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STATUS OF FAST REACTOR ACTIVITIES IN RUSSIA

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ABSTRACT

This paper outlines state-of-the-art of the Russian nuclear power as of 1997 and its prospects for the nearest future. Results of the BR-10, BOR-60 and BN-600 reactors operation are described, as well as activity of the Russian institutions on scientific and technological support of the BN-350 reactor.

Analysis of current status of the BN-800 reactor South-Urals NPP and Beloyarskaya NPP designs is given in brief, as well as prospects of their construction and possible ways of fast reactor technology improvement.

Studies on fast reactors now under way in Russia are described.

CONTENTS

1. NUCLEAR POWER STATE-OF-THE-ART AS OF 1997

In 1997 in Russia there were 29 nuclear power units in 9 NPPs having total installed capacity of 21.242 GW, namely:

- 13 power units with vessel type VVER reactors (including 6 VVER-440 power units and 7 VVER-1000 power units);

- 15 power units with uranium-graphite channel type reactors (including 11 power units with the RBMK reactors and 4 power units with EGP-6 reactors);

- 1 power unit with the fast reactor (BN-600).

27 power units having total capacity of 19.242 GW were in operation in 1997, while 2 RBMK-1000 reactor power units were modified (one power unit in Leningradskaya NPP, and the other in Kurskaya NPP[1]).

The total amount of energy produced by the NPPs in 1997 was $108.286 \cdot 10^6$ MW· hours, this value being 99.5% of the previous year production.

Average load factor value for all NPPs was 58.2% in 1997 as compared to 58.32% value of 1996. Values of scheduled and unscheduled losses of electricity production in 1997 were respectively 29% and 13%, total losses amount being 42%.

The total number of incidents occurred on NPPs during 1997 was 79, only 3 of this number being safety related events. These three incidents are attributed to the first level of INES international scale.

Diagrams presented below are showing NPP incidents, including those safety related, over the previous 6 year period (Fig.1) and their correlation to the INES scale (Fig.2). These diagrams demonstrate stable tendency of NPP incidents number decrease.

During 1997 there were 18 unscheduled disconnections of the power units from the grid, 9 out of them being caused by the reactor scrams.

Operating reliability of Russian NPPs is third following Japan and Germany.

"Program of Nuclear Power Development in the Russian Federation for 1998-2005 Period and Prospects up to 2010" is now considered by the Russian Government.

According to this Program the following activities are planned to be completed until 2000:

- modernisation of existing NPPs in order to continue their safe operation;
- resumption of construction of the frozen NPPs.

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[1]Kurskaya NPP power unit was put into operation in December 1997.

Fig. 1. Trends of operational events at NPPs in Russia during 1992-1997

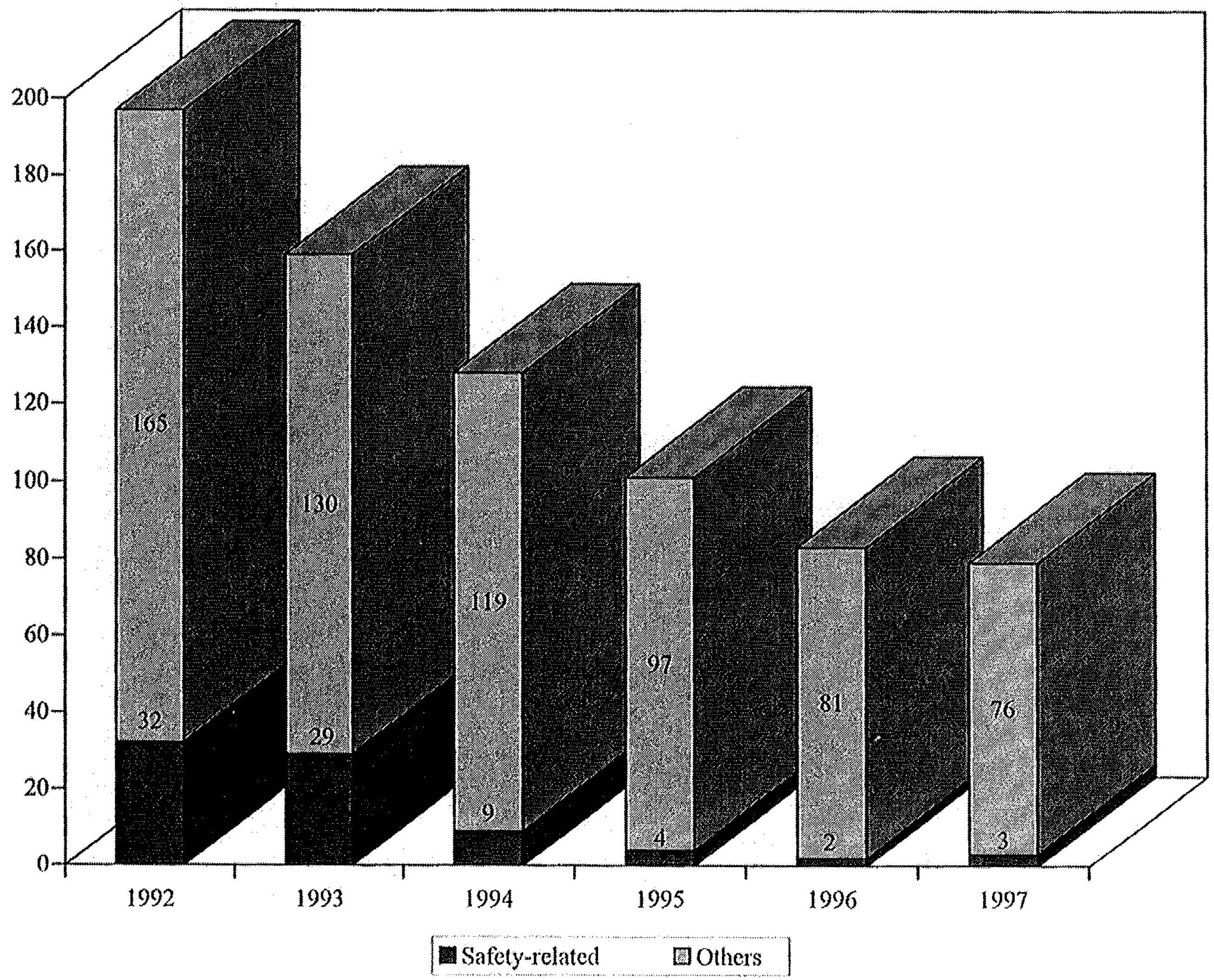
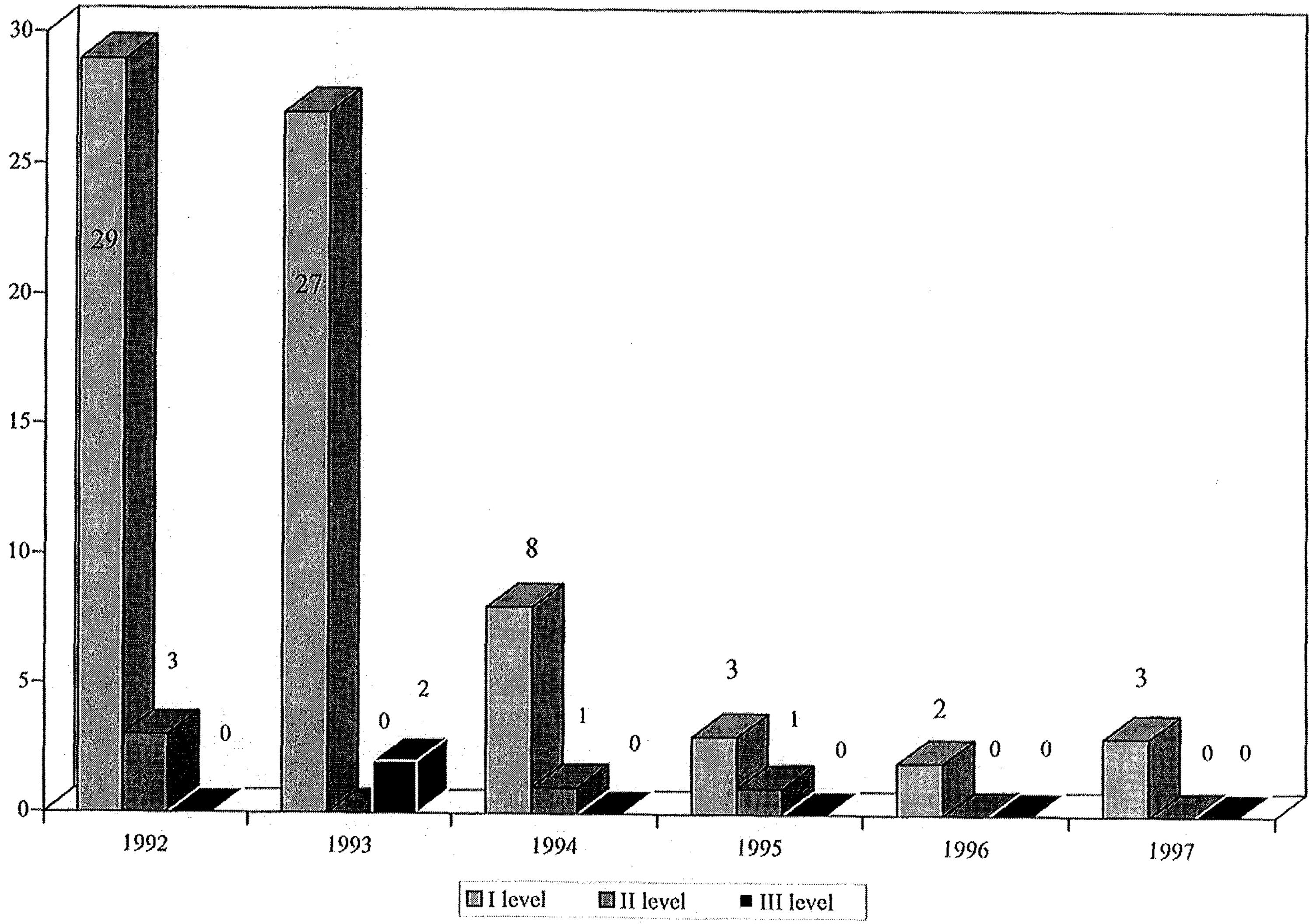


Fig. 2. Trends of safety-related operational events at NPPs during 1992 -1997



Construction of the front-end new generation power units with VVER-640 reactor (Sosnovy Bor) and VVER-1000 reactor (Novovoronezhskaya NPP-2) is to be completed by 2005.

Also, the license has been issued for the construction of the BN-800 reactor power units on Beloyarskaya NPP and South-Urals NPP.

2. FAST REACTOR OPERATING EXPERIENCE

Three fast reactors are currently in operation in Russia, namely BR-10, BOR-60 and BN-600 reactors. The BN-350 fast reactor NPP is operated in Kazakhstan under the scientific and technological support of the Russian specialists.

2.1. BN-600 reactor NPP

Histogram of the BN-600 reactor operation in 1997 is shown in Fig. 3.

There were two reactor shutdowns during 1997, caused by the core refuelling and scheduled maintenance works.

During 1997 no incidents occurred resulting in the abnormal operating conditions of the NPP, its load factor (73.0%) being the highest of those of all NPPs. Some reduction of the load factor value as compared to that of 1996 (76.3%) was caused by the increased scheduled maintenance period (90 days in 1997 as compared to 74 days in 1996).

The main characteristics of the BN-600 reactor over the previous 5-year period are presented in Table 1.

Table 1

Characteristic	Units	1993	1994	1995	1996	1997	From the start of operation up to 01.01.98
Electricity production	10^6 kW·hr	4220	4110	3695	4022	3835	65278
Heat supply	10^3 Gcal	–	–	258.7	312.8	313.5	2674.0
Load factor	%	80.3	78.2	70.31	76.3	73.00	70.00
Number of power unit shutdowns		2	2	3	2	2	79
Number of loop outages		0	0	6	3	0	66

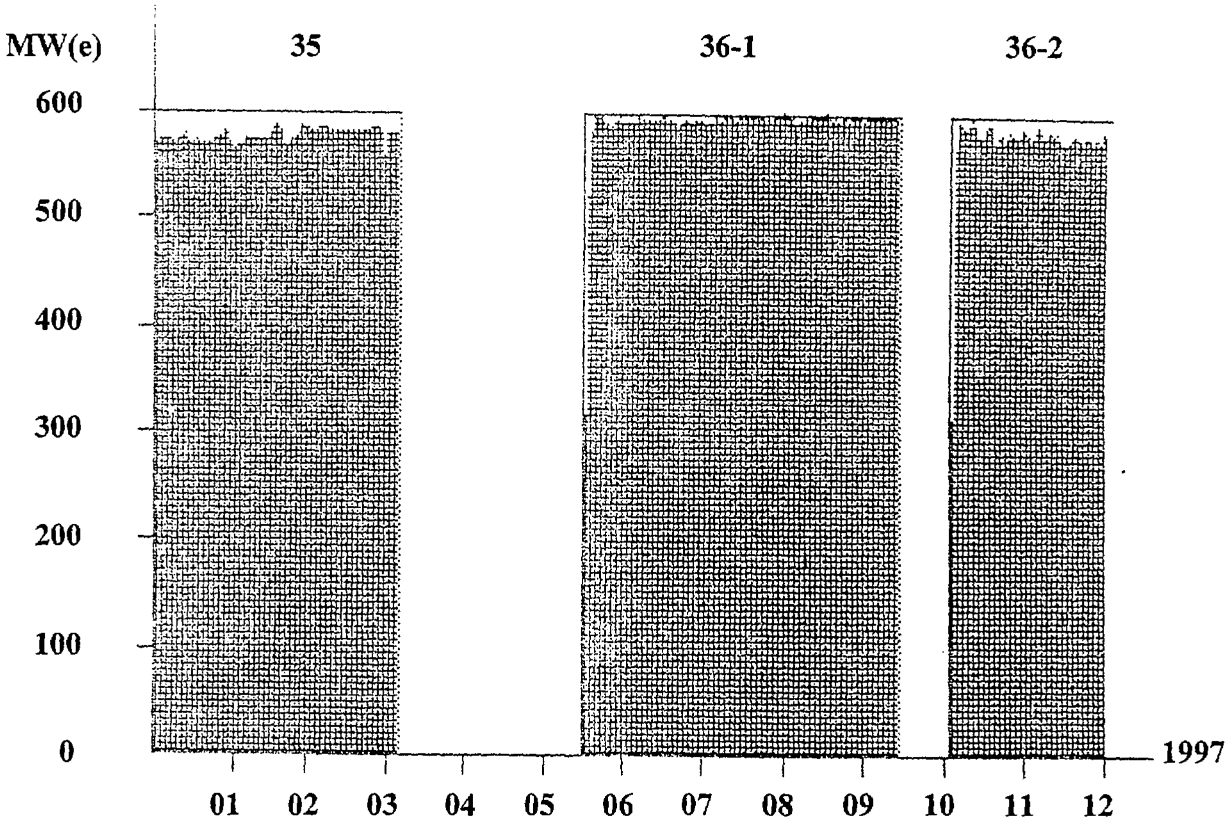


Fig. 3. BN-600 Operating Histogram

Crisis Centre has been established by "Rosenergoatom" Concern in order to render special support to the nuclear power plants of the Concern in the radiological dangerous situations. Local crisis centres are established at the NPP, and special engineering centres are established at some large research institutions for the support of the Crisis Centre. In particular, local crisis centre has been created at the Beloyarskaya NPP, and special engineering centres have been set up at the IPPE and OKBM to render timely support to the BN-600 reactor NPP in case of emergency.

In the framework of the establishment of these centres the following organising activity was fulfilled:

- expert groups were formed;
- on-line communication was organised providing transmission of ~200 parameters related to the NPP safety;
- total archives of the BN-600 reactor design and operating documents is being set up, some part of it being transformed into the electronic version.

The establishment of the engineering support centres is supposed to improve scientific maintenance of the BN-600 reactor operation.

2.2. BOR-60 experimental reactor

During 1997 the power of the BOR-60 reactor varied within the range from 48 MW to 55 MW.

The main operating parameters of the reactor for 1997 and for its whole life are presented in Table 2.

Table 2

1	2	3	4	5	6	7	8
Parameters	Units	1 quarter	2 quarter	3 quarter	4 quarter	1997	From the start of operation up to 01.01.98
Time of reactor operation on the power over minimum detectable level	hours	2146	1607	1632	1386	6771	153731
Reactor availability	–	0.99	0.735	0.743	0.627	0.774	

1	2	3	4	5	6	7	8
Maximum reactor power	MW	55	50	48	55	55	
Production of: heat electricity	 MW·hr	 112475 15098.4	 68480 11932.8	 60331 12556.8	 68909 10437.6	 310195 50025.6	 6359219 1042012.4
Time of steam generator operation: SG-1 SG-2	 hours	 2141 2141	 1595 1595	 1597 1597	 1374 1374	 6707 6707	 86343 39311
Heat supply of the consumers	Gcal	41481	10855	1189	19392	72917	321724

During 1997 tests of fuel, absorber and structural materials were in progress under conditions of high coolant temperatures and neutron flux.

Standard subassemblies with the vibropacked oxide fuel (UO_2, $UPuO_2$) were installed in the core.

Maximum fuel burn-up in the standard and experimental subassemblies was respectively 15.5 % h.a. and 26 % h.a. In some experimental fuel elements (3 fuel elements installed in a special demountable device ~32 % h.a. maximum fuel burn-up was achieved.

In one subassembly containing UPu-10 % Zr sodium bonded fuel elements ~8 % h.a. burn-up was achieved.

2.3. BR-10 experimental reactor

In 1997 the BR-10 reactor was in operation during 4674 hours on different power levels, the time of its operation on the rated power (7.6 - 8.0 MW) being equal to 2830 hours.

The following works were done on the reactor:

- target irradiation for the production of the isotopes required for pharmaceutical agents manufacture;
- irradiation of specimens of four types of structural materials used for the fuel element cladding;
- irradiation of the lavsan film by the fission products in the thermal neutron beam in order to produce hyperfine membranes;

- irradiation of oncological patients by the fast neutron beam.

During reactor operation on power permanent monitoring of the fuel element cladding integrity was carried out by measuring activity of delay neutron predecessors in the primary coolant and gaseous fission products in the cover gas of the primary pump vessels.

Four fuel element failures accompanied by the gas release were detected.

Fast neutron fluence (E_n>>0.1 MeV) on the reactor vessel was as high as $5.95 \cdot 10^{22}$ n/cm^{2,} design value being equal to $7.0 \cdot 10^{22}$ n/cm^2.

Development of basic design of the BR-10 test reactor decommissioning is under way.

2.4. Co-operation with the Republic of Kazakhstan on the BN-350 reactor NPP

Since the design lifetime of the BN-350 reactor was expired in 1993, permissions for continuation of the reactor operation are obtained annually.

Experts from Russian institutions participate in the activities on the improvement of safety systems and the NPP as a whole to meet the requirements of current regulatory documents, in the issuing of annual safety report, proving NPP safe operation in the next time period.

In 1997 maximum permissible power level of the reactor was 420 MW on condition that five loops are available for the reactor decay heat removal.

3. FAST REACTOR PROSPECTS

Fast reactor prospects can be divided into the near-term and long-term categories. The former includes plans for the BN-800 reactor NPP construction on Beloyarskaya and South-Urals sites (2005 - 2010), while the latter implies development of design of advanced NPP with the new generation fast neutron reactor having safety characteristics inherent to LMFRs and competitive engineering and cost parameters (2020 - 2030).

3.1. BN-800 reactor NPP

As it was mentioned above, a license for continuation of South-Urals NPP and fourth power unit of Beloyarskaya NPP with the BN-800 reactor was issued by the Russian regulatory authorities in December 1997.

The South-Urals NPP construction is oriented mainly to the budget financing, the Beloyarsk NPP 4 unit construction is planned to continue on joint-stock basis. The BNPP-2 joint-stock company has been created, the founders of which are Sverdlovsk region government, "Sverdlovskenergo" power concern, Rosenergoatom concern, machine-building, construction and installation enterprises of Sverdlovsk region, and BNPP.

In November 1997 agreement was signed based on the results of the meeting of the Minister of Atomic Energy and the Governor of Sverdlovskaya region on corrections to 'The Program of Development of Nuclear Power in Russia", and further it was corrected so that the construction of Beloyarskaya NPP fourth power unit with the BN-800 reactor should be completed by 2005.

3.2. Studies on advanced fast reactors

In spite of the current decrease in the rate of commercialisation process of fast reactors (FRs), there is a common opinion that their large-scale introduction would be required after 2030-2050 in order to provide fuel for the nuclear power [1]-[2].

Construction of single commercial NPPs with the fast reactors is possible in some countries during the previous period.

Analysis of the process and nature of FR development and their prospects would lead to the conclusion given below.

Now the first stage of sodium cooled FR technology development has been completed, the basic result of which is the demonstration of their satisfactory reliability and safety. This is a principal result showing the validity of the principal engineering solutions.

The tasks of the second stage which is previous to the large scale FR introduction into the nuclear power structure can be considered as follows:

- assimilation of the experience gained, improvements to be made on its basis and development of the new technologies in the FR area;
- development of FR designs having higher safety, reliability and technical and economical parameters;
- expansion of FR application area.

Among the innovative advanced FR technologies lead and lead-bismuth alloy coolants are considered.

In the process of analysis of different options some issues should be taken into account.

Experience has been gained on the use of lead-bismuth alloy in the submarines nuclear power plants. The possibility of this coolant application for the large scale nuclear power will be finally determined after the development of related technology taking into account the requirements to the commercial power engineering.

As regards the reactors cooled with lead, design studies have been made in this area, requiring related experimental approval.

As far as the sodium cooled fast reactors are concerned, it can be stated that their potential is far from its exhaustion. As it has been shown by the studies, technical and economical parameters of the sodium cooled fast reactors can be significantly improved on condition that the required reliability and safety are provided.

The development of the concept of advanced NPP design with the sodium cooled fast reactor has been initiated recently by the Russian specialists using positive experience gained on the development, construction and operation of the BR-10, BOR-60, BN-350 and BN-600 reactors as well as on the BN-800 and BN-1600 reactors design development.

In connection with the complexity of tasks lying ahead in scientific, engineering and economical aspects it could be rather useful to combine efforts of the countries-participants of nuclear community.

There is a need to create under the aegis of IAEA a long-term (7-10 years) work program, the objective of which will be the elaboration of agreed viewpoint on extension of application area, main directions of improvement of future NPP with fast reactors and the methods of their realisation.

This program including technical committee meetings, specialist meetings, conferences, fulfilment of joint R and D studies under agreed international programs etc will benefit the increase in international activity of the countries possessing fast reactor technology, as well as the involvement of countries interested in FR introduction into their future nuclear power.

The basis for the program mentioned realisation might be the scientific and engineering potential of France, Japan, Russia as well as United Kingdom, Germany and USA with participation of India, China, Korea and other countries.

It is believed that a gradual realisation of R and D studies in the framework of unified international program will allow to eliminate the major at present disadvantage of sodium cooled FR – higher cost as compared to thermal reactors. In doing so the scientific and engineering experience having been gained during 40 years in many countries will be used to the maximum extent which will noticeably decrease total expenditures for future NPP construction.

The FR prospects are as well rather promising if the expansion of their application is considered. Below the main possible FR applications are given.

1. <u>Industry:</u>

- production of high specific activity isotopes and unique isotopes obtained in fast neutron flux;

- fast reactor high-temperature heat utilisation for technological purposes (for example, production of motor fuels from coal and petroleum residue).

2. Energetics:

- fast reactor utilisation with extra long lifetime for autonomous and unattended small and medium power units.

3. Ecology:

- efficient plutonium utilisation (without breeding blankets, with MOX-fuel on the basis of increased enrichment (by plutonium) fuel or fuel without uranium-238 based on an inert matrix);
- efficient minor actinide burning (homogeneous and heterogeneous MA location in core and blankets when using different fuel types);
- use of FR for long-lived fission product burning.

4. R&D ACTIVITIES

In 1997 scientific research and design development works on many issues of the FR area were in progress. Below the brief description of some of these works is presented.

4.1. BN-600 reactor hybrid core design

Within the framework of utilisation of weapons plutonium stocks in Russia it is supposed to use the BN-600 reactor as plutonium burner.

For this purpose hybrid core design of the BN-600 reactor is under development.

This work was initiated in 1996 in the co-operation with the US Department of Energy and CEA of France.

The main directions of this work are as follows:

- development of the hybrid plutonium containing core design and comparative analysis of its characteristics with those of existing uranium fuel core;
- safety approval of the BN-600 reactor hybrid core, taking into consideration beyond design accidents.

The following requirements are to be observed in the BN-600 reactor hybrid core design:

- initial design of the core SA should be kept;
- zero value of the sodium void reactivity effect (SVRE) should be provided in accordance with the requirements of the Russian regulatory authorities.

The studies have shown that the latter requirement can only be met if some part of the uranium SAs are replaced by those containing MOX fuel with 21% of plutonium. High enrichment zone SAs are replaced in this case.

The studies were performed for two hybrid core designs, namely:

- core with fertile radial blanket containing depleted uranium dioxide, similarly to the existing design;
- core with non-fertile radial blanket made of steel and natural boron carbide.

Core design with the non-fertile blanket was chosen for the further development.

Both Russian and American computer codes are supposed to be used for the safety approval of the BN-600 reactor hybrid core. In this some American codes on fast reactor safety approval were handed over to the Russian specialists. Two training courses on SAS4A/SASSYS code for the Russian specialists were held at the ANL. By now the input data base on the BN-600 reactor hybrid core has been prepared for the SAS4A code calculation.

4.2. Training simulator centre for the BN-600 reactor

According to the TACIS program (1992 budget), works on the training simulator for the third power unit of Beloyarskaya NPP were initiated in September of 1996 by the Russian specialists in co-operation with the Siemens/Corys/Belgoatom consortium.

This simulator is intended for the modelling of all three circuits of the power unit, turbo-generator, electric systems and technological parameters automatic control system (TPACS). Neither auxiliary systems of the power unit, nor reactor refuelling system are assumed to be incorporated into the simulator. The simulator concept is based on the display interface supplemented with the mimic panel and the limited number of control keys.

By December 1997 the following works had been performed (and accepted by the NERSA supervising organisation):

1. Package of technological parameters on all simulated technological systems.
2. Models of all technological systems, namely:

- reactor - three-dimensional synthetical analysis of the core neutron profile, calculation of the core thermal characteristics using channels averaged over 9 radial flow profiling zones, 5 vertical zones and 3 angular zones;
- primary circuit - one-dimensional thermohydraulic calculation;
- secondary circuit - one-dimensional thermohydraulic calculation taking into account transfer of both dissolved and gaseous hydrogen, with partial modelling of he argon gas system and sodium tanks;

- steam generator - 4 sections of one half of one steam generator (SG) are simulated, while the second half of this SG as well as the halves of all other SGs are simulated as the single sections; modelling of all operating modes, from SG filling with water to the rated power operation, is provided in two-phase approach;
- third circuit - one-phase one-dimensional thermal hydraulics analysis of heat exchangers, turbo-generator and some auxiliary systems required for the start-up and shut-down operations;
- electrical systems - modelling of turbo-generator, auxiliary power supply systems and relaying devices connecting power unit with the grid;
- TPACS - modelling of the main system logic and main types of abnormal operation in all systems.

3. Approval procedures with the description of power unit operation during ~ 100 hours of the main transients.

The assembly of the simulator was planned to start on Siemens site in January 1998. However the work has been suspended because of the lack of the financing.

4.3. ULOF accident benchmark for the BN-800 type reactor

Comparative analysis of severe accidents in the BN-800 type reactor went on in 1997 using input data prepared at the IPPE.

Specialists from Germany, France, Italy, Japan, India and Russia are involved in this work, which was supported by the IAEA and CEC. Mr. G. Van Goethem and Mr. A. Rinejski have contributed much to the organisation and fulfilment of this work.

In the previous year reports were prepared on the steady state core pre-accident condition (FZK, Karlsruhe is responsible), input data on the core neutronics for the accident analysis (IPPE, Obninsk is responsible) and on the initial stage of the accident (IGCAR, Kalpakkam is responsible).

It should be noted that the work carried out in the framework of the group mentioned above, has made considerable influence on the software development for the fast reactor safety analysis in the Russian Federation. The experience gained by the West European, Japanese and Indian specialists, and their critical approach to the results obtained turned out to be useful and valuable for the Russian specialists.

It is very important to keep the fruitful co-operation of the large group of scientists formed during six years. The Russian party applies to the IAEA and CEC to continue support of this group.

4.4. Studies on the unauthorised withdrawal of absorber rod

In the framework of co-operation between Minatom (RF) and CEA (France) works are carried out on the improvement of the existing approach to the approval of safety in case of the absorber rod unauthorised withdrawal (ARUW) accident in the LMFR type reactor.

Modification of the approach to this design basis accident is made in two directions, namely:

- development of methods and related tools for identification of suspicious SAs and related values of maximum linear power as a function of time duration of the ARUW accident;
- analysis of conservatism extent in the assumptions used for the development of algorithm of calculation of the probability to avoid fuel melting in the end of the accident.

On the basis of RBR-3D three-dimensional code for the reactor analysis calculation tool has been developed at the IPPE in order to form the maps of potentially dangerous SAs in the ARUW accident, changes of maximum linear power in time being registered.

Evolution of the power profile during the reactor run is evaluated taking into account both movement of the absorber rod lattice and change of the fuel isotope composition. This approach allows decreasing uncertainty of the maximum linear power value.

In order to verify developed method of analysis of the accident under consideration, experiments on its modelling were carried out on the BFS facility.

The work on this method improvement is continued.

4.5. BFS critical facility studies

During the first half year of 1997 works went on at the BFS-2 critical facility on the preparation of experimental studies on the simplified model of the BN-800 reactor core with the increased plutonium content and non-fertile blankets.

In the second half-year critical assembly creation was started, its completion being planned for the first six months of 1998, when the experimental studies are to be initiated.

In the beginning of the last year, the standard part of the experimental program was completed on the BFS-71-2 critical assembly of the BFS-1 critical facility. The insert containing about 14% of neptunium dioxide replacing partially removed depleted uranium dioxide was places in the central part of the assembly. The results of measurements are being processed and analysed.

Plutonium content in the BFS-71-1 and BFS-71-2 critical assemblies MOX fuel is about 55%.

Starting from March studies were in progress on the BFS-1 critical facility using preliminary (conceptual) model of the KALIMER Korean fast reactor under the contract with the KAERI. Single-zone model of the reactor was assembled on the basis of 20% enrichment metal uranium fuel, and the following characteristics were measured:

- criticality;
- central spectral indices (using different methods and detectors);
- central reactivity coefficients of samples of different reactor materials;
- radial and axial distribution of fission numbers;
- Doppler effect on the samples; etc.

Results of measurements have been handed over to the customer.

Tests were completed on the FCA critical facility in Japan carried out within the framework of the international Program of studies on the efficient delay neutron fraction. After the results are summarised they will be presented at the special meeting.

In the framework of the program of control, accountancy and physical protection of nuclear fissile materials operation and modification of some elements of the system, created at the BFS facility continued.

4.6. Sodium fire studies

Studies of sprayed sodium fires were under way. These tests were carried out either in the closed chamber or in the ventilated room, their volumes being respectively 8 m^3 and 220 m^3. Ventilation ratio was about 5 hr^{-1}.

The sodium outflow parameters were as follows:

- sodium outflow velocity 2 m/s;
- sodium temperature 500°C;
- amount of outflowing sodium 10 kg;
- outflow hole diameter15 mm.

The results obtained are of preliminary nature. The experiments are in progress.

In addition, works continued on the development of analytical methods of the sodium fire studies, in particular, on modification of BOX and AERO computer codes.

4.7. Analytical and experimental studies of the coolant boiling

Several tests were carried out to study the boiling process of liquid metal coolant (sodium-potassium eutectic) and its stability using fast reactor SA model under natural flow conditions.

Experimental model is shown schematically on Fig.4.

The methodology of the experiments was based on the increase of power rating of the SA with stationary coolant which would result in appearance of the coolant natural flow within the closed circuit. As the SA power increased the coolant temperature also increased reaching saturation temperature in the heated area and coolant boiling started.

Tests were carried out for mass flow velocities varying within the range from 20 $kg/m^2 \cdot s$ to 700 $kg/m^2 \cdot s$, heat flux density reaching 260 kW/m^2 value.

It was determined that three boiling modes existed, namely bubble, slug and annular-dispersion modes. The latter is the ultimate boiling mode, providing SA cooling.

As it has been shown by the experiments, the slug boiling mode is characterised by the fluctuations of the coolant flow rate within the range limited by values typical for the bubble mode and the annular-dispersion mode. When changing from the bubble mode to the annular-dispersion mode about three time increase of the coolant flow rate occurred in the circuit. When the heat flux exceeded 250 kW/m^2 value, the coolant flow rate decrease was detected in the circuit accompanied by transition from the annular-dispersion boiling mode to the dispersion mode (heat transfer in the beyond-crisis area).

The analysis of the boiling stability of the coolant natural flow in the circuit has shown that there are two most typical kinds of dynamic thermohydraulic flow instability, namely:

- instability, observed in the region of low outlet vapour content values, when the variations of the gravitation related component of the pressure drop value are of great importance;
- instability, observed in cases of high and intermediate vapour content values, i.e. when the key component of the pressure drop is that caused by friction.

On the stage of tests preparation and planning and their methodology development computer code was worked out for the analysis of the liquid metal boiling dynamics. It was based on the one-dimensional mathematical model of two-phase flow in the closed circuit in one-dimension approach taking into account phase sliding.

Both qualitative and quantitative similarity of sodium-potassium eutectic and sodium thermophysical characteristics makes it possible to transfer experimental results, obtained for the eutectic, to the sodium.

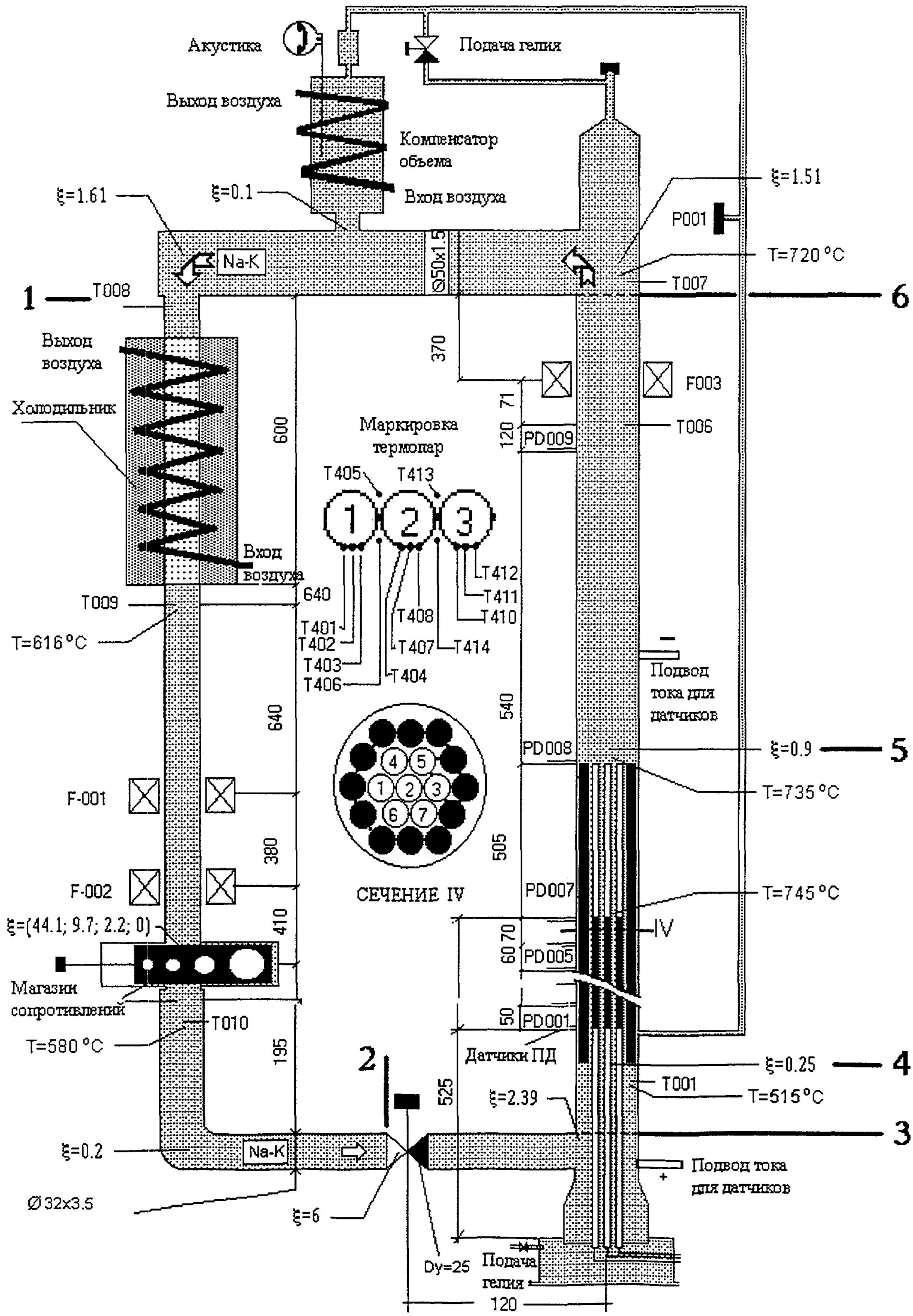


Fig. 4. Scheme of test facility for coolant boiling

Continuation of both experimental and analytical studies is planned in order to provide detailed and comprehensive approval of the possibility of stable heat removal from the fast reactor core under conditions of the natural convection flow of the boiling coolant that can take place in the reactor decay heat removal circuit in case of beyond design accident.

4.8. Studies on modelling of decay heat removal processes in fast reactor vessel

Studies are carried out at the IPPE on the development of theoretical foundations of using water for the modelling of the liquid metal coolant natural convection flow in the upper plenum of fast reactor in the decay heat removal mode. In particular, work is under way on the modification of the approximate modelling method proposed in [3], and development of the new approach to its basic principles interpretation.

In order to verify some principles of water models application for thermohydraulics studies of the sodium natural flow, rather simple experimental rig made of the acrylic plastic was constructed (Fig.5), on which the following studies are carried out:

- testing the possibility of method [3] application for studying transients in the stable natural convection flow;
- checking influence of the Reynolds number change (accompanied by the similar change of the Peclet number) on the modelling results and determining causes of the self-similarity appearance when certain values of these numbers are reached [4];
- determining relationship between the combined convection and the stable natural convection flows in various designs of immersed heat exchanger, etc.

Experimental studies carried out at the IPPE have confirmed the possibility of modelling process of transition to the developed natural convection for sufficiently high Reynolds number values. Analytical studies have shown the possibility of rather significant errors caused by the application of experimental data for the evaluation of the real reactor characteristics, if the hydraulic resistance coefficient is not taken into account as a function of Reynolds number according to the method [3].

It has been shown that the difference between resistance laws for the reactor and the model is caused by some modelling problems.

The works on this issue are in progress at the IPPE.

Water model tests are planned to study the reactor decay heat removal processes occurring in the reactor vessel as a whole, including those in the core, taking into account inter SA gap coolant flow, upper mixing plenum, intermediate heat exchangers and decay heat removal heat

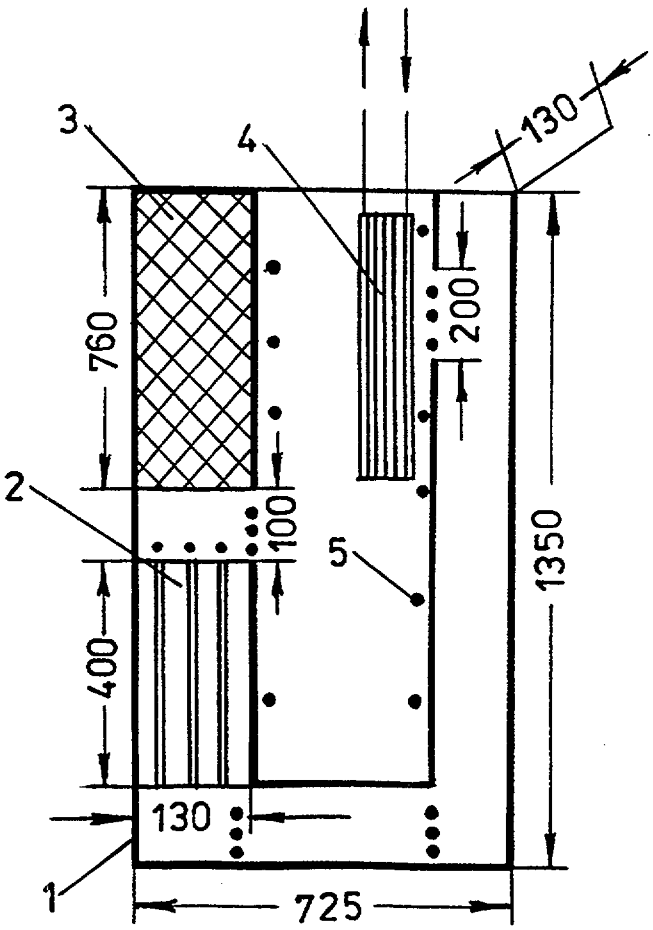


Fig. 5. Scheme of water experimental rig

1 - vessel,	2 - 16 heating rods,	3 - "displacers",
4 - cooler,	5 - thermocouples	

exchangers, coolant path from the intermediate heat exchanger to the core diagrid, and the reactor vessel cooling circuit.

In addition to the criterion-based approach to the modelling of decay heat removal process in the reactor vessel, three-dimensional computer code is supposed to be used for the coolant temperature and flow patterns evaluation.

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