers stabilized along length for smooth rods are described by the following relation (Fig. 12):

$$Nu = 7.55\,(s/d) - 14\,(s/d)^{-5} + 0.007\ P_e^{\,0.64+0.246\,s/d}\ , \tag{3}$$

$$1.28 \le s/d \le 1.5\ ,\ 1 \le P_e \le 2300\ .$$

For rods with lateral spacer grids (Figs. 13 – 15):

$$Nu = 7.55(s/d) - 14\,(s/d)^{-5} + a\,P_e^{\,0.64+0.246\,(s/d)}\ , \tag{4}$$

where a = 0.01 and 0.009 for ε_p = 10 and 20% respectively,

$$1.25 \le s/d \le 1.5,\text{and } 1 \le P_e \le 2300.$$

The availability of spacer grids does not result in additional overheating of rod covers. Moreover the surface rod temperature is reduced in the area of the spacer grid (lower with increase ε_p) due to high equivalent thermal conduction of rods of length of model rod *(*ε = 1.3) for model assembly with s/d = 1.28 and Pe = 675.

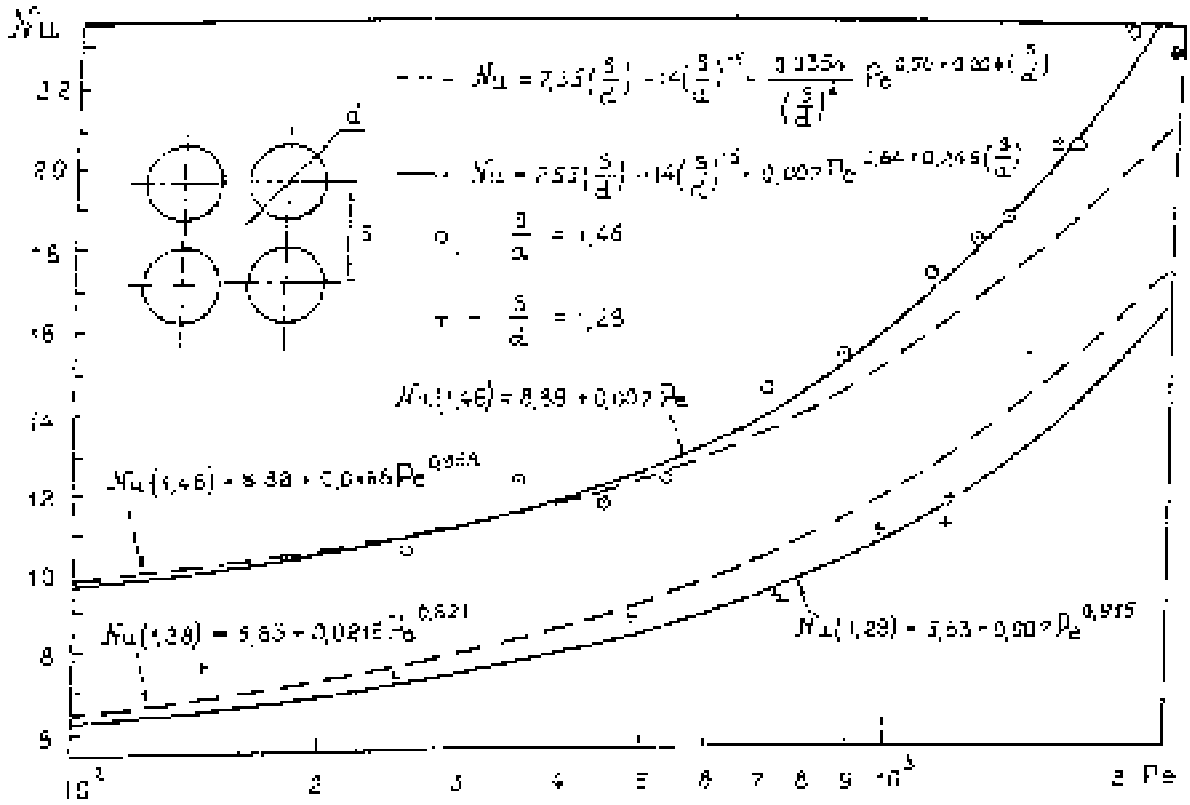


FIG. 12. Comparison of Nu numbers with available relations (○, + experiment, - calculation using proposed formulae, --- calculation using available relations).

The BREST-type reactor and increase of coolant velocity in a passage cross-section of a spacer grid (Figs 13 and 15).

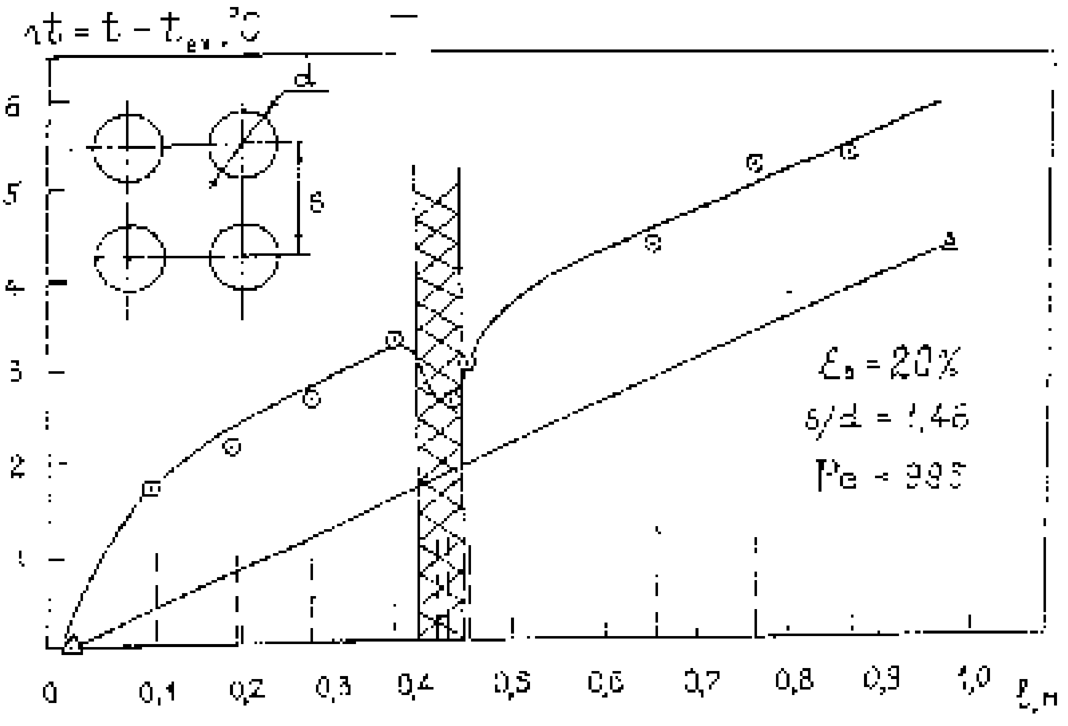


FIG. 13. Temperature field for model rod with spacer grid.

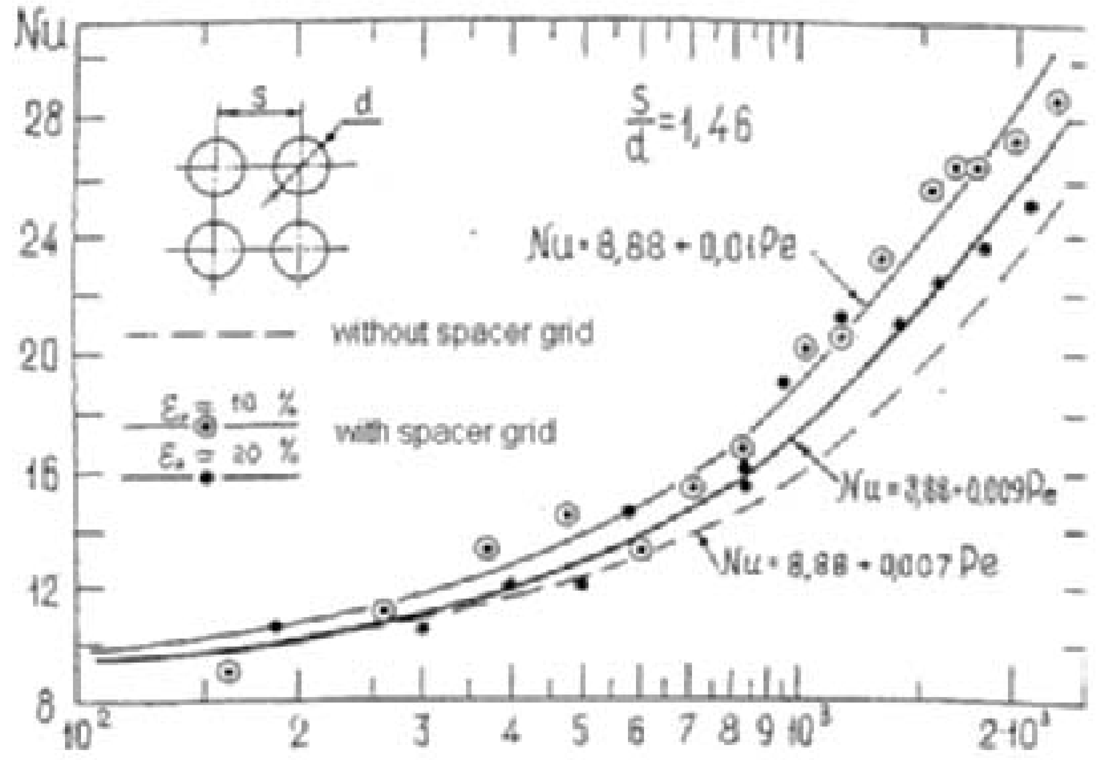


FIG. 14. Experimental data for Nu numbers in the smooth rods assemblies (s/d = 1.46) (---) and with spacer grids with overlapping of passage cross–section ε_p = 10% (○), and ε_p = 20% (●).

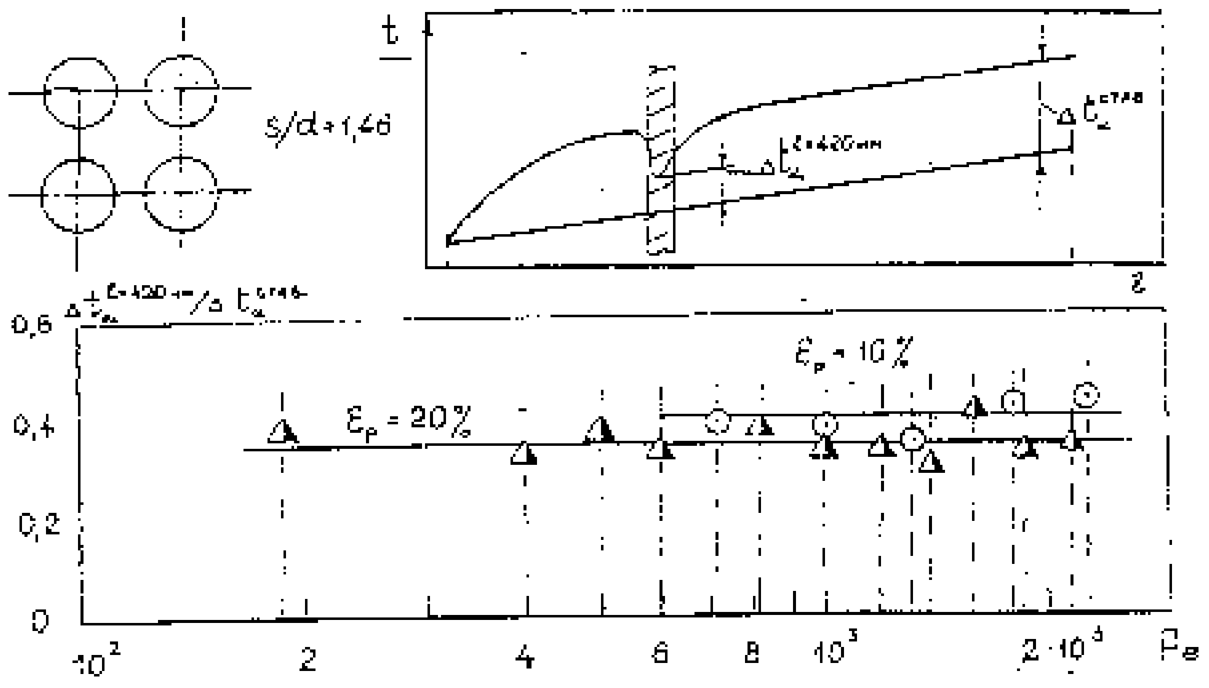


FIG. 15. Comparison of thermal pressures in the area of spacer grid ($\Delta t_\alpha^{\,l = 420\ mm}$) with stabilized values of thermal pressures (Δt_α^{stab}) for different Pe numbers in the assemblies with spacer grids ε_p = 20 (Δ) and 10% (○).

Thermohydraulic nonuniformity in the area of energy release jumps in rods with identical and different diameters are represented in Fig. 16. In this figure, the scheme is illustrated to evaluate the maximum temperature of rod cover for the BREST-OD-300 reactor. Along perimeters of rods located in the energy release zone an additional temperature nonuniformity.

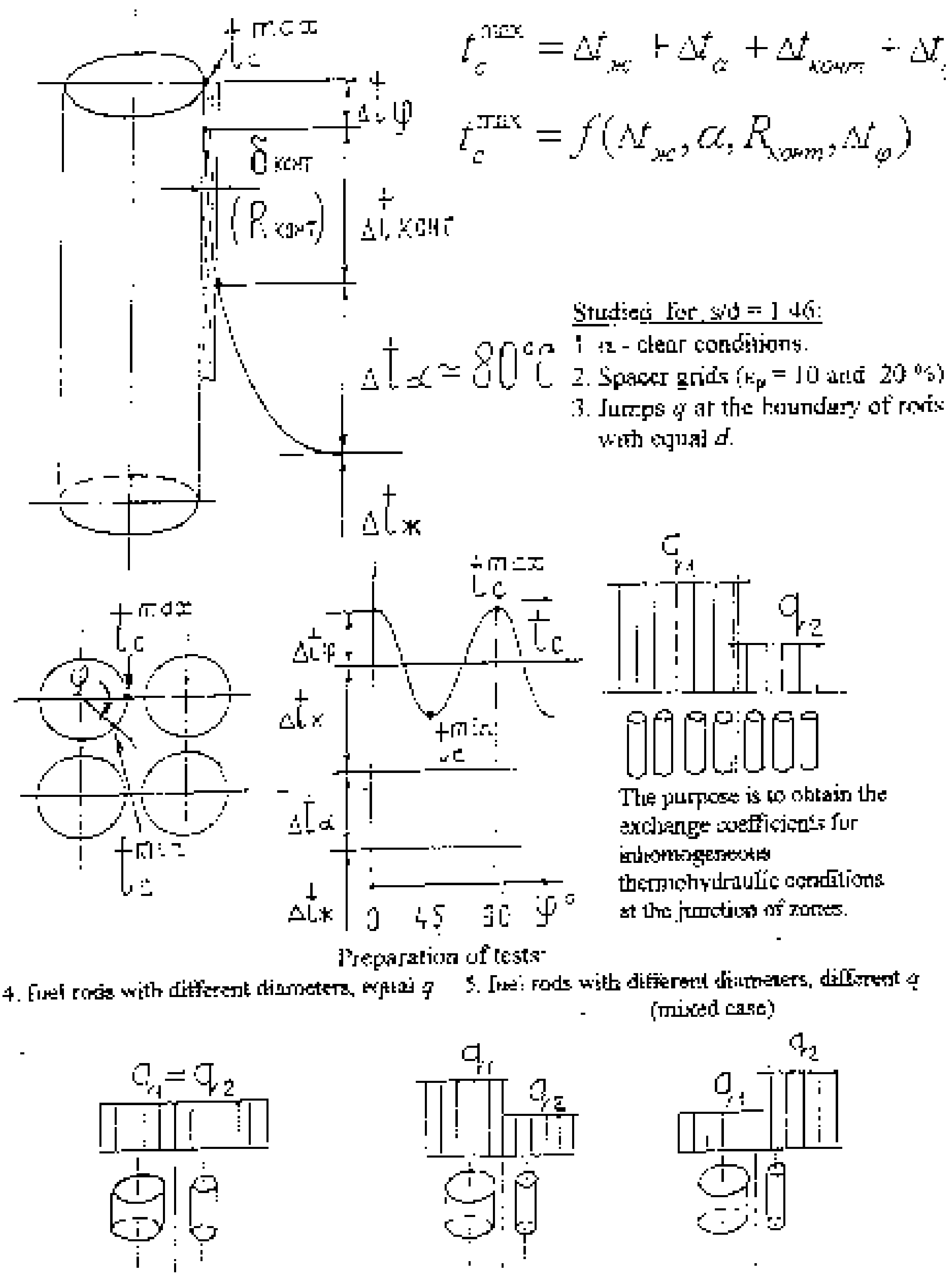
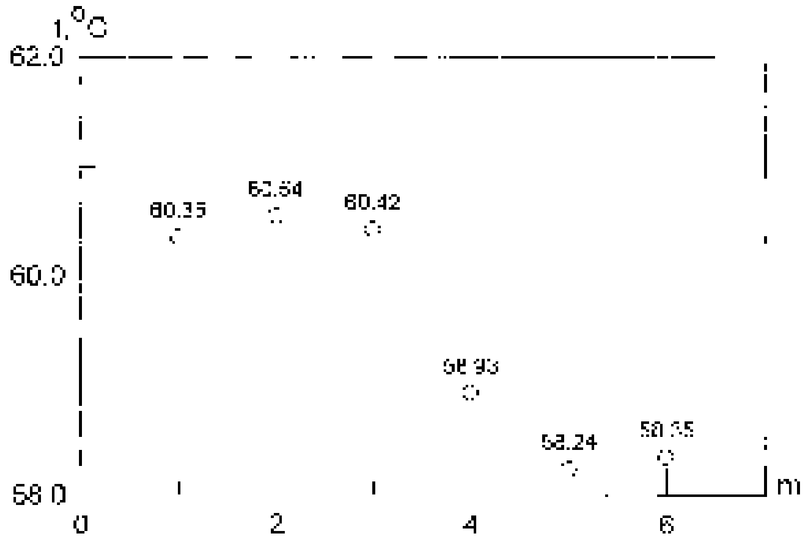


FIG. 16. Thermohydraulic factors for evaluation of temperature maximum of rod cover (t_c^{max}) for the BREST-type reactor.

appears due to different heating of the coolant in the cells surrounding the rods (Fig. 17). This temperature nonuniformity reduces heat transfer coefficients (Fig. 18). The recommendations are given to calculate heat transfer coefficients for various values of energy release jumps for identical and distinguishing diameters of adjacent rod zones.

E = 4.5 MW, N = 18.5 kW, n = 2, t_{in} = 55.48°C, t_{out} = 59.28°C

	60.02		58.24	58.60	
60.65	-----	60.16	59.15	57.99	57.75
60.20	60.31	60.86	58.67	58.19	57.65
60.45	61.40	60.64	59.13	58.30	58.16
60.11	59.90	60.02	58.78	58.50	59.86
	60.20	60.09	59.62	58.19	



E, mW	V, m^3/h	G, kg/h	$\overline{w}$, m/s	$\overline{Re} = \frac{\overline{w} d_r}{\nu}$	$\overline{Pe}$	Q_{el}, kcal/h	Q_t, kcal/h	$\varepsilon = \frac{Q_{el} - Q_t}{Q_{el}}$, %
4.5	20.475	17557	1.064	31160	782	15910	15345	3.5

FIG. 17. Coolant temperature in the areas of the modelled rods with distinguishing energy release.

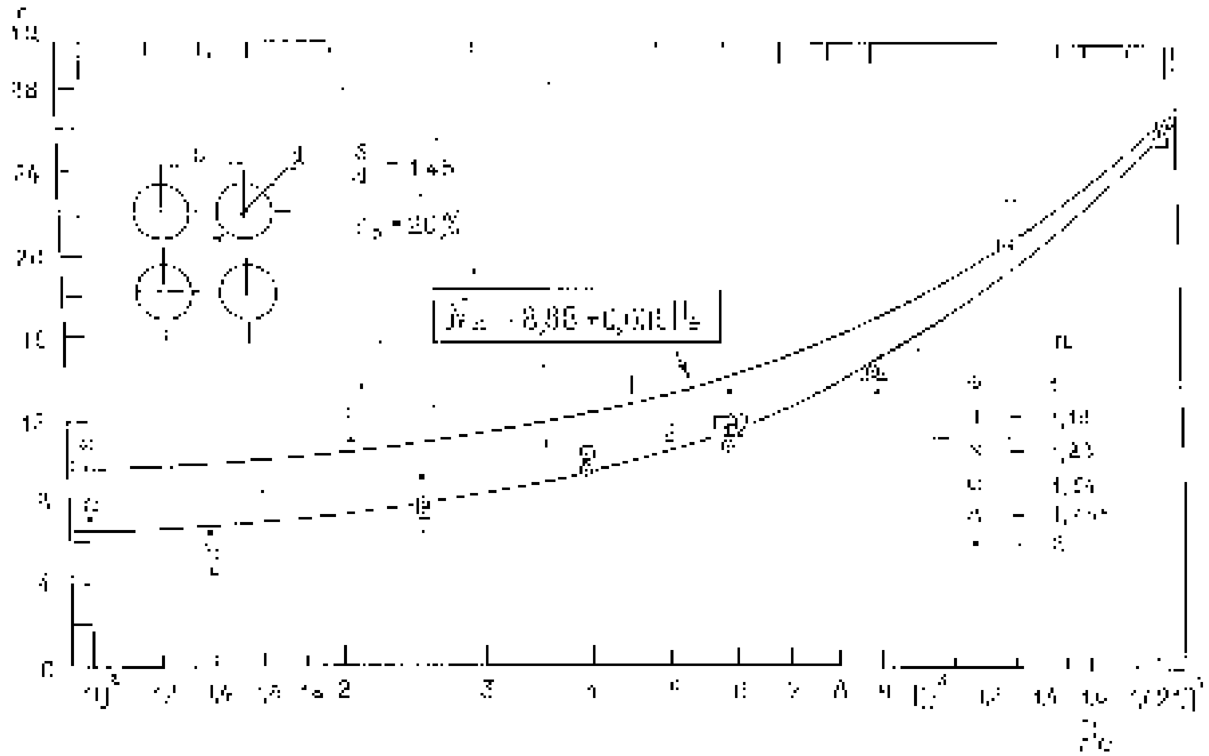


FIG. 18. The Nusselt numbers for uniform energy release in the model assembly cross-section (1) and for energy release jump (2).

4. COMPUTATIONAL ANALYSIS

The analysis of the codes which are available in the publications for calculation of thermohydraulic performances of liquid metal cooled reactor core was conducted; it was accepted that code development for the BREST reactor should be based on subchannel technique. The system of closing relations has developed in connection with fuel assemblies with square arrangement of rods. The system includes the relations for:

– Factors of hydraulic resistance;
– Factors of inter-channel exchange;
– Factors of heat exchange;
– Nonuniformities of temperature along perimeters of rods.

On the basis of subchannel technique the computational code has developed. This code takes into account features of the BREST–300 reactor (a square rod arrangement, high equivalent thermal conduction of rods, large porosity of the rod lattice, large temperature pressures "wall–liquid" and essential influence of variable energy release on numerical values of these pressures, variable rod diameters in the core cross-section and energy release jumps at the junction of rod zones with different diameters, etc.), and calculates the following characteristics with input of basic data (Table I):

– Geometric parameters of channels with selection of characteristic types of cells;
– Velocity distribution in the cross-section;
– Distribution of rods energy release and energy release incoming into channels;
– Distribution of coolant and rod covers temperature, etc.

TABLE I. CHARACTERISTICS OF THE CALCULATED REGIME

Denomination of magnitude	Dimensionality	Value
Pitch, s/d	-	1.46
Specific heat flux, q	W/m^2	41447
Average in the assembly, $\overline{w}$	m/s	1.67
Coolant velocity at the inlet assembly, T_{in}	^oC	38.96
Peclet number calculated with regard to velocity in the central cells, Pe	-	1390
Prandtl number, Pr	-	0.0282
Reynolds number calculated with regard to velocity in the central cells, Re	-	49318
Length of an initial thermal section, l_H	Mm	402

Code verification was carried out using the above–described experimental material. The comparison of experimental and computational results has shown the satisfactory agreement (Fig. 19).

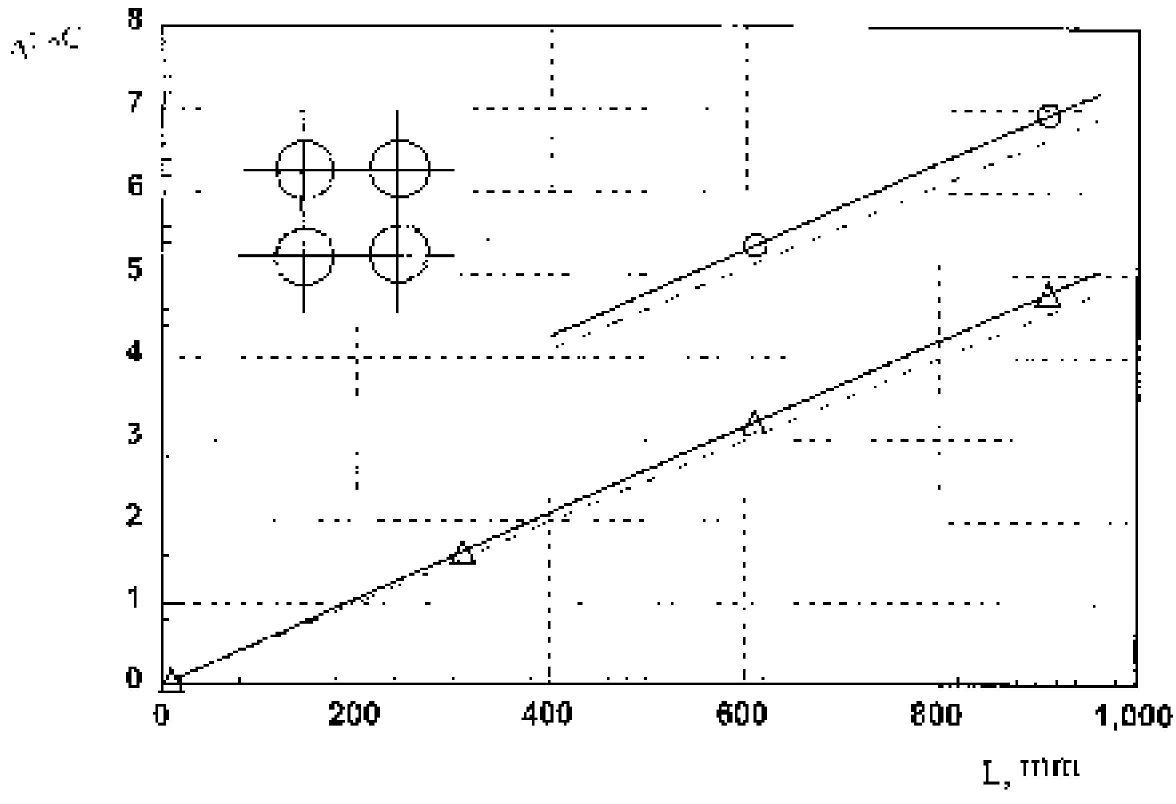


FIG. 19. Comparison between calculated and measured temperature field along the model rod length of the model assembly; - calculation ($\Delta t_\alpha = 2.15^o$C experiment $\Delta t_\alpha = 2.10^o$C).

5. PERSPECTIVE OF FURTHER RESEARCHES

The obtained data allow to conduct evaluations of temperature modes in the BREST-type reactors. However, in order to calculate thermohydraulic characteristics more strictly, further experiments should be carried out to study such factors as:

– Contact thermal resistance;
– Azimuth nonuniformities of rod temperature;
– Temperature fields and mass transfer in areas of thermohydraulic heterogeneities;
– Thermohydraulics of irregular channels;
– The overheating factors, etc.

For thermohydraulic substantiation of BREST–type reactor cores it will allow to reach the level of thermohydraulic substantiation of cores for sodium-cooled BN-type reactors. The additional information on problems under consideration is given in the works [3, 4].

REFERENCES

[1] ORLOV, V.V., New Stage of the Nuclear Power and Fast Reactors with Lead Coolant, Information bulleting **10** (3) SU Nuclear Society (1991) 6.

[2] ADAMOV E.O., ORLOV. V.V., Development of Nuclear Power on the Basis of New Concepts of Nuclear Reactors and Fuel Cycles, Program and Abstracts of Presentations, TZhT-98 Conference, Obninsk, 5–9 October 1998 (1998) 15.

[3] ZHUKOV, A.B., SOROKIN, A.P., KUZINA, U.A., SMIRNOV, V.P., Thermal Fields and Heat Transfer in Fast Reactor Cores Cooled by Liquid Lead, Abstracts of Presentations, The 10th Annual Conference of Nuclear Society of Russia, 28 June – 2 July 1999, Obninsk (1999) 86.

[4] KUZINA, U.A., SMIRNOV, V.P., ZHUKOV, A.V., SOROKIN, A.P., Studies of Thermal Fields and Heat Transfer on Mock-Ups of Fuel Assemblies of the Fast Reactor with Liquid Lead Coolant, Proceedings, Scientific Session "MIFI-2000", **8** Moscow (2000) 108–110.

CORROSION RESISTANCE OF STRUCTURE MATERIALS IN LEAD COOLANT WITH REFERENCE TO REACTOR INSTALLATION BREST-OD-300

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Abstract

Systematic researches for lead coolant have been performed from the beginning of 1990s within the framework of development of activities on the BREST-OD-300 reactor installation. The main type of corrosion damages in lead and lead–bismuth coolant is the local dissolution of structure materials (steels) and their components and then in those structure materials and components. The main results for ensuring high corrosion-erosive stability of materials in heavy metal coolant are given by the application of silicon-containing steels, the passivation oxygen, and the application of additional corrosion barrier as oxide films on the surface of structure elements. Various researches of corrosion behaviors of steels in lead coolant based on those results are under way. The experience gained so far allows to forecast the absence of corrosion damages of steels in lead–bismuth coolant up to 620 ^{o}C on the basis of tens thousand hours that was proved by successful experience of reactor installation activities. The obtained results from the test at higher temperature showed a good fuel-cladding compatibility.

The significant amount of experimental data of corrosion resistance of structure materials in heavy liquid metal coolant (Pb, Bi, eutectic Pb–Bi alloy), including the data of foreign sources obtained before the mid of 1960s, are adduced in the scientific literature. These data demonstrate, that the main type of corrosion damages in liquid Pb, Bi and Pb–Bi is the dissolution in them of structure materials (steels) and their components. The kinetic features of process of dissolution can be of different nature. For example, in some cases the dissolution is localized on boundaries of grains, causing inter-crystal infiltration of liquid metal (Pb, Pb–Bi) into steel.

The corrosion resistance study of structure materials in liquid Pb, Bi and Pb–Bi are systematically carried out in the Institute for Physics and Power Engineering (IPPE), Russia since the mid of 1950s. There was created and started up the unique complex of the experimental installations, including convection ampoules, loops, also non-isothermal circulating loops with forced circulation of coolant and reactor loops.

The basic varied parameters, on which the researches were carried out:

– Component and impurity composition of steels, the types of melting, influence of heat treatment of steels, condition and types of treatment of surfaces of steels and their weld joints;
– Type, structure and technology of protective antirust coats of steels. Such technologies, as alitizing, chromizing, nitriding, molibdenizing, oxidizing, and also lot of other kinds and types of coatings, including coatings created directly in liquid metal due to applying of dissoluble inhibitors were studied;
– The influence of temperature, and also gradient of temperature on liquid metal loop;
– The influence of coolant flow, conditions and flow regimes, especially of reactor core;
– Long-lived resource characteristics on parameter of corrosion resistance;
– The influence of stresses, thermocircles and vibrations.

The systematic researches for lead coolant were carried out from the beginning of 1990s, within the framework of development of activities on reactor installation BREST.

The main and the most dangerous kind of corrosion damages of materials both in Pb–Bi and in Pb coolants is the local corrosion of a material appearing as the separate corrosion–erosive centres (pittings). Local through corrosion damages of structure elements may appear at temperatures higher than 550°C after some hundreds hours under following conditions: unbalance of alloying and impurity elements in steel, poor quality of metal, absence of the control of quality of coolant, non–optimal coolant flow regimes. The typical corrosion rates in such cases are valued by values 2.55 mm/year.

The principle solutions ensuring high corrosion resistance of structure materials in heavy liquid metal coolant, were found by using of oxygen dissolved in coolant.

It was shown, as a result of long–term researches, that the corrosion resistance of structure materials in Pb and Pb–Bi essentially depends on concentration of oxygen, dissolved in liquid metal. At a definite level of concentration of dissolved oxygen the development of corrosion processes is locked due to protective oxide film formed on a surface of steel. At high temperatures an indispensable condition of blocking of corrosion processes is the presence of silicon in steel as additional alloyed element. The contents of silicon in steels used for heavy liquid metal coolant, are varied in limits from 1 up to 3.5% depending on application of steel.

Key role in solution of a problem of hyper-thermal corrosion resistance of materials was also played by developing and operational using of preliminary protective coatings on steels. These coatings are put on a surface of the most important structure elements of a reactor installation (fuel rod claddings, steam generators) at finishing stages of technological factory process of manufacturing of the suitable workpieces. The additional barriers on internal surfaces of liquid metal loop are formed also directly in coolant when reactor installation is prepared to operation (modes of running-in).

The best results on preliminary oxidizing of structure elements in a mode of factory cycle of their manufacturing were obtained by using of environments with low partial pressure of oxygen: Pb–Bi–O, $H_2O + H_2$ and CO_2. The applying of such technologies allows, first of all, to avoid kinetically critical stage of preliminary passivation process of the unsheltered reactor installation steels for the most important installation elements, and allows essentially to extend range of a permissible decrease of concentration of oxygen in coolant.

Thus, main factors ensuring high corrosion–erosive stability of materials in heavy liquid metal coolant (Pb, Pb–Bi) are:

– Application of silicon containing steels;
– Passivation of oxygen, regime application for coolant;
– Application of additional corrosion barriers as oxide films formed on a surface of structure elements during modes of running-in.

The gained experience allows with confidence to forecast absence of corrosion damages of steels in Pb–Bi coolant up to temperature 620°C on the basis of tens thousand hours, that is proved by successful experience of reactor installations activities.

As a result of the carried out recently researches of corrosion behavior of steels in lead coolant with rather high contents of oxygen was shown, that in temperature range 550 – 650°C on the basis of tests during 8500 hours the liquid metal corrosion of core materials of BREST reactor installation was also not watched. The tests were carried out near to high passivation boundary of ferrite–martensite 12%Cr-Si steel of the EP823 type, i.e., in range $4 \times 10^{-5} - 1 \times 10^{-6}$ mass% of oxygen dissolved in lead. At the same time, as was established as a result of these tests, protective oxide layers formed on steel are not optimal on their thickness. The researches for refinement of boundaries of passivation modes of activities of materials of BREST reactor installation are organized and carried out in IPPE.

The high reserve of corrosion resistance of steel EP823 was proved by last researches of fuel–cladding compatibility. The obtained results of research of compatibility of fuel rod materials (cladding of EP823 steel, lead contact heat–transfer layer, uranium mononitride fuel), tested at temperatures 650°C and 700°C during 9400 hours, were shown:

- The products of corrosion interaction and their thermal transfer on altitude of fuel rod were not revealed;
- There were not revealed infiltration of lead inside of a cladding, formation of corrosion centres on EP823 steel;
- The mechanical properties of fuel rod claddings were not changed as a result of a contact to a lead heat-transfer layer.

MULTIPURPOSED SMALL FAST REACTOR SVBR-75/100 COOLED BY PLUMBUM-BISMUTH

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Abstract

Currently the nuclear power (NP) development meets significant difficulties in many countries. First of all it relates to complicating and cost rising of nuclear power plants (NPPs) due to essential enhancing the safety requirements. The possibility and expediency of developing the NP based on unified small power reactor modules SVBR-75/100 with fast neutron reactors cooled by lead–bismuth eutectic alloy is substantiated for the nearest decades in the Paper. Based on those modules the following designs can be realized: renovating of the NPP units which operation term has been exhausted; regional nuclear heat power plants of 100–300 MW power which need near cities' location; large power modular NPPs (~1000 MW) like US concept PRISM or Japanese concept 4S; nuclear power complexes for sea water desalinating in developing countries which meet non-proliferation requirements, reactors for plutonium utilization and minor actinides transmutation.

1. INTRODUCTION

Today, despite phasing out a lot of programs from nuclear power (NP), the interest in the nuclear technology based on using lead–bismuth liquid metal coolant for reactor cooling is enhanced. In our country this technology has been developed during several decades. Eight nuclear submarines (NS) with reactors cooled by lead–bismuth coolant (LBC) were constructed. The total operation time is about 80 reactor-years. The innovative nuclear power technology which has no analogs in the world has been demonstrated in our industry [1].

The interest in this technology is explained by the fact that the coolant natural properties enable to design the reactor installation (RI) with very high safety level. Besides, the simultaneous improving of technical and economical characteristics can be expected. Now the conditions for implementing this technology into civilian NP have been arisen.

2. SUBSTANTIATION OF CHOOSING THE FAST REACTOR COOLED BY LBC

Liquid metal cooled fast reactors are classified as RIs which safety is ensured mainly due to their inherent safety. It is associated with a number of their internal features.

Lack of poisoning effects in the fast reactor (FR), low value of negative temperature reactivity coefficient, compensation of fuel burn-up and slagging processes by plutonium generation enable to ensure the operative reactivity margin to be less than delayed neutron share and to eliminate the prompt neutrons runaway in the reactor under operation.

Liquid metal coolant (LMC) used for FR cooling considerably motivates the RI design and hydraulic scheme, the NPP technical and economical characteristics. Among the LMCs used in NP sodium is the most commonly used.

Choosing this LMC for fast reactors was caused by its opportunity of intensive heat removal due to its good thermal and physical properties. It enabled to provide short plutonium doubling time that was the obligatory requirement at the early phases of designing the fast breeder reactors in the 60s and 70s years, and was caused by the fact of unproved forecast of NP's very high development rate and, therefore, need for fuel self-providing. That was the reason why in the 50s Academician A.I. Leypunsky, who considered various LMCs for cooling the FRs, preferred sodium, though LBC was initially considered for these purposes [2].

Today and in the foreseeable future there is no need for such short plutonium doubling time that can maintain the sodium cooled FRs. The necessity for designing fast breeder reactors has been postponed to many decades [3]. It enables to use the opportunity of using LBC for FR cooling.

It should be highlighted that the experience of using sodium coolant has been gained in conditions of NPPs' power reactors industrial operation and could be used immediately, whereas the experience of using LBC has been acquired in conditions of NSs' RIs operation, which were different from those at NPP in design and operation regimes, and so this experience requires applicable adaptation to new conditions. However, these circumstances should not be the reason for not using LBC in NP due to weighty backgrounds.

These backgrounds are as follows:

- Enhancing the reactor safety due to:
  (1) Negative total void reactivity effect which is typical for small power reactors of SVBR-75/100 type and elimination of the possibility of realizing the local positive void reactivity effect. The last relates to impossibility of coolant boiling up in the most heat stressed fuel subassemblies (FSA) even in cases of severest accidents (LBC boiling point is ~1670°C, sodium boiling point is ~870°C);
  (2) Use of chemically inert LBC which eliminates arising of explosions and fires if the coolant comes in contact with water and air which is possible in emergency situations;
- Improving technical and economical parameters due to using two-circuit scheme of heat removal, eliminating some safety systems, systems of accident localization, simplifying the technology of managing the spent nuclear fuel (SNF);
- Solving the number of principal problems of LBC, which determine RIs reliability and safety: ensuring the proper coolant quality and its maintenance in the course of operation, ensuring radiation safety associated with forming alpha-active polonium-210 radionuclide, etc.;
- Closeness of RI SVBR-75/100 scale factor to that of NS's RI that enables to use many of technical solutions tried out by practice.

3. SAFETY ENSURING CONCEPT

High boiling point and latent evaporation heat, which are LBC natural properties, practically eliminate the possibility of the primary circuit over-pressurization and reactor thermal explosion at any considerable accidents because in this case no pressure increase arises.

Impossibility of coolant boiling up enhances the reliability of heat removal from the core and safety due to lacking the heat removal crisis phenomenon.

For proposed integral design of the RI the loss of coolant with its circulation interruption through the core caused by tightness loss of the reactor main vessel (postulated accident) is eliminated by introducing the safe-guard vessel, small free volume between the main reactor vessel and the safe-guard vessel and impossibility of coolant's boiling-up off in case of primary circuit gas system tightness loss.

In the case of failing all systems of emergency cooling down (postulated accident), elimination of core melting down under heat decay effect and keeping intact the vessel of small and medium-sized power reactors are ensured completely by passive way with large margin to boiling due to heat accumulation in internal reactor structures and in coolant with short-time increase of its temperature. In this case heat is removed through the reactor vessel (which temperature increases correspondingly) and the air gap to the water storage tank around the reactor vessel and into the ambient air at its natural circulation after boiling away the water (in case of full de-energizing and stopping the operation of cooling down systems for five days or more).

In the case of emergency overheating and simultaneous postulated failure of emergency protection systems (EPS), the reactor power decrease down to the level which does not cause the core damage is ensured by reactivity negative feedbacks.

The coolant itself reacts with water and air very slightly. The emergency processes caused by primary circuit pipes tightness loss and steam generator (SG) inter-circuit leaks occur without hydrogen release and any exothermic reactions. Within the core and RI there are no materials releasing hydrogen as a result of thermal and radiation effects and chemical reactions with coolant. Thus, the possibility of arising chemical explosions and fires caused by internal reasons is eliminated completely.

The elimination of water or steam penetration into the core caused by full rupture of SG tube and consequent over-pressurizing the reactor vessel designed for maximal possible pressure for that accident are ensured by coolant circulation scheme. In this scheme steam bubbles and water drops are thrown out on the free coolant level by upgoing coolant flow. Thereby steam effective separation occurs in the gas space of the primary circuit above the coolant level, whence steam goes to the passively operating emergency condensers system, and in case of their postulated failure or simultaneous rupture of several SG tubes it goes to the bubbler through the rupture membranes.

The operation experience of the RI using LBC at the NS has revealed the possibility of RI safety operation during some time in conditions of small SG leakage, which does not cause any significant deviations of the designed technological parameters. This fact allows to realize necessary repair works not urgently but at the convenient time.

Chemical inertness and impossibility of coolant's boiling up in case of the primary circuit tightness loss and its property to retain iodine, which radionuclides, as a rule, represent the major factor of radiation risk just after the accident, as well as the other fission products (inert gases are an exception) and actinides, reduce sharply the scale of radiation consequences of that accident.

The containment above the reactor serves as the additional safety system barrier. Its main purpose is the protection against external effects. Low storage of potential energy in the primary circuit restricts the RI destruction scale caused by over-standard external effects to only external impact forces.

As computations reveal, extremely high safety potential peculiar to this type of RIs is characterized by the fact that even when such initial events as containment destruction and primary circuit tightness loss coincide (that is possible in case of diversion or military attack), neither reactor runaway, nor explosion and fire occur, and the radioactivity exhaust is lower than that which requires the population evacuation. Taking into account that energy stored in the coolant (heating, chemical and compression potential energy) is minimal in comparison with other coolants used and previously mentioned physical special characteristics of fast reactors and RI integral design, one could look forward to designing the RI of extremely high safety level. On the base of those reactors NP would become not only socially acceptable for population but also socially attractive if it gained economical competitiveness with heat electric power plants using organic fuel (we have all backgrounds for it).

4. MULTI-PURPOSED REACTOR MODULE SVBR-75/100

RI SVBR-75/100 is designed for generating steam which parameters enable to use it as working medium in thermo-dynamical cycle of turbogenerator installations. It is possible to vary the steam parameters according to the needs. Today the base variant of RI SVBR-75 has been developed for generating the saturated steam under the pressure of 3.24 MPa, i.e., the pressure which is produced by the SG of the Novovoronezh NPP (NVNPP) second unit, and when turbogenerator with intermediate steam superheating is used, this enables to generate electric power of about 75 MWe when working under condensation regime.

The design of RI SVBR-75 module has two-circuit scheme of LBC heat removal for the primary circuit and steam–water for the secondary circuit. The integral design of the pool type is used for the RI primary circuit (see Fig. 1). It enables to mount the primary circuit equipment inside the one vessel. RI SVBR-75 module includes the removable part with the core (the reactor itself with control rods), 12 SG modules with compulsory circulation over the primary circuit and natural circulation over the secondary circuit, 2 main circulation pumps (MCP) for LBC circulation over the primary circuit, devices for controlling the LBC quality, the in-vessel radiation shielding and buffer chamber (pressurizer) which are the parts of the main circulation circuit (MCC).

The scheme of coolant moving within the MCC is as follows: through the windows of reactor outlet chamber the coolant heated in the core flows to the inlet of the SG twelve modules which have parallel connection. It flows from top to bottom in the intertube space of the SG modules and is cooled there. Then the coolant penetrates into the intermediate chamber, from which it moves in the channels of in-vessel radiation shielding, cooling it, to the reactor vessel upper part and there it forms the free level of “cold” coolant (peripheral buffer chamber), further from the reactor vessel upper part the coolant flow moves to the MCP suction inlet.

The adopted circulation scheme with free levels of LBC existing in the reactor vessel upper part and SG module channels, which contact the cover gas medium, ensures the reliable separation of steam–water mixture out of coolant flow when the accidental tightness loss of SG tube system occurs, and existing of gas medium ensures the possibility of coolant’s temperature changes.

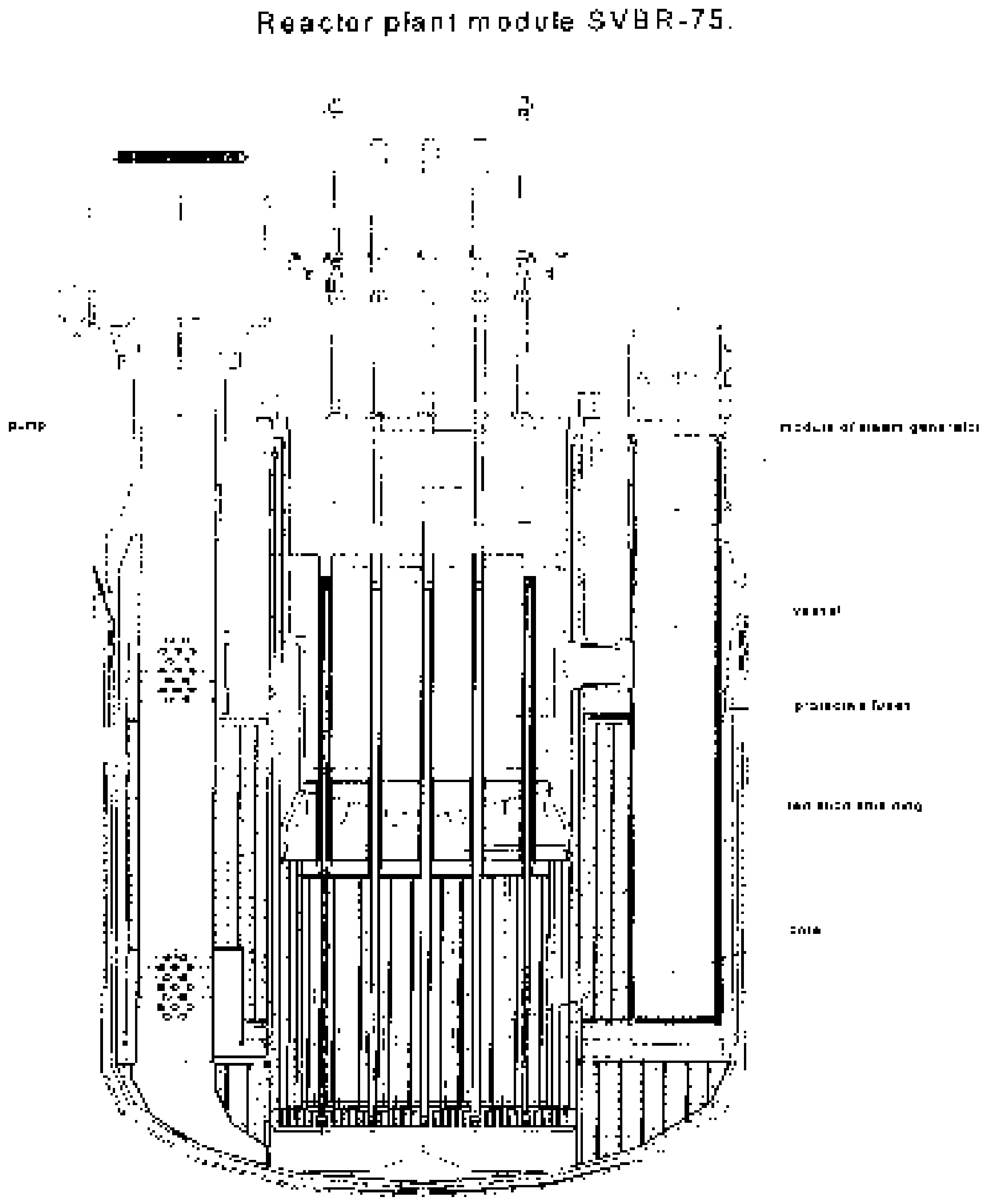


FIG. 1. Reactor general view.

Reactor vessel is placed in the tank and is mounted there (see Fig. 2). The tank is filled by water and is designed for cooling the RI in case of beyond design accidents. The gap between the main vessel and safe-guarding one is chosen to ensure the circulation circuit disrupture in case of accidents related to the tightness loss by the reactor vessel major vessel.

The secondary system is designed to operate the steam generator producing saturated steam with multiple natural circulation through the evaporator-separator circuit, as well as to provide the scheduled and emergency RI cooling by using steam generator.

The design provides three systems of heat removal to the heat sink both in scheduled and emergency reactor core cooling.

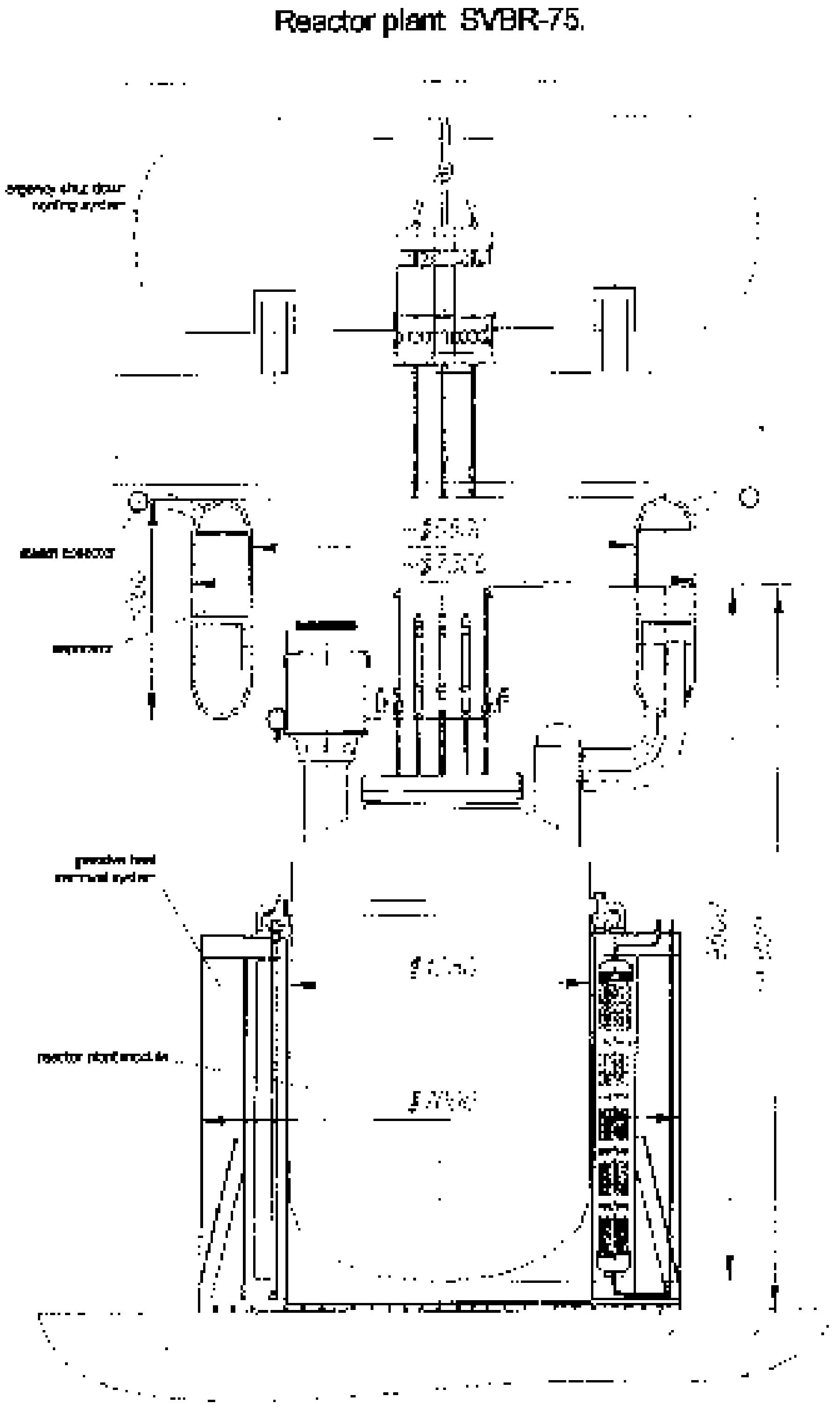


FIG. 2. Reactor plant SVBR-75.

First system includes normal operation RI and turbine equipment and systems. The system is cooled by the primary heat removal via steam generator heat exchange surfaces, steam being dumped to the turbine generator systems (TGS).

The second heat removal system is an independent cooling system (ICS), which includes, besides a part of primary and secondary circuit equipment, a separator condenser (circuit) with natural circulation. Via this circuit the heat is removed to the intermediate circuit water. This system ensures independent (from the turbine generator systems) reactor cooling and independent reactor plant operation at a power level up to 6% N_{nom} at the nominal steam pressure. Connection/disconnection of ICS is realized with no operator action and without using external power supply systems.

Third core heat removal system is a passive heat removal system (PHRS). The heat is removed from the reactor to the water storage tank located around the reactor vessel. This system ensures the reactor core cooling in case of postulated maximal accident with all secondary circuit equipment failed, reactor protection system failure and total de-energizing of the NPP.

The principal technical parameters of RI SVBR-75 are presented in Table I.

TABLE I. PRINCIPAL TECHNICAL PARAMETERS OF RI SVBR-75

Parameter	Value
Number of reactors	1
Rated heat power, MW	268
Electric power, Mwe	75
Steam production rate, t/h	About 487
Steam parameters:	
– Pressure, Mpa	3.24
– Temperature	238
Feed water temperature, ºC	192
Primary coolant flow rate, kg/s	11180
Primary coolant temperature, ºC	
– Core outlet	439
– Core inlet	275
Core dimensions, D × H, m	1.65 x 0.9
Average value of specific volume core power, kW/dm^3	135
Average value of specific linear core power, kW/m	~22
Fuel:	
– Type	UO_2
– U-235 mass loading, kg	1476
– Average U-235 enrichment, %	15.6
SG numbers	2
Evaporator numbers in SG	6
Evaporator dimensions D × H, m	~0.6 × 4
Numbers of MCPs	2
MCP electric driver power, Kw	400
MCP head, Mpa	~0.5
Primary circuit coolant volume, m^3	18
Major reactor vessel dimensions, D × H, m	4.53 × 6.92
Designed earthquake of magnitude (MSK)	9
Designed construction terms, months	36

RI SVBR-75 operates for eight years without core refueling. During this period there is no need in carrying out fuel works. At the initial stage the use of mastered oxide uranium fuel in the uranium closed fuel cycle is provided similar to that in reactor BN-600. Further the use of dense uranium and plutonium nitride fuel is possible. In this case the core breeding ratio is more than 1 and in the plutonium closed fuel cycle the reactor would operate by using only depleted waste pile uranium.

The refueling is performed after lifetime ending. Refueling means the complex of works on restoring the reactor full power resources which includes core replacing works, as well as dismantling and mantling works associated with them. FSA by FSA fuel unloading out of the reactor vessel is provided and loading of the fresh core as a part of new removable part is provided as well. Core refueling is realized by using special refueling equipment. Unloaded FSAs are placed in special capsules with liquid lead which solidifies further.

RI SVBR-75/100 is designed on the construction base of RI SVBR-75 and distinguishes from it only by SG operating in one through regime and generating the superheated steam of 400°C temperature and 9 MPa pressure. Thus the electric power is ensured to be of about 100 MWe.

5. AREAS OF USING MULTIPURPOSED REACTOR MODULE SVBR-75/100

5.1. Renovation of NPP's units with exhausted lifetime

The number of NPP's units which lifetime has been exhausted is growing in the NP of many countries. It needs for huge expenditures on decommissioning the units out of operation and constructing the new ones for replacing the removing power capacities. At the same time there is an opportunity for untraditional solving this task by using NPP's units renovating. Renovating means replacing the RIs for units with exhausted lifetime by new RIs, using the NPP's existing buildings and structures with full replacing of removed power capacity. However, this way of replacing removed power capacities rigidly restricts the type of the RI used for renovation:

– The possibility of mountling the RI in existing rooms after dismantling the equipment of the "old" RI (the reactor which dismantling is followed by great radiation doses is an exception);
– Satisfying the regulation requirements for safety including "old" units without containment.

That way of replacing the removing power capacities of the second, third and fourth units of Novovoronezh NPP (NVNPP) based on the unique Russian lead–bismuth technology has been proposed by SSC RF IPPE together with "Hidropress" Design Bureau and "Atomenergoproekt" State Planning and Design, Research and Survey Institute and has been developed according to the task of concern "Rosenergoatom".

SVBR-75 nominal power is chosen to be 75 MWe due to limited dimensions of NVNPP renovation units' SG compartments which do not enable to install the large power module, necessity for ensuring the equality of generated steam and feeding water flow rates for SVBR-75 and RI VVER-440 SG, possibility of reactor module complete plant fabrication and its transportation by the railway, as well as closeness of the scale factor to NS's RIs that enables to use some developed technical solutions and reduce R&D. For replacing the power capacities of the second unit four SVBR-75 modules are installed in SG compartments, and six modules are installed for each of third and fourth units.

Arrangement of modules in SG compartments of the MCP of NVNPP's second unit is presented in Fig. 3.

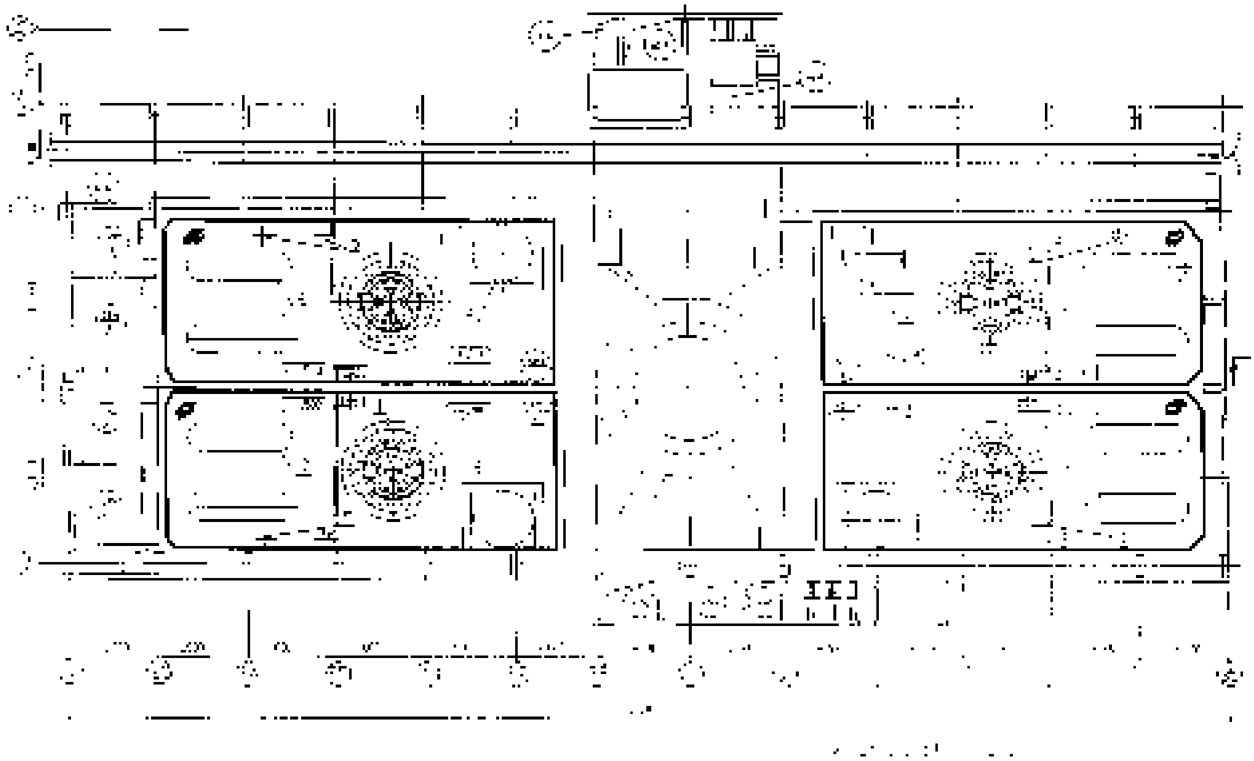


FIG. 3. SVBR-75 reactor modules arrangement at the NVNPP's second unit.

The great economical efficiency of renovating the "old" NVNPP units is expected: specific capital renovation cost is $560 per kWe, that is half as many as that for constructing the new NPP unit [5].

5.2. Regional NHPP based on module SVBR-75/100

In the world many regions and first of all medium-sized and large cities face with serious difficulties in energy supplying especially heat supplying in winter.

The way for solving this problem is use of nuclear heating power plants (NHPP). However, use of traditional type RIs (which use water under the high pressure for reactor cooling) for these purposes needs designing the number of additional safety systems if compared with those accepted for NPPs situated at the distance of 25 km or more from the cities. This results in going up the NHPP cost and at the same time does not eliminate the principal possibility of scarcely probable nuclear accident but with severe consequences because the high pressure in the reactor, which is the internal cause of its arising, is not eliminated.

The high level of SVBR-75/100 reactor module inherent safety makes it possible and expedient to use it simultaneously for producing electric energy at the NPP and for heat generating at the NHPP which needs near city's location. So we eliminate the possibility of arising severe accidents accompanied by explosions, fires with prohibited radioactivity exhaust requiring population evacuation beyond the NHPP site not only if there are personnel's errors and equipment failures but if they coincide, if there are terrorist groups' actions. It is possible to construct NHPPs of 100–300 MWe by using these standard modules.

For RI SVBR-75/100 the principal design solutions for the unit's main building of regional NHPP distinguish from those for traditional type reactors. Small dimensions of SVBR-75/100 module and developed properties of inherent safety of FRs with LBC require the RI protection from only those external effects: aircraft falling, shock waves, maximal earthquake. There is no need for designing the tight shell withstanding significant internal pressure. Small dimensions of protected reactor compartment and simple scheme of RI enable to reduce the terms of NHPP unit construction and significantly reduce the construction cost.

The simplicity of automated control system (ACS) conditioned by using passive systems for cooling down the RI facilitates the construction cost reducing.

As it has been assessed by experts, the economical competitive ability with HPP using fossil fuel will be ensured due to the following facts: almost lack of expenditures on nuclear fuel transportation; long lifetime of the fast reactor core, which ensures without refueling RI operation during about 8 years; low cost of spent nuclear fuel storing; almost lack of liquid radioactive waste and expenditures for its conditioning; complete plant fabricating of the RI and possibility of its transportation by truck, railway or sea to the NHPP constructing site that reduces the constructing terms and approaches them to those of traditional HPPs and reduces the investment cycle; sharp reduction of expenditures for constructing safety ensuring systems due to RIs' high inherent safety; high commercial production due to great demand for these NHPPs; possibility of export delivery of these RIs. The cost of these NHPPs is one-fifth to one-tenth as many as that of large NPPs.

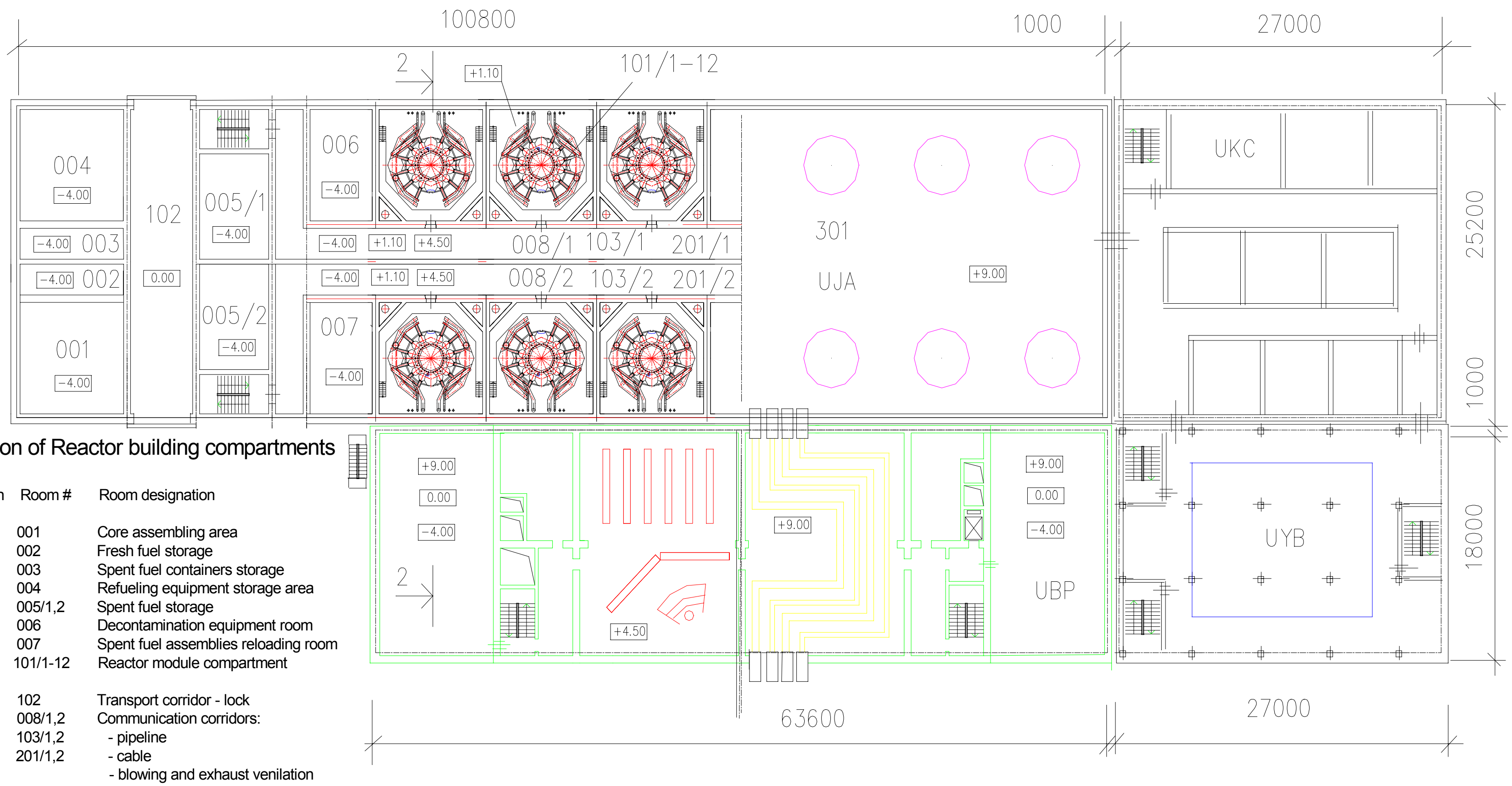
5.3. Large power modular NPP (Figs 4 and 5)

On the basis of commercially produced modules SVBR-75/100 it is expedient to develop the design of modular NPP of large power (1 GWe and more at one unit). The prospect for that principle of designing NPP is shown in conceptual design developed in the USA (PRISM) [6] and in Japan (4S) [7]. However, use of this principle for LBC cooled reactors is the most effective.

The economical gain is achieved due to: constructing bulks reduction because of eliminating the number of safety systems and localizing systems, reducing the specific material expenditure (including bismuth demands) as compared to traditional reactors of large power, reducing the fabricating cost due to high commercial production, reducing the NPP constructing terms when reactor modules are delivered to the constructing site in high plant readiness. It enables to improve the conditions of credit receiving and repayment and to increase the competitive ability of NPP. The preliminary estimations have revealed that the specific capital cost of constructing that NPP is expected to be not more than that for NPP's VER-1000 unit.

5.4. Dual-purpose nuclear desalinating power complex for developing countries

Many developing countries in Africa and Asia suffer from deficiency of fresh water and electric energy. The majority of these countries do not have sufficient own resources of fossil fuel, which can meet their demands. In some countries fuel transporting is difficult, there are no powerful electric power transmission lines. The marketing researches conducted recently by IAEA [8] have revealed that in many cases small sized nuclear power sources of 100 MWe can be used economical effectively for these purposes.



Explication of Reactor building compartments

Elevation	Room #	Room designation
-4.00	001	Core assembling area
	002	Fresh fuel storage
	003	Spent fuel containers storage
	004	Refueling equipment storage area
	005/1,2	Spent fuel storage
	006	Decontamination equipment room
	007	Spent fuel assemblies reloading room
-4.00 +1.10	101/1-12	Reactor module compartment
0.00	102	Transport corridor - lock
-4.00	008/1,2	Communication corridors:
+1.10	103/1,2	- pipeline
+4.50	201/1,2	- cable
		- blowing and exhaust venilation
+9.00	301	Reactor hall

FIG. 4. Nuclear island plan.

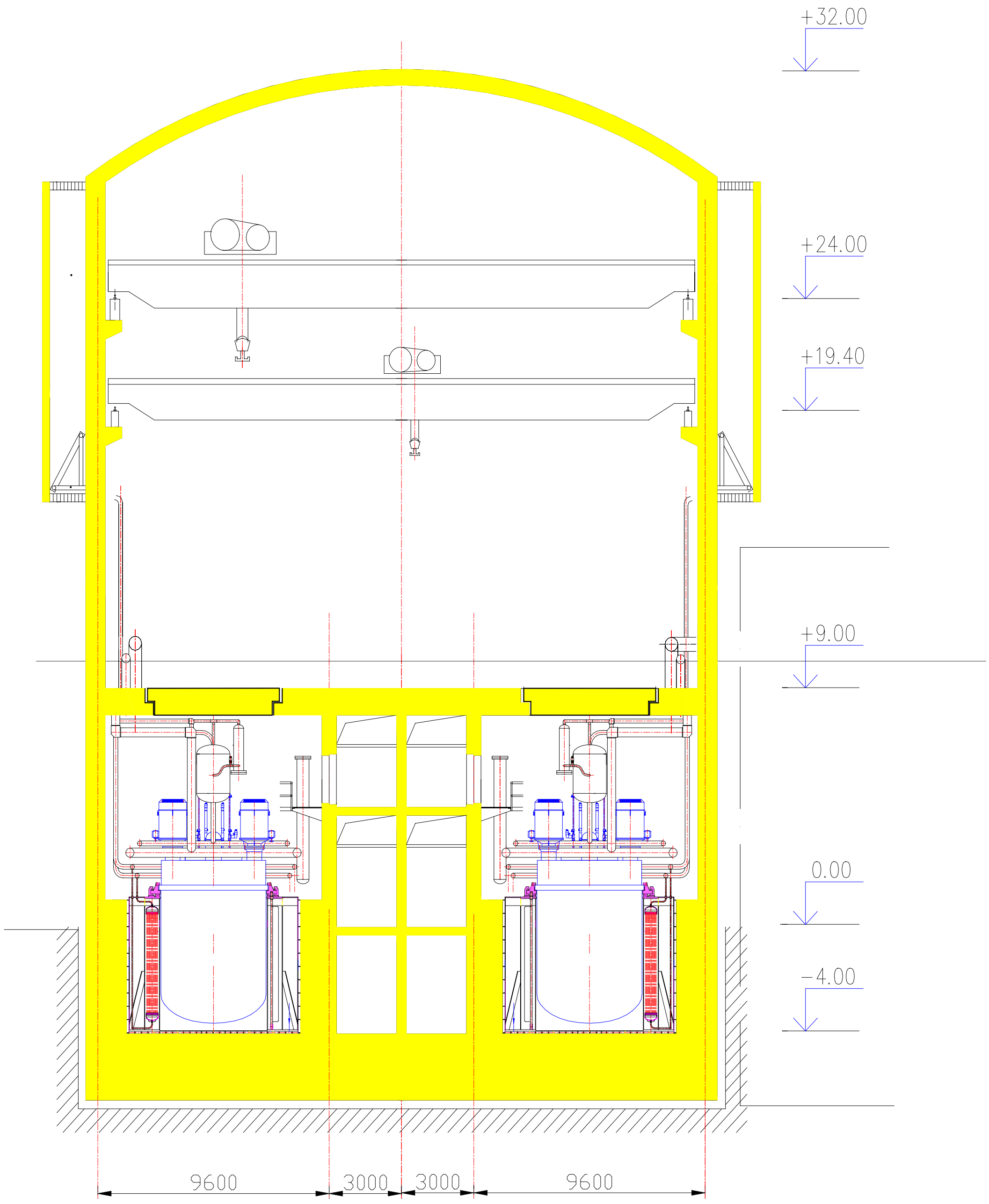


FIG. 5. Reactor compartment's sectional view.

However, the developing countries' particularities concerned with not sufficiently high level of education, technical culture, social and economical development, as well as possibilities of arising local military conflicts, put forward special requirements to the nuclear power technology which are stricter than those for developed countries.

First of all, these requirements are RI inherent safety against severe accidents that is based on RI inherent properties ensuring safety not only in cases of personnel's errors and multiple failures of technical systems coincidences, but in cases of sabotage terrorist actions, etc. Besides, they must meet strict non-proliferation requirements [9], including that refueling in the country-user must be eliminated and due to this fact the lifetime duration must be 10 years or more. The opportunity for reactor unit transportation to the country-manufacturer in the condition of nuclear and radiation safety for refueling and then transporting it to the country-user again must be ensured. Thefts of fuel must be technically eliminated as well. Besides, the competitive ability to the alternative resources of receiving fresh water and electric energy must be ensured.

RI SVBR-75/100 (Fig. 6) meets these requirements the most completely. It has extremely high safety potential, lifetime duration needed, ensures the regime of non-proliferation due to the following:

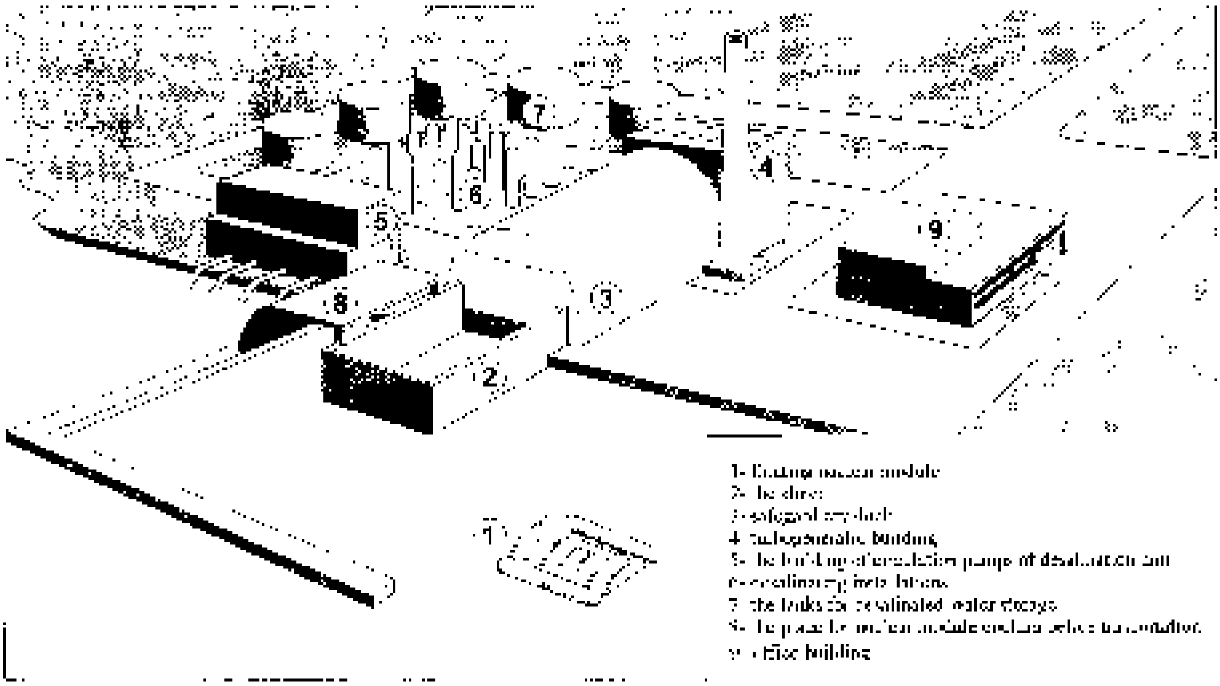


FIG. 6. General view of NPP site. Variant with electric supply and sea water desalination.

- Use of uranium with enrichment less than 20%;
- Lack of refueling in the country-user;
- Opportunity of transporting the reactor module after ending its lifetime to the country-manufacturer in the condition of nuclear and radiation safety with LBC “frozen” in the reactor.

5.5. Plutonium utilization and long-lived minor actinides transmutation

In different countries the policy on managing plutonium, the quantity of which increases steadily, is different. It is determined by the fact that on the one hand Pu is very valuable fuel for the future NP using FRs. On the other hand it can be used for political and military purposes. Besides, Pu is high radiotoxic material and is considered to be NP dangerous radioactivity waste. Long-lived minor actinides belong to it, too.

For the first case the issue of Pu managing (both extracted one and that is contained in spent nuclear fuel) results in long reliably controlled storing.

The second case results in the task of Pu transmuting into the form that reduces the risk of its unauthorized proliferation or its complete burning up and minor actinides nuclear transmutation.

For solving this task the works on mastering the new nuclear technology using accelerator-driven systems are carried out. The main stimulus for developing this technology is Pu and minor actinides blanket subcriticality that eliminates the prompt neutrons runaway nuclear accident.

Along with it, this task can be solved on the basis of already mastered technology. For example, during eight years one reactor module SVBR-75/100 can transmute about 1000 kg of Pu (weapons- or reactor grade one) into the form protected against unauthorized proliferation ("spent fuel standard") at reducing its quality as a weapon material compared to the weapon Pu. In terms of 1 GWe·year 1.25 t of Pu will be utilized in those reactors. If minor actinides (first of all americium) is introduced into fuel, their transmutation into short-lived radioactive wastes will take place.

The safety level needed will be ensured due to developed propertied of RI SVBR-75/100 inherent safety which have been mentioned above.

6. THE PROBLEMS OF INCREASING THE OVERALL BISMUTH PRODUCTION AND EXPLOITING THE LEAD COOLANT

The factor limiting LBC use in large-scale future NP depends on deficiency of today's bismuth manufacturing, which has been determined by its consumption level.

Today's bismuth situation looks like uranium one during the period from 1906 to 1940 when only 4000 t of uranium were mined over the 34 years. But in 1980 the world uranium mining (except the ex-USSR) reached 40 000 t per year [10]. The explored uranium resources increased to the great extent too.

One should point out that bismuth content in the earth's crust is one-fifth as much as that of lead [11]. However, deposits of bismuth with high content of about 5–25% are very rare and locate in Bolivia, Tasmania, Peru and Spain. That is why 90% of world's bismuth has been manufactured from the wastes of lead-refining, copper-smelting and tinning plants.

The experience of the USA and Japan has revealed that equipping copper-smelting, tungsten and lead plants by dust-catching and dust-utilizing systems can essentially enhance bismuth extraction and has great significance for the environment.

In Russia that work can be launched at MINATOM enterprises being the members of AO "Atomredmetzoloto". According to VNIPI Promtechnologia information, the production of bismuth of about 2500–3000 t per year, together with gold and other metals, can be organized in the south-east of Chita Region where the resources of gold–bismuth ore have already been explored. That volume of bismuth production will ensure the year input of 2.5–3.0 GWe for the NPP using RI SVBR-75/100. It is necessary to carry out bismuth geological works, develop existing and new mines, implement the technology of deep bismuth extracting.

Though the cost of bismuth is ten times as much as that of lead it is a very small part of capital costs for NPP construction. It should be taken into account, that coolant is not spent and could be used again in other RIs.

RDIPE proposes to consider lead coolant [12] as an alternative to lead–bismuth alloy because the scales of lead production do not limit the rate of large-scale NP development.

However, use of lead coolant is associated with some engineering problems. Due to the higher lead melting point (it is 327°C against 125°C for eutectic alloy), lead coolant temperature must be increased significantly. It complicates the solution to the problems on coolant technology, structure materials corrosion resistance and mass transfer. Being applied to the lead–bismuth coolant, the problem has taken about 15 years for its solving. Besides that, it results in more complicated RI operation because of great opportunity of forming solid "sows" in the primary circuit under transitive regimes, accidents, repairs, refueling and not clear today the opportunity of safe for RIs melting the "sows". Nowadays the works on mastering the lead coolant are at their initial phase.

Taking into account all mentioned above, use of lead coolant would be justified only if the rates of power capacities increase for the NPPs with the RIs considered were high enough, and expenditures for increasing the annual bismuth production and its cost were put up as much that the increase of specific capital costs for NPP construction would not be economically reasonable.

As bismuth has been in deficiency, one can consider non-eutectic alloy with bismuth content decreased up to 10% (versus 56% in eutectic alloy). Being compared with lead coolant, its melting temperature is decreased by 77°C (to 250°C) and that facilitates RI operation and reduces maximal temperatures of fuel elements claddings up to the values tested for eutectic alloy under the conditions of long-term operation tests.

7. CONCLUSION

The experience gained in the course of designing the LBC cooled NS RIs enabled to design multi-purposed small size reactor module SVBR-75/100, which ensures the most complete realization of inherent safety principle for severest accidents and their deterministic elimination due to using the fast reactor, LBC and the primary circuit integral design. Based on these modules, renovation of NPP's units exhaust their lifetime can be carried out, regional NHPPs for energy shortage regions can be built, large power NPPs of modular type can be constructed, meeting IAEA requirements power complexes for electric energy producing and seawater desalinating can be built in developing and other countries, reactors for utilizing Pu and transmuting the long-lived minor actinides can be designed.

REFERENCES

[1] GROMOV, B.F., TOSHINSKI, G.I., STEPANOV, V.S., et al., Use of Lead Bismuth Coolant in Nuclear Reactors and Accelerator-Driven Systems, Nuclear Engineering and Design, **173** (1997) 207–217.

[2] TOSHINSKY, G.I., A.I Leypunsky: Selected Works, Reminiscences, Naukova Dumka Publishing House, Kiev (1990) 225.

[3] HIPPEL, F., Evolutionar Approach to Fission Power, Proceedings, International Conference on Evaluation of Emerging Nuclear Fuel Cycle Systems Global-95, **1** (1995) 380–387.

[4] STEPANOV, V.S., KLIMOV, N.N., KULOKOV, M.L., LEGUENKO, S.K., CHITAYKIM, V.I., GROMOV, B.F., TOSHINSKI, G.I., GRIGORIEV, O.G., Application in Electric Power Industry Technology of Transport Reactors with Lead-Bismuth Coolant, Presentation, Advisory Group Meeting on Propulsion Reactor Technology for Civilian Applications, 20–24 July 1998, Obninsk, Russian Federation.

[5] IGNATENKO, E.I., KORNIENKO, A.G., ZRODNIKOV, A.V., GROMOV, B.F., TOSHINSKI, G.I., GRIGORIEV, G.I., CHITAYKIN, V.I., STEPANOV, V.S., KOOCKLIN, V.Z., et al., Renovating the First Generation NPP's Units Removed from Operation after Exhausting Their Service Life by Placing Them in Steam Generator Boxes of SVBR-75 RIs Using Liquid Metal Lead-Bismuth Coolant, Technical Report, 8th Russia Nuclear Society Annual Conference, Ekaterinburg-Zarechny, Russian Federation (1997).

[6] BERGLUNG, R.C., TIPPETS, F.E., PRISM, the Plant Design Concept for the U.S. Advanced Liquid Metal Reactor Program, Technical Report, 6th American Power Conference, Chicago (1989).

[7] HATTORI, S., MINATO, A., A Large Modular LMR Power Station which Meets Current Requirements, Proceedings, ICONE-3, **2**, Kyoto, Japan (1995) 787–790.

[8] INTERNATIONAL ATOMIC ENERGY AGENCY, Potential for Nuclear Desalination as a Source of Low Cost Potable Water in North Africa, IAEA-TECDOC-917, Vienna (1996).

[9] BROWN, N.W., HESSBERGER, J.A., New concepts for small power reactors without on-site refueling for non-proliferation, Presentation, Advisory Group Meeting on Propulsion Reactor Technology for Civilian Applications, 20–24 July 1998, Obninsk, Russian Federation.

[10] SINEV, N.M., The Nuclear Power Economy, Energoatomizdat, Moscow (1987).

[11] Encyclopedia of Inorganic Materials, Main Editorial of Ukrainian Soviet Encyclopedia, Kiev (1977).

[12] ADAMOV, E.O., ORLOV, V.V., Requirements to a New Nuclear Technology for Large-Scale Power Industry, Proceedings, International Topical Meeting on Advanced Reactors Safety ARS'94, Pittsburgh, USA, **2** (1994) 636.

CONCEPTUAL DESIGN STUDIES ON VARIOUS TYPES OF HLMC FAST REACTOR PLANTS

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Abstract

To seek for a promising concept of a heavy liquid metal coolant (HLMC) fast reactor plant, Japan Nuclear Cycle Development Institute (JNC) and the electric utilities conducted conceptual design studies on various types of plant concepts and compared these concepts based on technical feasibility and economical perspective. Finally, Pb–Bi cooled medium tank type reactor was selected as the most promising concept.

1. INTRODUCTION

Japan Nuclear Cycle Development Institute (JNC) and the electric utilities have been performing feasibility studies on commercialized fast breeder reactor system [1]. In this research studies, we conducted conceptual design studies for several kinds of FBR plants with not only sodium coolants but also heavy liquid metal coolant (HLMC), gas (CO_2, He) and water (pressurized light or heavy water, super critical water). HLMC such as lead (Pb) and lead–bismuth (Pb–Bi) have some advantages as follows:

- *Chemical inertness;*

  This means the reaction with water is less violent and less harmful compared with sodium coolant. It is easier to adopt a simplified heat transport system in which heat is transported from primary coolant to steam generators directly without any secondary heat transport systems. This simplification can reduce the plant construction cost. Also, the reaction between HLMC and air is mild and this character leads to simplification of countermeasures for coolant leakage compared with sodium cooled plants.

- *Excellent neutronic characters* [2];

  HLMC has excellent neutronic characters such as less moderating effect and higher scattering effect than those of sodium. Due to its less moderating effect, the neutron spectrum becomes harder and breeding parameters would be improved. The higher scattering effect decrease the neutron loss from the core and also decrease the burnup reactivity loss over the refueling interval. These neutronic characters can allow us to design lower pressure drop core with increasing the pin pitch-to-diameter ratio. This can enhance the natural circulation ability in the primary coolant circuit.

- *High boiling point;*

  The boiling points of HLMCs are higher than that of sodium. This gives us wider allowable coolant temperature rise in case of accidents and enhances the passive safety characteristics with negative reactivity feed back.

Of course, there are some disadvantages such as high load conditions on the structure due to HLMC's high density, the problem of ^{210} Po, and less compatibility with structural material that needs countermeasures for corrosion and erosion. To seek for a promising concept of a HLMC fast reactor plant, we conducted a conceptual design studies on various types of plant concepts and compared these concepts based on technical feasibility and economical perspective.

2. THE SCOPE OF THIS STUDY

Figure 1 shows the scope of this conceptual design study. Four types of plant concepts listed below are examined:

- Concept 1: Large-scale pond type reactor with Pb as coolant;
- Concept 2: Large-scale loop type reactor with Pb as coolant;
- Concept 3: Medium-scale module tank type reactor with Pb as coolant;
- Concept 4: Medium-scale module tank type reactor with Pb–Bi as coolant.

Concepts 1 and 2 are selected to seek for scale merit on economical aspect. In Concepts 3 and 4, we tried to reduce the inventory of HLMC and to ease the load conditions on the structures and seek for competitiveness with module effect such as mass production and learning effect. Through this conceptual design study, we identified some technical features on the each concept and roughly evaluated economical competitiveness based on the total weight of Nuclear Steam Supply System (NSSS). These considerations are described in Section 3. In addition to the conceptual design studies, we investigated the amount of Bi natural resources and the results of this estimation are explained in Section 4.

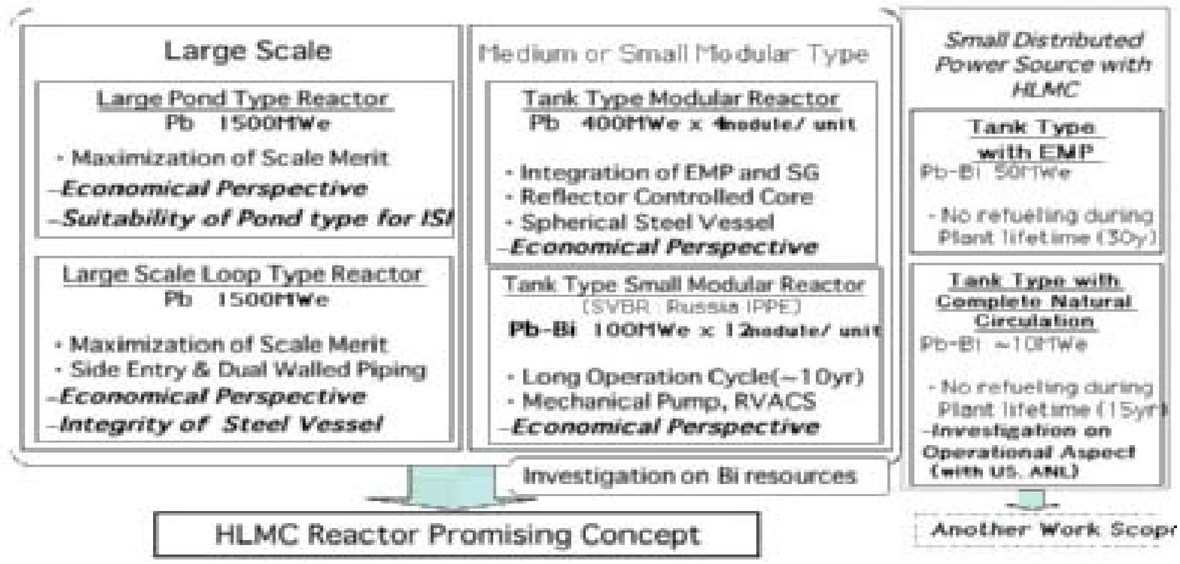

FIG. 1. Work scope.

3. SURVEY OF VARIOUS HLMC REACTOR CONCEPTS

3.1. Large-scale pond type reactor with Pb coolant (Concept 1)

The design concept is shown in Figs 2.1 and 2.2. The coolant system is a pond type and the plant weight is supported by concrete that surrounds vessels. Referring the Russian BREST design [3], we tried to accommodate the concept to both the Japanese seismic design conditions and the regulatory requirement for in-service inspection on the coolant boundary. As a result, the structure of the coolant boundary is different with that of the BREST concept. To keep the concrete temperature below the allowable limit (< 65°C), we adopted thermal shielding and concrete cooling system on the surface of the concrete. The reactor power was assumed to 1500 MWe to seek for scale merit on economical aspect. As a result, huge amount of NSSS materials (nearly 9 500 t; ~6.4 t/MWe) is needed. One of reason of this huge amount of materials is the aseismatic structure under the Japanese seismic condition. The other reason is the amount of the thermal shielding structure (~3000 t). Comparing the up to date sodium cooled fast reactor design concept in JNC (1500 MWe, the weight of the NSSS; 2700 ~3300 t), this concept seems to be little economical advantage based on the Japanese seismic design standard.

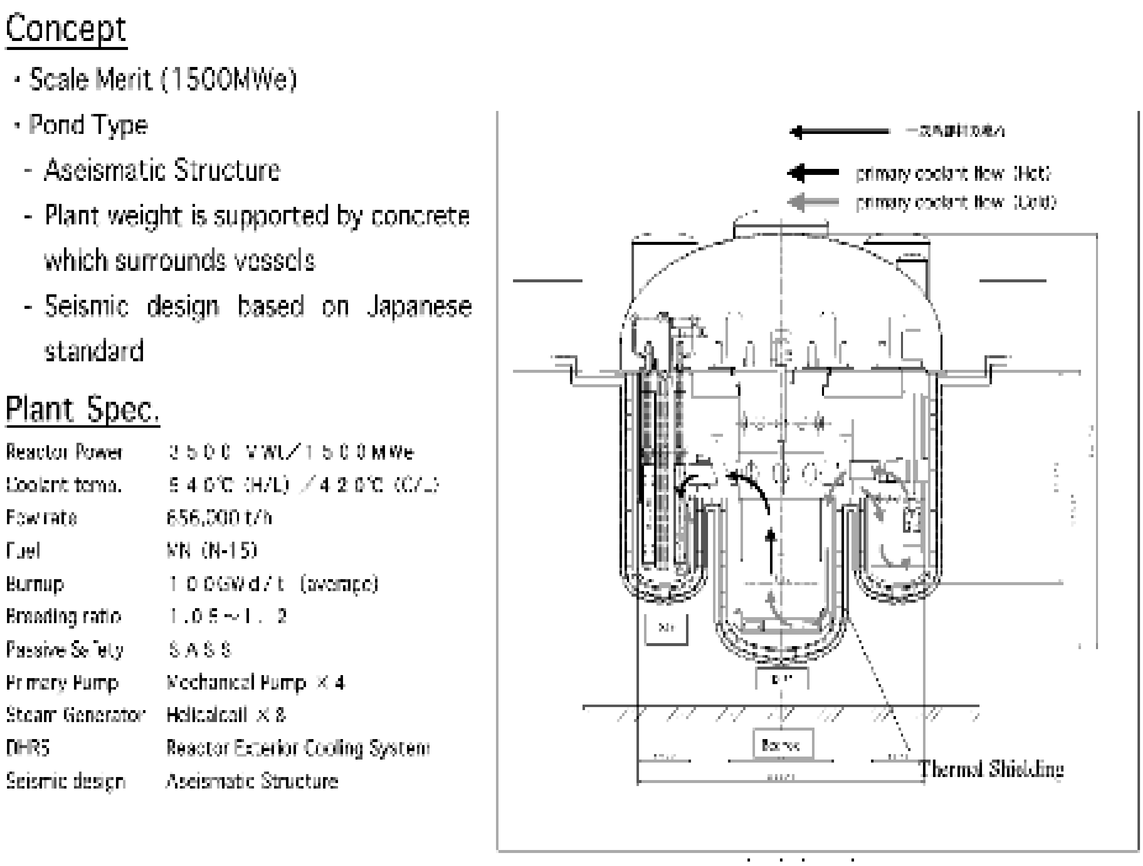

FIG. 2.1. Large scale pond type reactor (Pb) (2/1).

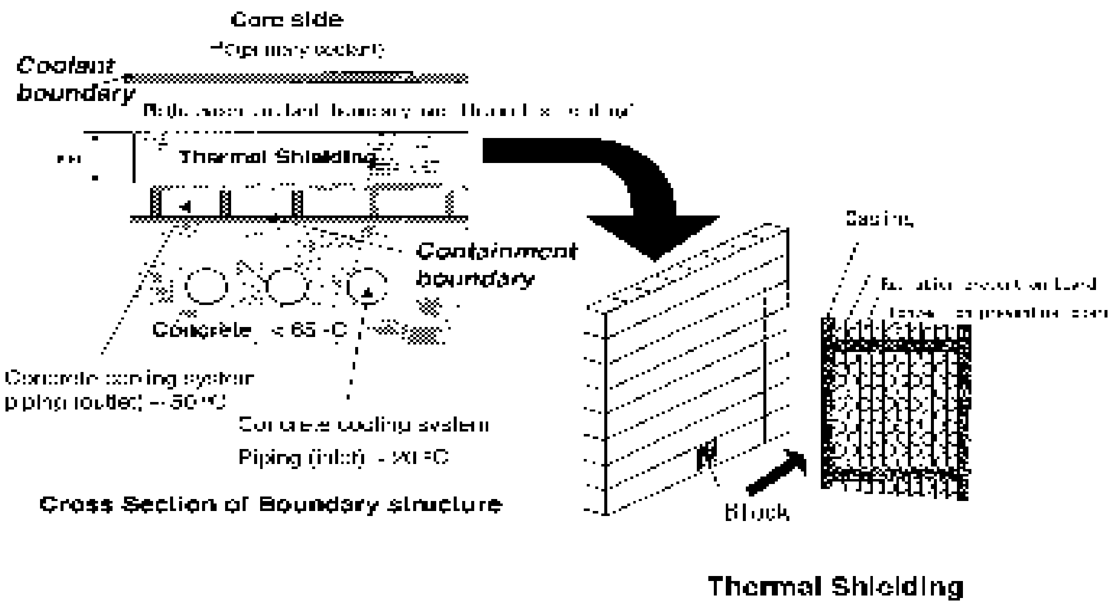

FIG. 2.2. Large scale pond type reactor (Pb) (2/2).

3.2. Large-scale loop type reactor with Pb coolant (Concept 2)

This concept (Fig. 3) was also considered to seek for scale merit (1500 MWe). To reduce the coolant inventory, we adopted a loop type concept. Based on the consideration on the type of piping system, we selected the side entry and dual walled piping concept with slide-joint inner wall to cope with thermal expansion on the piping system. With this new idea and the seismic isolation system, the amount of NSSS structural materials can be reduced to about 3300 t. This weight of the NSSS is comparable to that of the sodium cooled tank type fast reactor concept (1500 MWe, 3100 ~3300 t) but somewhat larger than that of the sodium cooled loop type fast reactor concept (1500 MWe, 2700 t). However, the concept of the dual walled tube has some engineering difficulties as follows:

- Inspection of inner boundary of dual walled piping;
- Tricky structure such as slide-joint is inevitable;
- Incompatibility between the slide-joint and the corrosion prevention measures; mechanical tearing of the oxide thin layer.

Further R&Ds are needed to solve the above issues.

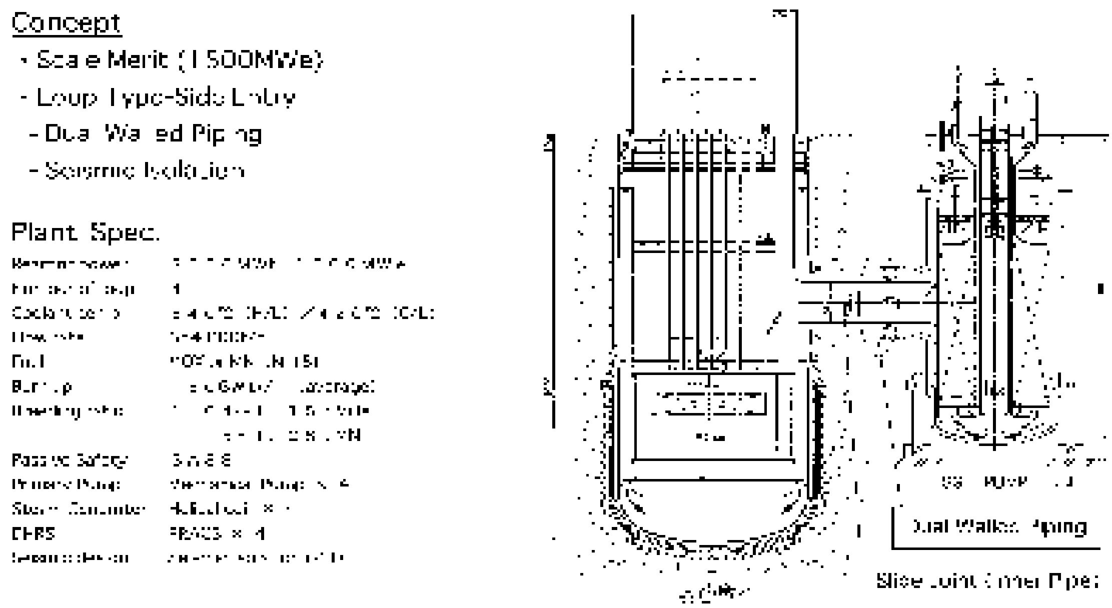

FIG. 3. Large scale loop type reactor (Pb).

3.3.Medium-scale module tank type reactor with Pb coolant (Concept 3)

Figure 4 shows the plant concept of the medium-scale module tank type reactor (4 module (400 MWe × 4 module)/plant). In this concept, we tried to reduce the inventory of HLMC and to ease the load conditions on structures and seek for competitiveness with module effect such as mass production and learning effect. The Pb coolant is circulated by the Pb electromagnetic pumps (EMPs). The EMP and the steam generator (SG) are integrated into one component to simplify the coolant circuit in the reactor tank. The reflector control system is employed as the reactivity control system; the absorber rods are located only around the circumference of the core. This idea can simplify the upper core structure and increase the flexibility on the design of fuel handling mechanism in the reactor tank. The amount of NSSS structural materials was evaluated to about 1496 t (~3.7 t/MWe). The estimated construction cost is about

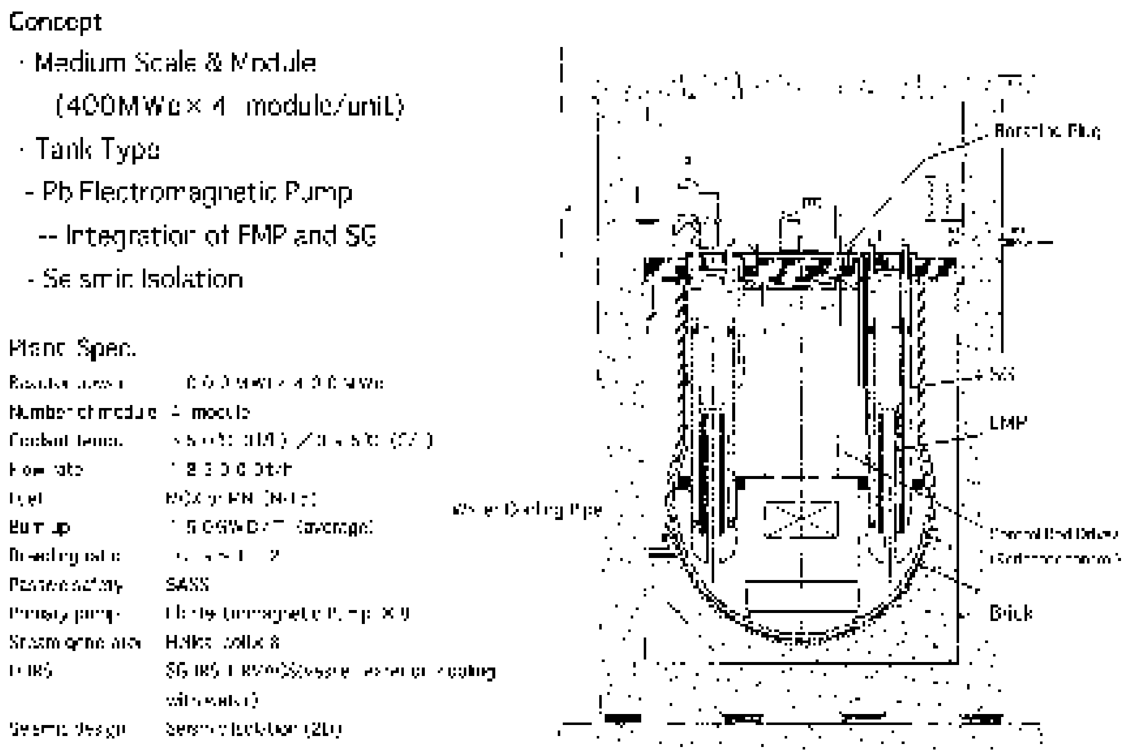

FIG. 4. Medium scale module tank type reactor (Pb).

300 000 ¥/kWe. But this estimated value has not reached our requirement (200 000 ¥/kWe). It is desired to make further efforts for reduction of the amount of structural material. Also, there are some disadvantages with EMP as follows:

– The EMP has been combined with SG. This makes the repair work difficult;
– The large-capacity Pb electromagnetic pump is supposed to need long R&D term.

3.4. Medium/small-medium/small-scale module tank type reactor with Pb–Bi as coolant (Concept 4)

Referring to the Russian SVBR concept [4], we considered the technical feasibility and economical perspective of this type of HLMC reactor. Figure 5 shows this concept. One of the advantages of this module tank type reactor concept (100 MWe x 12 module/unit) seems to be minimization of the coolant inventory. Small amount of the coolant inventory makes the NSSS weight decrease and the earthquake resistance can be improved. Also, it is advantageous from the viewpoint of the Bi resource problem. The estimated weight of the NSSS per unit power is around 2.8 t/MWe and is larger than that of sodium cooled fast reactor concept (1500 MWe, pool type; ~2.0 t/MWe, loop type; ~1.7 t/MWe). It seemed necessary to further optimize the design concept. To achieve further construction cost reduction, there are some ways as follows.

– Seeking for scale merit with increasing the reactor power;
– Reconsidering the number of SG unit and optimization of its heat transfer tube type (bayonet or helical coil).

It is important for a Japanese plant user to make sure that the seismic design satisfies the Japanese design standard.

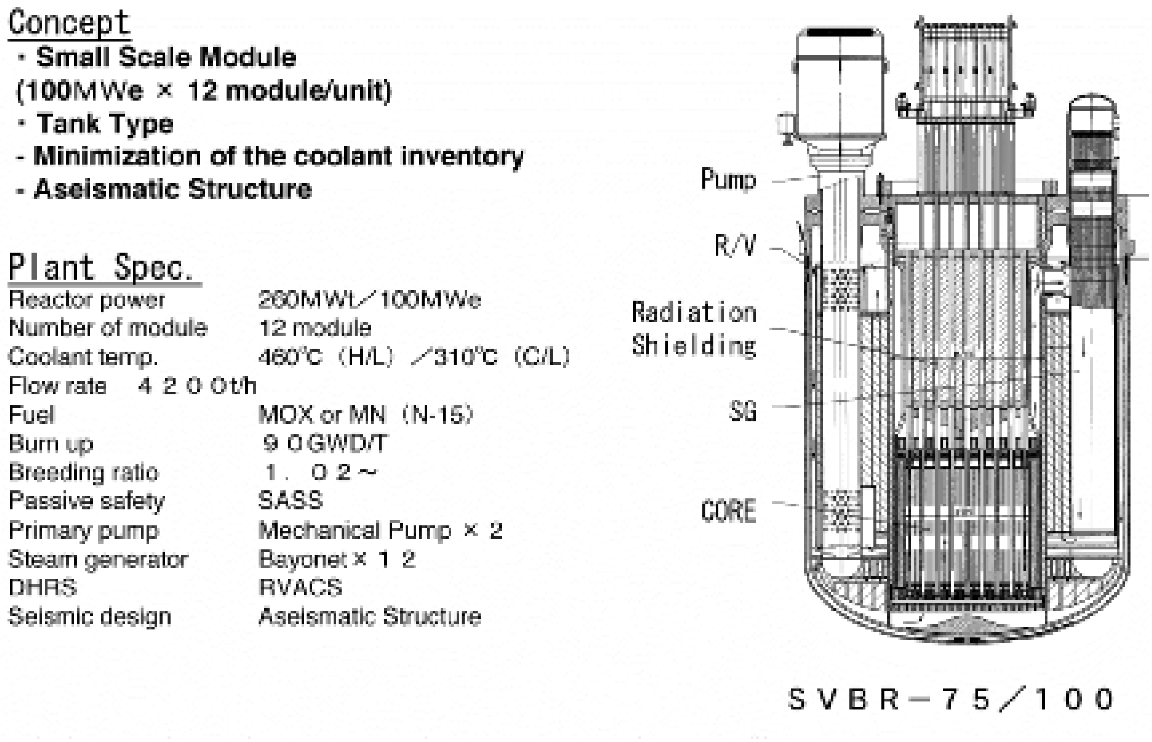

FIG. 5. Medium/small scale module tank type reactor (Pb–Bi), referring to Russian SVBR concept [4].

4. CONSIDERATION ON THE PB–Bi AS REACTOR COOLANT

4.1. Estimation on the amount of Bi natural resources

According to the US geological survey published as "Minerals Yearbook Mineral Commodity Summaries 1999" [5], the amount of Bi resources is estimated 260 000 t. This estimated value is mainly based on existing mines. We examined the total amount of Bi resources taking into account the undeveloped area on the earth. To estimate the amount of the natural resource value, we assumed some geological conditions listed below with some extent of conservatism:

- Total mass of the earth's crust form surface to 3 km depth: 1.2×10^{18} t;
- Possibility of mine development: 20% of the upper part of the crust;
- Possibility of mining & ore dressing: 0.01% of the total mineral in the crust;
- Abundance of Bi in the earth's crust: 0.2 ppm;
- There was a tendency of concentration near the surface due to relatively low melting point of Bi.

Multiplying all of the above numbers and rates, we can roughly estimate that the amount of available Bi resources is around 5 000 000 t. Based on the conceptual design study described above, Bi inventory can be supposed to be 1100 ~18 000 t/GWe. Therefore, the expected power by Pb–Bi cooled power reactor is deduced by the below calculation

(5 000 000 t) / (1100 ~ 18 000 t/GWe) = 280 ~ 4500 GWe.

On the other hand, the future demand for nuclear power in Japan at 2100 is estimated about 120 GWe. Based on the above consideration, the amount of Bi resources seems not to be a significant issue. But the price of Bi depends on not only the amount of the Bi resources but also the principle of the market such as demand and supply. Further consideration may be needed for the cost aspect of Bi resources.

4.2. Comparison between Pb and Pb–Bi

Pb–Bi coolant with lower melting point seems to be more attractive than Pb coolant because:

- There is no need for maintenance work under high temperature (~400^{o}C);
- Lower temperature of SG feed water (FW) gives us the possibility of more compact SG and better controllability of FW system;
- Wider operational temperature range has better compatibility for passive decay heat removal system (DHRS).

According to the Russian researcher [6], Amount of Po in the cover gas system in a Pb–Bi cooled reactor plant may not so higher than that of a Pb cooled reactor because lower shutdown temperature of a Pb–Bi reactor may mitigate the amount of Po^{210} release during maintenance and refueling duration. Further consideration is needed based on the specific plant design and operating condition.

5. CONCLUSION

From the consideration described above, we concluded:

- In general, large-scale type concepts have little economical advantage because of its huge amount of material needed for its severe load conditions (Concepts 1 & 2);
- Under the Japanese seismic condition, aseismatic structure makes the amount of material increase. For the large-scale pond type reactor, total amount of the thermal shielding material became huge (Concept 1);
- For the large-scale loop type reactor, we selected a side entry and dual walled piping concept with slide-joint inner wall to cope with thermal expansion on the piping system. However, there seemed to be difficulty with compatibility between the slide-joint and oxide film corrosion prevention measures (Concept 2);
- The medium modular types seemed to be preferable to large types but the total weight of NSSS was lager than that of the evolutional sodium cooled fast rector concept that had been studied at the same time. It is necessary to further optimize the design concept (Concepts 3 & 4);
- Lower melting point of Pb–Bi coolant makes it more attractive than Pb coolant. Based on roughly but conservative estimation, the amount of Bi resources seems not to be a significant issue.

One of the important ways to attain competitiveness is simplification of NSSS and Balance-of-plant (BOP) design by employing passive features. As described in Section 1, compared with sodium cooled core, the neutronic characters of HLMC can allow us to design lower pressure drop core with less deteriorating the core breeding parameters. This advantage leads to the idea of a complete natural circulation cooled tank type reactor concept. We are now performing design study on this type of reactor with Pb–Bi coolant. To avoid the high load conditions on the structural materials, we selected medium-seized and modular type. The DHRS are also natural circulation type. Eliminating the coolant circulation pumps from the reactor vessel makes the heat transport system simple and can decrease the amount of the weight of NSSS. The capacity of BOP system such as the electricity supply system, the HVAC system and the component cooling water system would be reduced because there is no need to supply electricity for the circulation pumps and no need the circulation pump cooling system. The reduction of the amount of BOP leads to the reduction of the volume of the buildings that contain these systems and components. Therefore, the reduction of the plant construction would be expected.

REFERENCES

[1] NODA, H., Current Status of Fast Reactor System in Japan, Presentation, ICONE-8, 2–6 April 2000, Baltimore, USA.

[2] SPENCER, B.W., The Rush to Heavy Liquid Metal Reactor Coolants-Gimmick or Reasoned, Presentation, ICONE-8, 2–6 April 2000, Baltimore, USA.

[3] ORLOV, V.V., et al., 300 and 1200 MW(th) BREST Reactors Design, HLMC-98, (1999).

[4] ZRODNIKOV, A.V., et al., Multi purposed Reactor Module SVBR-75/100, Presentation, ICONE-8, 2–6 April 2000, Baltimore, USA.

[5] USBM (United States Department of Interior Bureau of Mines), Minerals Yearbook Mineral Commodity Summaries 1999 (1999).

[6] GROMOV, B.F., et al., Lead-Bismuth as a Perspective Coolant for Advanced Reactors and Accelerator-based Plants, Proceedings, International Topical Meeting on Advanced Reactor Safety, ARS’94, Pittsburgh, USA, 1994.

STATUS REPORT ON THE ADS RESEARCH ACTIVITY IN ITALY

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Abstract

The Italian activity on Accelerator Driven Systems (ADS) is divided into two interacting programs based on the C. Rubbia design of an Energy Amplifier (EA). The first is a fundamental R&D program named TRASCO (Italian acronym standing for waste transmutation: "TRAsmutazione SCOrie") and the second is an industrial program related to system design. The TRASCO program is again divided in two parts. The first part, led by INFN (the Italian national research institute for nuclear physics), is devoted to the development of a 30 mA·1 GeV proton linac accelerator based on Radio Frequency Quadrupole (RFQ), Drift Tube Linac (DTL) and Superconducting Cavities (SC) technologies. The second part is devoted to the subcritical system. In this framework the CHEOPE and LECOR facilities built at ENEA (the Italian national research agency for energy, environment and new technologies) Brasimone site are described. They will be used for testing the basic technology related to Pb–Bi coolant (loading/unloading operations, oxygen control, instrumentation, filtering) and the compatibility of structural materials.

1. INTRODUCTION

Starting from 1995 a growing interest on ADS has taken place in Italy, following the ideas of Carlo Rubbia's Energy Amplifier concept [1] and has given origin to two different programs:

- Basic R&D program named TRASCO started in 1997 under the leadership of INFN (for the accelerator part) and of ENEA (for the sub-critical system). The TRASCO program promoted collaboration among groups of different competencies (accelerator, reactor physics, plant design), and was aimed at providing relevant results in support of an industrial program;
- The industrial program involving Ansaldo and CRS4, devoted to make a preliminary design of an ADS experimental plant, considered the "Italian proposal" for a European Roadmap [2] for developing an ADS demonstration facility. The preliminary design will be completed in the first half of 2001.

2. THE TRASCO PROGRAM

The program, devoted to study the physics and to develop the technologies needed to design an ADS for nuclear waste transmutation, was prepared with close reference to Carlo Rubbia's Energy Amplifier proposal. It consists of two main parts regarding, respectively, the accelerator and the sub-critical system.

2.1. Accelerator

The aim of the project is to design a 1 GeV, 30 mA continuous current, super conducting proton linac. The resulting power, 30 MW is the one needed to drive a plant with a thermal power of 1.5 GW and a multiplication coefficient k ~0.95. The R&D extends in three main directions:

- Design and construction of the proton source, with a $\varepsilon_{RMS} \leq 0.2\pi$ mm mrad and of high availability and reliability. It is currently being assembled and tested;

- A reference preliminary conceptual design of the medium energy section (100 MeV), which includes an RFQ and a DTL (352 MHz). A more detailed design work of the RFQ has been done and a 3 m long aluminium model of the RFQ was built for Radio Frequency (RF) field stabilisation tests. Construction of a full power 2 m section is in progress. Studies of an Independently Phased Superconducting Cavity Linac (ISCL) to be used instead of the traditional DTL have also been done;
- Design of the high energy SC linac (100 MeV-1 GeV, 352 MHz) including the design of the SC RF cavities (similar to the ones used in the CERN LEP accelerator) and construction of prototypes.

2.2. Sub-critical system

The part relative to the sub-critical system, managed by ENEA, was structured in the following fields:

- General safety criteria and classification;
- Nuclear data;
- Neutronics;
- Thermal-hydraulic;
- Beam window technology;
- Materials technology and compatibility with lead and/or lead–bismuth alloy.

2.2.1. The LECOR facility

This LECOR facility (LEAd CORrosion) has been designed by ENEA in collaboration with Fabbricazioni Nucleari s.p.a. and has been put into operation at the end of 2000. The objective of the facility is the study of the corrosion of steels and other materials for ADS in presence of flowing Lead–Bismuth Eutectic (LBE).

The experimental campaigns will qualify components and instrumentation such as the oxygen meters and the oxygen control system. The total LBE inventory in the loop is 6000 kg, the installed electrical power is 150 kW and it can work between 350°C and 550°C at a maximum flow-rate of 4 m^3/h. Linear velocity of Pb–Bi in contact with the specimens will be between 0.5 and 1.0 m/s. A picture of the facility is shown in Fig. 1.

3. THE CHEOPE FACILITY

The CHEOPE facility (CHEmistry and OPErations) has been built in ENEA for preliminary testing of operation using liquid metals and as a small test facility for different European projects studies (such as MEGAPIE [3] and MHYRRA [4]). It consists of a tank filled with 400 L of LBE at a design temperature of 450°C. The flow-rate is only 1 L/s for a maximum pump pressure of 6 bar and the installed electric power supply is 180 kW (including test section requirements). The main objectives of the facility are:

- Characterisation of alloy furnace;
- Filtration tests of alloy, through porous media, in different oxidising conditions;
- Assessment of different level meters;
- Characterization of bubbling pressure meters;
- Characterization tests of flow-meters based on magnetic flux distortion;
- Characterization tests of rotary induction pumps (not requiring wall wetting);

- Level measurements and observation of free surface in view of windowless configuration studies (in collaboration with the MHYRRA project);
- Thermal characterisation of MEGAPIE cooling pins;
- Characterization tests of the measurements and control system for the oxygen concentration;
- Corrosion tests in a controlled oxygen environment.

FIG. 1. The LECOR test facility at ENEA Brasimone.

The picture of the test facility is shown in Fig. 2. The facility has been put into operation in July 2000.

Fig. 2. Cheope Test Facility At Enea Brasimone.

4. THE INDUSTRIAL PROGRAM

The industrial program is devoted to make a preliminary design of an ADS 80 MW_{th} demonstration facility [5], considered the "Italian proposal" for a European Roadmap for developing an ADS experimental plant. The purpose of the work was to define/confirm the main technical features applicable to an ADS demonstration prototype, in particular:

- Plant safety and functional requirements;
- Accelerator configuration (including proton beam transport tube to the sub-critical reactor);
- Target/window configuration;
- Fuel element type, composition and thermal-hydraulic evaluation;
- Core and supporting structures configuration;
- Core reference cycle;
- Primary coolant fluid and circulation;
- Structural materials;
- Plant thermal cycle and heat removal system (normal and emergency) configuration;
- Fuel handling system and storage configuration;
- Containment system configuration;
- Main components preliminary design;
- Primary system auxiliaries configuration;
- Instrumentation and control architecture;
- Plant building preliminary layout (plot plan and general arrangements);
- Civil structures preliminary design.

The results obtained so far, though preliminary and not exhaustive allow outlining a consistent demonstration prototype configuration, whose main key technical features, with the corresponding reference solutions, are reported in Table I.

The choice of the power (80 MW_{th}) is motivated by the fact that this is the minimum consistent with a representative core characterised by an annular configuration. Moreover, the accelerator needed to operate the subcritical reactor requires a beam power up-scaling factor of 2–4 with reference to existing ones and, at last, the decay heat can be removed by Reactor Vessel Air-Cooling System (RVACS), with negligible creep damage, according to RCC-MR.

Though the process of selection of the accelerator type for the Demonstration Facility is on-going, the basic scheme assumed is a reasonable extrapolation of the operating Paul Scherrer Institute (PSI) facility and is based on a three-stage system capable of suppling a proton beam of a few mA (up to 5–6 mA) at 590 MeV. The first stage, made of the proton source and a small cyclotron, supplies the low energy pre-injection beam (5–6 MeV). The following accelerator stages are provided by two separated-sectors cyclotrons. The intermediate-stage cyclotron provides a low-medium acceleration with extraction at 100 MeV. The final stage cyclotron (the so called "ring cyclotron") boosts the proton beam to the final energy of 590 MeV. The booster is an 8-sector magnets cyclotron with 6 RF cavities operating at 50 MHz – 1MV.

Molten Pb–Bi eutectic has been chosen as primary coolant. From the neutronics point of view it behaves like pure lead — which was the first candidate for the Energy Amplifier — but it allows lower operating temperature of the reactor (300°C at the core inlet and 400°C at the core outlet). Moreover the important experience on its use achieved by the Russians with the reactors for submarine propulsion can be exploited.

TABLE I. MAIN CHARACTERISTICS OF THE DEMONSTRATION PROTOTYPE DESIGN

Power	80 MW_{th} subcritical system controlled by a 600 MeV, 6 mA proton beam
Plant safety	Full passive system
Target / window	Pb–Bi target with 2 options under evaluation: a) Window target b) Windowless target
Core	0.97 (at beginning of cycle) < k_{eff} < 0.94 (at end of cycle) at full power
Fuel	U and Pu MOX
Primary coolant	Pb–Bi (300°C at core inlet, 400°C at core outlet)
Primary coolant circulation	Circulation enhanced by gas injection in a natural-circulation reactor configuration
Secondary coolant	Low vapour pressure organic diathermic fluid (280–320°C)
Normal power removal	Air coolers (system designed to use six air coolers available at ENEA)
In-vessel fuel handling	Two rotating plugs, one fixed arm and one direct lifting machine
Safety decay heat removal	Reactor Vessel Air Cooling System (RVACS)
Earthquake protection	Common basemat on antiseismic support
Reactor roof	Metallic plate
Main vessel and safety vessel	Hung, 316L steel
Hot shut down temperature	280°C
Cold shut down temperature	200°C
Pb–Bi purification	Internal
Economics	Under evaluation

The reactor has been designed in the pool-type configuration because of the possibility to contain within the main vessel all the primary coolant with the highly active polonium originated by bismuth, and because of the large experience acquired with the design and operation of sodium cooled pool-type reactors. In the EA concept proposed by Carlo Rubbia, the lead coolant operates in natural circulation, driven by the density difference between the riser and the down-comer of the primary circuit. For the Demonstration Facility the lead–bismuth circulation is enhanced by a flow of 80 NL/s Argon cover gas, injected into the bottom part of a circular array of identical pipes (0.2 m internal diameter), which make up the riser.

The target material is also molten Pb–Bi eutectic, separated from the primary coolant. Pb–Bi has good spallation efficiency and neutronics properties and low melting temperature. Two options have been considered: (i) a "windowless" design, where the free surface of the spallation material is the interface with the void of the beam transport line, and (ii) a "window" design, where a physical separation is made by means of a 9Cr-1Mo-V-Nb Martensitic steel hemispherical window. Both designs have advantages and drawbacks. The beam window is a delicate element whose lifetime is at currently difficult to assess, since it is affected by the combined action of liquid metal corrosion, radiation damage (induced by protons and high energy neutrons interactions) and thermal fatigue (induced by stress cycling due to beam trips and beam interruptions). The windowless option is less sensible to radiation damage, but its design is more complex for the presence of a free-surface flow and of Pb–Bi vapours in the beam pipe.

The proposed core is made of 120 Sub-Assemblies (SA) with Superphenix (SPX) type fuel pellets with enrichment in Pu of 23.25%. The fuel pellets have been assumed of the same geometry as for SPX. A mock-up of the fuel SA is under construction to perform hydraulic tests for pressure loss measurements in water and operability tests in air of the fuel foot locking system.

The secondary circuit is a closed loop that, in normal operation, dissipates the generated heat from the reactor to the atmosphere. It is made up of four Intermediate Heat Exchangers (IHX), arranged in parallel, and of six Air-fin Heat Exchangers (AHX, re-useable from the ENEA-PEC reactor), arranged in series, two circulation pumps in parallel and of the interconnecting piping. The secondary coolant is a low vapour pressure diathermic fluid, that is fully compatible with the thermal cycle (280–320^{o}C) and ensures, in case of leak, no fast chemical reaction with lead–bismuth or air.

5. THE CIRCE FACILITY

ENEA has started to design and build CIRCE (Italian acronym standing for eutectic circulation: “CIRColazione Eutettico”) a pool type facility with internal LBE circulation, for separate-effects and integral system tests, to be located at Brasimone (Italy).

CIRCE features a full-length, reduced-diameter mock-up of the Demonstration Facility primary vessel filled with 100 t of molten LBE. The 1.1 MW thermal power, supplied to the primary loop by electrical heaters, is removed by an intermediate heat exchanger looped on a circuit filled with organic diathermic fluid and with an air cooler. Test vessel size is 1.2 m diameter per 9 m height and the maximum operating temperature is 500^{o}C.

The facility is designed for accommodating devices for testing such different topics as:

- Oxides control and integral LBE eutectic purification systems;
- Natural and enhanced (by gas injection system) circulation through dummy core elements;
- Hydraulics of the window and windowless target configurations;
- Intermediate heat exchanger performance;
- Hydraulic behaviour of the integral system at normal and transient conditions;
- Actual fuel assembly pressure loss, bypass flow and heat transfer;
- Instrumentation in aggressive and opaque medium;
- Kinematics links of the fuel handling machine in cover gas and in the melt;
- System inspection technology.

The construction of the CIRCE facility should be completed by the end of 2001. A sketch of the facility is shown in Fig. 3.

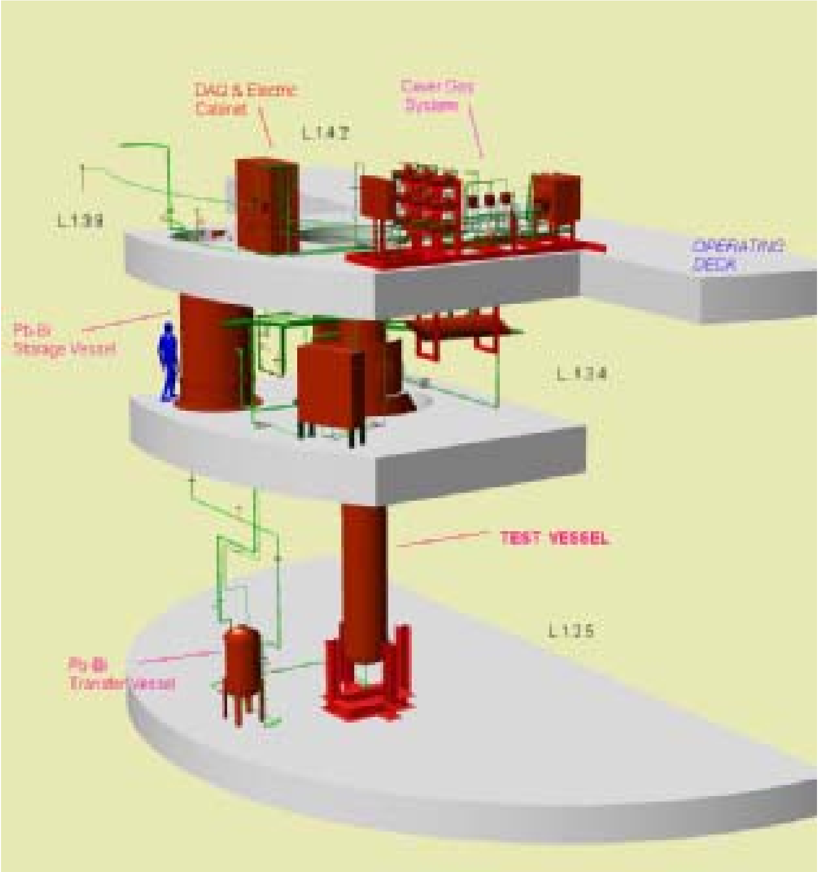

Fig. 3. Sketch Of The Circe Facility In Brasimone.

REFERENCES

[1] RUBBIA, C., et al., Conceptual Design for Fast Neutron Operated High Power Energy Amplifier, CERN Report, CERN/AT/95-44 (ET), Geneva, Switzerland (1995).

[2] ENTE PER LE NUOVE TECHNOLOGIE, The European Working group on ADS, A European Roadmap for developing Accelerator Driven Systems (ADS) for Nuclear Waste Incineration, ENEA, Rome, Italy, ISBN 88-8286-008-6 (2001).

[3] SALVATORES, M., et al., The MEGAPIE initiative, PSI Bericht N. 00-05, ISSN 1019-0643 (2000).

[4] http://www.sckcen.be/research/reactorsafety/fuel/myrrha/myrrha_home.html

[5] Energy Amplifier Demonstration Facility Reference Configuration, Summary Report, Ansaldo Nucleare, EA B0.00 1 200 (1999).

THE INP (KAZAKHSTAN) ACTIVITY IN THE FIELD OF ACCELERATOR-DRIVEN SUBCRITICAL REACTORS

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Abstract

Experimental works have been performed to provide information data on radiation resistant properties of structural materials, fission cross sections, and angular distributions of fragments, all of which can be used for development and design of ADS. New results have been obtained on alterations in structure, short-time mechanical properties, and swelling for structural materials irradiated in the fast reactor BN-350. It was found that the main contribution in nuclide production in a thick spallation target of ADS is conditioned by the evaporation residues and the fission of target nuclei in the reactions with secondary nucleons with energies less than those of primary protons. In connection with this experimental work, a new method for calculation of fragment mass distributions from fission of preactinide compound nuclei at excitation energy up to 70 MeV has been developed. A modal approach has been applied to obtain physically adequate description of these nuclei fission cross sections and angular distributions of fragments in a wide range of incident nucleon energies.

1. INTRODUCTION

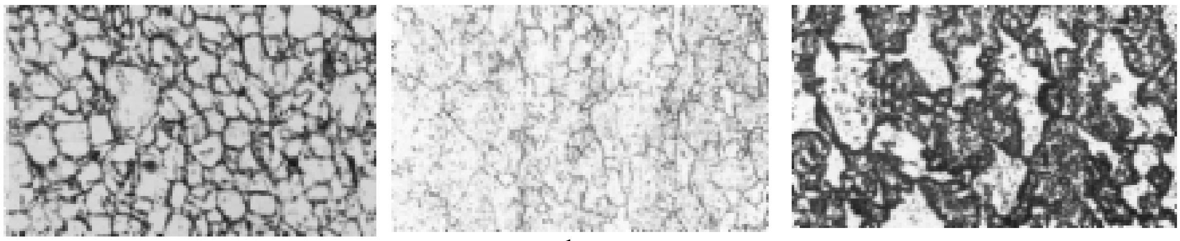
For 40 years the Institute of Nuclear Physics (INP) in Almaty, Republic of Kazakhstan, has dealt with basic researches on nuclear physics and irradiated material science. In past years extensive experimental information allowed to solve a number of important scientific questions has been accumulated. In our opinion, a part of the experimental data and knowledge acquired at realization of the basic researches can be used for solving the problems arising at development and design of the Accelerator Driven subcritical System (ADS). In particular, practical realization of the ADS will inevitably need new information on radiation resistant properties of materials and fission cross sections, mass, charge, energy distributions of fission fragments in a wide range of incident particle energies and nucleon composition of fissile nuclei.

2. IRRADIATED MATERIAL SCIENCE

At the development and design of the ADS one should overcome such material science difficulties as radiation embrittlement, swelling, creep, and corrosion resistance of structural materials. A promising opportunity on this way could be obtained from the analysis of radiation resistance of the same structural materials operated under severe radiation conditions (damage dose up to 83 dpa at irradiation temperature ~ 300°C) in fast breeder reactors, for instance, BN-350 (Aktau, Kazakhstan). In this report we present some experimental data on alterations of microstructure, mechanical properties, corrosion characteristics and swelling of stainless steels of austenitic and ferritic-martensitic types, which are basic structural materials of shrouds used in this reactor. It is known that one of the most important features of fast reactor with a sodium cooled BN-350 is very low inlet coolant temperature. At the inlet of the reactor core the temperature is equal to 280°C For comparison, in the reactor of the similar class EBR-II (USA) this temperature is close to 370°C. It should also be noted that the structural materials exploited in reactor BN-350 were up-to-date the most promising stainless steels of three generations, including the well-performed ferritic-martensitic steel EP-450 (1Cr13Mo2NbVB). Due to these circumstances all scientific results in material science investigation carried out with the materials of reactor BN-350 become unique. Especially, it is related to the experimental data obtained at studying such practically important effects as radiation swelling, creep, hardening and embrittlement.

By now in the INP the preliminary studies have been carried out with the materials of BN-350 fuel assembly shrouds made of stainless steels: 12Cr18Ni10Ti, 08Cr16Ni11Mo3, 1Cr13Mo2NbVB. The structure of these steels is presented in Fig. 1. Several data about the studied spent fuel assemblies are presented in the Table I. The maximal damage doze was ~84 dpa.

To investigate the features of alteration of strength and ductility characteristics for steels the INP has facilities provided for necessary radiation safety that allows to deal with high radioactive materials (see Fig. 2).



a b c

FIG. 1. Microstructure of stainless steels used as materials for fuel assembly shrouds
a – 12Cr18Ni10Ti; b – 08Cr16Ni11Mo3; c – 1Cr13Mo2NbVB (× 500).

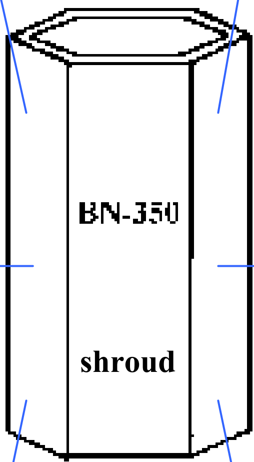
TABLE I. IRRADIATED STEELS AVAILABLE FROM BN-350

Steel composition	Specific component	Storage condition	Operating temperature (^{o}C)	Maximum dose (DPA)	Specimen configuration*
1Cr13Mo2NbVB	Assembly duct	Water	290–465	50	Rectangular plates (2-mm think)
1Cr13Mo2NbVB	Same	Same	290–465	57	Same
Cr18Ni10Ti+0.1%Sc	Same	Same	290–465	44	Same
08Cr16Ni11Mo3	Same	Same	290–465	Unknown	Same
05Cr12Ni2Mo	Same	Same	290–465	84	Same
12Cr18Ni10Ti	Same	Same	280–420	20	Same
12Cr18Ni10Ti	Sb–Be source	Same	280–465	25	Same
Cr16Ni15Mo3Nb	Cladding	Argon atmosphere	280–465	50	Dumb-bell shape 3-mm diameter 30-mm long
Cr18Ni10Ti +0.1%Sc	Welding material	Same	280–465	1	Same
08Cr16Ni11Mo3 +0.1%Sc	Welding material	Same	280–465	1	Same
Cr18Ni9	Reactor vessel	Same	280–400	0.2–0.5	Same
12Cr18Ni10Ti	Same	Same	280–420	20	Same

* Rectangular plates machined from duct in hot cell; dumb–bell specimens pre–machined before irradiation

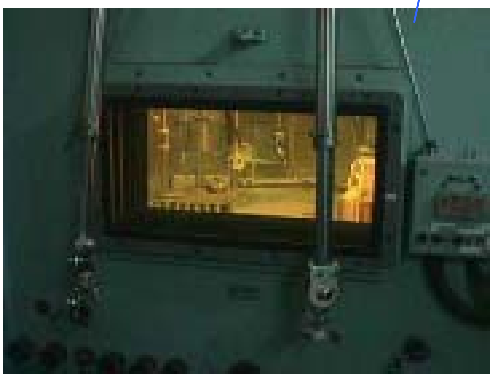
Universal testing machine "Instron-1195", provided with optical-electron extensometer

Calvet-calorimeter combined with the micro-destruction machine



Shear punch facility arranged at base of testing machine FP-100/1

Electron microscope JEM-100CX



Testing machine and device for the cutting of shroud located in the "hot" cell

Optical microscope Neophot 2

FIG. 2. Test facility for fuel assembly shrouds.

These facilities are:

- Universal testing machine “Instron-1195” supplied with optical-electron extensometer (for determining true characteristics of strength and ductility) and device for magnetization map-recording in the process of deformation of austenitic stainless steels;
- Testing machine and device for the cutting of shroud located in the “hot” cell;
- Calvet-calorimeter combined with the micro-destruction machine (for the investigation of thermal effects in the deformation and destruction processes as well as for the determination of latent energy);
- Shear punch facility arranged at base of FP-100/1 machine (for the mechanical tests of the high radioactive micro-samples punching from TEM-objects);
- Equipment for micro-hardness measurement;
- Complex of in-reactor facilities for studying radiation creep;
- Electron (TEM, REM) and optical microscopes (for studying microstructure of irradiated materials).

2.1. Radiation hardening and embrittlement

For austenitic steels, the analysis of the results obtained in the mechanical tests revealed the strong decrease of short-term characteristics of strength and ductility with growth of swelling, which can lead to fracture of fuel assemblies, for instance, during transportation (see Fig. 3).

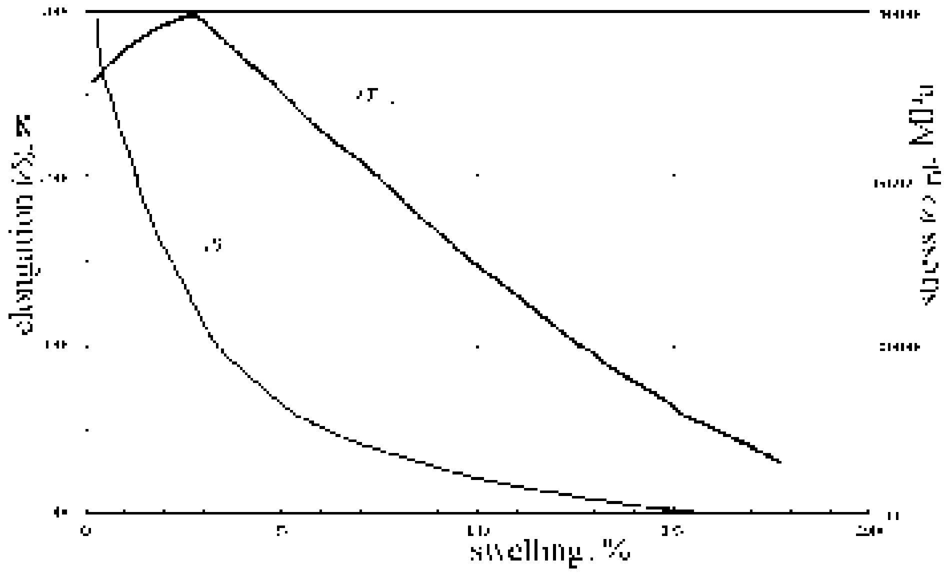


FIG. 3. The dependency of short-term mechanical properties for the steel 08Cr18Ni11Mo3MTO (the shrouds of the second type) against swelling.

2.2. Swelling

To evaluate the swelling level in structural materials of shroud tubes of the spent fuel assemblies, such procedures as measuring a “width across sides”, hydrostatic weighing and TEM-studies were used.

Figure 4 gives the profilograms of fuel assembly shrouds for 12Cr18Ni10Ti and 08Cr16Ni11Mo3 (with the damage dose up to 45 dpa) obtained before wet storage. From these profilograms one can conclude that swelling of austenitic chromium-nickel steels of types 16-11 and 18-10 does not exceed 2.5%.

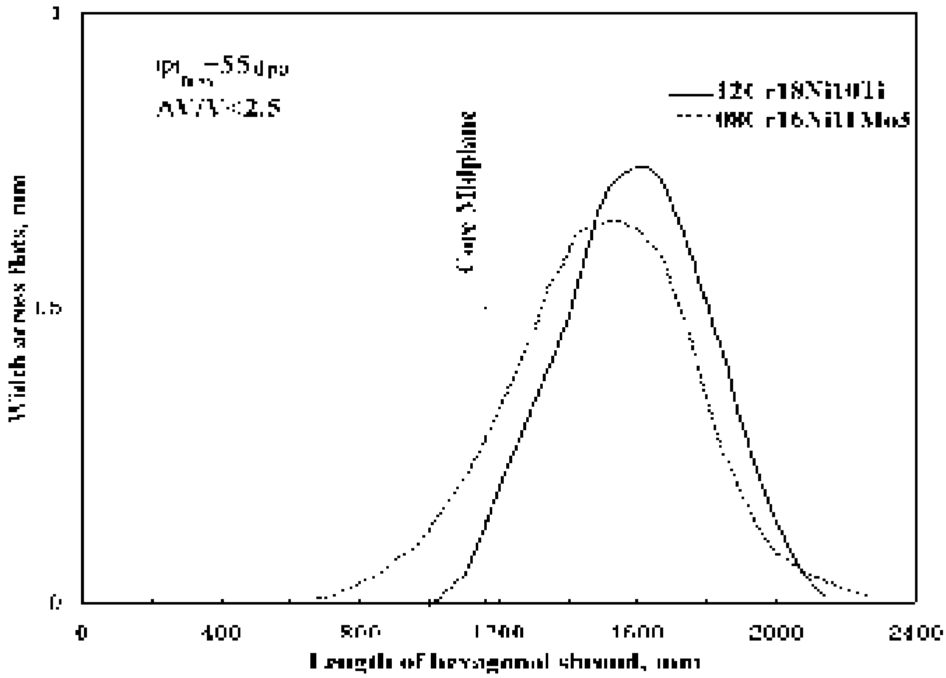


FIG. 4. Forming of fuel assembly shrouds of the reactor BN-350 for the first and second types of loading.

At the same time, if the same materials were stored in a water pool for a long time, the samples cut out from experimental hexagonal shrouds at measuring by means of the hydrostatic weighing demonstrated the density alteration ranging from 6 to 10%.

Pores were found in the steel EP-450 at the label "-375" by means of the transmission electron microscopy (Fig. 5).

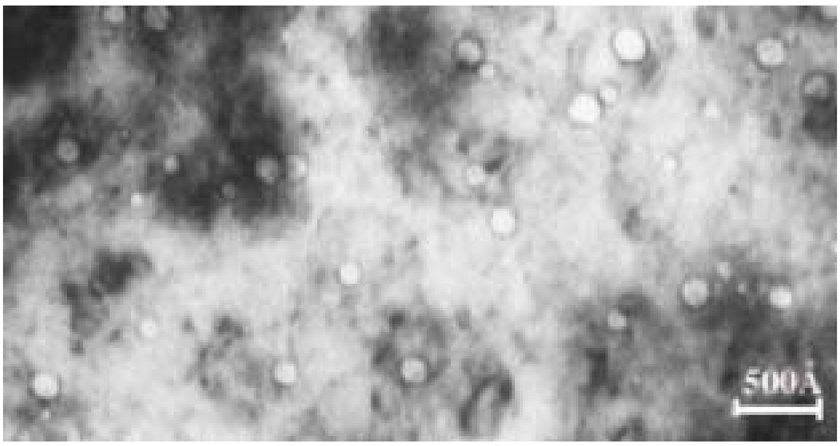


FIG. 5. Pores in steel EP-450 irradiated in the reactor BN-350 (label-375, T_{irrad} ~623 K, ~83 dpa).

2.3. Corrosion

To characterize a corrosion state in stainless steels after irradiation and long-term wet storage the spent fuel assemblies, which were kept under the worst conditions of irradiation (50 - 70 dpa) and following wet storage (for 5–20 years), were selected (see Table I). The typical examples of corrosion effects in steels are shown in Fig. 6.

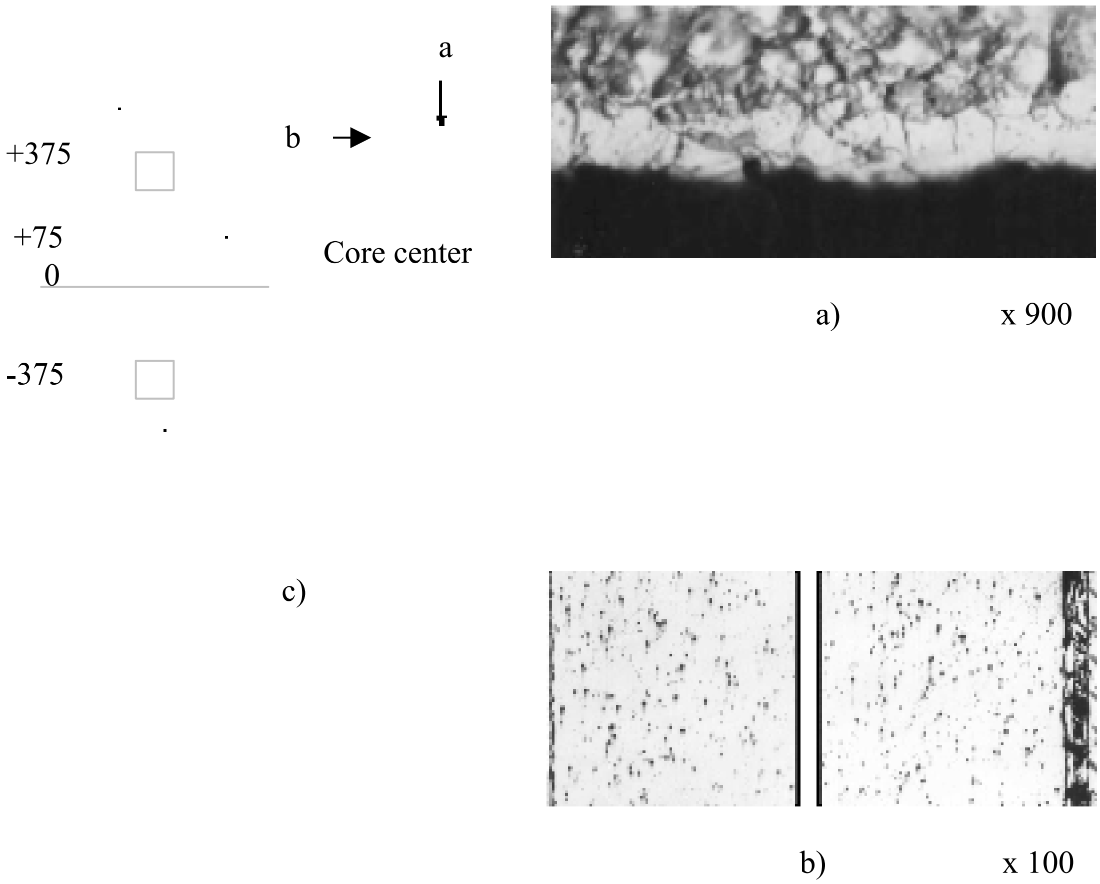


FIG. 6. Corrosion in stainless steels: a) steel 12Cr18Ni10Ti, 0.5% burnup, storage in water for 6 years at 293 K (an appearance along the arrow a; b) steel 1Cr13Mo2NbVB, 7.4% burnup, storage in water for 7 years at 293 K (an appearance along the arrow b); c) a shroud of a fuel assembly of the reactor BN-350.

It was found that the widths of a corrosion layer on inner and outer sides of a hexagonal shroud from steel EP-450 are different. For the inner side, which was intensively passed overby sodium during operation and stored in slow water, the width of the oxidized film is essentially smaller than that of the outer side equal to 50 µm. It should be noted that some assemblies from austenitic steel 08Cr16Ni11Mo3 kept in water after irradiation for a long time have destroyed at attempt to take them out. The fracture has been observed near to a weld. One of the supposed causes for the fracture is deeply penetrated corrosion. Figure 7 shows the SEM-image of the metallic shroud fracture surface. One can see, in particular, that on one side of the shroud the corrosion layer is wider than on the other.

3. NUCLEAR FISSION

3.1. Fission of pre-actinide nuclei

In the ADS of various purpose the reactions of light charged particles with heavy nuclei at kinetic energies of collisions ~1 GeV are supposed to be used for creation of powerful primary sources of fast neutrons with energies up to 150 MeV.

At such energies (Fig. 8) a collision of particles and heavy nuclei through intra-nuclear nucleon-nucleon collisions accompanied by emission of secondary particles leads to formation of pre-fragments with a wide spectra of excitation energies. Then these pre-fragments either become spallation residues by evaporating nuclides, nucleons and γ-rays or go fission. So, the production of radioactive waste in a thin spallation target is defined by both the spallation residues and fission products.

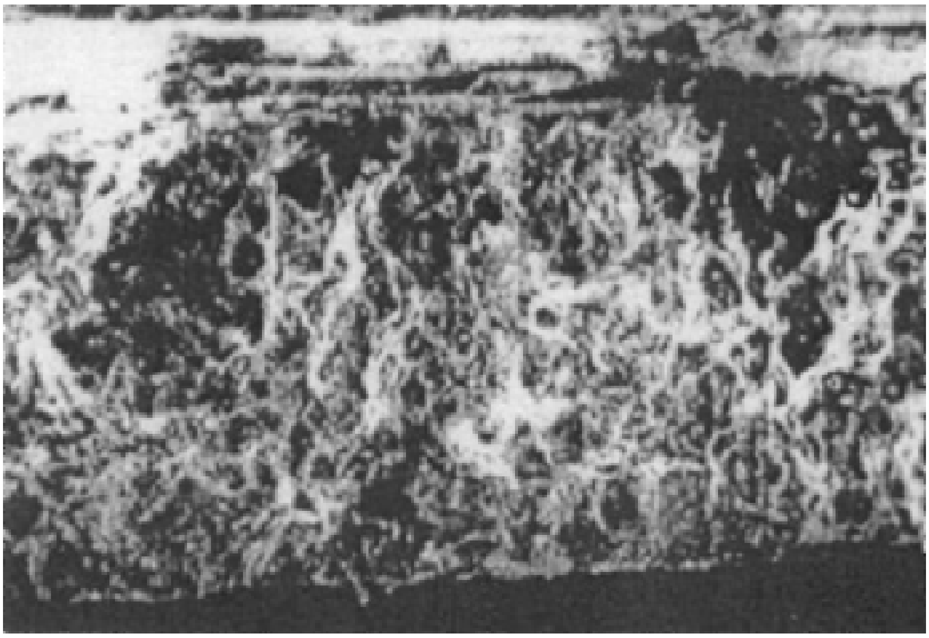


FIG. 7. The SEM-image of a fracture surface for a fuel assembly shroud from steel 08Cr16Ni11Mo3.

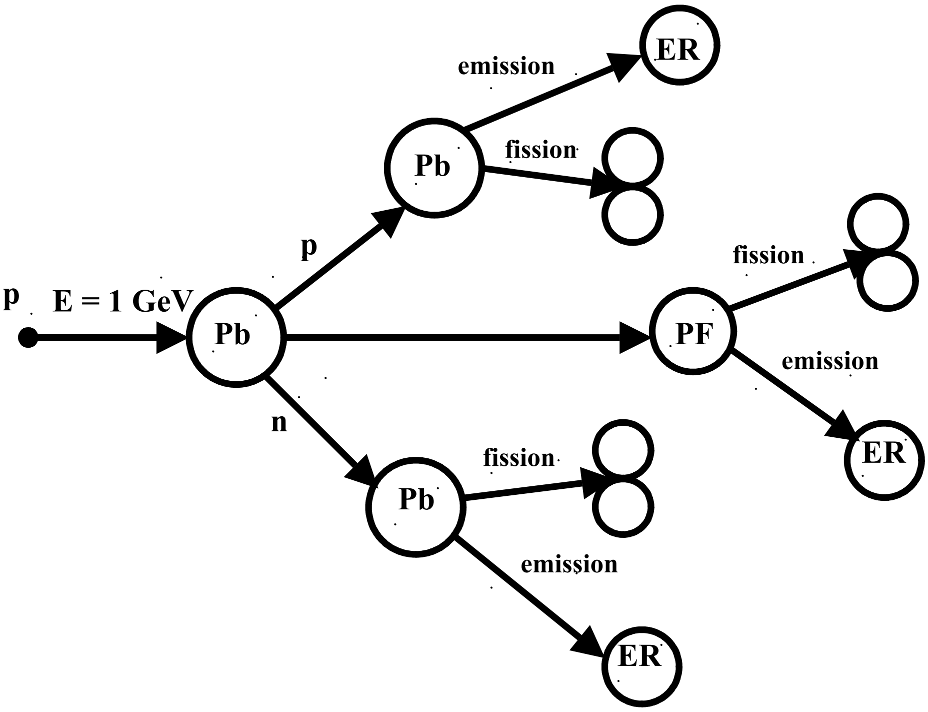


FIG. 8. Physical scenario of nuclear processes in the ADS spallation target.

In a thick target the lead nuclei and nuclei lighter than lead with excitation energies above the fission barrier will be formed not only as a result of spallation reactions on the lead target nuclei but also in the interaction of secondary nucleons (protons and neutrons) with lead nuclei. The number of secondary particles generated in the spallation reactions of nuclei with high-energy protons exceeds the number of primary protons as a factor of 30–40, and, though only some part of them have kinetic energies enough to cause fission of lead nuclei, the contribution of reactions with secondary particles in a general production of radioactive nuclides (GPRN) in the thick spallation target of the ADS will be predominant. Besides, the strong competition of fission process with the particle emission (first of all, neutron emission) at high initial excitation energies leads to that in a reality a nucleus passes over the fission barrier at rather small excitation energies (<100 MeV). Under this circumstances, in order to calculate the GPRN one should involve the information on fission cross sections and fragment yields from reaction of pre-actinides with particles having comparatively low kinetic energies.

During last 30 years unique experimental data on fission cross sections, angular, mass and energy distributions of fragments in the fission of pre-actinides from Yb to At at excitation energies up to 70 MeV have been accumulated in the INP.

The fission probabilities extracted from this data [1] could be used either directly in semi-empirical calculations or for testing the parts of existing models. For instance, Bertini-Dresner [2], Cugnon-Schmidt [4], etc., are responsible for the calculations of the competition between evaporation channel of the pre-fragment de-excitation and fission.

Recently the analysis of the accumulated experimental information on fragment mass distributions in the fission of pre-actinides allowed us to develop a semiempirical model for calculating the distributions based on the statistical approach to the description of the level

density with phenomenological taking account the shell effects and effects of pair correlation [6]. The results of fragment mass distribution calculations within the framework of this model are presented in Figs 9 and 10. Figure 9, as an example, shows the calculation results and experimental mass yields Y(m) for ^{201}Tl in the dependence on the excitation energy above the fission barrier U. One can see that the model reproduced both the Gaussian-like shape experimental data at higher U, where the influence of shell effects could be neglected, and the strong deviations from the Gaussian shape interpreted as the manifestation of two deformed neutron shells formed in the fragments with $N \approx 52$ and 68. The model also succeeded to reproduce sharp alterations of mass distribution shape at low U caused by the shells mentioned above in the dependence on nucleon composition of fissioning pre-actinides from ^{187}Ir to ^{213}At as performed in Fig. 10.

3.2. Fission of actinide nuclei

The fission of actinide nuclei is also an objective of the investigations carried out in the INP. The practical need for precise information on fission cross sections and methods for calculating mass, charge and energy distributions of fission fragments of these nuclei in a wide range of excitation energies for the ADS, transmutation of minor actinides, etc. is rather great.

The development of the calculation methods, which allow to describe the fission fragments yields of fissile nuclei, could be accomplished taking account of systematics based on a wide totality of experimental data and semi-empirical models which are also based on the experimental data, but additionally taking into account the main physical factors influenced on the formation of these yields. One of the main property of fission fragment yields for these nuclei is their multimodality. This property consists in that the experimentally observed yields manifest itself as a superposition of yields of several independent fission modes. For the fission of actinide nuclei four fission modes are characteristic:

– A symmetric mode (S) is characterized by strongly elongated shapes of a fissioning nucleus in the vicinity of the scission (rupture) point;

– An asymmetric mode (standard 1 or S1) is due to a close-to-sphere shell formed in heavy fragments with average mass M ≈ 132–134;
– An asymmetric mode (standard 2 or S2) is conditioned by shell effects corresponding to comparatively small deformations of heavy fragments with average mass M ≈ 138–140;
– An asymmetric mode (standard 3 or S3) is due to a spherical neutron shell N = 50 formed in light fragments.

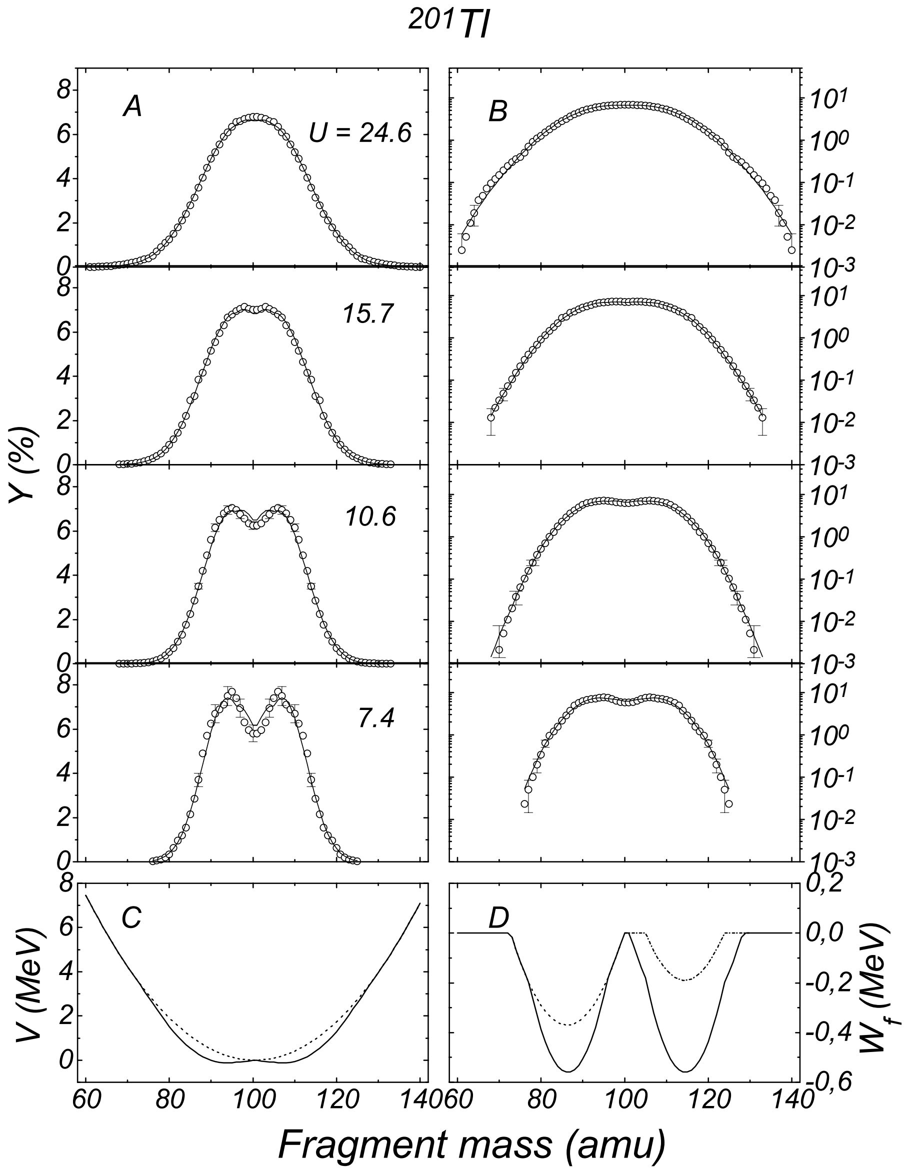


FIG. 9. Relative yields Y, their description, calculated potential energy and its constituents versus fragment mass for ^{201}Tl at different excitation energies. Part A: (o) the experimental yields Y and (-) the proposed description in a linear scale; Part B: (o) the experimental yields Y and (-) the proposed description in a logarithmic Scale; Part C: full (-) and liquid-drop (...) potential energies; Part D: calculated shell corrections in the light fragment group W_L (...), in the heavy fragment group W_H (- . -), and for the whole nucleus W_f (-).

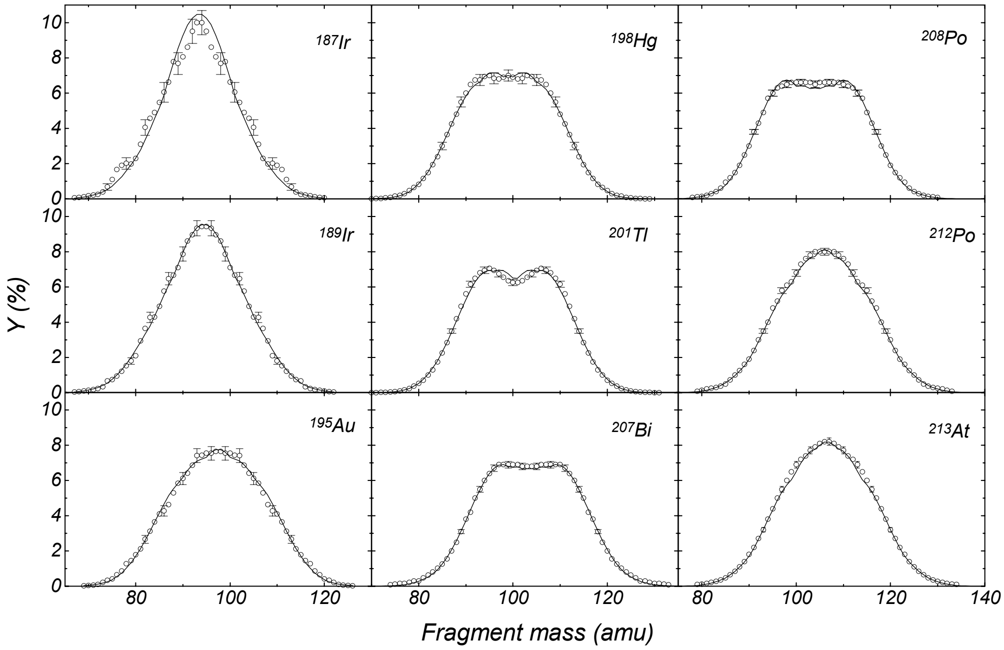


FIG. 10. Relative yields Y and their description versus fragment mass for nuclei from ^{187}Ir to ^{213}At at an excitation energy U ≈ 10 MeV. The experimental data (o) and the proposed description (-) are shown.

According to our expectations, the basic characteristics of a distinct mode have more simple dependencies on nucleon composition and excitation energy of a fissioning nucleus than those of the independent modes superposition observed in an experiment. Thus, in our opinion, one of the most perspective directions towards the development of the semi-empirical models for describing the yields mentioned above is an approach based on a systematic study of the basic characteristics of independent fission modes, including the data on reactions with light charged particles.

One should note that in the fission of actinide nuclei the characteristics of distinct fission modes are determined on base of the multi-component analysis. Recently we have developed a new method for analyzing the experimental fission fragment mass and energy yields that has some important advantages in the comparison with the analogous ones [7]. In contradiction to the widespread analytical multi-component methods using the Gaussian distributions for parameterization of distinct fission mode mass distribution (MD), the new method is free from any parameterization of the MD of independent mode and, therefore, allows both to investigate the MD shape for every fission mode and to avoid biased estimations of basic characteristics of modes conditioned by rough parameterization of their MDs.

The results of this method application to the analysis of the MEDs from fission of nuclei from low energy fission of ^{226}Th to ^{245}Bk are partly presented in Fig. 11. In this figure, the experimental relative mass yields Y(M) and the MDs for independent fission modes are presented.

The basic characteristics of modes are partly shown in Fig. 12. In part A, the average masses of heavy-fragment group of experimental distributions and average masses of heavy fragments for modes S1 and S2 (mode S3 with relative yield ≤ 1% and mode S with average M = A/2 according to definition are missing) from the proton induced fission of target nuclei ^{232}Th are performed in the dependence on the proton energy. One can see that the dependencies of modal average masses are more simple than that of average masses for experimental mass yields. This confirms our preliminary expectations that the modal approach to describing the MEDs from fission of actinide nuclei is very perspective from the viewpoint of developing a semi-empirical model of the MEDs applicable to this practically important region.

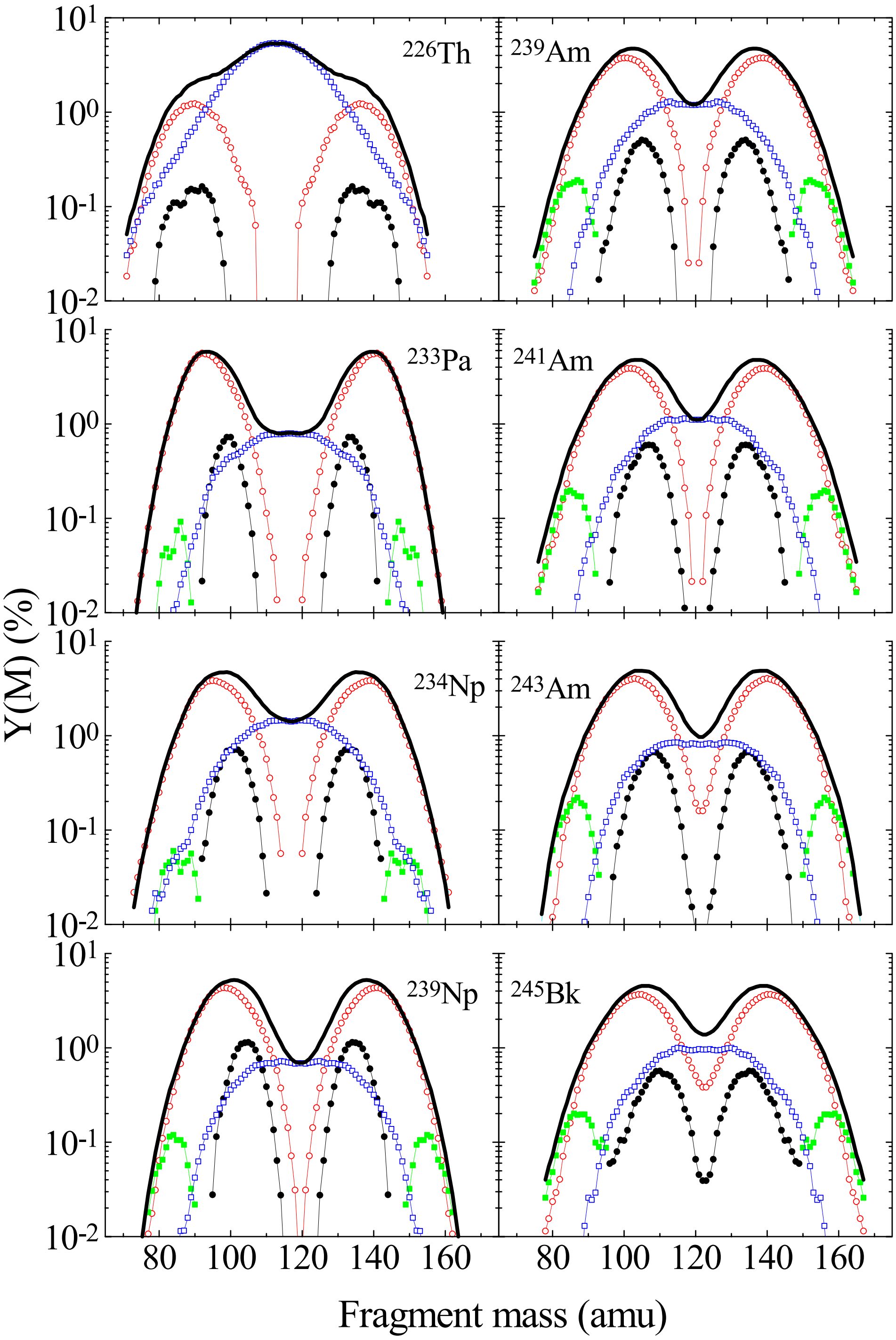


FIG. 11. Mass distributions and results of their decomposition in the fission of nuclei from ^{233}Pa to ^{245}Bk.

(–) – the experiment, (•) – S1; ( ) – S2; (■) – S3; ( ) – S

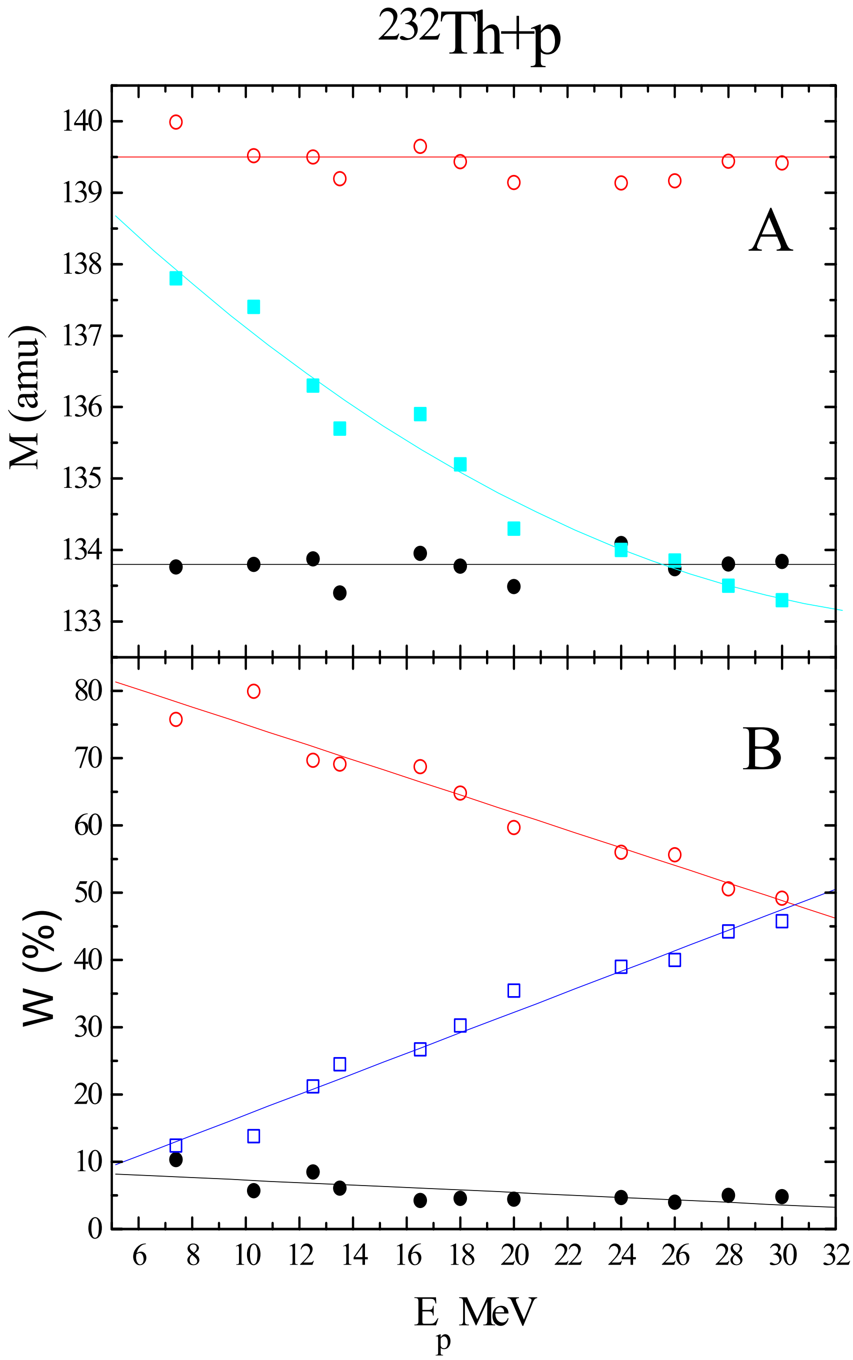


FIG. 12. The experimental and modal averaged masses M of heavy fragments (Part A) and contributions W of independent modes (Part B) in the dependence on incident proton energy E_p from the fission of target nucleus ^{232}Th.

(■) – the experimental average masses, (•) – S1; ( ) – S2; ( ) – S

In part B of Fig. 12, the relative contributions W of modes in experimental mass yields in the dependence on incident proton energy for the target-nucleus ^{232}Th are shown. One can see that the contributions W of asymmetric modes S1and S2 conditioned by shell effects decrease with growth of proton energy, and this value for mode S increases. Therefore, at low energies of incident particles the fission cross sections for actinides are defined by the fission through asymmetric modes, and at high energies the cross sections are defined by the fission through mode S. Different modes are characteristics of different fission barriers and spectra of excited states that make the fission cross section and angular distribution of fission fragments: it means nobody will succeed in giving physically adequate description of these practically important quantities without taking into account the fact that actinide nuclei go fission through independent modes.

4. SUMMARY

New results have been obtained on alterations in structure, short-time mechanical properties, corrosion depth, and swelling for structural materials irradiated in the fast breeder reactor BN-350.

The main contribution to nuclide production in a thick spallation target of the ADS will be conditioned by the evaporation residues and the fission of target nuclei in the reactions with secondary nucleons with energies less than those of primary protons. In this connection, the calculations of fission product and evaporation residues yields in the ADS spallation target should involve the information on the reactions of pre-actinides with particles measured at low and intermediate incident particle energies.

The new method for calculation of fragment mass distributions from fission of pre-actinide compound nuclei at excitation energies up to 70 MeV has been developed.

The modal approach to studying the fission of actinide nuclei gives an opportunity for development of semi-empirical models for calculations of actinide nuclei fission fragment mass and energy distributions applicable to subcritical blankets of the ADS, fast reactors and transmutation of minor actinides. This approach is the only way to obtain physically adequate description of the nuclei fission cross sections and angular distributions of fragments in a wide range of incident nucleons energies.

REFERENCES

[1] IGNATYUK, A.V., et al., Fiz. Elem. Chastits At. Yadra **16** (1985) 709 (in Russian).
[2] BERTINI, H.W., Phys. Rev. **188** (1969) 1711.
[3] DRESNER, L., Report, ORNL-TM-196, Oak Ridge National Laboratory (1962).
[4] CUGNON, J., VOLANT, C., VUILLIER S., Nucl. Phys. **A620** (1997) 457.
[5] JUNGHANS, A.R., et al., Nucl. Phys. **A629** (1998) 635.
[6] MULGIN, S.I., et al., Nucl. Phys. **A640** (1998) 375.
[7] MULGIN, S.I., et al., Phys. Lett. **B462** (1999) 29.

MYRRHA: A MULTIPURPOSE ACCELERATOR DRIVEN SYSTEM FOR RESEARCH AND DEVELOPMENT

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Abstract

SCK•CEN, the Belgian Nuclear Research Centre, and IBA s.a., Ion Beam Application, a world leader in accelerator technology, are designing an ADS prototype, the MYRRHA Project. The partners are foreseeing MYRRHA as a first step towards the European ADS-Demo facility. The MYRRHA system should become a new major research infrastructure for the European partners presently involved in the ADS Demo development. The MYRRHA concept, as it is today, is based on the coupling of an upgraded commercial proton accelerator with a spallation target surrounded by a subcritical neutron-multiplying medium. A cyclotron, based on positive ion acceleration technology brings the protons up to an energy level of 350 MeV. The nominal current is 5 mA of protons. The spallation target system is a circuit with, at the upper part, a free surface in contact with the incoming proton beam. This windowless concept distinguishes MYRRHA from other ADS designs. The subcritical assembly consists of fuel rods of fast breeder type, with Pu contents between 20 and 30%, in a Pb-Bi environment. This zone is very suitable for material testing and transmutation studies due to the high fast flux attainable. The desired performances in the fast core in terms of the fast flux are 10^{15} n/cm²·s, for a total power output of 25 MW. The reactor presents itself as a single vessel of about 4 m diameter and 5 m high. The primary heat exchangers, the primary pumps, the spallation loop are also vertically inserted in cover penetrations, at the vessel periphery, very much like it is done in fast reactors. Smaller penetrations are foreseen at different radial positions, in order to provide irradiation positions. This paper will present the current status of the MYRRHA project, the problems identified so far in the design. It will describe the hardware (pumps, heat exchangers, spallation loop, handling tools) as they are now foreseen, and list the cooperations which will be needed in those developments.

1. INTRODUCTION

SCK•CEN, the Belgian Nuclear Research Centre, and IBA s.a., Ion Beam Applications are developing jointly the MYRRHA project, a multipurpose neutron source for R&D applications on the basis of an Accelerator Driven System (ADS). Both partners are intending to fit the MYRRHA project into the European strategy towards the ADS Demo facility for waste transmutation.

The R&D applications that are considered in the MYRRHA project can be grouped in three blocs: (i) continuation, and later on extension towards ADS, of the ongoing R&D programmes at SCK•CEN in the field of reactor materials, fuel and reactor physics research; (ii) enhancement and triggering of new R&D activities such as waste transmutation, ADS technology, liquid metal embrittlement; (iii) initiation of new competencies such as medical applications (proton therapy, PET production, etc.).

The MYRRHA concept, as it is today, is based on the coupling of an upgraded commercial proton accelerator with a spallation target surrounded by a subcritical neutron-multiplying medium. Its design is determined by the versatility of the applications it would allow. Further technical and/or strategic developments of the project might change the concept.

The design of MYRRHA needs to satisfy a number of specifications such as:

- The achievement of the neutron flux levels required by the different applications considered in MYRRHA: $\Phi_{>0.75\ MeV} = 10^{15}$ n/cm²·s at the locations for minor actinides (MA) transmutation, $\Phi_{>1\ MeV} = 10^{13}$ to 10^{14} n/cm²·s at the locations for structural material and fuel irradiation, $\Phi_{th} = 2 \times 10^{15} \sim 3 \times 10^{15}$ n/cm²·s at locations for long-lived fission products (LLFP) transmutation or radioisotope production;
- The subcritical core total power: ranging between 20 and 30 MW;
- The safety: $k_{eff} \leq 0.95$ in all conditions as in a fuel storage to guarantee its inherent safety;
- The operation of the fuel in safe conditions: average fuel pin linear power < 500 W/cm.

2. MYRRHA PRESENT DESIGN STATUS

In its present status of development, the MYRRHA project [1] is based on the coupling of an upgraded commercial proton accelerator with a liquid Pb–Bi windowless spallation target, surrounded by a subcritical neutron multiplying medium in a pool type configuration (see Fig. 1).

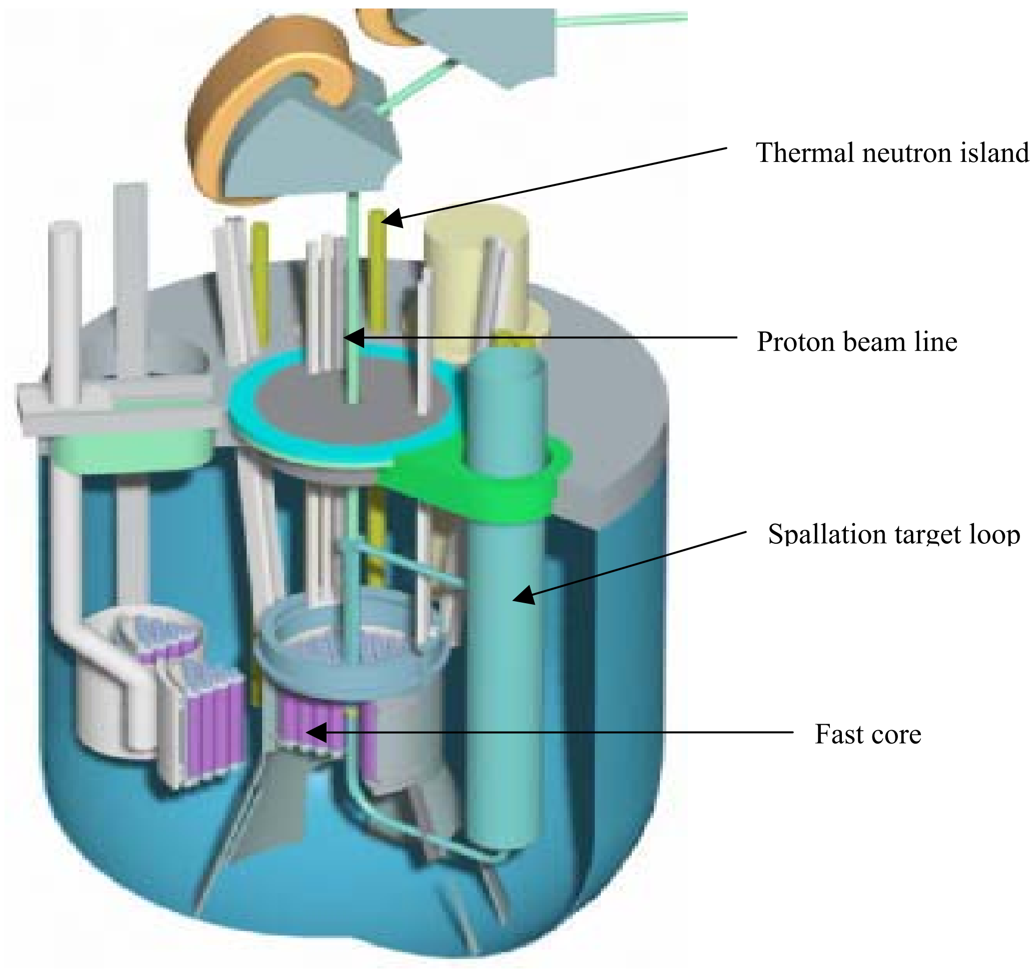

FIG. 1. Global view of a possible design for MYRRHA.

The spallation target circuit is fully separated from the core coolant as a result of the windowless design presently favoured in order to utilize low energy protons without reducing drastically the core performances.

The core pool contains a fast spectrum core, cooled with liquid lead–bismuth (Pb–Bi) or Pb, and several islands housing thermal spectrum regions located in In-Pile Sections (IPS) at the periphery of the fast core. The fast core is fuelled with typical fast reactor fuel pins with an active length of 600 mm arranged in hexagonal assemblies of 122 mm plate-to-plate. The central hexagon position is left free for housing the spallation module. The core is made of the two following crowns containing in total 18 assemblies, of which 12 have a Pu content of 30% and 6 a Pu content of 20%.

The MYRRHA design is determined by the requirement of versatility in applications and the desire to use as much as possible existing technologies. The heat exchangers and the primary pump unit are embedded in the reactor pool. The accelerator is to be installed in a confinement building separated from the one housing the sub-critical core and the spallation module. The proton beam is impinging on the spallation target from the top.

2.1. Accelerator

The present design of the subcritical core requires the accelerator to deliver a 350 MeV, 5 mA proton beam. This 1.75 MW CW beam has to satisfy a number of requirements, some of which are unique in the world of accelerators up to now. At this level of power it is compulsory to obtain an extraction efficiency above 99.5% and a very high stability of the beam, but on top of that, the ADS application needs a reliability well above that of common accelerators, bringing down the beam trip frequency (trips longer than a few tenths of a second) to below 1 per day.

The design principles are based on the following lines of thought:

- Statistics show that the majority of beam trips is due to electric discharges (both from static and RF electric fields). Hence the highest reliability requires to minimize the number of electrostatic devices, which favours a single stage design;
- In order to obtain the very high extraction efficiency, 2 extraction principles are available: through a septum with well-separated turns, or by stripping;
- The beams are dominated by space charge. Therefore one needs careful transverse and longitudinal matching at injection, and avoiding cross talk between adjacent turns (by an enhanced turn separation) if a separated turn structure is required for the extraction mechanism.

The space charge dominated proton beam needs a 20 mm turn separation at 350 MeV if a septum extraction has to be implemented. This solution requires the combination of a large low field magnet and of very high RF acceleration voltages for realizing such a large turn separation, and also an electrostatic extraction device. In view of what precedes, it is considered that this solution is not well suited for very high reliability of operation. Extraction by stripping does not need separated turns. It may be obtained by the acceleration of H^- ions, but the poor stability of this ion makes it extremely sensitive to electromagnetic stripping (and hence beam loss) during acceleration. The use of H^- would therefore lead to the use of an impractically large magnetic structure. The other solution is to accelerate 2.5 mA of H_2^+ ions up to 700 MeV, where stripping transforms it into 2 protons of 350 MeV each, thus dividing the magnetic rigidity by 2 and thereby allowing to extract. This solution reduces the problems related to space charge since only half the beam current is accelerated. However, the high magnetic rigidity of a 700 MeV H_2^+ beam imposes a magnetic structure with a pole radius of almost 7 m thus leading to a total diameter of the whole cyclotron of close to 20 m. The cyclotron would consist of 4 individual magnetic sectors, each of them spanning 45 degrees.

2.2. Spallation target

The spallation target is made of liquid Pb–Bi. The liquid Pb–Bi is pumped up to a reservoir from which it descends, through an annular gap ($Ø_{outer}$ = 120 mm), to the middle of the fast core. Here, the flow is directed by a kind of nozzle into a single tube penetrating the fast core ($Ø_{outer}$ = 80 mm). At about the position of the nozzle a free liquid metal surface is formed, which will be in contact with the vacuum of the proton beam guideline. No conventional window is foreseen between the Pb–Bi free surface and the beam in order to avoid difficulties in engineering this component and to keep the energy losses at a minimum. When the Pb–Bi has left the fast core region, it is cooled and pumped back to the reservoir.

The MYRRHA windowless spallation module is given particular attention in the present pre-design phase because of its particular features, as illustrated in [2].

2.3. Subcritical system

The design of the subcritical assembly is application dependent. Indeed, it should yield the neutronic performances and provide the irradiation volumes needed for the considered applications. In order to meet the goals of material studies, fuel behaviour studies, radioisotope production, transmutation of MA and LLFP, the subcritical core of MYRRHA must include two spectral zones: a fast neutron spectrum zone and a thermal spectrum one.

2.3.1. Fast zone description

The fast core will be placed centrally in a liquid Pb–Bi or Pb pool, leaving a central hexagonal assembly empty for housing the spallation target. It consists of hexagonal assemblies of mixed oxide (MOX) fast reactor (FR)-type fuel pins with a Pu-content, Pu/(Pu + U), ranging from 20 to 30%, arranged in a triangular lattice with a pitch of 10 mm. The fuel pins have an active fuel length of 60 cm and their cladding will likely consist of 9%Cr martensitic steel. The fuel pins are arranged in typical FR fuel hexagonal assemblies with an assembly dimension of 122 mm plate-to-plate. The fast zone is made of 2 concentric crowns of respectively 6 highly enriched (30% Pu content) and 12 fuel assemblies, of which 6 are 30% enriched and 6 are 20% enriched.

Alternative core configurations are envisaged, with smaller fuel assemblies (around 65 mm plate-to-plate) in order to simplify the fuel management. In addition, this would help to reduce the bending of the fuel assembly caused by the differential swelling of the assembly case, submitted to a high radial flux gradient.

Neutronic calculations coupling the high energy transport code HETC and the lower energy neutron transport deterministic code DORT, have been carried out for simulating typical configurations of the fast core and led to encouraging results showing that the targeted performances could be achieved. Table I illustrates the preliminary results we obtained for a particular configuration with an active length of 50 cm but where the fuel assembly is not well simulated [3].

TABLE I. ACHIEVABLE PERFORMANCES IN THE MYRRHA SUBCRITICAL CORE

Spallation source parameters	E_p = 350 MeV	
	I_p = 2 mA	I_p = 5 mA
Source intensity (E < 20 MeV) (× 10^{16} n/s)	4.9	12.3
Core parameters:		
MF = 1 / (1–K)	19.15	19.15
K	0.948	0.948
Thermal power, MW	10.0	25.0
Avg. power density, W/cm^3	87	218
Peak linear power, W/cm	191	477
Max. flux > 0.75 MeV (× 10^{14} n/cm^2·s)	4.5	11.2
No. of fuel pins (MOX 30 and 15%)	2646	2646
MOX-30% zone ID, cm	12.8	12.8
MOX-15% zone ID, cm	34.2	35.2
Fast core OD, cm	55.5	55.5

2.3.2. Thermal zone description

As compared to the initial design that we were considering (with a water pool surrounding the fast core zone and housing the thermal neutron core zone), the design of this part of MYRRHA has been completely changed for evident safety reasons (water penetration into the fast zone). In the present approach the thermal zone will be kept at the fast core periphery but it will consist of various In-Pile Sections (IPS) to be inserted in the Pb–Bi liquid metal pool from the top of the reactor cover. The IPS could be a pressure tube in which water is circulated. The water would act as a moderator and as a coolant for the samples. The samples could be steel samples, LWR fuel, or transmutation specimens. The IPS concept allows to perform the irradiations at temperatures different from the operation temperature of MYRRHA. As an alternative, the IPS could contain a solid matrix made of moderating material (Be, C, $^{11}B_4C$) on which a total leakage flux of 1 × 10^{15} to 3 × 10^{15} n/cm²·s will impinge Local boosters made of fissile materials can be considered depending on the particular performance needed in the thermal neutron IPS. Black absorbers settled around the IPS could ensure the neutronic de-coupling of the thermal islands from the fast core. The design of these thermal islands is still in a preliminary phase.

Next to the spallation target, the fast core and the thermal islands, the pool will contain other components of a classical reactor such as heat exchangers, circulation pumps, fuel loading and handling machines, emergency-cooling provisions.

2.4. Hardware

2.4.1. Coolant

The design of the reactor is governed by the main requirements presented in Section 1. We opted for liquid metal as a coolant because it allows the high fast flux and the high power density requested with a relatively low power machine.

Lead-bismuth (Pb–Bi) was chosen because it is compatible with the fluid chosen for the spallation source, it does not react violently with air or water has a very high boiling point and a reasonable melting point.

Nevertheless, the fluid poses several challenges and exhibits some unusual characteristics:

– The fluid melts at 123.5°C;
– Its density is so high that steel floats;
– Its density and low vapour pressure should be favourable for the cavitation aspect. But if it occurs, cavitation might be devastating;
– Due to the fluid density, erosion might be intense at points of velocity change (deceleration, or directional change);
– The compatibility of Pb–Bi with cladding or structural materials is mostly unknown — at least in the West. Research is undertaken to determine the adequate choice;
– The ability of Pb–Bi to be used as a lubricant has to be demonstrated;
– Pb–Bi generates Polonium (Po) under irradiation. Polonium is an extremely radiotoxic α-emitter;
– There are no Western suppliers with an experience of equipment for Pb–Bi service.

Other coolants (gas or other liquid metals, water is ruled out for neutronics reasons) were also envisaged, but up to now, no one showed a definite advantage on Pb–Bi.

2.4.2. Vessel configuration

To take profit of the thermal inertia provided by a large coolant volume, we opted for a vessel-type reactor, that is a single vessel in which the components of the primary loop (pumps, heat exchangers, handling tools, etc.) are inserted in penetrations in the reactor cover (Figs 2, 3 and 4). This configuration is clearly inspired from the design of FBRs. It is considered that it will help to rule out the LOCA case in the Safety Study and to reduce the volume of zones where α-contamination would be a concern. The configuration will also limit the exposition of the vessel material to the neutrons and reduce its embrittlement, as compared to a smaller vessel. Table II shows general characteristics obtained for the vessel configuration.

2.4.3. Vessel cover

The reactor cover is a thick slab with a great number of penetrations. The slab could possibly be composite, that is an arrangement of layers of steel, concrete, water, polyethylene or other neutron absorbing materials. The thickness has still to be determined but is anticipated to be between 1 and 2 m.

There are a few large penetrations for the main equipment, such as the primary pumps, the heat exchangers, the spallation loop or the handling tools. There are numerous smaller penetrations in which “In-Pile Sections” or IPS can be inserted.

The cover provides the shielding against the neutrons escaping from the core through the Pb-Bi pool. Additional shielding must be provided in the area of the particles beam tube and its bending magnets, where there is a direct escape path for neutrons and possible activation of the upper core structures. Of course, each device inserted in the cover must be designed to fill the gap it makes in the shielding.

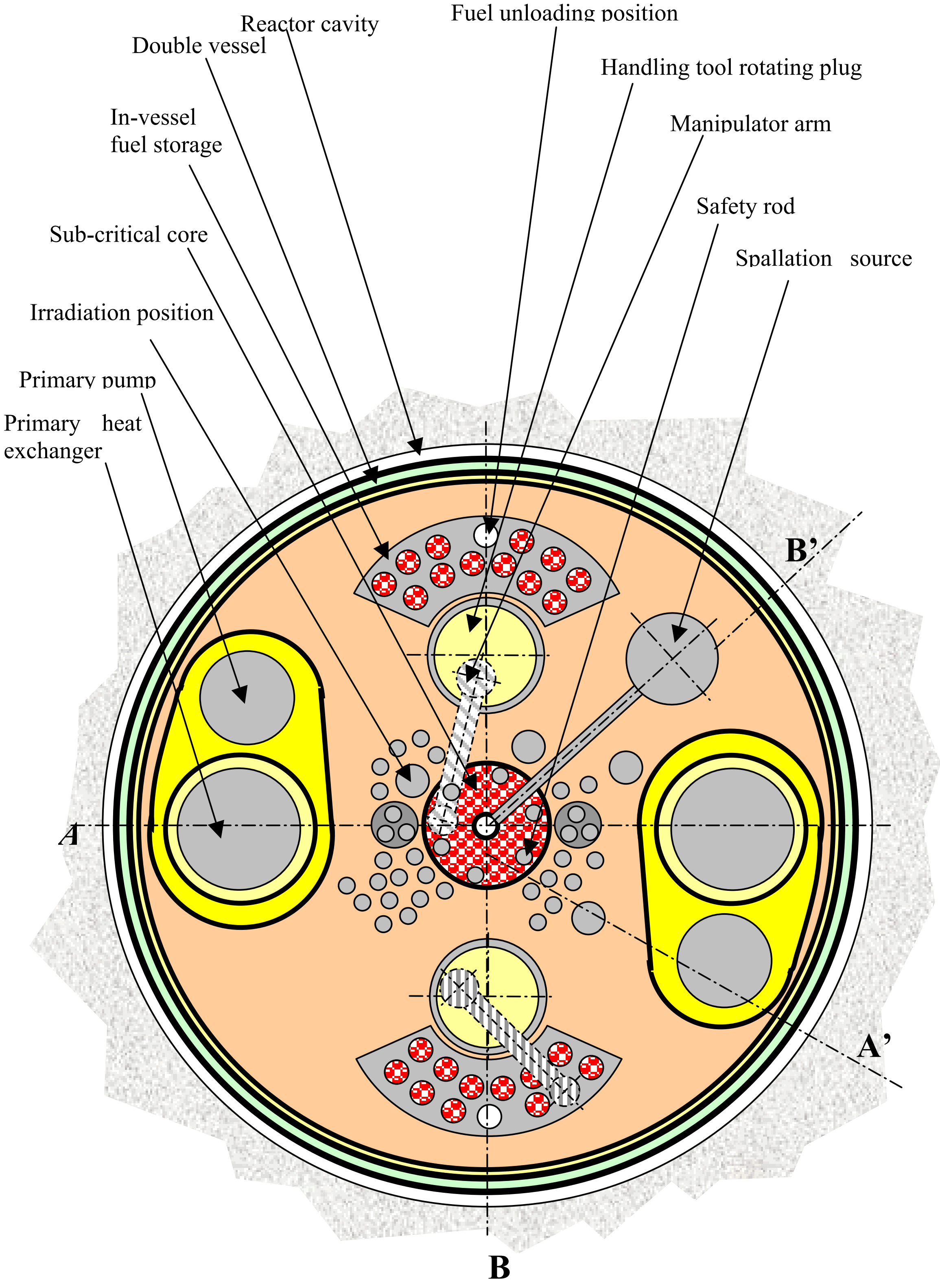

FIG. 2. MYRRHA vessel configuration.

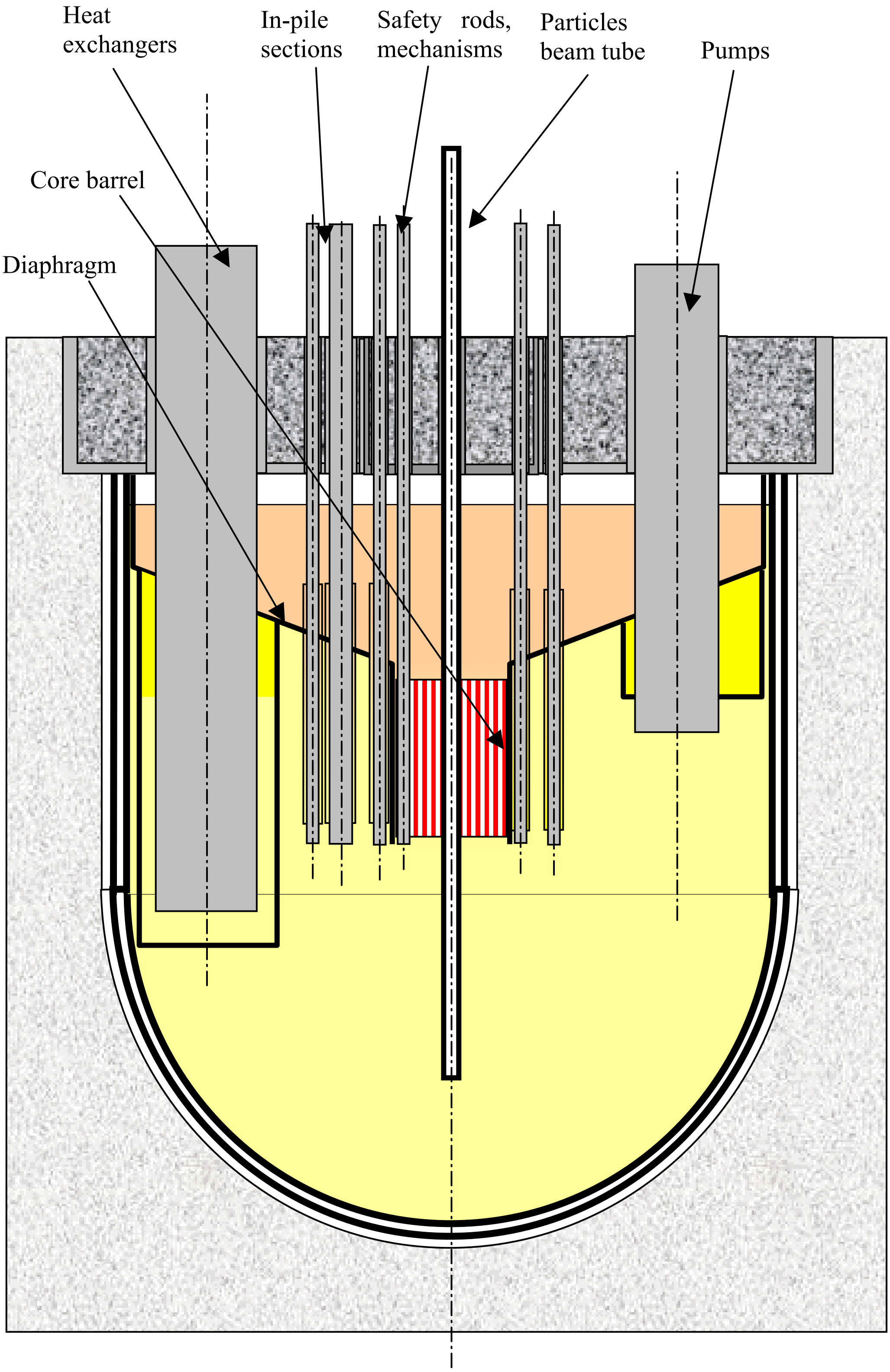

FIG. 3. MYRRHA vessel configuration - AA' view.

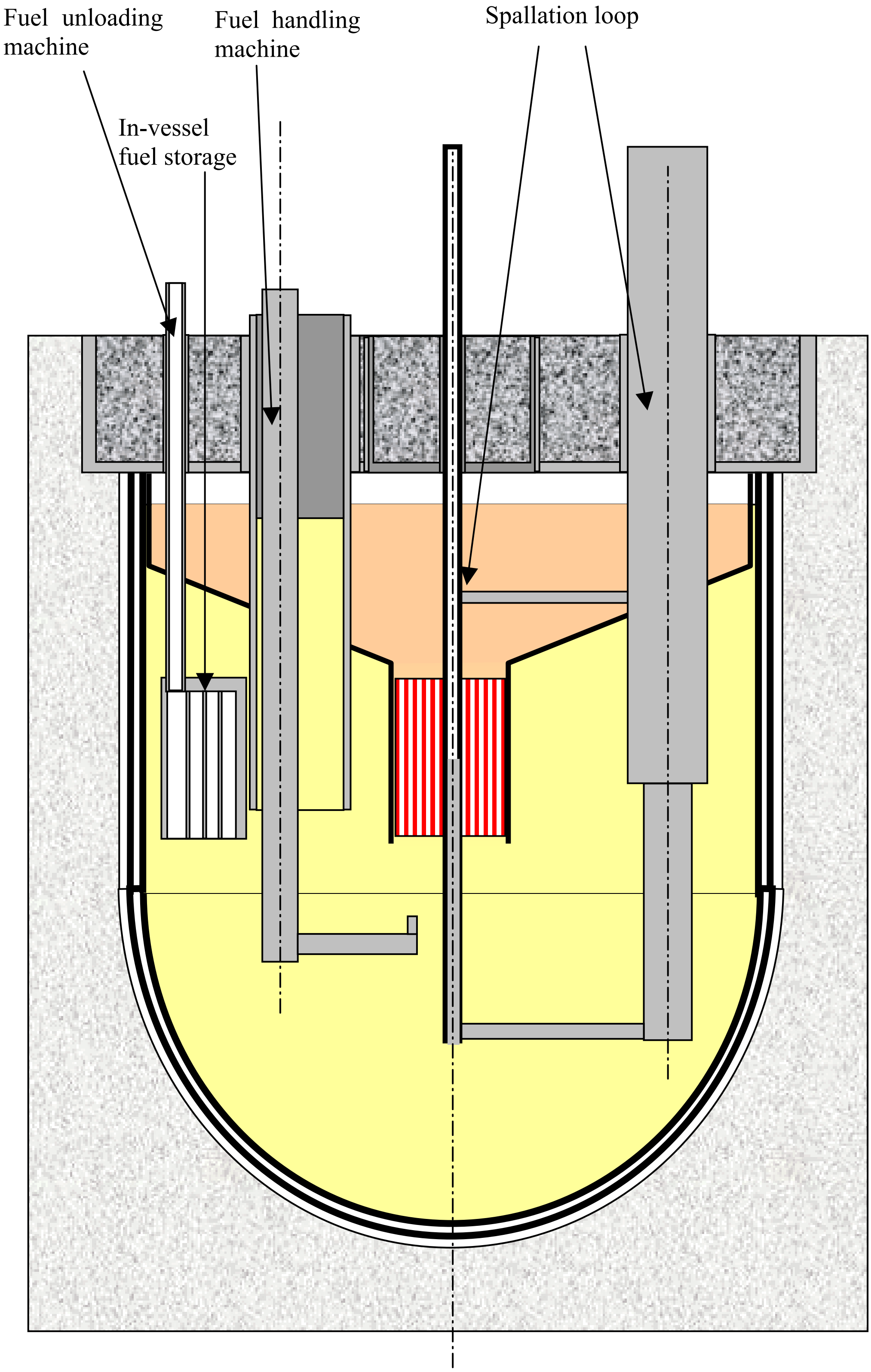

FIG. 4. MYRRHA vessel configuration – BB' view.

TABLE II. GENERAL CHARACTERISTICS

Dimensions:	
Core external diameter	900 mm
Core height	1300 mm
Fuelled length	500 mm
Vessel diameter	4000–5000 mm
Vessel total height	7000 mm
Vessel cover thickness	1000–2000 mm
Gas plenum above the coolant (if any)	0–500 mm thickness
Parameters:	
Nominal power	30 MW
Primary coolant	Pb–Bi
Coolant pressure	Atmospheric and hydrostatic
Core inlet temperature	$200^{o}C$
Core outlet temperature (max.)	$350^{o}C$
Coolant velocity in the core	2 m/s
Primary coolant flow rate (nominal)	1400 kg/s
Secondary coolant	water or possibly steam

2.4.4. Internals

The sub-critical core has been described in Section 2.3. The fuel assemblies stay under a support plate by the buoyancy force. The sub-critical core is surrounded by a cylindrical piece, the core barrel, whose function is to prevent the fuel assemblies to move away from each other. The core barrel is connected to the central hole of a conical piece, the diaphragm, which separates the lower part of the vessel, at ‘low’ temperature, from the upper part at ‘high’ temperature. The diaphragm has numerous penetrations, for the large components and for the IPSs. When an IPS is not inserted, the corresponding diaphragm penetration, and the reactor cover penetration as well, must be closed by a plug.

2.4.5. Spallation loop

In the conditions foreseen for MYRRHA, the spallation process generates about 2 MW in the target material. To avoid overheating of the latter, it was chosen to use liquid Pb–Bi as spallation material and to circulate it in a loop. To avoid ageing problems of a heavily loaded material at a critical location, there is no physical separation between the Pb–Bi of the spallation target and the high vacuum particles beam, but instead, there is a free surface. The design is therefore called “windowless”.

The loop (Figs 5 and 6) is installed in a vessel, close to the core, and contains a feed tank, a main circulation pump, an MHD flow regulation pump, a secondary tank for the collection of the excess Pb–Bi, an auxiliary drain pump and the required instrumentation. Because of the free surface interface between the spallation target and the particles beam tube, the whole spallation vessel must also be under vacuum, to avoid the contamination of the beam tube by gas.

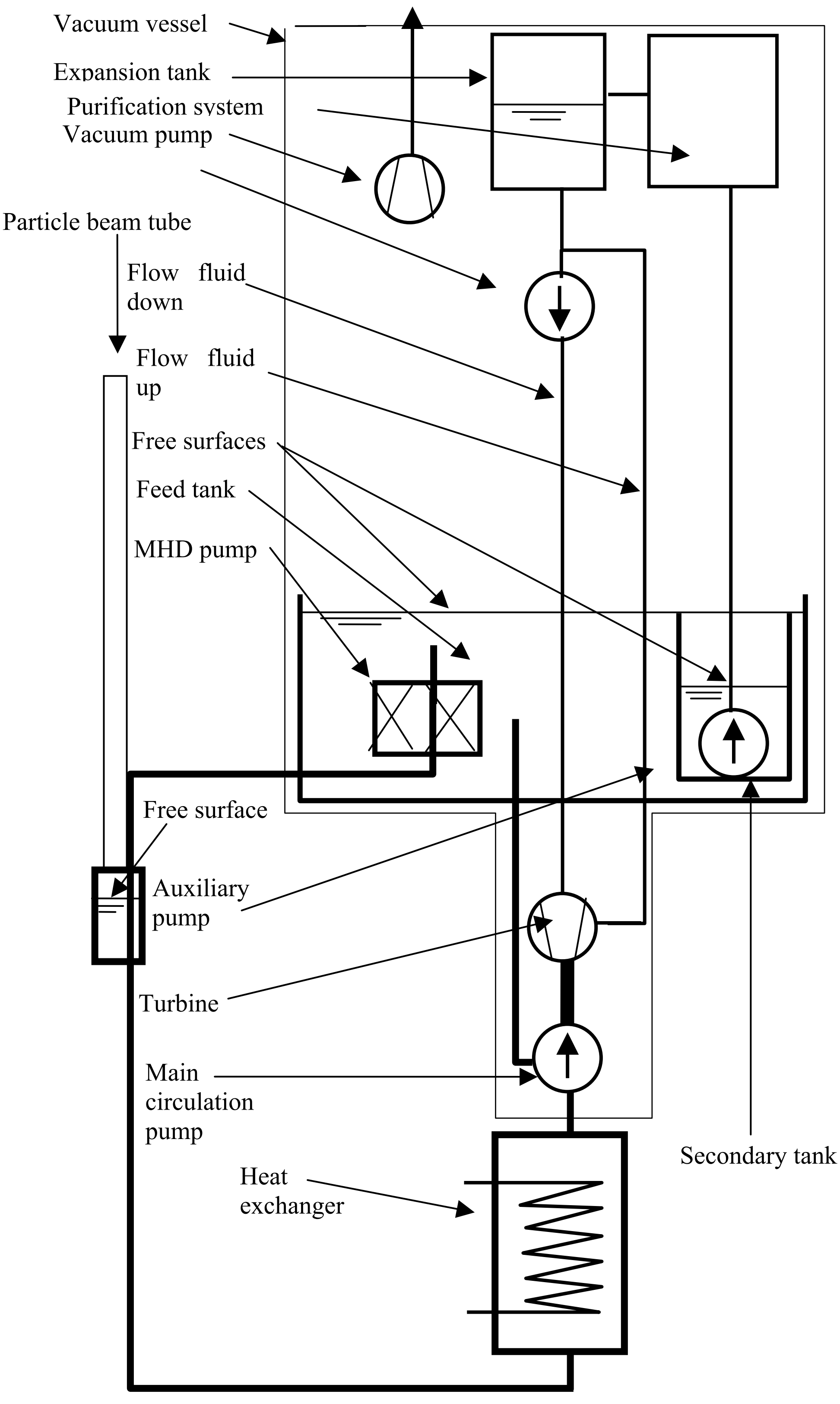

FIG. 5. MYRRHA spallation loop flow sheet (turbine pump option).

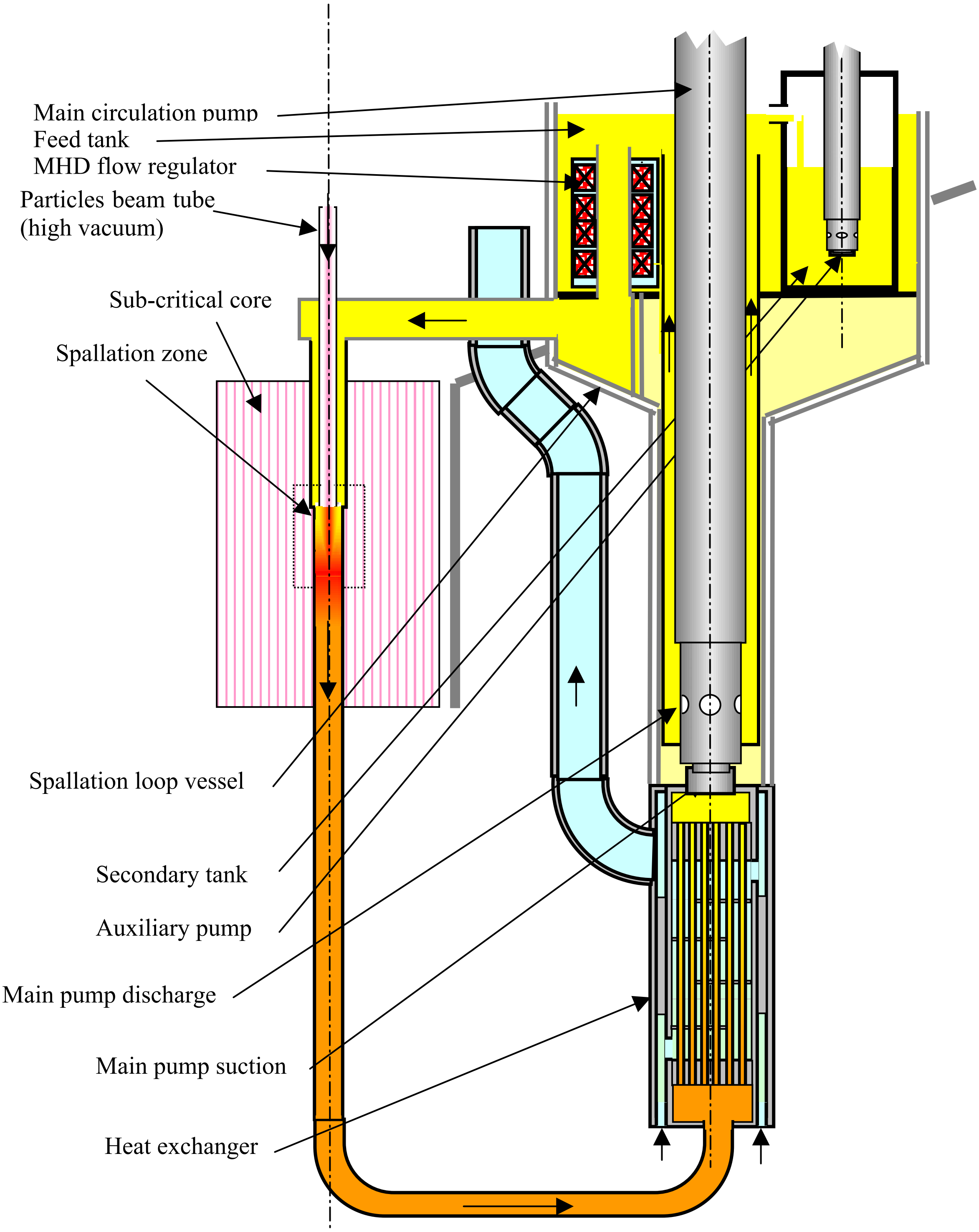

FIG. 6. MYRRHA spallation loop.

The Pb–Bi in the target is recirculated by the main circulation pump, which pumps it back to the feed tank. A liquid metal/liquid metal heat exchanger is installed in the main circuit and removes the 2 MW deposited by the spallation process. The location of this heat exchanger (HX) (upstream or downstream the main circulation pump) has to be determined, in function of the room available and of the pressure drop generated by the HX. The heat sink of the HX is provided by the primary coolant of the reactor, that is Pb–Bi at around 200°C.

TABLE III. PRIMARY HEAT EXCHANGER CHARACTERISTICS

Heat removal capacity	30 MW per HX - two identical units		
Service heat removal	15 MW		
Available Pb–Bi depth	4.5 m max.		
HX type	straight tubes, counter-current		
Secondary connection	two central concentric channels		
HX diameter	1 m max. (shell included)		
Total HX length	Vessel cover thickness: 2 m Gas plenum (if any): 0–0.5 m Upper tubesheet level: 2–3 m Bundle length: 5.5–6.5 m		
Design conditions (30 MW)	**Primary side**	**Secondary side**	
		Reference design	**Alternative design**
Fluid	Pb–Bi	water	water
Inlet temperature	350°C	160°C	50°C
Outlet temperature	200°C	170°C	120°C
Flow rate (indicative)	1400 kg/s	700 kg/s	110 kg/s
Velocity	2 m/s max.	10 m/s allowed	10 m/s allowed
Pressure	Hydrostatic	15 bar	
or	Hydrostatic + 5 bar		5 bar
Allowed ΔP	1 bar		
Service conditions (15 MW)	**Primary side**	**Secondary side**	
Fluid	Pb–Bi	water	water
Inlet temperature	350°C	160°C	50°C
Outlet temperature	200°C	TBD°C	TBD°C
Flow rate (indicative)	1400 kg/s	TBD kg/s	TBD kg/s
Secondary coolant inlet temperature		150°C min.	50°C
Secondary coolant service Pressure		15 bar	5 bar

The level in the feed tank is kept constant by pouring the excess fluid in a secondary tank, from where it is removed by the auxiliary pump. The Pb–Bi contained in the feed tank flows by gravity in an annular tube surrounding the particle beam tube. The flow rate is determined by the tube geometry and by the height difference between the free surface in the feed tank and the free surface in the spallation target. In addition, a MHD pump is foreseen to provide the fine tuning of the feed flow. A detection system (e.g. a LIDAR) measures the vertical position of the free surface and adjusts the flow of the MHD pump in order to keep constant the position of the free surface.

2.4.6. Primary heat exchangers

The coolant circulation will be classically performed upwards in the central sub-critical core region, and downwards at the periphery. The region located above the sub-critical core will contain instrumentation, like the measurement of the temperature at the outlet of the fuel assemblies.

Two groups pump–heat exchangers are installed at the periphery of the vessel, and are rated for 30 MW each.

The heat exchangers will be of the straight tubes, single pass, counter-current type. The secondary coolant is water, chosen mainly because its technology is widely available. Several alternatives are presently considered, each having its own benefits and drawbacks.

In the so-called Reference Design, the water is at 15 bar, 160^{o}C and is circulated at high velocity. Because the secondary coolant is at a temperature higher than the freezing point of Pb–Bi, the risk of local freezing of the latter can be excluded, even during cyclotron trips or during plant black-out. In addition, water can be used to keep the bulk at temperature during the shut-downs and to compensate the heat losses. The main concern posed by this option is the water ingress in the core after the failure of a tube of the HX. If water is released in large amounts through the core, the risk of prompt-criticality cannot be excluded, even if a large part of the water should flash into steam. By careful design of the flow path, it should be possible to provide separation of the water and the Pb–Bi at the outlet of the HX. The large difference in density should help to achieve this, but it has still to be demonstrated.

An alternative design tends to avoid that risk by using water at 5 bar and only 50^{o}C inlet temperature, with the primary circuit also pressurised at 5 bar. The higher pressure on the primary side will prevent the ingress of water in the primary system. In addition, the higher temperature difference allowed in the HX leads to significantly smaller equipment.

Other alternatives consider the cooling by a steam generator. The idea is to reduce the amount of water present in the primary system, and to generate internally the power used by the cyclotron and the reactor.

2.4.7. Primary pumps

The primary pumps are installed besides the heat exchangers. In the Reference Design, there is one pump per heat exchanger and one single pump is able to deliver the Pb–Bi flow rate corresponding to the nominal power. An option considers an integrated group pump/heat exchanger which would allow to spare some room inside the vessel. A second option considers two pumps per heat exchanger.

The pumps are vertical units, of a design very similar to the one used in FBRs, that is with an impeller at the bottom end of a long shaft. The electric motor is located on the reactor cover, out of the neutron flux, at the upper end of the shaft.

2.4.8. Spent fuel handling

Another important parameter in the configuration is the choice of the system for the handling of the fuel elements. The fact that the fuel assemblies float in the Pb–Bi is a quite significant difference regarding the FBRs and calls for an original design: the handling system cannot simply drop the fuel assemblies on a support plate. In addition, the room situated directly above the core will be occupied by instrumentation and safety rods mechanisms, with which the fuel handling will interfere if performed from the top of the core. Worse, the spallation loop and its particles beam tube run through the core and are not withdrawable. Therefore, an eccentric rotating plugs system, as used in FBRs, is not applicable.

TABLE IV. PRIMARY PUMPS CHARACTERISTICS

	Reference design 1	Alternative design 2	Alternative design 3
	1 Pump per HX	Group Pump-HX	2 Pumps per HX
Number of pumps	2	2	4
Impeller depth	1m min.	3.5 m min.	1 m min.
Shaft length min.	3.5 m	6 m	3.5 m
max.	6 m	6 m	6 m
Operation pressure	hydrostatic head	hydrostatic head	hyrdrostatic head
Suction head	0.5 bar min.	0.5 bar min.	0.5 bar min.
Discharge head	5–10 bar	5–10 bar	5–10 bar
Nominal flow (per pump)	700 kg/s	700 kg/s	350 kg/s
Maximum flow	1400 kg/s	1400 kg/s	700 kg/s
Temperature at the impeller	200°C (pump after the heat exchanger)		
or	350°C, if the pump is installed upstream to the heat exchanger		
Temperature max. along the shaft	350°C		

In Fig. 1, an early concept is shown, in which the fuel assemblies are grouped in clusters. One cluster forms one sixth of the core and can be removed laterally. This system allows to unload the core in only six handling operations. However, to recover one particular fuel pin in a given assembly, the cluster must be transferred in a hot cell and dismantled. This is a lengthy process and is definitely not a desirable feature for a research reactor, which should provide a high degree of flexibility.

Finally, we opted for a system where the fuel handling is performed by the bottom of the core (Fig. 3). In this concept, the fuel assemblies rest by buoyancy under a support plate. Two handling systems are inserted in a penetration of the reactor cover on opposite sides of the core. Each system is a rotating plug with an offset arm. The arm can rotate in the rotating plug, and so has access to half of the core. The arm can move up and down by about 2 m, to extract —– downwards — the assemblies from the core. The extremity of the arm has a wrist, which gives the adequate orientation to the fuel assembly before its insertion into the core.

The handling could possibly be performed by a single arm, provided it is fitted with an elbow. Such an arm would able to reach fuel assemblies behind the spallation loop. However, to avoid the complexity and the loss of positioning accuracy introduced by the elbow, we preferred the double system.

2.4.9. In-vessel fuel storage

Spent fuel still generates decay heat and must remain in the coolant for some time after the reactor is shut down. To avoid excessive delays between two operation cycles, it was chosen to store the spent fuel at the periphery of the reactor, in a dedicated zone. Because of the double fuel handling system chosen, there are two such zones. A fuel storage zone is a steel rack, with sufficient storage positions to store two core halves. The design of the zone is such that all storage positions are within the reach of the handling system. Of course, the rack is open at the bottom and fuel assemblies simply float against the top of the rack.

One position of the rack is located under the loading machine, which extracts the assemblies from the reactor and transfers them, via a transfer cask, to the spent fuel cells where the assemblies are dismantled and the samples recovered. Of course, the same path can be followed, in reverse order, by fresh assemblies.

It has been shown that there is no neutronic coupling between the core and the spent fuel storage zones, and therefore no modification of the k_{eff} of the sub-critical core. The residual flux seen by the stored assemblies is too low to allow heat generation by fission in the storage zone. On the other hand, the storage zone increases significantly the flux on the reactor vessel. The uneven dose on the vessel could possibly pose problem.

2.4.10. In-vessel shielding

To limit the activation of the structures external to the core, the sub-critical core is surrounded by dummy fuel assemblies. Those assemblies can be handled like real fuel assemblies, and possibly replaced, at some locations, by IPSs.

2.4.11. Components handling and maintenance

The large components, including the spallation loop, are built in such a way they can be extracted from the reactor and transferred to a cell where they can be inspected and repaired if necessary. Considering the service conditions of the spallation loop, in the spallation region, it is certain that the components in-core of the loop will have to be replaced at regular intervals.

2.5. Confinement building

Parallel to the core and the spallation module design, attention is given to the confinement building (Fig. 7) where the MYRRHA subcritical reactor including the spallation module will be located. The accelerator is kept in a separate confinement building to keep the maintenance and inspection procedures of the accelerator unchanged.

For the sub-critical reactor building, three options are to be assessed:

– Re-using an existing confinement building where the operators are not allowed to enter during the operation of the system, i.e. a BR3-like situation as illustrated in Fig. 3;
– Re-using an existing confinement building where the operators are allowed to enter during the operation of the system, this means that the dose exposure is less than 10 μSv/h. A preliminary assessment showed that a lateral shielding of 1 m steel followed by a 2 m heavy concrete would be necessary for achieving such a radiation level due to the very high neutron leakage. These preliminary estimates are based on analytical estimates as well as on MCNP modelling [4];
– Designing a completely new building with the above two options considered above.

FIG. 7. MYRRHA in a not accessible confinement building during operation.

3. MYRRHA ASSOCIATED R&D PROGRAMME

For the period 1999–2000 the management of SCK•CEN and IBA have mandated the MYRRHA project team to perform a detailed conceptual design and to complete the needed R&D effort to assess the main technical risks of this design for the most important parts of the system.

3.1. Accelerator

IBA is conducting preliminary design studies on the accelerator required for MYRRHA [1]. The negative ion cyclotron technology is well known by IBA, which pioneered it in its CYCLONE 30, the world reference cyclotron today for radioisotope production. At the higher energies required for MYRRHA, however, negative ion cyclotrons become unpractical large and therefore too expensive. The choice was made to use positive ion acceleration in a separate sector cyclotron.

3.2. Spallation source

The choice of a windowless design has been influenced by the following considerations:

- At about 350 MeV, an incident proton delivers 7 MeV kinetic energy per spallation neutron. Almost 85% of the incident energy exit the target in the form of "evaporation" energy of the nuclei. The addition of a window would diminish the fraction of the incident energy delivered to the spallation neutrons [5];
- A windowless design avoids vulnerable parts in the concept, increasing its reliability and avoiding a very difficult engineering task;
- Because of the very high proton current density (>130 μA/cm²) and the low energy protons we intend to use, a window in the MYRRHA spallation module would undergo severe embrittlement.

The project team has identified three main risks to be assessed for this windowless design:

3.2.1. Basic spallation data

Since the flux characteristics in an ADS are determined by the spallation neutron intensity and since there is a lack of experimental spallation data in the proton energy range considered, SCK•CEN has decided to assess, in collaboration with PSI (CH) and NRC Soreq (Is), the basic data of the spallation reaction when bombarding a thick Pb–Bi target with protons at energies close to the values that are considered for MYRRHA (E_p = 350 to 590 MeV). A joint team from the three institutes conducts the experimental programme at the PSI proton irradiation facility (PIF). The programme started in December 1998 and is due to finish by the end of May 2000 for the experimental part. The analysis of the data is still going on and expected to be finalised by the end of 2000. The expected data from this programme are:

- The neutron yield or the amount of spallation neutrons per incident proton (n/p yield);
- The spallation neutron energy spectrum;
- The spallation neutron angular distribution;
- The spallation products created in the Pb–Bi target.

3.2.2. Feasibility of the windowless design

The design of the windowless target is very challenging: a stable and controllable free surface needs to be formed within the small space available in the fast core ($Ø_{outer}$ = 120 mm). This free surface will be bombarded with protons, giving rise to a large and concentrated heat deposition (1.75 MW) dispersed over a 15 cm depth starting from the surface for a proton energy of 350 MeV. This heat needs to be removed to avoid overheating and possible evaporation of the liquid metal.

To gain confidence and expertise in the possibility of creating a stable free surface, SCK•CEN started in June 1999 an R&D program in collaboration with the thermal-hydraulics department of the Université Catholique Louvain-la-Neuve (UCL, Belgium). Within this R&D program, water experiments under atmospheric pressure on a one-to-one scale are performed. Water is used because of its good fluid-dynamic similarity with Pb–Bi. A stable and controllable free surface was established. To characterise the flow pattern, this programme has been complemented by velocity field measurements in collaboration with Forzungszentrum Rossendorf (FZR, Germany) using hot-wire and more successful ultrasonic velocity profile techniques. Laser Doppler measurements were successfully performed by UCL. A recirculation zone which leads to relatively large residence times in the region of heat deposition was found to be present. This recirculation zone could be minimised - however, not to the full extent because of air entrapment.

A confirmation experimental programme making use of Hg as a fluid under vacuum (and thus eliminating air entrapment), at the Institute of Physics at the University of Latvia (IPUL) at Riga, has been started in April 2000. First experiments will be performed in November 2000. These will lead to a fine-tuned design of the spallation target, adapted to the geometrical constraints imposed by the neutronics of the fast core.

As a final confirmation, we expect to run experiments with the real fluid at the actual temperatures and under vacuum. With a view to this, a collaboration with Forschungszentrum Karlsruhe (FZK, Germany) is set up aimed at inserting the MYRRHA spallation target head in their KALLA Pb–Bi-loop which has a working temperature of about 250^{o}C. These experiments were originally foreseen for the second half of 2000 but are postponed to a later date depending on the commissioning of the KALLA loop.

In parallel with the experiments, numerical simulations using Computational Fluid Dynamics (CFD) codes are performed, aimed both at reproducing the existing experimental results and giving input for the optimisation of the head geometry in the experiments. The CFD calculations will also be used to investigate the flow pattern and temperature profile in the presence of the proton beam, which cannot be simulated experimentally at this stage. At SCK•CEN the CFD modelling is performed with the FLOW-3D code which is specialised in free surface and low Prandtl number flow. This effort is being backed-up at UCL using the Fluent code. Moreover, a collaboration agreement with NRG (NL) has been set up for more CFD calculations with the Star-CD code [2].

3.2.3. Compatibility of the windowless free surface with the proton beam line vacuum

As the free surface of the liquid metal spallation source will be in contact with the vacuum of the proton beam line, SCK•CEN is concerned about the quantitative assessment of emanations from the liquid metal. These can lead to the release of volatile spallation products, Pb and Bi vapours and of Po, which will be formed by activation of Bi. These radioactive and heavy metal vapours can contaminate the proton beam line and finally the accelerator, making the maintenance of the machine very difficult or at least very demanding in terms of manpower exposure.

In order to assess the feasibility of the coupling between the liquid metal of the target and the vacuum of the beam line and to assess the types and quantities of emanations, SCK•CEN is preparing the VICE experiment (Vacuum-Interface Compatibility Experiment), studying the coupling of a vacuum stainless steel vessel containing 130 kg Pb–Bi, heated up to 500^{o}C, with a vacuum tube (10^{-4} ~10^{-6} torr) simulating the proton beam line. A mass spectrometer will measure the initial and final out-gassing of light gasses and the metal vapour migration. To protect the vessel from liquid metal corrosion, the possibility of Mo and W coating is currently being investigated. The full experiment is in a preparation stage and will be commissioned during the third quarter of 2000. First results are expected by the beginning of 2001.

4. MYRRHA INTERNATIONAL COLLABORATIONS

As one can conclude from the above, SCK•CEN considers the MYRRHA project as an opportunity for an international collaboration project in its design phase, during its construction and also in its future operational stage.

Collaboration agreements have been signed with:

- NRC Soreq (Is): basic spallation data;
- PSI (CH): basic spallation data, MEGAPIE;
- ENEA (I): spallation source thermal-hydraulics, core dynamics;
- UCL (B): spallation source design;
- IBA (B): cyclotron design and construction;
- FZR (D): instrumentation for the spallation target;
- FZK (D): spallation source testing with Pb–Bi;
- NRG (NL): CFD modelling and system safety assessment;
- CEA (F): subcritical core design, MUSE experiments, system studies and window design for the spallation target;
- ENEA (It): in association with ANSALDO, CRS4 and Politecnico di Torino, for Pb-Bi corrosion, windowless design, neutronics kinetic, Po contamination;
- IPUL-Riga (LT): spallation source testing with Hg.

Contacts with a view to collaborations exist with:

- Sweden: participation in MYRRHA;
- ISTC Project 559: Pb–Bi target design for LANL;
- USA: ATW roadmap international collaboration;
- AEKI (H): Hungarian Nuclear Energy Institute for the modelling of the spallation source;
- Belgonucleaire (B): core design and fuel loading policy and fuel procurement;
- Tractebel Energy Engineering (B): confinement building and auxiliary systems.

5. CONCLUSIONS

An accurate evaluation of the needed investment to build MYRRHA is one of the objectives of the present pre-design tasks to be fulfilled by the MYRRHA project team. The potential sources of funding should also be analysed by the end of the pre-design phase foreseen for end 2000–mid 2001.

The MYRRHA project could be attractive for several kinds of scientific and industrial groups at a regional, national and international level. SCK•CEN and IBA will seek for funding at stakeholders such as:

- Waste management agencies and producers of long lived radioactive waste (at the Belgian level: NIRAS/ONDRAF and the electricity utilities);
- Governmental authorities at the regional, national and international level, in charge of scientific policy, energy policy or industrial development. As was mentioned, MYRRHA is proposed as a first technical step in the development of a large scale demonstration model for the transmutation of radioactive waste in Europe. In this context, MYRRHA will be proposed for support to the European Union or specific member states;

– Industrial partners, for which a participation in the development of MYRRHA can be an important reference. This is obvious for IBA, but is also applicable to engineering companies, challenged by innovative technologies. These industrial opportunities may also attract public and private venture capital.

Accelerator Driven Systems can become an essential and very viable solution to the major remaining problems of nuclear energy production. The development of these systems requires a thorough study and experimental verification in which SCK•CEN and IBA can play a major role. Moreover, the MYRRHA system would provide the indispensable first ADS step towards a European ATW installation without forcing to freeze all options of the ADS (Pb–Bi versus gas, pool versus loop, sub-criticality level, mitigating tools for reactivity effects, etc.) from now (2000) on for the European ADS Demo for transmutation.

MYRRHA is an innovative project that will trigger different research and industrial activities in the fields of accelerator reliability, waste management (transmutation), development of new materials, environmental medicine, structural material corrosion and embrittlement, safety of nuclear installations. Increasing knowledge and know-how in these fields contributes to some aspects of sustainable development and offers a good potential for industrially applicable spin-offs.

REFERENCES

[1] Aït ABDERRAHIM, H., KUPSCHUS, P., JONGEN, Y., TERNIER, S., MYRRHA, A multipurpose Accelerator Driven System for R&D, first step towards waste transmutation, SCK•CEN, BLG 841, Mol, Belgium (2000).

[2] VAN TICHELEN, K., KUPSCHUS, P., Aït ABDERRAHIM, H., SEYNHAEVE, J.M., WINCKELMANS, G., JEANMART, H., MYRRHA: Design of a Windowless Spallation Target for a Prototype Accelerator Driven System, Presentation, ICENES-2000, 25–28 September 2000, Petten, The Netherlands.

[3] MALAMBU, E., Neutronic Performance Assessment of the MYRRHA ADS Facility, SCK•CEN, R3438, Mol, Belgium (2000).

[4] COECK, M., AOUST, Th., VERMEERSCH, F.M., Aït ABDERRAHIM, H., Shielding assessment of the MYRRHA accelerator driven system using the MCNP code, Presentation, MC2000 Conference — Advanced Monte Carlo for Radiation Physics, Particle Transport Simulation and Applications, 23–26 October 2000, Lisbon, Portugal.

[5] WACQUIER, W., Determination of the neutronic gain of a spallation source, Gent – Leuven, Universiteit Gent, Katholieke Universiteit Leuven, Belgium (1997).

DESIGN OF A MOLTEN HEAVY-METAL COOLANT AND TARGET FOR FAST-THERMAL ACCELERATOR DRIVEN SUB-CRITICAL SYSTEM (ADS)

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Abstract

Reactor physics design of a 750 MW_{th} one way coupled fast-thermal ADS was evolved in BARC earlier. This indicated that a fairly large thermal power output was possible with a proton beam of 1 GeV and current 2 - 3 mA. We also carried out preliminary studies on the molten Pb/Pb–Bi coolant and spallation-target system in the fast zone of this ADS. The thermal power in this zone was estimated to be about 109 MW. In these studies, analyses related to thermal hydraulics have been carried out for a buoyancy driven system to determine the coolant parameters for both lead as well as lead–bismuth eutectic. This design is similar to the passive lead coolant system design of CERN-EA. Appropriate equations for buoyancy pressure head, pressure drop in the reactor core; coolant velocity, heat transport etc. have been solved for different coolant inlet and outlet temperatures, fuel-pin pitch distances, fuel-power densities etc. The analyses show that coolant height required for generating buoyancy pressure head is a very strong function of power density of the fuel-pin and ΔT (difference between outlet and inlet temperature) of the coolant. Main advantage of lead–bismuth eutectic comes from the inlet temperature that can be significantly lower than that of lead; thus larger ΔT can be obtained which in turn will reduce required coolant height. This will result in saving of coolant inventory and cost. In this paper, preliminary engineering design of coolant and target for the 109 MW fast reactor zone is presented. In addition, the proposed Indian programmes to study thermal-hydraulics and materials for the technology development are discussed.

1. INTRODUCTION

Accelerator Driven sub critical Systems (ADS) have evoked considerable interest in recent years. The Energy Amplifier concept developed by C. Rubbia and others at CERN incorporates a buoyancy driven, lead-coolant primary system for extracting the heat generated in the fast reactor as well as that in neutron spallation target [1]. In earlier publications [2], our BARC group has proposed a one-way coupled booster reactor system which could be operated at proton beam currents as low as 1–2 mA for a power output of 750 MW_{th}. Here, the basic idea is to have a fast booster reactor zone of low power (~100 MW_{th}) which is separated by a large gap from the main thermal reactor zone. In this arrangement, the spallation neutron source feeds neutrons to the fast reactor zone where neutrons are further multiplied. Further in this system, the neutrons from the booster region enter the main reactor but very few neutrons from main reactor return to booster, thus ensuring one-way coupling. In earlier work, several possible configurations of the booster and thermal regions were presented. In the present work, we describe an engineering design particularly with respect to thermal hydraulics of lead/lead–bismuth eutectic coolant also acting as spallation neutron source.

This hybrid ADS reactor consists of fast and thermal reactor zones producing about 100 MW_{th} and 650 MW_{th} respectively. The schematic of the system is shown in Fig. 1. The fast core consists of 48 hexagonal fuel bundles each containing 169 fuel pins of 8.2 mm diameter arranged in 11.4 mm triangular array pitch. The average thermal power per fuel pin is about 13.46 kW. However, due to neutron flux peaking effect, the maximum fuel pin power can be up to 2.5 times this average power. The thermal reactor consists of heavy water as moderator and coolant similar to a typical CANDU type Indian PHWR except for fuel composition. Though the gap between fast and thermal zones essentially provides one way coupling of neutron flux, a thermal neutron absorber at the fast zone boundary is also being contemplated to ensure one way coupling.

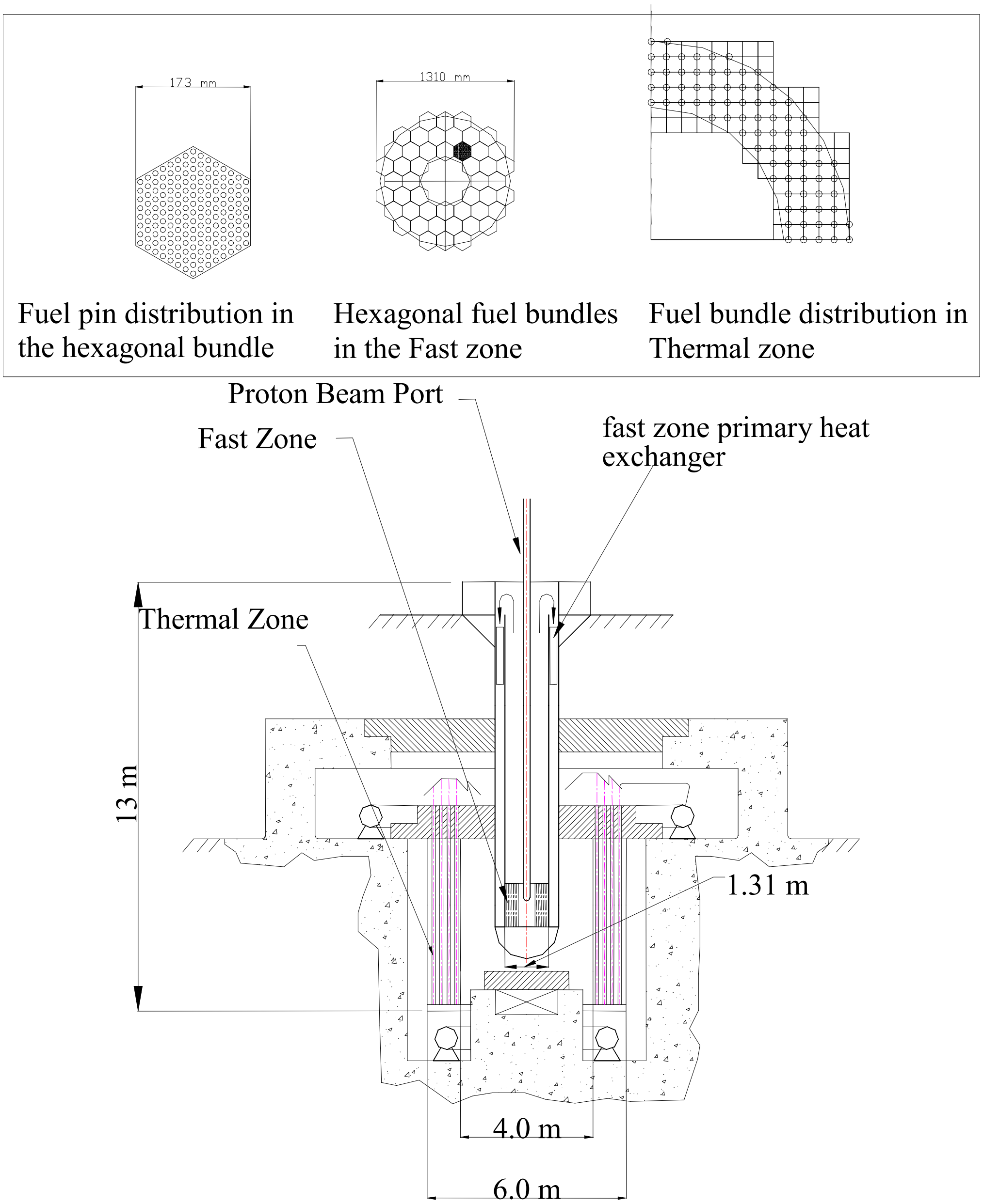


FIG. 1. Schematic of the 750 MW fast-thermal ADS.

2. HEAVY METAL COOLANT SYSTEMS

Various schemes of ADS have proposed coolant systems with molten lead (Pb) or lead-bismuth (45% lead and 55% bismuth by weight) eutectic and also these as spallation target materials. These schemes include common coolant-target circuit [1, 3], as well as isolated target and coolant circulation loops [4]. In addition, schemes have been proposed with proton window as well as windowless target systems [5–7]. CERN concept has two main attractions; buoyancy driven flow makes it highly reliable (no pump failure) and the same coolant is used for reactor as well as for removal of heat due to proton beam heating in spallation target and beam window. Buoyancy effect can be further enhanced by introducing a compatible gas into the coolant near the bottom of the riser pipe of the coolant [4]. Such schemes have also been extensively studied in liquid metal magneto-hydrodynamic power generation [8]. However, detailed analysis of this scheme is required before arriving at specific configuration for a particular ADS.

From thermal-hydraulics point of view, lead–bismuth eutectic (LBE) is a better coolant and target material than lead. This is primarily because of lower melting temperature (which can be as low as 127°C for LBE of 55% Bi and 45% Pb weight ratio) as compared to that of pure lead (327°C). Thus, it can be operated with larger temperature difference (ΔT) across inlet and outlet of the core and target zone. The main disadvantage in using bismuth as a target is the production of sizable amounts of highly volatile and radio-toxic polonium due to neutron capture. However, estimates made out by Russians [3] indicate negligible influence of Polonium in the overall radioactivity in the coolant-target of LBE. Further studies have to be carried out on this issue to arrive at a decision whether LBE can be used for ADS without significant safety hazards. If the reactor can be operated in the presence of Polonium then LBE will be a strong candidate for coolant and target material. In this paper, preliminary analysis related to thermal hydraulics has been carried out to determine the coolant parameters for the both lead as well as LBE for the fast reactor zone of our ADS.

3. DESIGN OF BUOYANCY DRIVEN SYSTEMS

Convective pumping in fast and spallation zones is realized with the help of a coolant column (h) in which hot coolant from the core rises as a result of thermal expansion coefficient (β). The coolant returns to the core after being cooled down to the lower inlet temperature in the heat exchanger. The schematic of the system is shown in the Fig. 2. The pressure head generated is given by [1]:

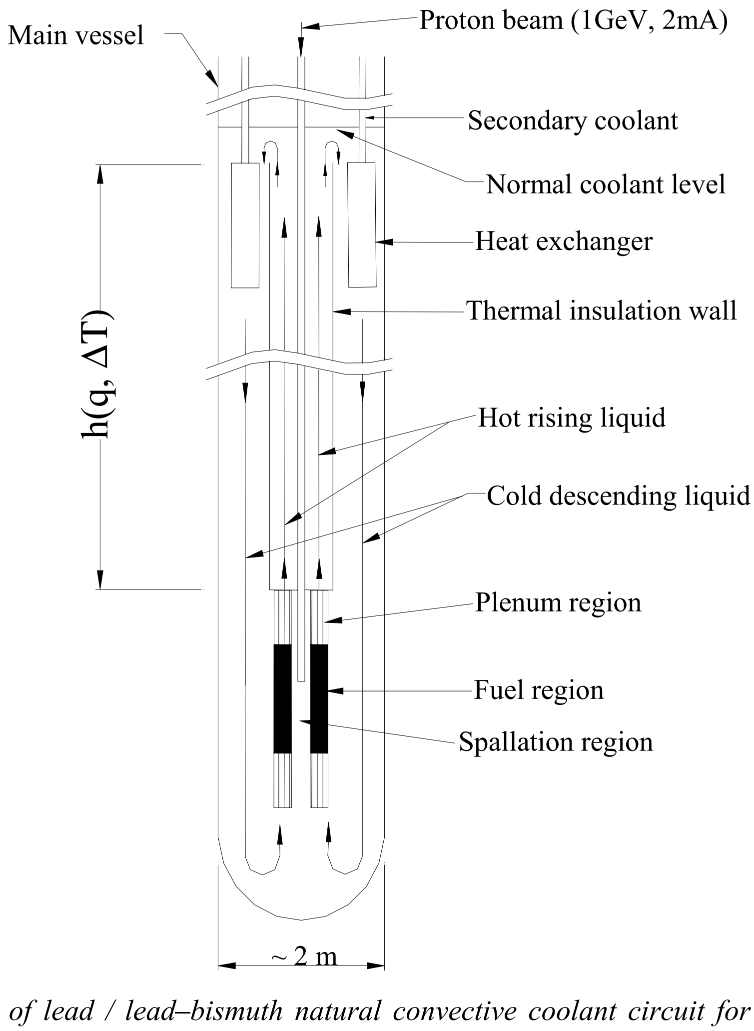


FIG. 2. Schematic of lead / lead–bismuth natural convective coolant circuit for fast reactor of the hybrid one-way coupled system

$$\Delta P_{buoy} = \beta \cdot \Delta T \cdot h \cdot g \ ,$$

where ΔT is the temperature difference between the inlet and outlet of coolant across the reactor zone. In order to dissipate a power q produced by nuclear reactions in the fuel pin with a resulting temperature difference ΔT, the coolant must traverse the core or fuel channel with a velocity v given by:

$$v = \frac{q}{f_f \, \Delta T \, \rho \, C_p} \ ,$$

where f_f is the flow area and ρ and C_p are density and specific heat of the coolant. For cylindrical pins of radius r_f of the fuel arranged in an infinite hexagonal lattice pitch of p, the flow area is given by:

$$f_f = \sqrt{3}\, p^2 / 2 - \pi \, r_f^2 \ .$$

Neglecting the end effects and the temperature dependency of the parameters, the pressure drop in the core, ΔP_{core} for the given flow velocity v in the fuel is given by:

$$\Delta P_{core} = \frac{2 \chi \eta \, l \rho \, v^2}{d} + \Delta P_{end}$$

$$\eta = 0.079 \left(\frac{\rho \, v \, d_f}{\mu} \right)^{-0.25}$$

$$d_f = \frac{4 \, f_f}{2 \pi \, r_f}$$

$$\chi = \frac{l_f}{l} + \frac{l_p}{l} \left(\frac{d_f}{d_p} \right)^{1.25} \left(\frac{f_f}{f_p} \right)^{1.75}$$

$$l = l_f + l_p \ ,$$

where χ is the geometry dependent factor, l_f and l_p are the lengths of the pin in fuel and the plenum sections respectively. η is the friction factor, μ is the viscosity of the coolant, d_f and d_p are the effective flow diameter of the fuel and plenum regions of the coolant. f_p is the effective flow area corresponding to plenum part of the fuel pin. ΔP_{end} is the additional pressure drop due to abrupt change in the flow area and is given by:

$$\Delta P_{end} = \frac{\rho \, v^2}{4} \left[\left(1 - \frac{f_p}{f_o}\right) \left(\frac{f_f}{f_p}\right)^2 + \left(1 - \frac{f_f}{f_p}\right) + 2\left(1 - \frac{f_p}{f_f}\right)^2 + 2\left(1 - \frac{f_o}{f_p}\right)^2 \left(\frac{f_f}{f_p}\right)^2 \right] ,$$

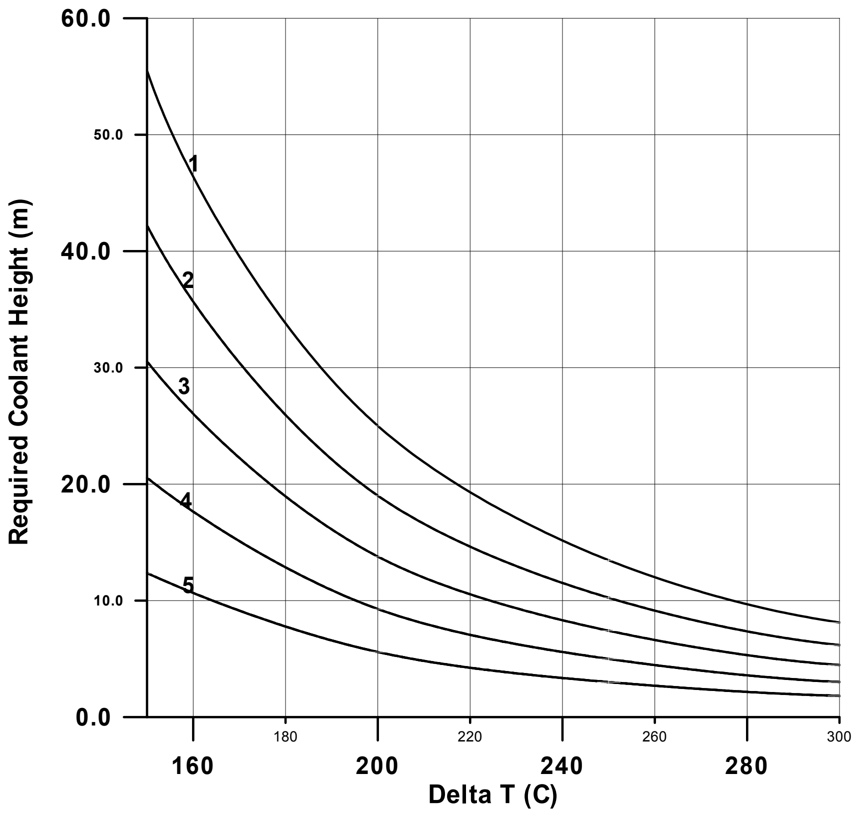
where f_o corresponds to the flow area in the absence of fuel pins.

The pressure head obtained due to buoyancy, ΔP_{buoy} has to provide sufficient pressure head to overcome ΔP_{core}, pressure drop across the heat exchanger and frictional pressure drop due to circulation in the pipe. Solving the above set of equations give complete details of the coolant parameters.

The above one-dimensional equations were solved to arrive at equation for h**,** the required height of the coolant for a given system. This analysis gives only approximate values and we have to take into account radial temperature variations due to unequal fuel heat generation, two-dimensional spallation heat distribution etc. to arrive at actual parameters. Here, it was assumed that typically we would require around 30% additional pressure head to overcome heat exchanger and frictional pressure losses in the loop [1]. Thus:

$$h=\left(\frac{0.023\chi\cdot l}{\rho\beta}\right)\left(\frac{\mu^{0.25}}{d_f^{1.25}(f_f\ C_p)^{1.75}}\right)\left(\frac{q^{1.75}}{\Delta T^{2.75}}\right)+\frac{\Delta P_{end}}{\beta g\Delta T}.$$

Based on the above equation, parametric analysis was carried out for different pitch of the fuel pin distribution, different coolant inlet and outlet temperatures corresponding to both lead and LBE flow systems.

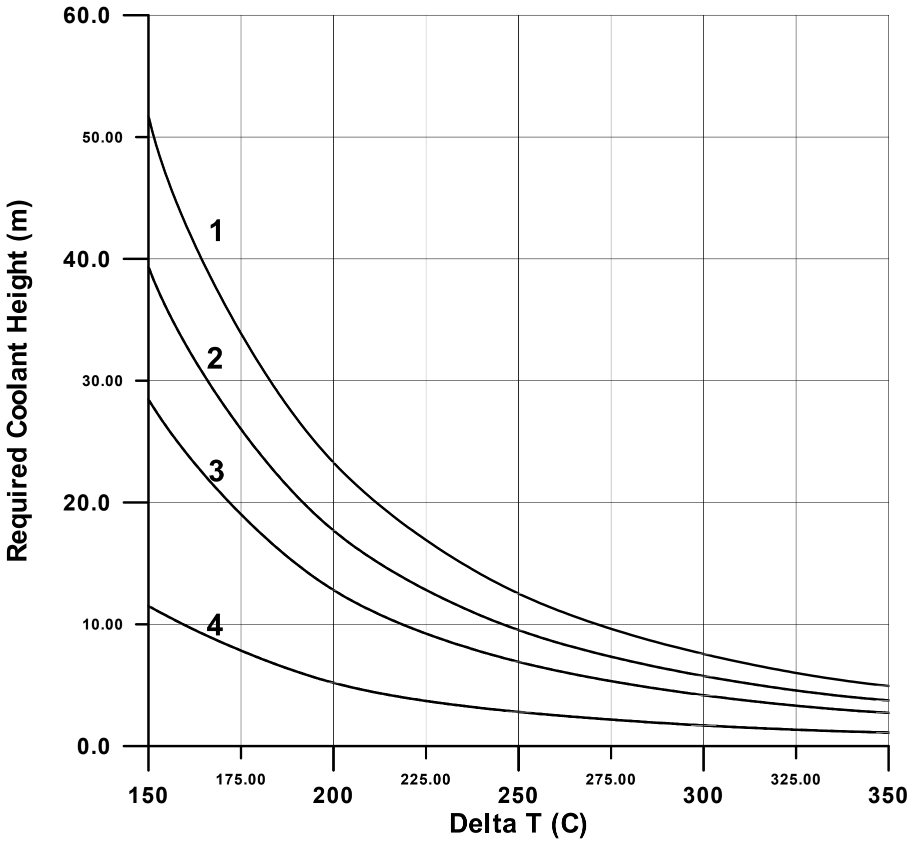


Required Height for driving coolant through Reactor Core, Heat Exchangers and other frictional losses as a function of temperature difference in the riser and downcomer

1- q = 35 kW (558 W/ cc) ; 2 - q = 30 kW (478 W/ cc)
3 - q = 25 kW (398 W/ cc) ; 4 - q = 20 kW (319 W/ cc)
5 - q = 15 kW (239 W/ cc)

FIG. 3. Lead coolant.

In Figs 3 and 4, the required height of the coolant column to provide adequate pressure head for different ΔT coolant values and fuel power densities are plotted for lead and LBE respectively. As expected, the required height is a strong function of both q and ΔT of coolant. For a ΔT of 200°C when q is reduced from 35 to 15 kW, the required column height is reduced from ~25 to ~7 m. Similarly, if ΔT is increased from 150 to 300°C, the required height is reduced from ~55 to ~9 m for the case of q = 35 kW (560 W/cc). In addition to the height determined above, we require some additional height to make coolant flow smoothly into the core. Reduction in the required height for the coolant by increasing ΔT leads to reduction in the requirement of the coolant inventory and lower operating pressure for reactor vessel and window. On the other hand, if the height is reduced too much sufficient riser length will not be available for the coolant emerging from fuel bundles with different outlet temperatures to reach temperature uniformity in the plenum. *(For more accurate analysis, we have yet to carry out further two-dimensional analysis.)* Since the lowest temperature for lead has to be more than about 350°C and there is little scope of increasing the outlet temperature unlike the case of LBE where lowest temperature can be about 150°C, we have chosen LBE for detailed analysis.



Required Height for driving coolant through reactor core, Heat Exchanger and other frictional losses as function of temperature difference in the riser and downcomer

1 - q = 35 kW (558 W/ cc) ; 2 - q = 30 kW (478 W/ cc)
3 - q = 25 kW (398 W/ cc) ; 4 - q = 15 kW (239 W/ cc)

FIG. 4. 45% Lead and 55% bismuth coolant.

Table I summarizes various parameters of the LBE coolant system of the 100 MW_{th} fast reactor part of our design. The required riser column height for buoyancy driven pumping is ~6 m for ΔT about 200°C. This is also the result of lower fuel pin power of our design which can be increased further by using higher accelerator current, if necessary. The flow velocity in the fast core is less than 1 m/s and that will reduce tendency for metal erosion/corrosion.

TABLE I. TYPICAL COOLANT PARAMETERS FOR A 109 MW_{th} FAST REACTOR OF THE HYBRID SYSTEM

Parameter	Value
Coolant	45% Pb–55% Bi alloy
Inlet temperature, °C	250
Outlet temperature, °C	450 Av. (~ peak 585)
$\Delta P_{buoyancy}$ (bar)	0.16
Height required for buoyancy, m	6
Coolant mass flow rate, t/s	3.8
Power from single fuel pin, kW	13.46 Av.
Pitch of the fuel pin matrix, mm	11.4
Average velocity of the coolant in the fuel pin	0.18
Number of pins	8100
Pins in single fuel bundle	169
Radius of fuel pin, mm	
Core	4.1
Plenum	2.5
Length of the fuel pin, m	
Core	1.3
Plenum	1.6

Preliminary studies indicate that smaller height reduces the excess coolant transient temperature over the steady temperature during start up. For a typical 30-second start up, the excess coolant temperature over the steady state reduces by a factor of more than two when the height is reduced from 25 to 6 m.

4. BUOYANCY ENHANCEMENT WITH GAS FLOW

The analysis carried out here is based on buoyancy pressure head created due to temperature difference between riser and downcomer of the coolant loop. Another effective way of increasing the buoyancy pressure head is to introduce suitable gas like Argon or Helium at the bottom of the riser of the coolant (at the exit of the reactor) [9]. The presence of the gas significantly reduces the effective density of the fluid in the raiser due to two-phase flow. Analysis of the flow is complex since various flow regimes may exist (bubbly, churn, slug, etc.) depending upon the void fraction and flow velocity [10].

5. THERMAL HYDRAULICS OF SPALLATION TARGET

The above analysis for reactor coolant was carried out on the basis of requirement for the fast-zone core cooling. However, in the scheme proposed here, the same coolant can also be used as spallation target. A proton beam of 1 GeV and CW beam current of 1 to 3 mA current would generate 1 to 3 MW of thermal power. However, this heat is generated in a small volume leading to large thermal power densities (~ GW/m^3) [11]. It has been shown that for a ΔT of 200°C (lead coolant), the velocity near the spallation region can be sufficiently high to remove the heat-generated by convection currents [6]. Adequate effective thermal hydraulic diameter should be provided (for a given pressure head, heat removal is proportional to hydraulic diameter) for heat removal due to spallation. Detailed thermal hydraulic studies have yet to be carried out to design the appropriate configuration for heat removal in the spallation region.

6. CONCLUSIONS

Preliminary thermal hydraulic analysis has been carried out for the 100 MW fast zone of the one-way coupled fast-thermal 750 MW reactor with lead as well as LBE as coolant and spallation target. LBE was preferable since the reactor can be operated with larger ΔT. Reactor coolant parameters with LBE for ΔT of 200°C have been calculated for buoyancy driven flow. The height of 6 m is adequate for coolant circulation. For exact design calculations, we have to carry out two-dimensional reactor coolant studies based on actual neutron flux distribution in the reactor. In addition system design requires detailed analysis of thermal hydraulics of spallation heat generation and proper flow configuration, which avoids local flow stagnation zones. Some of the other issues involved are the corrosion studies related to Pb–Bi and development of appropriate materials, which in fact decide the upper limit of the operation of the coolant. We are proposing to set up a mercury/LBE experimental loop with an appropriate spallation heat simulator to study the thermal hydraulics of the spallation target and window and corrosion related aspects of LBE system.

REFERENCES

[1] RUBBIA, C., et al., CERN-Group conceptual design of a fast neutron operated high power energy amplifier, Accelerator Driven Systems: Energy generation and transmutation of nuclear waste, IAEA-TECDOC-985, Vienna (1997).

[2] DEGWEKAR, S.B., SAHNI D.C., KAPOOR S.S., Analysis of a high gain energy amplifier based on the one way coupled booster reactor concept, Presentation, International Conference on Sub-critical Accelerator Driven Systems, 11–15 October 1999, Moscow, Russia.

[3] DEGWEKAR, S.B., SAHNI D.C., KAPOOR S.S., High gain energy amplifier employing a fast spectrum booster coupled one way to a thermal spectrum system, Presentation, Advisory Group Meeting on Review of National Accelerator Driven Systems (ADS) Programmes, 1–4 November 1999, Daejon, Republic of Korea.

[4] SHEDOV, O.V., et al., ADS Program in Russia, Accelerator Driven Systems: Energy generation and transmutation of nuclear waste, IAEA-TECDOC-985, Vienna (1997).

[5] Interim report of the technical working group on accelerator driven sub-critical systems, CERN Internal Report (1998).

[6] EFIMOV, E.I., et al., Concept of the neutron generating targets with molten metal circulation, Accelerator Driven Systems: Energy generation and transmutation of nuclear waste, IAEA-TECDOC-985, Vienna (1997).

[7] RUBBIA, C., CERN concept of ADS, Proceedings, International Atomic Energy

Agency Technical Committee Meeting on Feasibility and Motivation for Hybrid concepts for Nuclear Energy Generation and Transmutation, 17–19 September 1997, Madrid, Spain, Editorial CIEMAT, ISBN 84-7834-342-3 (1997) 29-172

[8] ORLOV, Ju.I., SOTOV M.I., POLETAEV, G.P., BEZNOSOV A.V., Experimental Investigations of hydrodynamics of liquid lead–bismuth target without 'Window', Proceedings, Second International Conference on ADTTR & Applications, Kalmar, Sweden (1996).

[9] SATYAMURTHY, P., VENKATRAMANI N., QURAISHI A.M., MUSHTAQ, A., Basic design of a prototype liquid metal magnetohydrodynamic power generation for solar and waste heat, Energy Conversion and Management, **40** (1999).

[10] An Energy Amplifier Demonstration Facility Reference Configuration, Summary Report, Ansaldo Nucleare No. EA B0.00 1 200 Rev. 0 (1999).

[11] SATYAMURTHY, P., et al., Two-fluid model studies for high density two-phase liquid metal vertical flows, Int. J. of Multiphase Flow 24 (1998) 721–737.

[12] DURY, T.V., et al., Design of European Spallation Source Liquid Metal Target Using Computational Fluid Dynamics, Nuclear Technology, **127** (1999) 218–232.

DESIGN PROGRESS OF HYPER SYSTEM

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Abstract

The Korea Atomic Energy Research Institute (KAERI) has been performing accelerator driven system related research and development, called HYPER, for the transmutation of nuclear waste and energy production through the transmutation process. The HYPER program is being performed within the framework of the national mid- and long-term nuclear research plan. KAERI is aiming to develop a system concept and type of roadmap by the year 2001 and complete the conceptual design of the HYPER system by the year 2006. Some major design features of the HYPER system have been developed. The burnable poison concept is being developed to keep the core reactivity swing less than 10%. In order to increase proliferation resistance, a pyro-chemical process is employed for the separation. The trade-off studies for fuel fabrication are being performed. A dispersion type fuel is believed to have advantages in terms of achieving high discharge burnup. The long-lived fission products such as Tc-99 and I-129 will be destroyed using localized thermal neutrons separately in the HYPER. A calcium hydride is employed as moderator. SSC-H (Super System Code-HYPER) is being developed to simulate the behaviour of coolant systems. The thermal hydraulic properties of Pb–Bi are implemented on the SSC-H. Design optimization of target and beam window is being performed using FLUENT and ANSYS computer codes. In addition, beam irradiation testing is performed to estimate the hardness of window material (9Cr-2WVTa) due to the proton using keV order accelerator. Beam diameter and window thickness are optimized based on the simulation results.

1. INTRODUCTION

An accelerator driven subcritical system named HYPER (HYbrid Power Extraction Reactor) is being developed within the framework of the national long-term nuclear research plan. Many types of transmutation systems were investigated in terms of transmutation capability, safety, and the proliferation resistance of the related fuel cycles. A simple stratum shown in Fig. 1 is supposed to be the most reasonable for transmutation in terms of the proliferation issues. The HYPER system is believed to have excellent compatibility with this single stratum.

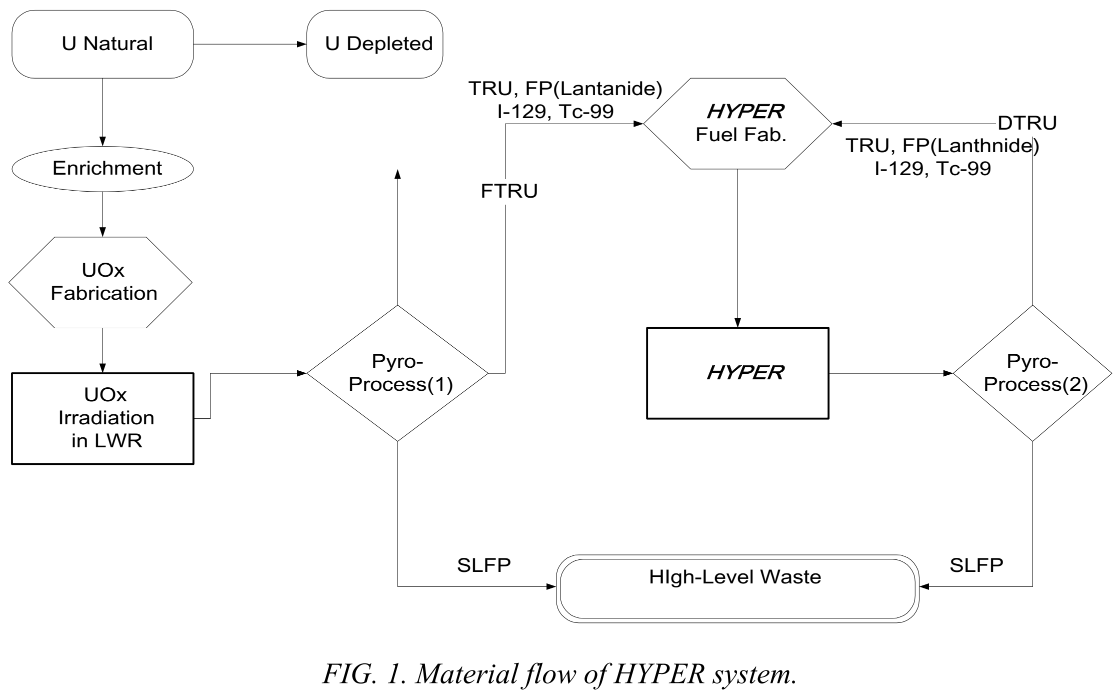

FIG. 1. Material flow of HYPER system.

The whole development schedule for the HYPER system is divided into three phases. The basic concept of the system and the key technical issues are derived in Phase I (1997–2000). Some experiments will be performed to confirm the key technical issues in Phase II (2001 - 2003). A thermal hydraulic test for the lead–bismuth (Pb–Bi), an irradiation test for the fuel and a spallation target test are the major experiments that KAERI is considering. In Phase III (2004–2006), a conceptual design for HYPER system will be finished by completing the development of design tools based on the experiments.

2. CORE DESIGN

The HYPER core adopts a hexagonal type fuel array to render the core compact and to achieve a hard neutron energy spectrum by minimizing neutron moderation. Table I represents the design parameters of the HYPER core [1]. The reference core consists of 183 fuel assemblies and 6 fission product burning target assemblies. In order to keep the radialassembly power peaking within the design target value of 1.5, the core is divided into three zones. A low transuranics (TRU) fraction fuel is designed to be loaded in the innermost zone and a high TRU fraction fuel is loaded in the outermost region. The refueling is to be performed based on scattered loading with 3 batches for each zone. The core configuration is shown in Fig. 2.

TABLE I. DESIGN PARAMETERS OF THE CORE

Parameter	Values
System:	
- Core thermal power, MW	1000
- Active core height, m	1.2
- Effective core diameter, m	3.8
- Total fuel mass, TRU·kg	2961
- System multiplication factor	0.97
- Accelerator beam power, MW	~6
- Average discharge burnup, at%	~25
- Transmutation capability, kg/y	380
- Number of fuel assemblies	183
- Average linear power density, kW/m	13.5
- Average neutron energy, keV	600
- Average neutron flux, $cm^{-2}sec^{-1}$	6×10^{15}
Assembly:	
- Assembly pitch, cm	19.96
- Flow tube outer surface flat-to-flat distance, cm	19.52
- Tube thickness, cm	0.3556
- Tube material	HT-9
- Rods per assembly	331

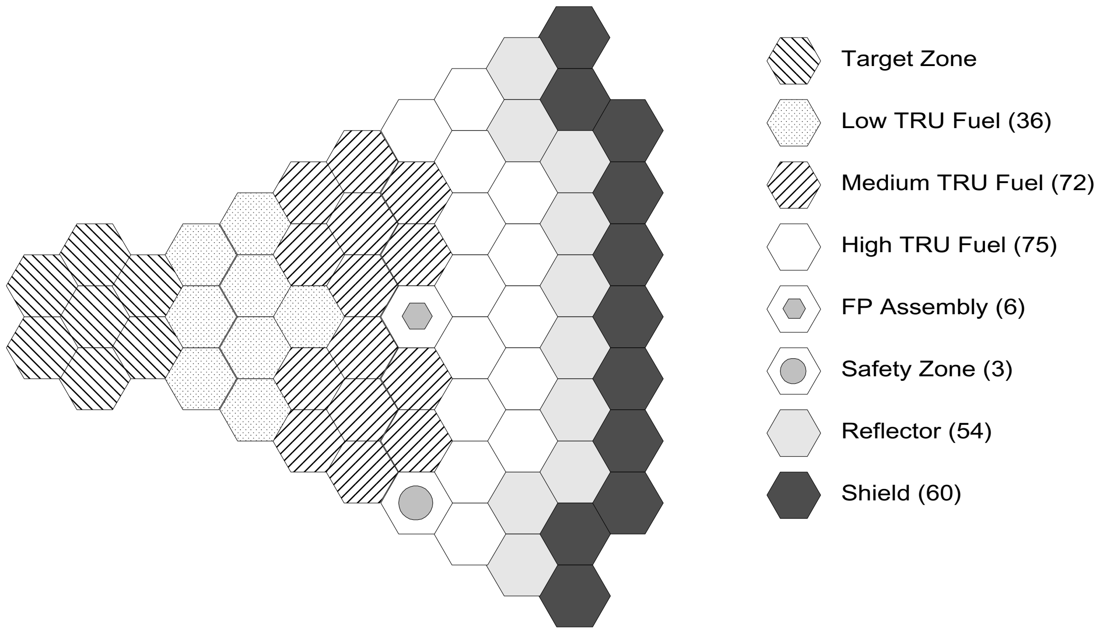

FIG. 2. Core layout.

The core is designed to produce 1000 MW_{th} with an average temperature rise of 170^{o}C. The active core height is 1.2 m and the effective core diameter is 3.8 m. In terms of neutron economy, Pb is estimated to be better than HT-9. As results, the reflector assemblies filled with liquid Pb are located at the core perimeter. HT-9 shield assemblies are located at the outermost core perimeter to prevent excessive irradiation damage to reactor structures and components surrounding the core.

The HYPER core has a relatively small amount of fertile nuclides. This raises two problems in terms of core neutronic behaviours. The first is small Doppler coefficient that contributes to making the fuel temperature coefficient negative. Preliminary calculations show a Doppler coefficient of about –0.36 pcm/K. The coolant void and temperature coefficients were also found to be negative though they are very small. The homogeneous void coefficient for BOC is about ~-140 pcm/%void. However, the local void coefficient in the central region of the core was evaluated as slightly positive. The coolant temperature coefficient is about -2.1 pcm/^{o}C. The HYPER system is supposed to have good dynamic stability. A detailed investigation will be performed on the dynamic behaviour of the core.

The second problem is the relatively large reactivity swing in the core. As the fuel burns up, the reactivity inside the core is reduced and more accelerator power is needed to maintain constant power. However, the reactivity runs down so quickly that the system cannot be operated effectively with a desirable cycle length (~1 year). The simulation shows that k_{eff} drops down from 0.97 to 0.90 in less than 10 months, which means the system needs more than 300% of the designed accelerator power. To avoid such a large amount of reactivity change and minimize the fluctuation of the required accelerator beam power due to the fuel burnup, the burnable absorber is employed. 90% enriched B_4C is used as a burnable absorber. Two different loading types of burnable absorber are being investigated. The first one coats the inside of the cladding with B_4C to a thickness of 0.002 cm. The second replaces some of the fuel rods with burnable absorber rods. Figure 3 shows the concepts of the two different loading methodologies. Figure 4 shows the preliminary results of the coating method. The coating method is evaluated to reduce the reactivity swing by about 38% compared to the non-burnable absorber cases for the depletion period of 180 days. Table II describes the variation in nuclide composition due to depletion in the non-burnable absorber and the coating cases. The coating of burnable absorber is believed to make the core neutron energy spectrum much harder.

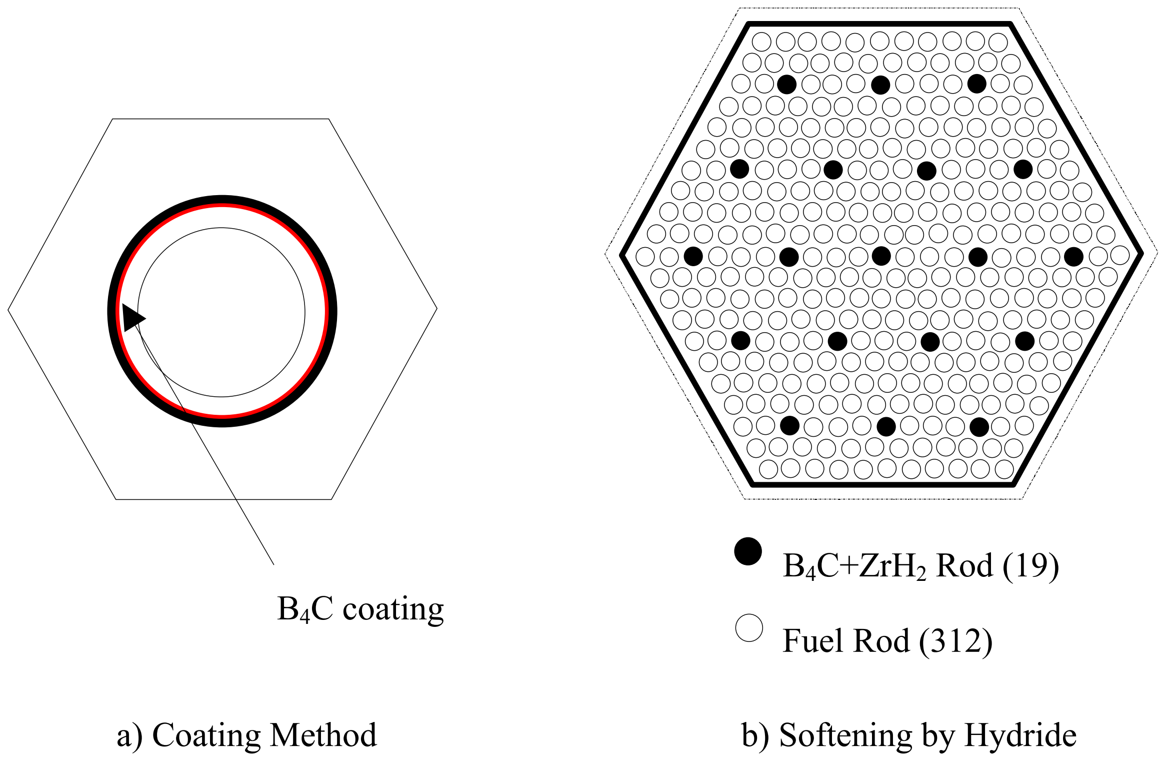

FIG. 3. Burnable absorber loading concept.

TABLE II. DEPLETION CHARACTERISTICS FOR COATED BORON ABSORBER

Nuclide	Mass, kg			
	No burnable absorber		Coated boron absorber	
	0-Day	180-Day	0-Day	180-Day
U-238	237.72	231.34 (-2.68%)	324.1	319.2 (-1.51%)
Np-237	118.56	103.74 (-12.5%)	161.6	149.0 (-7.80%)
Pu-238	26.41	40.07 (+51.7%)	36.01	46.54 (+29.3%)
Pu-239	1267.6	1084.6 (-14.44%)	1728.4	1558.9 (-9.81%)
Pu-240	577.76	588.69 (+1.89%)	787.8	787.8 (---)
Pu-241	71.68	76.74 (+7.06%)	97.73	99.02 (+1.32%)
Pu-242	109.30	110.01 (+0.65%)	149.03	148.89 (-0.09%)
Am-241	230.51	201.72 (-12.49%)	314.29	290.08 (-7.70%)
Am-243	20.23	21.46 (+6.08%)	27.59	28.38 (+2.86%)

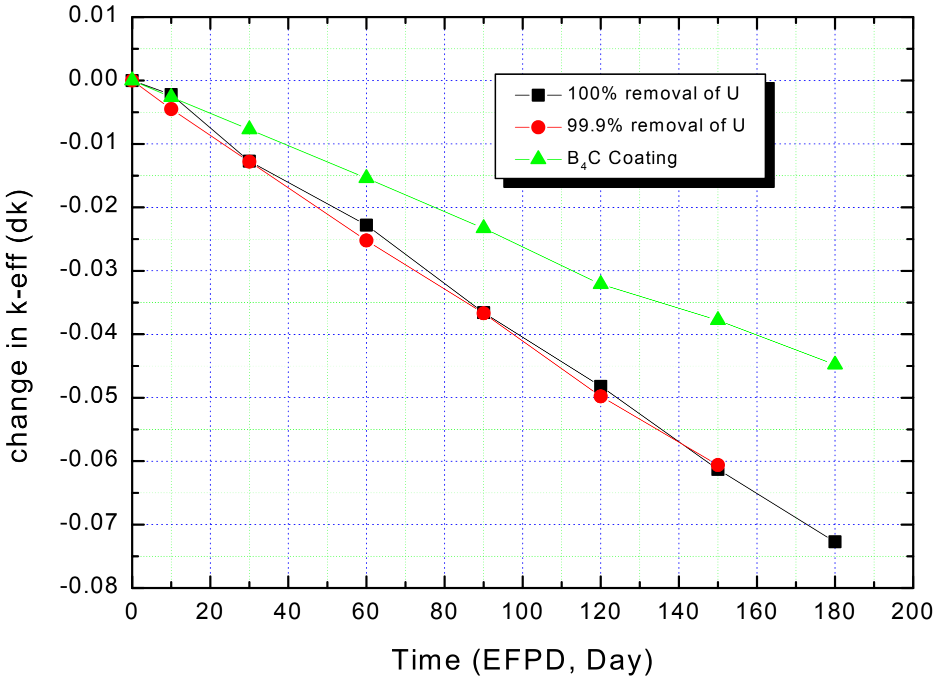

FIG. 4. Reactivity swing variation due to burnable absorber.

The HYPER core is to transmute about 380 kg of TRU a year. This corresponds to a support ratio of 5 ~6. The core-averaged neutron flux is about 6×10^{15} neutrons/cm^2 s.

3. FUEL/FP TARGET DESIGN

TRU and fission products (FP) are to be loaded into the HYPER system for incineration. They have different loading types and will be loaded into different sites because of their different neutronic characteristics. TRU is to be loaded as a fuel to drive the system. On the other hand, FP is just a target to be incinerated.

3.1. Fuel design

Either TRU-Zr metal alloy or (TRU-Zr)-Zr dispersion fuel is considered as a blanket fuel for the HYPER system [2]. In the case of the dispersion fuel, the particles of TRU-Zr metal alloy are dispersed in a Zr matrix. A blanket rod is made of sealed tubing containing actinide fuel slugs in columns. The blanket-fuel cladding material is ferritic-martensitic steel. Figure 5 shows a typical cross sectional view of alloy and dispersion fuel rods. It is expected that the dispersion fuel will generally withstand significantly higher burnup than alloy fuel. If the fuel particles are separated sufficiently, the areas damaged by fission fragments will not overlap and remains a continuous metal phase which is essentially burnup without significant swelling than is possible with alloy fuel. As a result, the undamaged by fission fragment. This relatively undamaged metal matrix can withstand higher dispersion fuel does not need as much gas plenum as the alloy fuel needs. Figure 6 shows a schematic view of alloy and dispersion fuel rod. The major features of alloy and dispersion fuel rods are described in Table III. It is not easy to control the vaporization of americium nuclides in the fabrication process of an alloy type fuel rod.

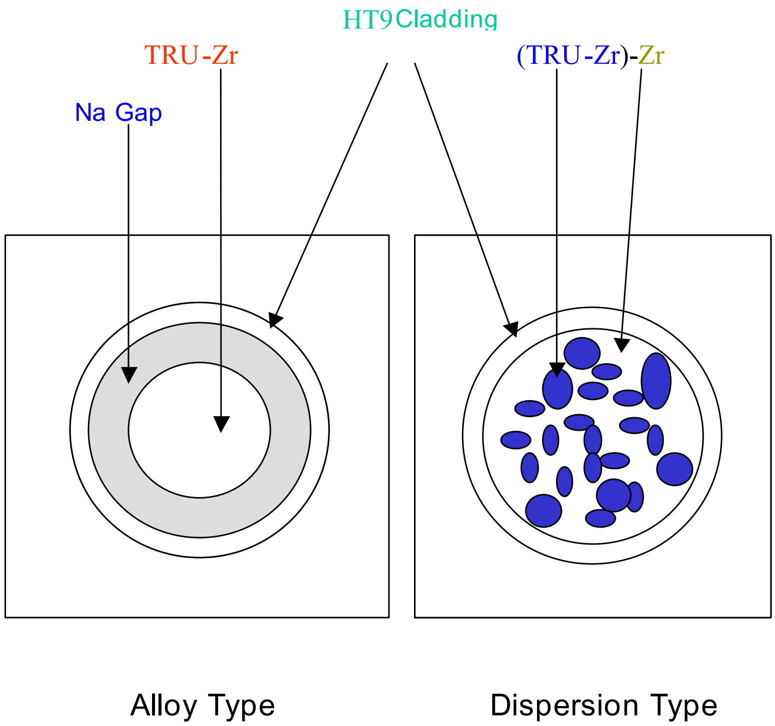

FIG. 5. Cross-sectional view of alloy fuel and dispersion fuel.

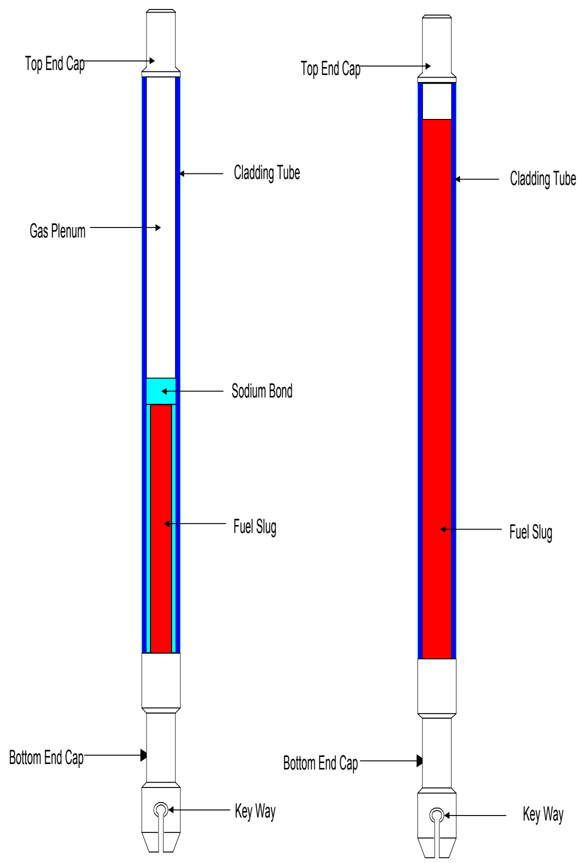

FIG. 6. Rod type of alloy and dispersion fuel.

TABLE III. GENERAL FEATURES OF ALLOY AND DISPERSION FUEL

Feature	Alloy type	Dispersion type
Development status	•Demonstration up to 15 at% burnup •Irradiation tests for about 0.16 million fuel rod •Reached max. 20 at% burnup	•Lack of DB •No experience for (TRU-Zr)-Zr dispersion fuel •Demonstrated the technique of dispersion fuel with Al matrix
Design analysis and in-reactor behaviour characteristics	•Excellent in-reactor behaviour •Need to clarify eutectic melting behaviour •Need irradiation test for optimization	•Excellent in-reactor behaviour expected •Need compatibility test between matrix and fuel particle •Need irradiation test for validation •High in-reactor behaviour safety
MA addition behaviour	•Am, Cm precipitate with rare element •Very high migration rate of Am under irradiation condition •Degradation of compatibility with SUS (FCCI)	•Less Am, Cm precipitate (comparing with alloy fuel) •No FCCI
License ability	•Basic DB accumulation very good	•DB accumulation needed •Compatibility test needed
Proliferation resistance	•Very good (pyroprocessing)	•Very good (pyroprocessing)
Thermal shock resistance	•Very good	•Very good
Heavy metal density (g/cm^3)	•12.36 (TD) •10.6 (SM: 75%)	•9.16
Melting point (K)	•1373 (for ternary)	•1373/1800 (particle/matrix)
Thermal conductivity (W/cm·K)	•0.5	•0.1 (Zr matrix)
Compatibility with Na	•Very good	•Very good
Compatibility with SUS	•Mechanical comp. is good •Eutectic melting (T > 973 K)	•Very good
Economics	•Very good	•A little lower than that of alloy fuel
Fabrication and reproducibility	•Very easy •Injection casting •Minimum facility •Some fabrication tech. proven •Some processing technology improvement needed •Am vaporization contamination problem	•Not proven for Zr matrix fabrication technology •Am vaporization is comparatively small •No gap between fuel and cladding (No need for plenum or smear density)

The maximum allowable TRU-Zr particle content is limited in the dispersion fuel because of fabrication and swelling problems. From a fabrication point of view, the maximum allowable content is 60wt% (TRU-Zr)-40wt% Zr based on the technical experience of silicide dispersion fuel of 38vol.% U_3Si-62vol.% Al. The maximum content of 60wt% (TRU-Zr)-40wt% Zr corresponds to 38vol.% (TRU-Zr)-62vol.% Zr. Therefore the minimum content of the Zr matrix is approximately 40wt%. This means that (TRU-Zr)-Zr dispersion fuel cannot be fabricated by the fuel core extrusion method if the Zr content is less than 40wt%. In other words, a content of 60wt% (TRU-Zr) is the maximum limit for (TRU-Zr)-Zr dispersion fuel. From a breakaway swelling point of view, the maximum content of 50wt% (TRU-Zr)-50wt% Zr is obtained when the performance experience of silicide dispersion fuel of 28.8vol.% U_3Si-71.2vol.% Al is directly applied for (TRU-Zr)-Zr dispersion fuel. The maximum content of 50wt% (TRU-Zr)-50wt% Zr corresponds to 28.8vol.% (TRU-Zr)-71.2vol.% Zr. Considering the above two factors, the maximum content of (TRU-Zr) fuel particle is approximately 50wt% in (TRU-Zr)-Zr dispersion fuel, and the minimum content of the Zr matrix is approximately 50wt%. Therefore, the maximum content of (TRU-Zr) fuel particle should be less than 50wt%. And the content of Zr matrix should be higher than 50wt%.

Table IV describes the fuel design parameters. There is no experience on TRU-Zr type metal fuel. Most experimental data and experience are with U-Zr or U-Pu-Zr types that have a Zr fraction of no more than 10%. A couple of fuel performance analyses have been done using the MACSIS-H code (alloy type fuel analysis) and DIMAC (dispersion type) developed by KAERI.

Figure 7 shows the thermal conductivity variations as a function of fuel temperature [3]. The alloy type has much better thermal conductivity as expected. Figure 8 is the radial temperature distribution variation at the burnup of 10 at%. In the case of the dispersion fuel, the temperature difference between the centre and the cladding surface is 110°C at a linear power of 21.6 kW/m. On the other hand, the alloy fuel shows the difference of 90°C.

TABLE IV. FUEL DESIGN PARAMETERS

Parameters		Alloy fuel	Dispersion fuel
Fuel slug	Fuel diameter, mm	4.58	5.18
	Composition	50wt%TRU-50wt%Zr	45wt%(TRU-10Zr)-55wt%Zr
	Density, g/cm^3	12.36	9.16 (TRU-10Zr: 18.37)
	TRU density, g/cm^3	6.18	3.7
Integrated gap between fuel slug and cladding, mm		0.7 (75% SD)	0.1 (engineering gap)
Cladding, mm	Inside diameter	5.28	5.28
	Outside diameter	6.68	6.68
	Thickness	0.7	0.7
LHGR (kW/m)		13.5	13.5

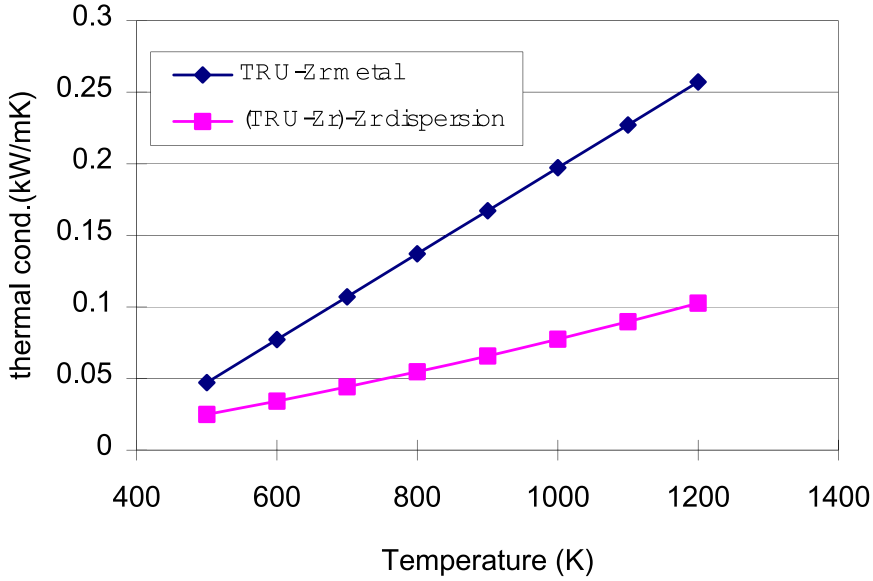

FIG. 7. Thermal conductivity variation.

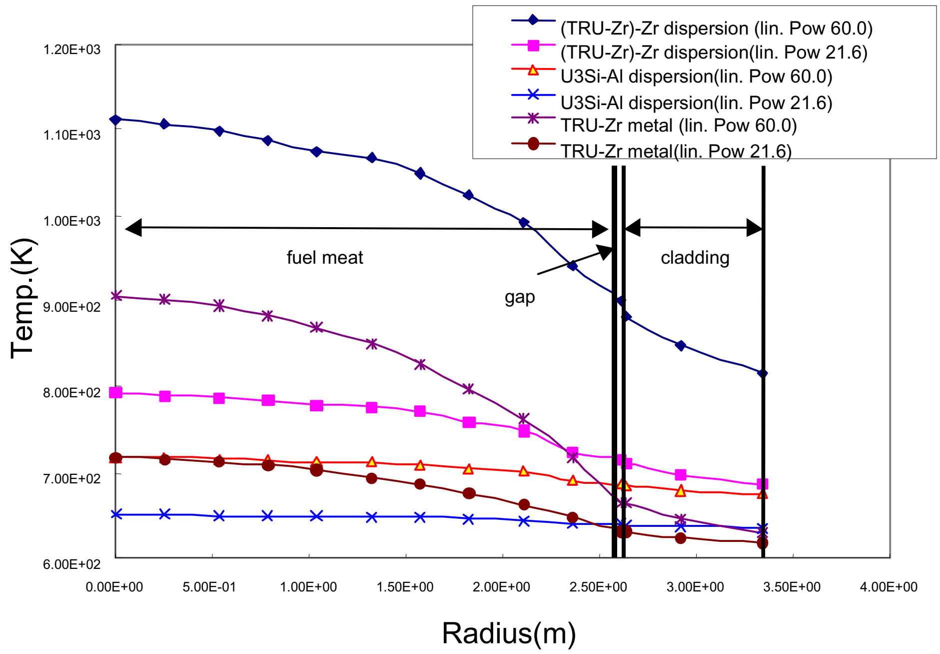

FIG. 8. Radial temperature distribution.

Figure 9 shows the fraction of the fission gas released as a function of fuel burnup for 4 different Zr fractions in the case of alloy fuel. More than 90 % of the fission gas is released at 5 at% burnup. The simulations predict that the fission gas release rate is almost independent of Zr fractions when the Zr weight fraction varies from 45 to 55%. Figure 10 shows the results of the cladding strain analysis. The cladding strains start to increase abruptly after 7 at% burnup. The total swelling of the dispersion fuel is comprised of three components; a volume change due to transformation to a higher Zirconium phase as a result of TRU burnup, a volume increase due to the accumulation of non-gaseous fission products, and a volume increase due to fission gas accumulation. Figure 11 shows the swelling variation of dispersion fuel.

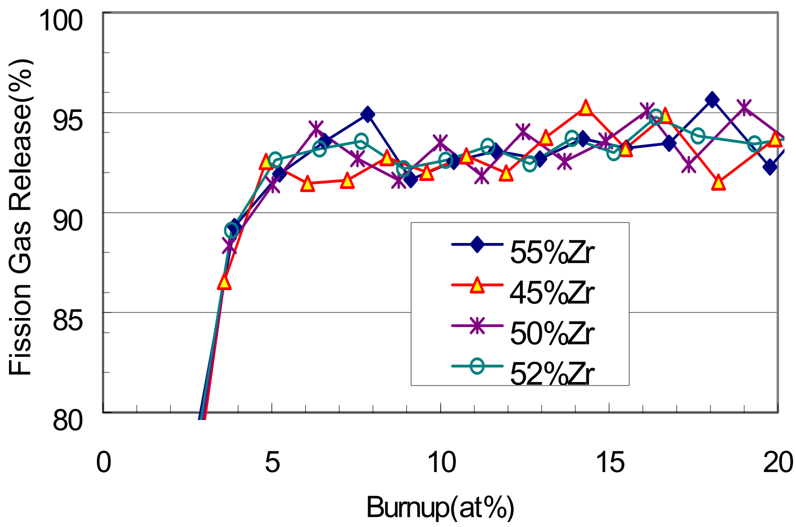

FIG. 9. Fission gas release rate.

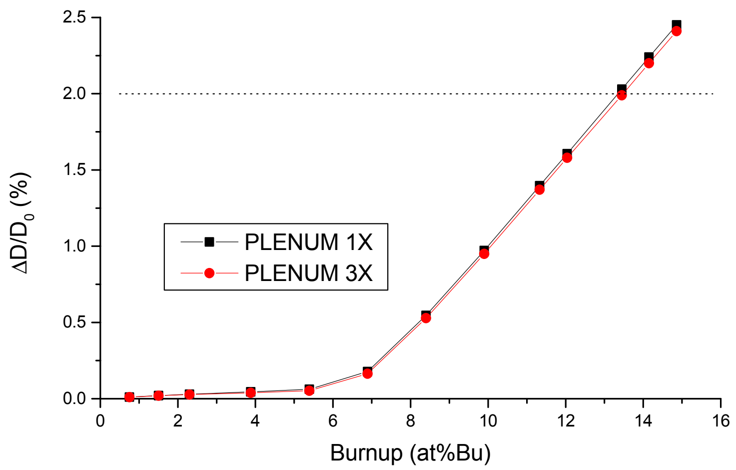

FIG. 10. Rod deformation rate.

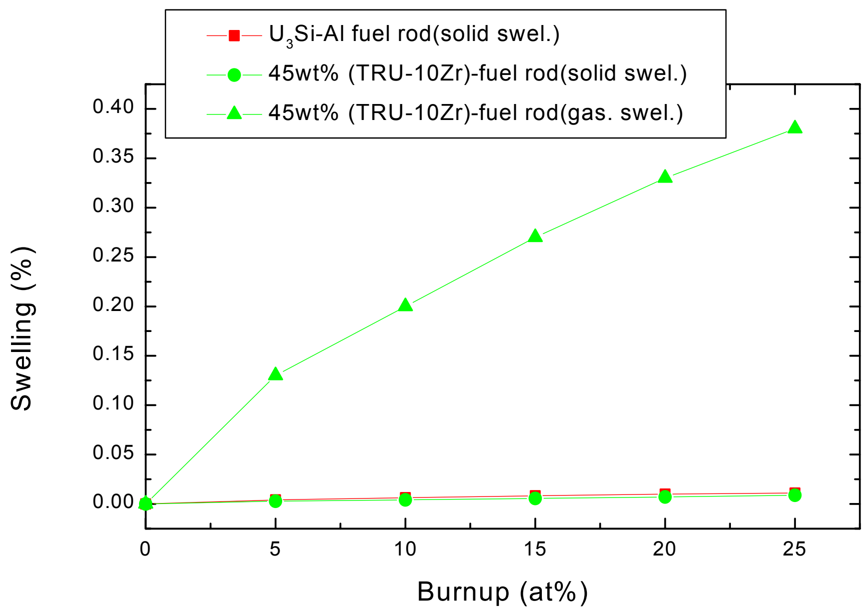

FIG. 11. Three component swelling as function of burnup.

3.2. FP Target design

Tc-99 and I-129 are to be transmuted among long-lived fission products. They will be irradiated in a special region of the reactor core (Fig. 2). However, the target forms are different for Tc and I.

The preliminary results of the basic material studies have shown that a pure metallic form is the most desirable one for the incineration of Tc-99: a fabrication route for casting the Technetium metal has been developed and irradiation experiments did not show any evidence of the swelling or disintegration of the metal. On the other hand, an elemental form was found not to be acceptable for Iodine because of its volatility and chemical reactivity. Thus, metal iodides are being considered. Sodium iodide (NaI) and calcium iodide (CaI_2) are the desirable forms. Sodium iodide is expected to have melting problems when the sodium is liberated from iodide due to the transmutation.

Design optimization of the FP Assembly (target) is being performed to maximize the transmutation of Tc-99 and I-129 in the HYPER system. The localized thermal flux is obtained by inserting some moderators such as CaH_2. Many types of assembly configurations were investigated. The configuration where Tc-99 is loaded as a plate type in the outermost region and I-129 is loaded as a rod type alternatively with a moderating (CaH_2) rod in the inner region is believed to be the most optimum in terms of transmutation rate and core power peaking increment (Fig. 12). The designed FP assembly configuration is estimated to have a transmutation rate of 6.41%/y and 13.88%/y for Tc-99 and I-129, respectively. About 6 FP assemblies are loaded into HYPER system in order to make the transmutation rate of FP equivalent to that of TRU.

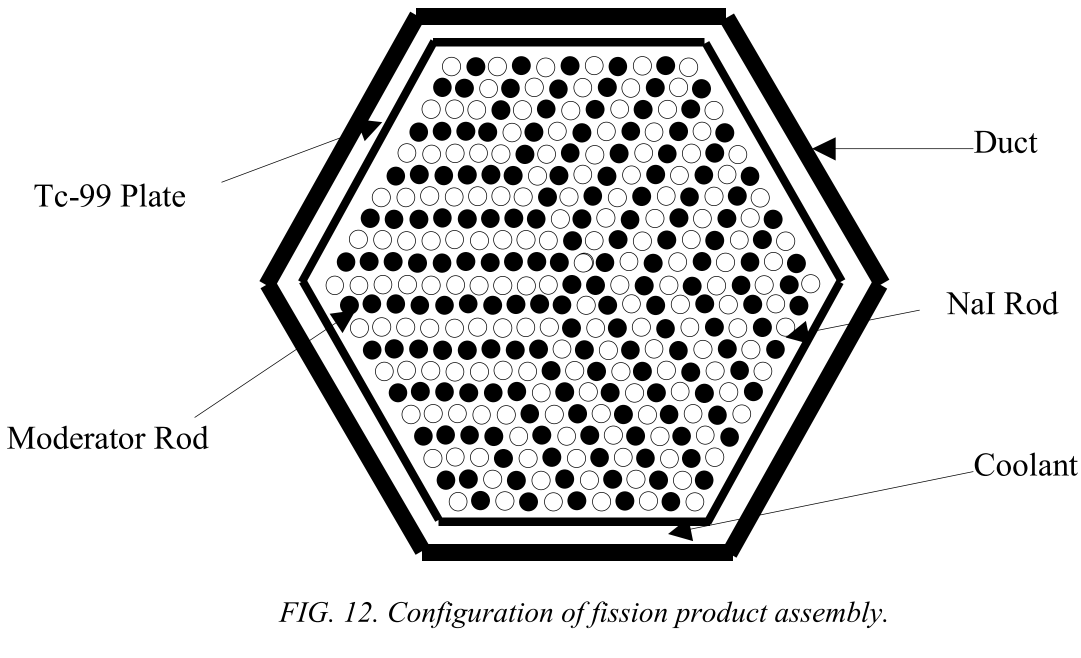

FIG. 12. Configuration of fission product assembly.

Figure 13 shows the neutron energy spectra in the region of FP assembly. The spectrum in the fuel assemblies surrounding FP assembly is relatively softer than that of normal fuel assemblies. This softened spectrum causes the increment of power peaking. The calculation produces the local power peaking of 1.232, which is within the acceptable range. In addition, the design configuration makes the core coolant void coefficients more negative but the doppler coefficient less negative.

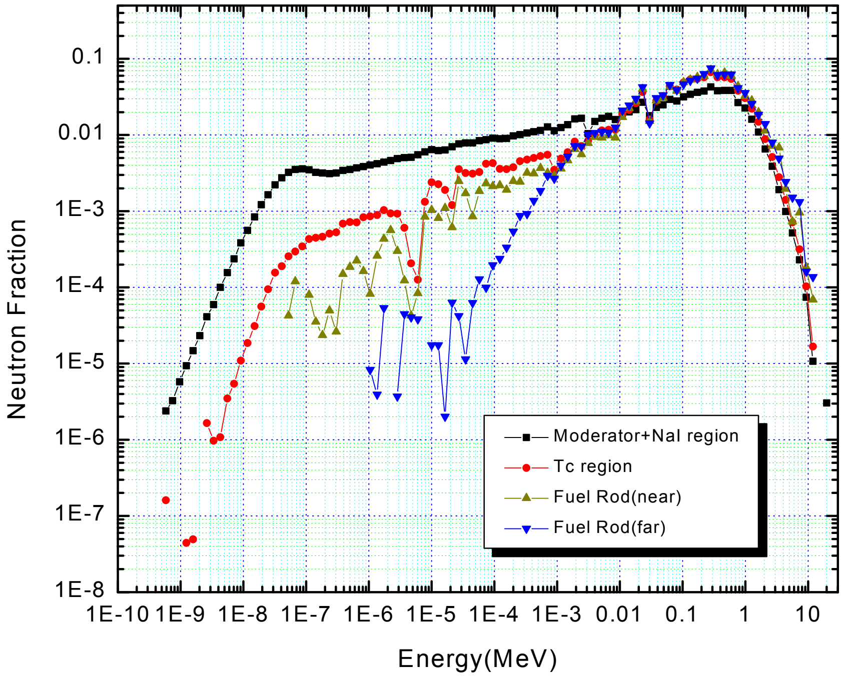

FIG. 13. Neutron energy spectra nearby FP assembly.

4. COOLING SYSTEM DESIGN

The thermal efficiency of the power cycle strongly depends on the temperature at which heat is supplied by the primary to the secondary coolant. It is obvious that coolant temperature should be set as high as possible for high efficiency. However, mechanical and corrosion characteristics of structural materials set the upper limit. According to Russian results, the maximum allowable temperature of Pb–Bi coolant is approximately 650^{o}C. The lower limit of the coolant temperature can be started from the Pb–Bi melting point, 125^{o}C. For safe operation, Pb–Bi temperature must be sufficiently above 125^{o}C. Therefore 125 and 650^{o}C can be the basic temperature limits of Pb–Bi coolant.

The core inlet and outlet temperature of Pb–Bi coolant were determined to be 340 and 510^{o}C, respectively, based on the Russian design experience. This temperature range marginally satisfies the basic temperature limits. Resulting core flow rate in order to cool 1000 MW thermal power is 46 569.0 kg/s.

Coolant velocity of primary cooling system can also cause a design constraint. Coolant velocity affects the integrity of structural materials and the pumping load. The primary cooling system of HYPER should be designed with low coolant velocity as long as it can satisfy another design requirements. Since Pb–Bi does not significantly absorb or moderate neutrons, it allows the use of a loose lattice which favours lower coolant velocity. The P/D (pitch-to-diameter) of the HYPER core is chosen to be 1.5 and the corresponding Pb–Bi velocity is 1.1 m/s, which is a relatively low coolant velocity compared to that of typical power reactors. Instead of wire spacer commonly used for tight lattice, grid spacers are suitable to ensure proper separation of the fuel rod.

Although the Pb–Bi velocity within the fuel channel is sufficiently reduced, the roughly estimated pressure loss across the reactor vessel is similar to that of typical power reactors. This is mainly due to the high density of Pb–Bi. The complex geometry of the inlet and outlet components of the core also contributes to the large amount of pressure loss. Therefore, it is expected that natural circulation does not play an important role in cooling the HYPER core under normal operating conditions.

A loop type configuration was selected for the preliminary design of the HYPER system and three-loop system was chosen as the optimal system for HYPER. The number of loop is determined by considering the coolant velocity and pressure drop across the loop. The mass flow rate flowing into a heat exchanger is 15 523.0 kg/s.

It is possible to eliminate the intermediate heat transport system with Pb–Bi coolant. A steam cycle is adopted for HYPER due to its long and successful experience. Figure 14 shows the overall view of the cooling system of HYPER.

5. BEAM TARGET

The core coolant is also used as the spallation target. Lead-bismuth comes from the bottom of the reactor and encounters the beam window before going out of the top of the reactor. A single beam window is adopted so that there is no independent window cooling system [4].

There are some design goals for the stable and safe operation of the target and reasonable lifetime of the beam window. We set the maximum allowable temperature and stress of the beam window at 700°C and 200 MPa, respectively. The temperature of Pb–Bi is set to be less than 600°C and the lifetime of the beam window is set to be 1 year. We performed calculations to optimize the appropriate design parameters and operating conditions for achieving the design goals. We used the FLUENT code for temperature calculations and the ANSYS code for stress analysis.

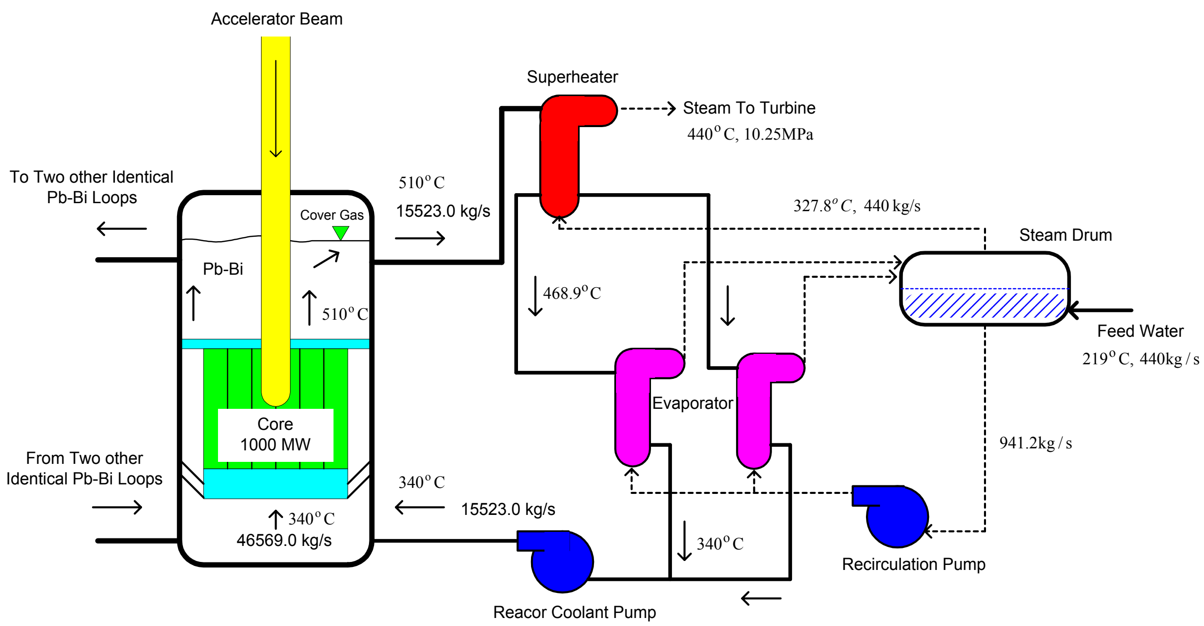

FIG. 14. Cooling system of HYPER system.

We assumed that the beam has a parabolic density distribution with a circular shape of 20 cm diameter. The inlet temperature and velocity of Pb–Bi coming from the bottom of the reactor are 340°C and 5 m/s, respectively. When the beam current is 6 mA (1 GeV·proton), the maximum temperature of the window is 651°C, which is within the limit. The maximum stress of the window turned out to be 157 MPa while the static pressure was 16 atm. Temperature and stress calculations are still under progress for other beam and cooling conditions. Our goal is to evaluate the maximum current that does not exceed our design limits.

The factors affecting the lifetime of the beam window are corrosion due to Pb–Bi and radiation damage. We expect that the effect of radiation damage is dominant in deciding the window lifetime. To predict the lifetime, first we should know the amount of radiation damage and then how the properties of the window material change as the radiation damage increases. Therefore we calculated dpa (displacements per atom) and He production rates using the LCS (LAHET Code System). When the beam current is 6 mA with a parabolic distribution and circular shape, the maximum radiation damage is 50 dpa per year at the window centre. About 20% of the radiation damage is caused by neutrons and the rest by protons. Most of the He production is caused by protons and is calculated to be 4500 appm per year at the window centre.

To find out if the window can be used for 1 year without an exchange, we are studying how the window properties change in 1 year based on available experimental data related to radiation damage. Table V and Fig. 15 show the basic characteristics of beam window and the temperature distribution nearby window, respectively.

TABLE V. CHARACTERISTICS OF TARGET DESIGN

Parameter	Characteristics
Beam window	
- Material	9Cr-2WVTa
- Structure	Cylinder with circular shape end (single window)
- Diameter/thickness	30 / 0.2 cm
Beam	
- Incoming beam	1 GeV, 6 mA Proton with 20 cm diameter
- Neutron production	1.1×10^{18}/s
- Average neutron energy	14 MeV
Radiation damage	
- Radioactive material production	5×10^{5} Ci
- Damage	50 dpa/y
- He production	4500 appm/y
Thermal damage	
- Static pressure on window	16 atm
- Maximum stress on window	157 Mpa
- Maximum temperature in window	651°C

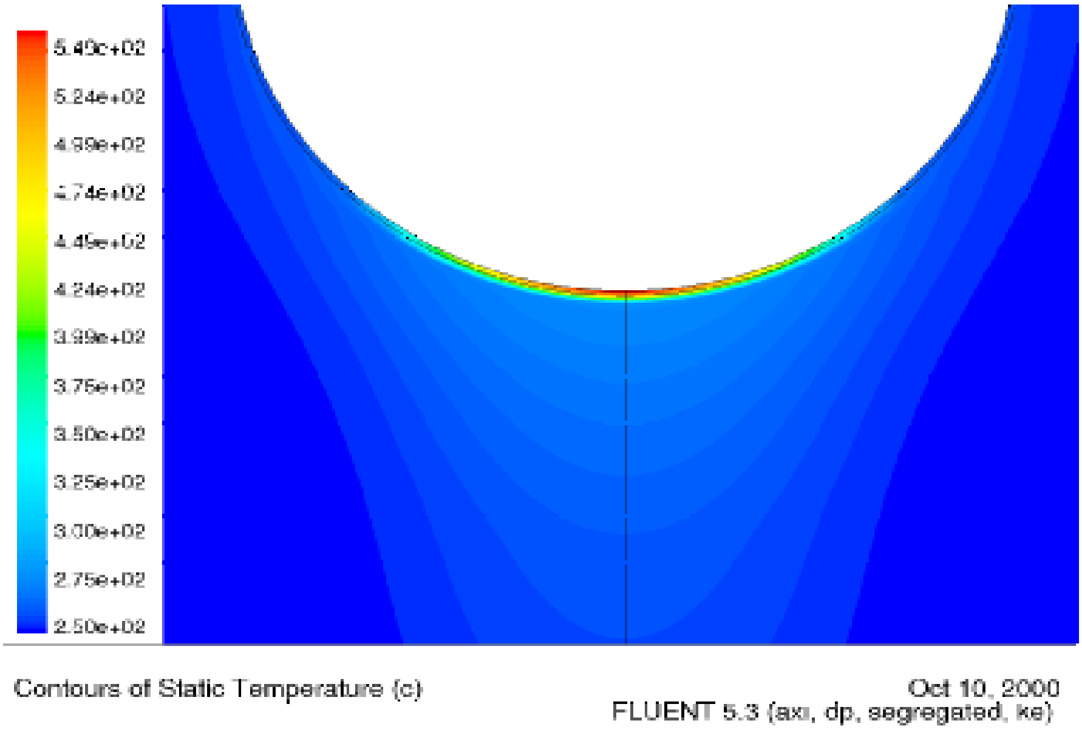

FIG. 15. Temperature distribution in target zone.

6. SUMMARY

The design goal of the HYPER system is to transmute nuclear waste. Much consideration has been given to maximizing incineration capability as well as power production efficiency. The HYPER core does have uranium as impurities. In order to lessen the reactivity swing due to the lack of uranium, burnable absorbers are being designed. Either alloy or dispersion type metallic fuel are being considered for the HYPER system. There are few experimental data available on TRU-Zr fuel. In addition, the development of Pb–Bi coolant/target technologies is also very challenging work. The HYPER system concept is to be finalized and key technical issues related to TRU-Zr fuel, and Pb–Bi coolant/target are to be derived by the end of 2000. The experiments to solve the key technical issues will be performed from 2001 to 2003 and the conceptual design will be completed by 2006.

REFERENCES

[1] PARK, W.S., et al., HYPER (Hybrid Power Extraction Reactor): A System for Clean Nucl. Eng. and Design, **199** (2000) 155–165.

[2] HWANG, W., et al., A Mechanistic Swelling Model of TRU Metal Dispersion Fuel for HYPER, Proceedings, Korean Nuclear Society Fall Meeting, 26-27 October 2000, Daejon, Republic of Korea (2000).

[3] LEE, B.W., et al., Modeling and Preliminary Analysis on the Temperature Profile of the (TRU-Zr)-Zr Dispersion Fuel Rod for HYPER, Proceedings, Korean Nuclear Society Fall Meeting, 26–27 October 2000, Daejon, Republic of Korea (2000).

[4] SONG, T.Y., et al., Transmutation Technology Development, KAERI/RR-2117/2000.

STUDY ON ADS Pb (Pb/Bi) SPALLATION TARGET

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Abstract

The neutron yield and double differential cross section of outgoing neutron produced in p+Pb spallation reaction up to 1.6 GeV have been investigated by some theoretical. Residual radioactivity, energy deposition and radiation damage of the Pb target were studied as well. A verification facility commented of a swimming pool reactor driven by a pulsed low energy high current accelerator (150–300 MeV/3mA) is planned in the next phase of our project. Pb–Bi eutectic is one of the options as the target material in the facility. In order to investigate the material compatibility and the thermohydraulics of Pb–Bi eutectic, a simplified small-scale loop is under designing.

1. INTRODUCTION

Since the nuclear power system of the accelerator-driven sub-critical facility (ADS) has obvious environment and source benefit, the nuclear energy scientists and engineers in China pay fully attention on the study of the ADS nuclear power system. Through the earlier stage study for several years, certain progress has been achieved. Some research achievements are published in the paper collection [1]. It is the basic research stage for the physics and technology of ADS system before 2004. The research work in this stage consists of four parts, which form a complete research system (Fig. 1). The first objective of this research plan is built-up of the comprehensive principle verification facility, which has the engineering practice significance, in the decade ahead. Some research achievements and work plan on the studies of the target physics and material will be introduced briefly in this paper.

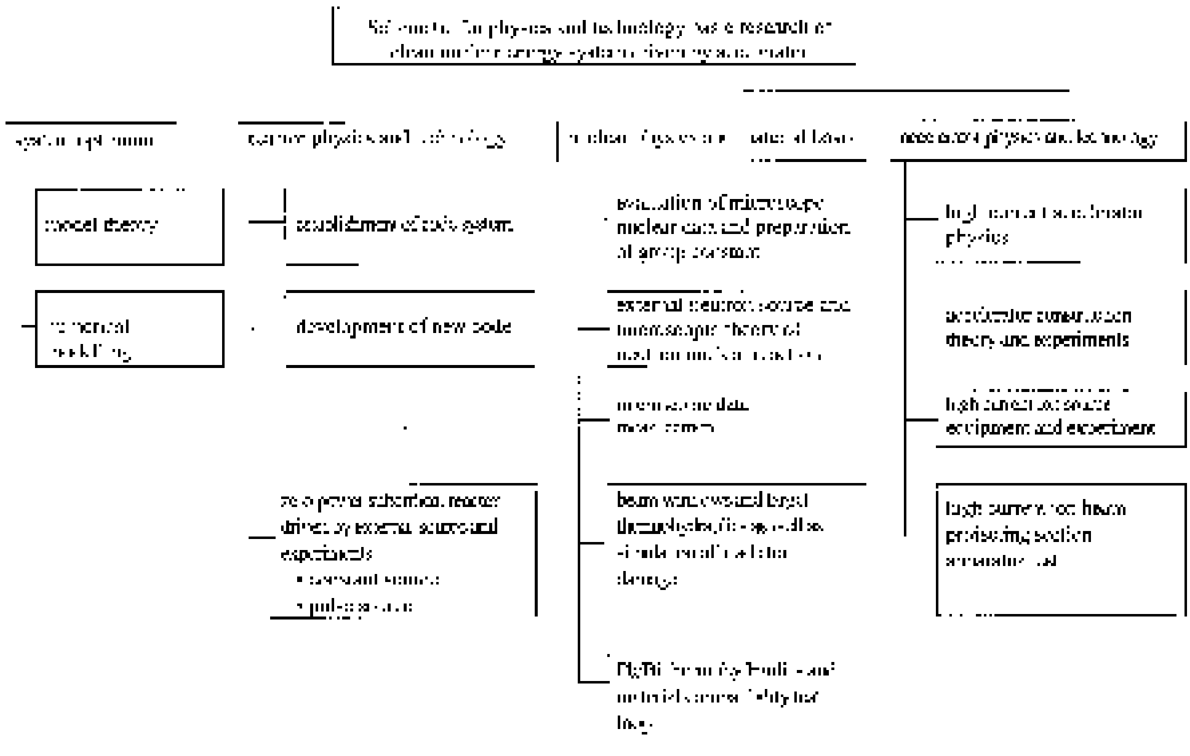


FIG. 1. Schematic of ADS basic research.

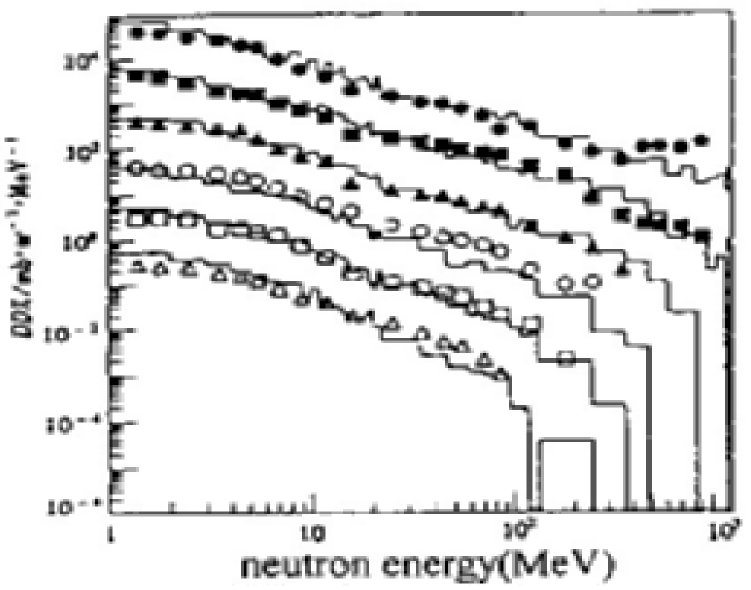


FIG. 2. Double differential cross section of neutron yield from Pb–Bi target.

● 15° ▲ 30° (× 10) ▲ 60°(× 1.0) 90°(× 0.1) □ 120°(× 0.01) 150°(× 0.001)

2. STUDY ON SPALLATION TARGET PHYSICS

The proton beam tube and the spallation target are the combination part of the accelerator and the sub-critical reactor, which is the important component of the ADS. The physical study on the spallation target is one of the basic studies of the ADS design. From the point of view on the engineering application, the basic problems such as spallation neutron yield, energy deposition, radiation damage, and radioactivity accumulation are firstly concerned.

The calculations and analyses to the thin target and the standard thick target were made based on the nuclear reaction models in this paper.

2.1. Study of the neutron double differential cross sections of thin target

The comparison of the neutron yield between the calculation results and the measured data of thin target can be used to distinguish that the theoretical models are reliable or not. Figure 2 shows the comparisons of the neutron double differential cross sections induced by 1500 MeV protons between the calculated values by the SHIELD code [2] and the experimental data [3] at 15°, 30°, 60°, 90°, 120°, and 150°. In summary, the predicted neutron double differential cross sections by SHIELD code is better than HETC code. The quantum molecular dynamics (QMD) model and the neutron double differential cross sections of the $p+^{208}Pb$ reaction for 600, 800, and 1500 MeV incident proton energies were calculated [3]. The calculated results for 800 and 1500 MeV are in good agreement with the experimental data, but the calculated results for 600 MeV incident proton energy and the small angle region (30–60°) are not good in accordance with the experimental data. The relative study work is in progress.

2.2. Study on standard thick target

The size of the standard thick target used in the international code comparison is Φ 120 × 60 cm cylinder. The lead and tungsten targets were chosen and two kinds of the calculation methods were used in our calculations. One is that the particle transport in the medium calculated by the Monte-Carlo method and the low and intermediate energy nuclear data taken from the experimental data or the empirical formulas based on the experimental data [4] another is by the SHIELD code. The comparisons and analyses of the calculated results by above two kinds of calculation methods and the study on the target physics are in progress.

2.2.1. Spallation neutron yield

The neutron yield per incident intermediate proton is the most interesting data for the practical application in the ADS study, which is related with the incident proton energy and the target material. Table I presents the neutron yield per incident proton for different target nuclei and different incident energies, it can be seen that our calculated results are in accordance with the results obtained in the international code and the experimental data.

TABLE I. COMPARISON OF NUMBERS OF NEUTRONS GENERATED PER PROTON FOR Φ 120 × 60 cm TARGET

Target	$E_{p.in}$/MeV	Average value from int. code comparison	Experimental values	This work
^{186}W	800	19.95 [9]		18.62
^{208}Pb	800	17.71 [9]	(16.6 + 8.9)/2 = 17.75 [10], 16 [11]	17.72
^{208}Pb	1000		19.5 [11]	22.74
^{208}Pb	1600		(33.5 + 37.8)/2 = 35.65 [10]	35.59

2.2.2.Energy deposition

Energy deposition is closely related to the target structure and the design of the cool system. The deposited energy in the spallation target comes from the loss energy of the incident protons since their reactions with the target nuclei and the reaction energy caused by the changing of the masses. The main physical mechanism of the energy deposition is: (1) ionization loss energy caused by that the incident protons and the secondary charged particles ionize the target atoms; (2) recoil energy of the target nuclei; (3) fission energy of the target nuclei. The magnitude of the contribution by the three kinds of the physical mechanisms is related with the target material and the incident proton energy. Figure 3 presents the calculated energy depositions by SHIELD code for 6 kinds of target materials with 1 GeV incident protons and Φ 120 × 60 cm cylinder target. The calculated results are compared with the experimental data [4] It can be seen from above results that the calculated values agree with the experimental data pretty well for those target material, which atomic order Z is larger, such as depleted uranium, lead, copper and are in bad agreement with the experimental data for those target material, which atomic order Z is less, such as carbon, beryllium since the number of the nucleons in the light nucleus is little and the statistical effect is not good.

The calculation results show that the ionization energy as well as the deposition energy increases with the atomic order Z increasing. The deposited energy in the depleted uranium target is 5 times of the deposited energy in the lead target since the fission energy would be released from the threshold fission reaction of ^{238}U. It means that taking the fissile nuclide, as the target material is not good considering the conduction heat out from the target region.

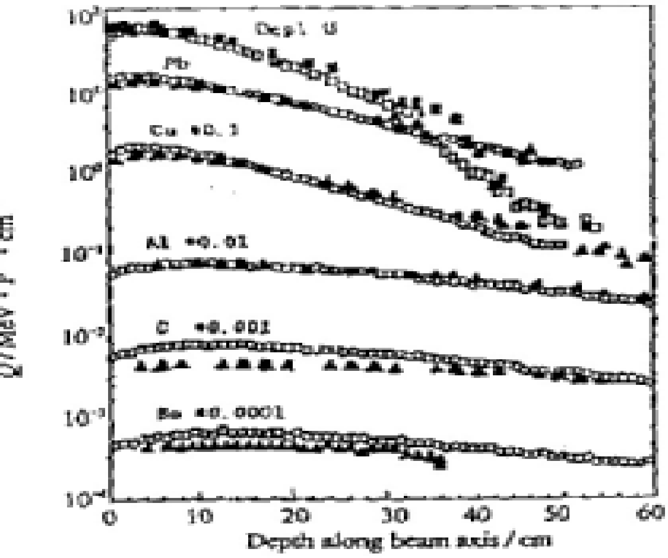


FIG. 3. Comparison of energy deposition in various targets at proton projection.

○ □ *calculated by SHIELD code;* ▲ ■ *measured by Belyakov-Bodin V.I.*

2.2.3. Radiation damage

The radiation damage problem of the spallation target in the ADS system is very serious than the normal reactor material because they undergo the intensive radiation by the intermediate energy protons and the secondary neutrons. Therefore study on the radiation damage of the target and the beam window has important significance for the life of the beam window and the safety of the ADS system. The reaction mechanism of the material radiation damage induced by the intermediate energy nucleons is mainly the elastic and inelastic scattering reaction, which can lead to the radiation damage effects, such as the atomic displacement of the target material, production of the gas in the light nuclei (mainly H_2 and He), etc. The radiation damage effect can be denoted by the radiation damage cross section and the production rate in the gas of the light nuclei. The radiation damage of the material tungsten and lead, which are given priority to the target material choice, was calculated by the SHIELD code. In order to compare with the previous calculated results, the thin target was calculated firstly. Figure 4 shows the comparisons between our calculated results by the SHIELD code and the calculated results [5] by the HETC code for $E_p = 600$ MeV as well as the calculated results [6] given by C. Rubbia group for $E_p = 800$ MeV. All calculated results in Fig. 4 show that the radiation damage cross sections increase basically in a line as the atomic order Z increasing.

The radiation damage of the standard thick target Φ 120 × 60 cm cylinder with 1600 MeV incident protons was calculated by the SHIELD code. Based on the thin target calculations, the total radiation damage induced by the incident protons, secondary protons, secondary neutrons above and below 14.5 MeV is mainly concentrated in front 30 cm at the target, the radiation damage in last 30 cm at the target is about 1/8 of the total one. The contribution to the total radiation damage by the secondary neutrons below 14.5 MeV is about 42%, that by incident protons is about 30%.

The production rate of the gas is the product of the particle flux and the production cross section σ_i of the gas. The calculated results by the SHIELD code show that under the above 400 MeV protons or neutrons bombard, the production cross sections of the helium gas increase rapidly with the incident energy for tungsten target, but slowly for iron target. Because the intermediate energy proton (p, He) cross section of the heavy nucleus is larger than that of light nucleus. It means that the heavy element should be chosen as the beam tube and window.

Obviously the life of the beam window and the solid target can be predicted by the combination of the radiation damage code described by the product of the radiation damage (including the production of the gas) cross section and the particle flux and the engineering design standard. This problem will be studied in the future.

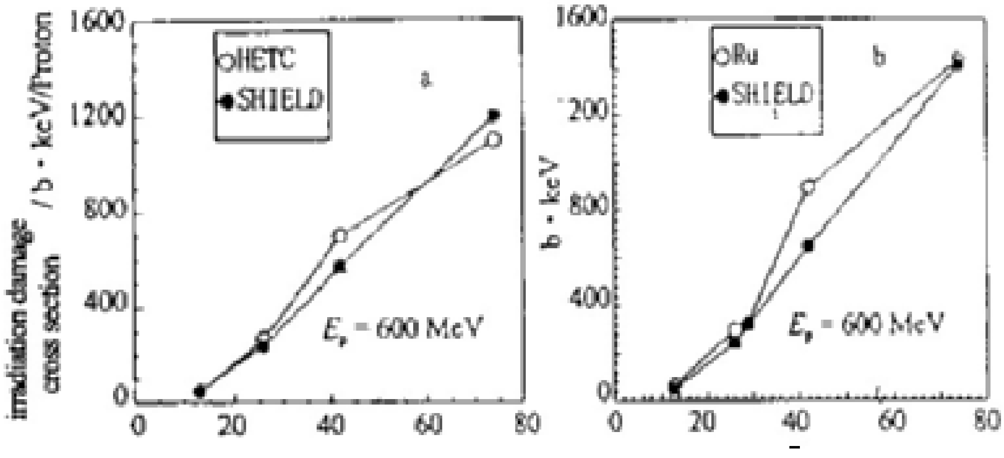


FIG. 4. Irradiation damage cross sections of Al, Fe, Mo, and W thin targetsinduced by incident protons with 600 MeV [5] and 800 MeV [6].

2.3. Study on the target radioactivity accumulation

While the protons bombard the Φ 120 × 60 cm cylinder solid lead target, the low energy neutrons will be produced and the radioactive nuclides will also be accumulated in the target. The accumulation of the radioactivity in the target was calculated after operation with 30 mA accelerator proton beam for one year. The situation for 1600 MeV incident protons was calculated. Figure 5 shows the relation of the total, neutron, proton, and neutron below 20 MeV radioactivity with time, respectively. Figure 6 presents the relation of the total radioactivity and the radioactivity of the several typical radioactive nuclides with time. The radioactivity is mainly caused by protons and neutrons (especially above 20 MeV neutrons) for 1600 MeV incident protons. The relation of the radioactivity with time is in basically accordance with the results for 1600 MeV incident protons [8]. The situation for 150 MeV incident protons was also calculated and the calculation results show that the radioactivity of the solid tungsten target for 150 MeV incident protons is lower 50 times than those for 1600 MeV incident protons. Figure 7 shows the relation of the total, neutron, proton, and neutron below 20 MeV radioactivity with time, respectively. Figure 8 presents the relation of the total radioactivity and the radioactivity of the several typical radioactive nuclides with time. It is interesting that the radioactivity for 150 MeV incident protons are mainly caused by protons, and ^{195}Au has no contribution to radioactivity in target. Furthermore the ratio of the magnitude of the radioactivity caused by protons and neutrons is also related with the size of the target.

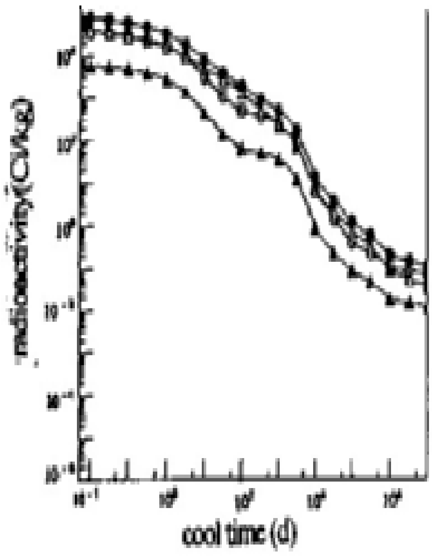


FIG. 5. Changes of total radioactivity and radioactivity induced by neutron, proton and neutron with E <20 MeV.

beam flux energy 1600 MeV.

● total radioactivity;

○ induced by neutron;

Δ induced by proton;

▲ induced by neutron with E<20MeV.

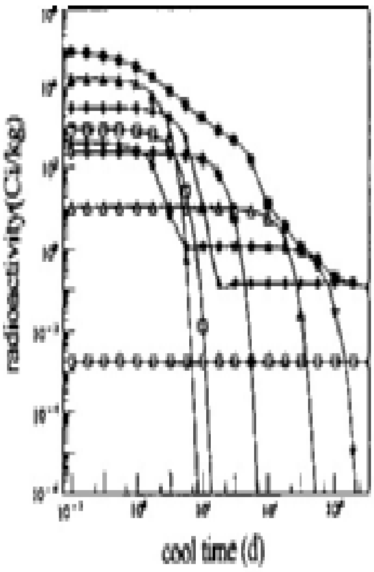


FIG. 6. Changes of total radioactivity and radioactivity from some typical radiation elements.

beam flux energy 1600 MeV.

● total radioactivity;

▲, ◊, □, , ✞, Δ, and ○, from ^{203}Pb, ^{202}Pb, ^{202}Tl, ^{206}Bi, ^{194}Au, ^{194}Hg, ^{193}Au, ^{207}Bi, and ^{205}Pb.

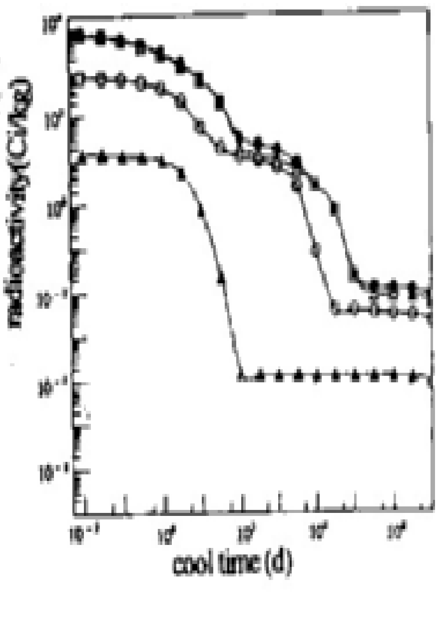


FIG. 7. Changes of total radioactivity and radioactivity induced by neutron, proton and neutron with E < 20 MeV.

beam flux energy 150 MeV

● total radioactivity;

○ induced by neutron;

Δ induced by proton;

▲ induced by neutron with E < 20MeV.

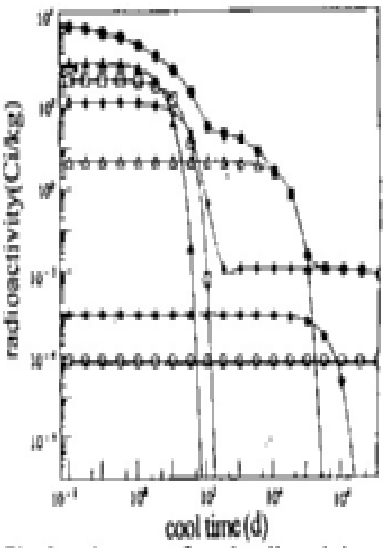


FIG. 8. Changes of total radioactivity and radioactivity from some typical radioactive nuclides.

beam flux energy 150 MeV

● total radioactivity;

▲, ◊, □, , ✞, Δ, and ○, from ^{203}Pb, ^{202}Pb, ^{202}Tl, ^{206}Bi, ^{194}Au, ^{194}Hg, ^{193}Au, ^{207}Bi, and ^{205}Pb.

3. CONSTITUTION OF VERIFICATION FACILITY AND APPLICATION CONSIDERATION

The ADS principle verification facility according to plan takes the pool light water reactor in the China Institute of Atomic Energy (CIAE) as the core and a proton accelerator with appropriate proton energy and beam intensity and a neutron production target in the centre of the reactor are assembled. The reason to choose the pool reactor is that it is existed and easy to change its core structure, such as changing the cell constitution in the reactor centre or placing the heavy material surrounding the target, so that two regions with different average neutron energies are formed. The average neutron energy in the heavy material surrounding the target is high and in the region far from the target is low since they are moderated by the water, which provides the condition for the principle test of the ADS. After the particle beam goes out the accelerator, it can be led to the core of the reactor and it can be also led to the multiple function experimental region, the study on the neutron physics, nuclear property of the target, thermohydraulic, and material as well as the study and production of the radioactive isotopes can be carried out.

Figures 9.1, 9.2 and 9.3 present the whole arrangement of the facility, the constitution of the accelerator (here taking the linear accelerator as an example), and the cell constitution in the core of the reactor, respectively. The relative parameters of every parts of the facility are given in the figures.

The construction of the verification facility is an engineering practice for developing the ADS system; the engineering experience on combination of the accelerator and the sub-critical reactor will be obtained. It is expected that the studies on the target physics and target material, accelerator physics and its reliability, reactor physics and engineering can be performed in this facility. Above experiments and studies will become the basis of the physics and technology for the construction of ADS engineering demonstration system.

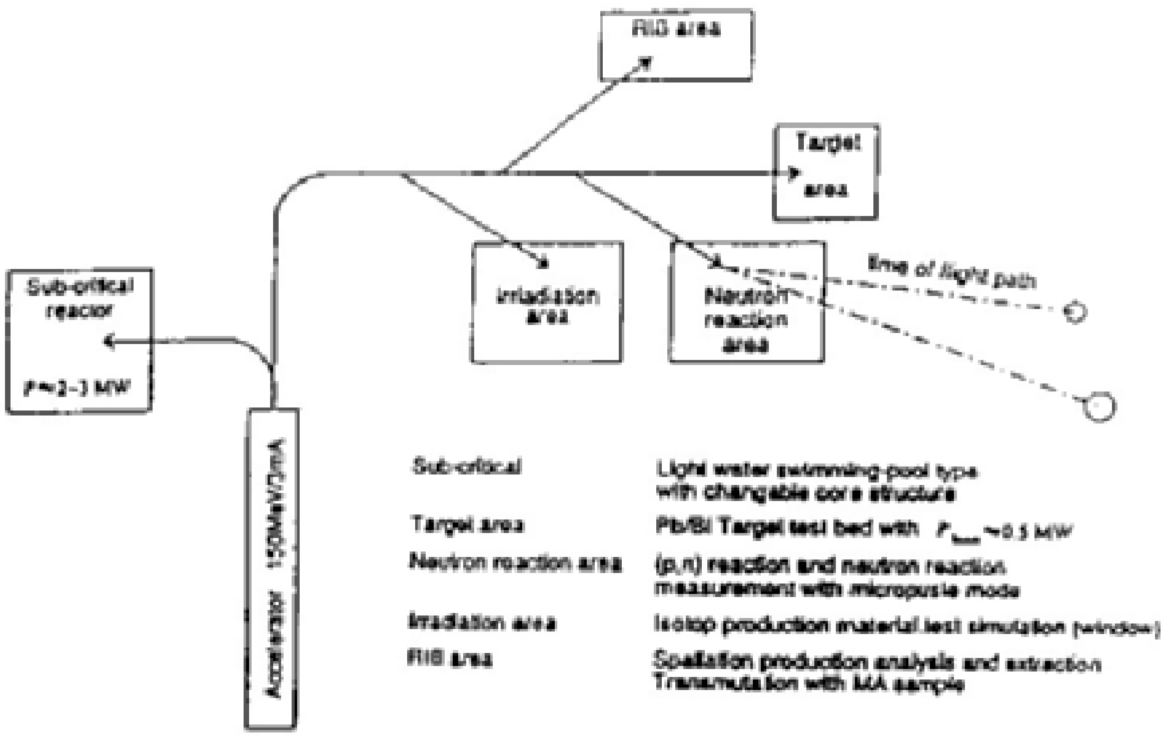
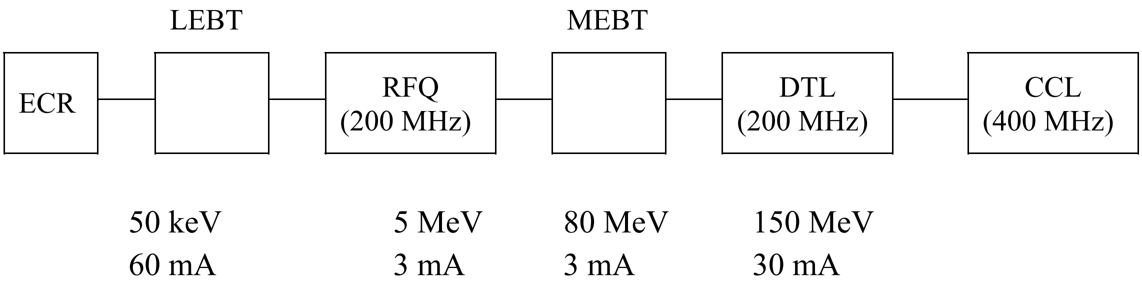


FIG. 9.1. Conceptual layout of verification facility for ADS.



a) Pulse mode
Repletion rate: 100 Hz
Duty factor: 10%
Average current: 3 mA

b) Micropulse mode
Pulse width: ~10 µA
Average current: ≤ 10 µA

FIG. 9.2. Arrangement of high current accelerator with low energy pulse operation.

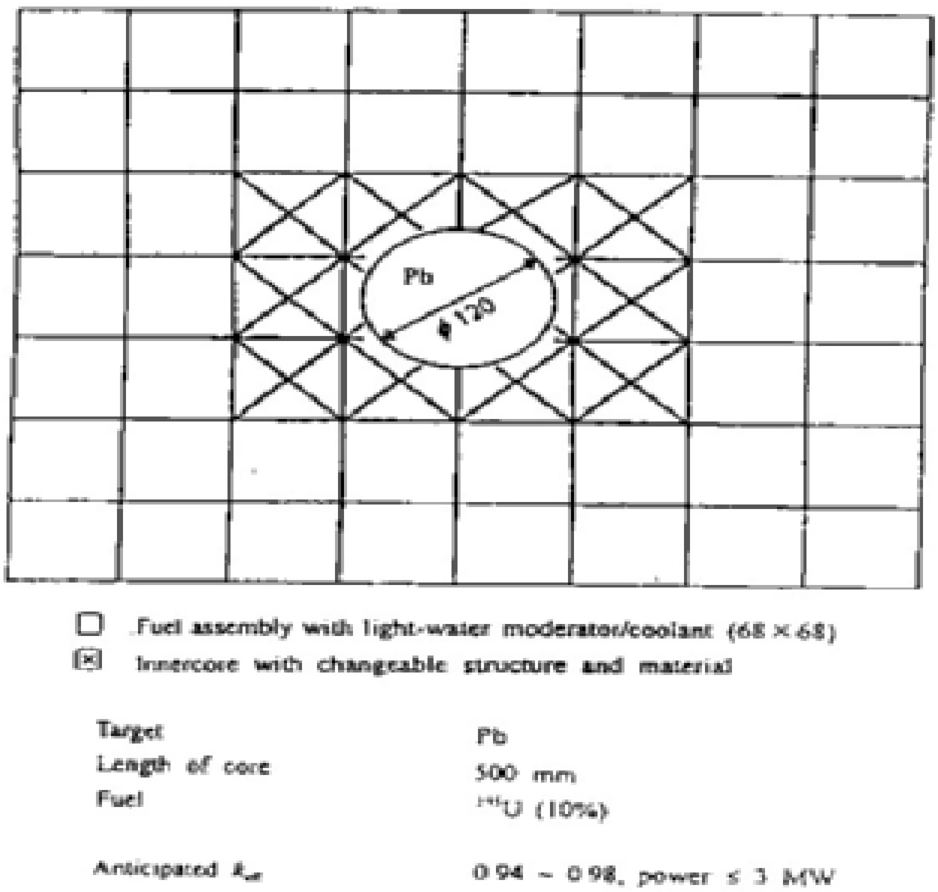


FIG. 9.3. Arrangement of pool light water reactor core.

4. INVESTIGATION OF CONSTRUCTION MATERIALS AND THERMAL PRODUCTION

Pb–Bi eutectic as the optimization material of the target and coolant for ADS has been generally acknowledged in nuclear energy circles [12]. However, liquid Pb–Bi eutectic is corrosive considerably to construction materials, and can be contaminated and oxidized easily by the impurities. The magnetic residues and oxides will cause plugs and stoppages of the circulation, thereby reduction of the system life.

These problems should be concerned and solved if Pb–Bi eutectic will be used as the target and coolant material.

The investigation on the construction materials and thermal properties of the liquid Pb–Bi eutectic system is a part of the program of ADS physics and essential technology. The following studies are being planned.

4.1. Compatibility of construction materials with Pb–Bi eutectic

4.1.1. Scanning of corrosion resistant to high temperature static Pb–Bi eutectic for the home-made materials

Evaluating the corrosion resistance of the homemade materials with the solubility rate in Pb-Bi eutectic, liquid metal embrittlement, corrosion morphologies and depth, one or two materials will be optimized, and their mass transfer effects will be studied further.

4.1.2. Tentative idea for construction of a small scale Pb–Bi thermal convection loop

For liquid Pb–Bi eutectic system, the mass transfer, especially temperature gradient mass transfer will be the most damaging type of all materials degradation phenomena. The thermal convection loop may offer several advantages as the mass transfer study apparatus, the small size and simplicity of design result in low cost and ease to fabrication. So that a number of tests can be performed and thereby many variables can be evaluated, and a considerable amount of information can be given. Hence, many thermal convection loops were constructed in come countries [12–17], a small scale Pb–Bi thermal convection loop is being thought of design in our institute now. The tentative idea of this loop is shown in Fig. 10, and following parameters of the loop are being considered:

Construction material:	Low carbon steel or 2.25Cr-1Mo steel
Size of loop:	High limit 90 cm and 60 cm wide
Max. operation temperature:	650^{o}C
Temperature gradient:	about 200^{o}C
Velocity of Pb–Bi eutectic:	some cm/s

According to the operation experiences from some countries [12, 16–18], the impurities in Pb-Bi eutectic rather than corrosion products appear to be the major factors resulting in plugging of the system pipe. Therefore, a trap containing a permanent magnet may be used in our thermal convection loop to alleviate early stoppages. The Pb–Bi will be melted into the loop without pretreatment, but will be followed by the addition of the magnesium and zirconium as the deoxydant and inhibitor, respectively.

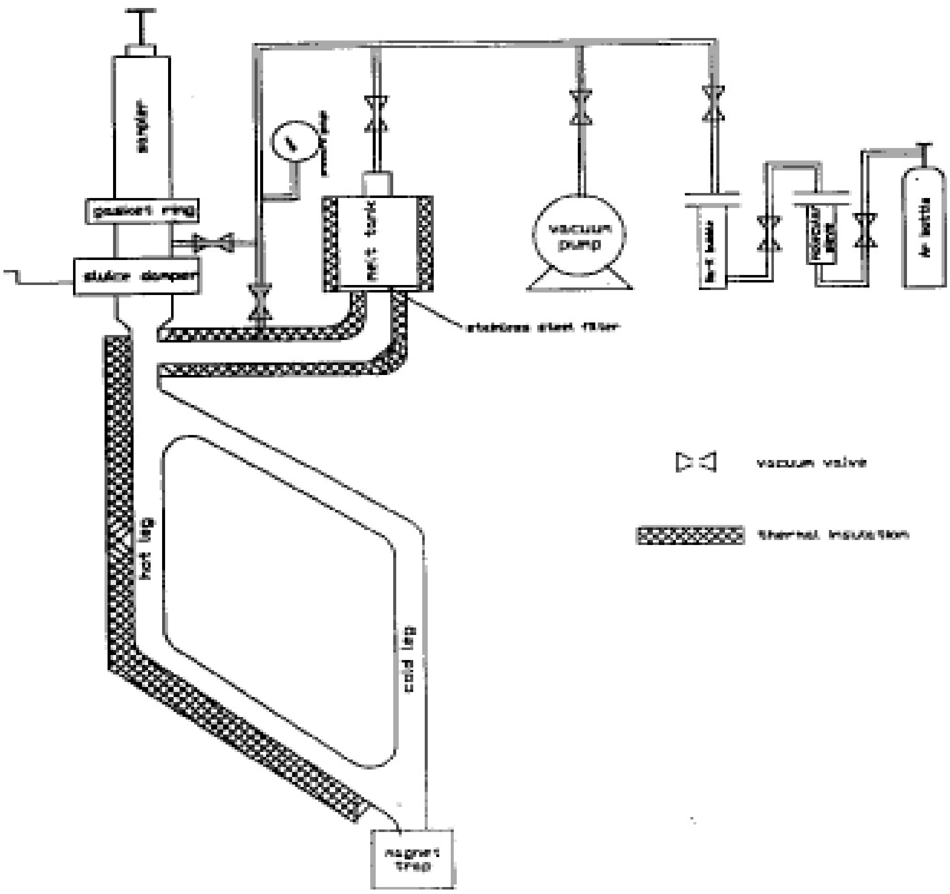


FIG. 10. Schematic diagram of Pb–Bi thermal conversion loop.

4.1.3. Contents and goal of the studies

- Making an investigation on effects of temperature and temperature gradient on mass transfer and liquid metal embrittlement, to define an appropriate maximum temperature and temperature gradient for Pb–Bi/home-made materials system;
- Making an investigation on effects of deoxydant and inhibitor concentration on mass transfer, to determine an optimum matching of Mg and Zr concentration in our thermal convection loop;
- Investigating mass transfer for test time dependent, to get the solubility rate of materials versus test time, and provide the reference data for evaluation of the ADS system life;
- Making a study of mass transfer characters for welding of materials, to explore the optimum welding process for the materials used in Pb–Bi eutectic.

4.2. Proton and neutron irradiation effects for construction materials

Because of the construction materials will be exposed to a significant flux of high current proton and high-energy neutron, the materials should be good performance under an intense proton and neutron environment. Following irradiation tests will be performed for some homemade materials.

4.2.1. Equipment

Using existent HI-13 tandem accelerator and cyclotron in CIAE and heavy ion irradiation simulation tests to investigate the irradiation effects of the proton beam, neutron beam, mixture beam of proton and neutron with different energy.

4.2.2. Study content and goal

- Making an investigation on macroscopic property changes and microscopic mechanism after irradiation for the materials, such as defects, swelling, fatigue, creep, plastic, embrittlement and resistance;
- Investigating irradiation effects for the particle energy, dose, temperature and material mechanical process dependent, to understand the improving methods on the irradiation resistant properties;
- To make an inference for life limit of the materials resulted from irradiation;
- Test and verification of the theoretical model.

5. CONCLUSIONS

- The results of the calculation and analysis on the target physics show that SHIELD code used in the study for the high energy can also be used for the investigation and design calculation of ADS target physics;
- In the view of the target cool, the heavy metals which may generate the fission reactions are not appropriate for using as the spallation target materials, though they have more neutron yields;
- The radioactivity accumulation in the spallation target will have an effect on the replacing of the target and the maintenance of the related components, this issue should be paid close attention;
- Due to the application properties of ADS are dependent on the material and physics character of the spallation target, the investigation on the material compatibility and thermal-hydraulics of Pb–Bi eutectic, irradiation resistance of the material as well are being planned together with the study and design of a verification facility in our ADS essential program.

REFERENCES

[1] DAZHAO, D., ZHIXIANG, Z., SHENG, F., QINGBIAO, S., et al., Proceedings, Symposium on Conception research of the clear nuclear energy system driven by accelerator, Beijing, Atomic Energy Publishing House, Beijing (2000) (in Chinese).

[2] MAO, X.C., et. al., High-energy physics and nuclear physics, **23**(4) (1999) 402 (in Chinese).

[3] SHENG, F., et al., Ph.D. Thesis, China Institute of Atomic Energy, Eur. Phys. J. **A4** (1999) 61–68 (in Chinese).

[4] BELYAKOV-BODIN, V.I, et al., Nucl. Instrum. Methods **A314** (1992) 508.

[5] LEDTELEV, S.G., et al., INR-0896/95-16 Moscow (1995).

[6] RUBBIA, C., et al., CERN/AT/95-45(ET) (1995).

[7] RICHARD, E., et al., LA-NR-3014 (1989).

[8] SHUBIN, Y.N., et al., Proceedings, 2nd Int. Conf. on ADTT Kalmar, Sweden (1996) 953.

[9] FIGLES, D., et al., NEA/NSC/DOC (95)2 (1995).

[10] Proceedings, ICANS-XI Report, KEK-90-25 (1990).

[11] LONE, M. A., et al., Nucl. Instr. Meth. In Phys. Research **A256** (1987) 135.
[12] PORK, J.J., BUTT, D.I., BEARD, C.A., et al., LA-13336-MS (1997).
[13] TAKEDA, Y., MANSUR, L.K., DAI, Y., et al., CONF-9603171 **P.A.1-A.3** (1996) 500.
[14] DISTEFANO, J.R, PAWER, S.J., DEVAN, J.H., ibid **P5.31-5.48**.
[15] CYGAN, R., NAA-SR-253 (1953).
[16] ROMANO, A.J., KLAMUT, C.J., GURINSKY, D.H., BNL-811 (T-313) (1963).
[17] CYGAN, R., FISKIN, J.M., IMLAH, K.J., et al., NAA-SR-1060 (1954).
[18] TOLSON, G.M., TABOADA, A., ORNL-3870 (1965).

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