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MATHEMATICAL, PHYSICAL & ENGINEERING SCIENCES

The Present Status of Fast Breeder Reactors in the U.S.S.R. [and Discussion]

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The present status of fast breeder reactors in the U.S.S.R.

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Because of the present conditions – the rate of growth of nuclear energy is slowing down in the U.S.S.R. – the large-scale deployment of commercial LM fast breeder reactors (LMFBRS) is being postponed. Notwithstanding that, experience has been gained through research and development and through the operation of FBRS, and both grants planning and the implementation of the next steps for the promotion of LMFBRS are under way. A brief analysis of the current status and prospects for further development of the FBR programme is given.

1. Nuclear power and fast breeders

The nuclear industry in the U.S.S.R. is at present going through some hard times. The Chernobyl disaster has aroused a wide public renunciation of nuclear power and a distrust of arguments for its safety. A torrent of emotional pseudo-scientific discourse has gushed into press, television and radio programmes, nourishing distrust of both specialists and scientists.

Still, one should admit that striving for rapid development of nuclear power has been accompanied, at times, by an incomplete accounting for all factors important for the proper choice of nuclear power plant (NPP) sites. More stringent regulations, concerning the siting of NPPS, has led lately to the abandonment of several construction sites.

Many opponents of nuclear power argue that more intensive energy conservation may ensure not only a decrease in demands on nuclear energy, but also on conventional energy sources as well. These and other factors act positively against nuclear power. Other factors, nevertheless, are in favour of the further deployment of the nuclear energy, and these factors have been and still are quite imperative and do not lose their importance.

Estimates of primary resources, outputs and energy production show that even with a significant decrease (by a factor of about two) of the specific energy consumption related to the gross national product, the nuclear energy constituent in the balance of primary energy should inevitably grow to the year 2010, from *ca.* 4% to *ca.* 16% (Kruglov 1989). The share of oil and gas (together) should decrease to *ca.* 65%; the share of coal may slightly increase (from *ca.* 21% to *ca.* 24%). Seemingly, any inadequacy in the fulfilment of the energy conservation programmes will necessitate some additional increase of the nuclear energy share in the balance of the primary energy in the country. To slacken the pace of nuclear energy now may create difficulties in the future.

At the beginning of 1989, installed capacities of NPPS amount to 34.7 GW_e. Electricity production for the year 1988 totals 215×10^{11} kWh, or 12.7% of the total electricity production in the country.

Prognosis of the installed capacities for the year 2000 is still unstable. The Minister for Atomic Power gives a figure of 110–120 GW_e as a feasible and indispensable level, adequate for the future demands on electricity production. His estimates take into account installed capacity renovation, upgrading and modernization of the existing fossil power units, progress in energy conservation, etc.

Equally, there are also very conservative opinions arguing that in the year 2000, nuclear energy capacities might total a significantly lower figure, between 55 and 80 GW_e. Anyway, even the highest estimate differs drastically from forecasts done in the beginning of the 1980s. In those days capacities of 200 GW_e, and even higher, had been mentioned.

This changing situation implies that the time when economy of uranium will become an urgent need and a wide-scale construction of LMFBRS will become indispensable is postponed. On the other hand, this means that before the series production of FBRS there is time for a critical analysis of the experience gained, for a thorough selection of new solutions and for implementation of additional R&D programmes. Though, from the point of view of fast reactor enthusiasts, the most efficient way to improvement is to construct workable – rather than 'paper' – power units, to obtain practical experience, even if not on a large scale.

2. BN-350 and BN-600 reactors

The BN-350 reactor has been in operation for almost 16 years now. For a long time the facility represented a significant source of energy for the region. Presently, because of the growth of energy supplies on the Mangyshlak peninsula, the importance of the reactor as a source of energy has declined, and experimental activities and investigations have begun to play the major role at the reactor. (See table 1.)

Table 1. Main characteristics of BN-350

thermal capacity/MW		750
sodium temperature/°C		
primary circuit	inlet	280
	outlet	430
secondary circuit	inlet	256
	outlet	415
steam temperature/°C		410
steam pressure/atm[a]		48

[a] 1 atm ≈ 10^5 Pa.

During the past few years the reactor has operated with a high load factor, which is about 90% in the steam-generation mode.

The originally accepted lifetime of most components of the reactor equipment has already been exceeded. Presently, the task of residual lifetime assessment for major components and the reactor as a whole has been set. It should be taken into account that the reactor design was begun in 1960, and this design followed general-purpose industry requirements and old nuclear safety standards.

The modern, stringent safety requirements now necessitate the fulfilment of a number of serious arrangements. A comprehensive study of the design, with regard to its compliance with contemporary standards, is now being carried out, and on the basis of this analysis a decision will be taken about the prospect for the reactor's future operation.

The 44th run of the core is now close to its end (each run lasts for about 100 days). The peak burn-up achieved in the driver core is about 9% heavy atoms (HA), with no failure of fuel elements.

Our aim is to achieve peak burn-up of 12% HA, and we plan to change over to ferritic–martensitic (13% Cr) steel for the fuel assembly wrapper. Assemblies with wrappers of such steel have been tested successfully in the BN-350, and two assemblies (254 pins) were tested with fuel claddings of this steel.

Tests of fuel assemblies with mixed-oxide fuel (MOX) manufactured by various technologies are being carried out.

We have recently come across problems with the steam-generating part of the reactor. In January 1989, two leakages of sodium into water occurred in steam generators of CSSR design, which had operated for 46000 and 63000 h, respectively. The loops involved were cut off and drained. At the moment the reactor is operated with four loops, at 520 MW_{th} power level. What can be stated now is that the origin of these failures is corrosion; nevertheless, neither the intrinsic correlations of the event, nor its causes have yet been investigated in detail and the overall picture of how these defects occurred has not yet been constructed.

The BN-600 reactor has been in operation for nine years. This power unit operates solely in an electricity production régime. By 1 January 1989, the power unit had produced 29.95×10^9 kWh_e and in 1988 its power production was 4.04×10^9 kWh_e (a load factor of 76.5%). (See table 2.)

TABLE 2. MAIN CHARACTERISTICS OF BN-600

thermal capacity/MW		1470
electrical capacity/MW		600
sodium temperature/°C		
primary circuit	inlet	355
	outlet	550
secondary circuit	inlet	320
	outlet	520
steam parameters		
temperature/°C		505
pressure/atm[a]		140

[a] 1 atm ≈ 10^5 Pa.

The average load factor of the reactor for the period until 1 January 1989 is about 66.5%; for the year 1988 it was 76.5%, as has been mentioned above. The average efficiency factor in 1988 was 41.6%.

Up to February 1989, 21 runs of the core had been completed (the length of a run is about 165 full-power days). The modernized core operating now (Lukonin 1988) exhibits good performance. By using cold-worked austenitic steels for both cladding and wrapper, the peak burn-up achieved is 8.3% HA (67250 MWd t^{-1}) without fuel failures.

The next stage will be a transition to ferritic–martensitic steel as the wrapper material, to achieve burn-ups of 10 and 12% HA. Experimental fuel assemblies with this wrapper material and advanced austenitics for cladding have been tested in the BN-600 up to burn-ups of 11% HA (90 displacements per atom). Tests of vibropacked MOX are carried out in the BN-600 core as well.

During the routine shutdown period in the summer of 1988, maintenance of the machine hall equipment, inspection and servicing of other components and equipment were performed. Specimens were cut from evaporating tubes. Corrosion damage of a 'pitting' nature was found; however, the depth of such defects was, in general, less than 0.3 mm. The inspection allowed us to prolong the originally accepted service life of the evaporators from 50000 to 75000 h. A further programme of sampling and inspections is planned, to establish permissible limits of the service life of components.

During the February 1989 shutdown, a routine cleansing of one of the loops was performed. Normally, one loop is cleansed during each shutdown; by experience the interval between such operations is 14000 h. In total, 10 cleansings have been done since the commissioning of BN-600. The operation includes two stages: removal of copper deposits and dissolution and removal of iron oxides. The operation is carried out without sodium drainage from the circuit.

Good radiation conditions are permanently maintained at the facility. Average individual annual exposure in 1988 was at the level of about 0.2 rem (2 mSv). Releases of fission product gases into the atmosphere did not exceed 1.5 Ci (55 GBq) per day. More often the releases were less than some tenths of a curie per day.

Seismic analysis and development of an auxiliary emergency heat removal system with air heat exchangers are being carried out.

The BN-600 reactor has been included in the list of facilities, proposed to the International Atomic Energy Authority (IAEA) for the application of international safeguards. The IAEA has already informed the U.S.S.R. about its intention to apply safeguards to the facility. Some preliminary discussions have already taken place, with regard to various aspects of safety at the facility.

To facilitate the development of safeguards for FBRS, the U.S.S.R. carries out a number of programmes in addition to those of the IAEA. Also note that the BOR-60 research reactor is in the list too, and it is used for IAEA inspectorate training purposes (an advanced training course for Agency inspectors is held at the facility on a regular basis, together with studies on non-destructive assay applications in the safeguarding of nuclear materials).

3. Research work at laboratories and experimental reactors

(a) *Fast-reactor physics*

The basic processes of neutron interactions with fast-reactor materials are now well understood. Less reliable is the knowledge of nuclear constants for abnormal hypothetical situations and for calculations of some isotope transmutations. Further work is necessary for the reconciliation of estimated nuclear data files and group constants of various detailed levels.

It is necessary to upgrade the accuracy of fast-reactor-core calculations, by means of non-diffusion two- and three-dimensional methods of analysis, both for statics and dynamics.

Critical assemblies are still allowing us to define more accurately some effects and constants. For example, investigations of absorption in nickel and chromium disclosed that the neutron absorption in these elements was overestimated, by comparison with figures accepted earlier. The same situation was discovered when absorption in stable fission products was investigated.

With regard to measurement metrologies, a comparison of methods for reactivity measurements under the influence of certain essential spatial effects was performed in cooperation with specialists from the Argonne National Laboratory (U.S.A.) and Cadarache (France), as well as comparative measurements of reaction rates in the 'Mazurka' assembly. A joint programme on B_{eff} measurements on the large critical assembly (BFS) is envisaged, together with specialists from France and the G.D.R.

Researches on a large heterogeneous core continue at the BFS assembly. In the near future most efforts there will be given to safety problems. Some time and resources will also be given to modelling metal fuel cores.

(*b*) *Thermophysics and hydraulics*

Experimental and theoretical studies of heat-exchange phenomena in regular and deformed pin bundles over many years has resulted in development of efficient methods and codes for modelling fuel assembly thermohydraulic behaviour, followed by development of thermo-mechanical methods of calculations of fuel assemblies with radiation-induced deformations. In 1988, the regulatory *Methodological guidelines and recommendations for thermohydraulic calculations on fast reactor cores* was issued. Methods of detailed calculations of the thermohydraulic behaviour of heat-exchanging equipment are being developed as well. Investigations on flow characteristics in diagrid pressure chambers are in progress.

The similar problems should be solved also for dynamic régimes. In progress is research work on transitory régimes in reactors, in particular for cases of emergency shutdowns accompanied by possible boiling of the sodium. For the moment, some new research work on the process of the sodium boiling is envisaged. The role of spacing wire may be very essential, as the wire itself is a potential centre for intensive vaporization. The presence of such centres may influence the initial margin in case of sodium overheating. This research work also imposes some serious experimental and calculational problems.

(*c*) *Sodium technology*

All principal problems in the area of sodium technology seem to be resolved. Systems developed for monitoring sodium impurities operate successfully, as do devices for leakage detection in steam generators. The same is true for sodium purification systems. Procedures for washing equipment of residual sodium and for decontamination of the primary circuit components have been mastered at all sodium reactors.

With regard to the technology of large non-isothermic sodium circuits, one might note that some more attention should be paid to the processes of deposition and accumulation of impurities in 'back-water' zones, cold parts of the circuit, in cover gas, and to their future behaviour. Kinetics of impurities should also be studied at temperatures higher than operational ones, to find their behaviour in abnormal situations.

Non-sampling coolant monitoring is well worthwhile being developed and upgraded. Prevention of sodium leakages into the environment and sodium fires are problems, the solutions of which will require even more inventiveness from designers and technologists.

Decommissioning of nuclear power plants will pose, with regard to FBRs, a specific problem of utilization or safe disposal of primary circuit sodium. This task is also accounted for in our plans.

(*d*) *Work at the BR-10 and BOR-60 reactors*

In January 1989 the BR-10 reactor had its 30th anniversary. Twice in its lifetime the reactor has undergone modernization. The second upgrading involved the replacement of the reactor vessel and some primary-circuit components.

The BR-10 reactor has proved to be a good instrument for in-pile material testing under controlled loading. A tradition of the BR-10 reactor research programme is the study of the transfer of radioactive impurities in the primary circuit. Biologists from the Obninsk Institute of Medical Radiology have been working at the reactor for many years. Since last year neutron therapy of cancer patients has been carried out with the fast neutron beam. More than 130 patients have received effective courses of neutron treatment.

The reactor has been operated on nitride fuel since 1983. The planned burn-up (8.0%) has been exceeded and is now 8.3% HA.

Recently, the processes of fission product release from various fuels with artificial defects and the product's migration through the circuit have been studied. A total of 12 experimental assemblies with faulty fuel elements with MOX and mixed carbide fuels, uranium nitride and metallic uranium have been studied. Various types of defects (holes, slits) were introduced in different parts of the fuel elements. These studies of activity release dynamics for the delayed neutrons' predecessors, gaseous and solid fission products, provide the possibility for developing models of propagation of activity in the circuit and its monitoring.

Since 1988, a special experimental loop in the primary circuit has been operated; it incorporates a tritium detector and instrumentation for studying the behaviour of corrosion products.

The BR-10 reactor is used for irradiation of materials, testing of various instruments and activation analysis.

The BOR-60 reactor provides, mainly, a base for material irradiation studies. It is generally accepted that new materials are subject to testing in this reactor before being cleared for tests in the BN-350 or BN-600.

In the BOR-60 reactor, fuel elements with vibropacked MOX fuel have undergone full testing. This fuel, together with the pellet-type fuel, will be further tested in the BN-600 reactor.

A large-scale testing programme was carried out on structural materials, in particular on low-swelling ferritic steels.

At this reactor, caesium traps with graphite sorbent for sodium purification were tested and mastered. A certain grade and grain size of the graphite were selected and the purification process conditions were determined. Similar equipment was developed and used for sodium purification at the BN-350 and BN-600 reactors.

The BOR-60 reactor has been in operation for nearly 20 years. Proposals for its updating are now under consideration.

4. The development of the BN-800 and BN-1600 reactors

The BN-800 reactor design is based on many solutions adopted earlier for the BN-600. The approach adopted both to succession and variations in the design has been presented at various conferences (Troyanov *et al.* 1987; Anon. 1988).

Briefly, the approach and its results can be summarized as follows.

In the vessel of the same size as of BN-600, 2100 MW_{th} instead of 1470 MW_{th} can be produced. The vessel design was somewhat refined in general. In-vessel shielding was reduced in mass by taking into account results of the BN-600 studies and operation. The core size was increased as compared with the BN-600 (516 assemblies instead of 369).

Emergency cooling after a loss of power and water supply is ensured by air heat exchangers. Such a solution is also to be realized for BN-600 when it is updated. In the heat removal system no intermediate steam superheating (reheating) by the hot sodium is provided. Secondary superheating is performed in the separator–reheater by the steam coming out from the turbine.

Each steam generator section consists of two modules – evaporator and superheater – the same pearlitic steel being used for the superheater and the evaporator. To ensure the pearlitic steel steam generator's performance, the temperature of the secondary sodium was slightly reduced as compared with the BN-600 (505 °C against 520 °C). The steam temperature is also somewhat lower (490 °C instead of 505 °C).

In the BN-800 only one turbine is used. Working drawings of the BN-800 are already at the 'Atommash' fabrication plant. Construction work has started at the Beloyarsk and South-Ural sites. For the moment, work is being carried out on a small scale.

The BN-1600 reactor is considered to be the prototype of future commercial reactors. In 1988 the results of its development were reviewed and it was decided to extend its design approach with the aim of finding some additional solutions to provide higher safety and better economics. The list of problems to be considered includes, among others, the following:

(i) the complete preclusion of the radioactive sodium piping beyond the reactor vessel boundaries;

(ii) the use of special built-in heat exchangers for emergency heat removal directly from the reactor vessel by means of air heat exchangers;

(iii) the use of additional (based on passive principles) independent emergency shutdown rods;

(iv) minimization of the positive component of the sodium void reactivity effect;

(v) ensuring reliable natural circulation for the decay heat removal;

(vi) an increase of the core burn-up up to 12–15% HA;

(vii) a possibility of using both conventional and heterogeneous core designs;

(viii) a reduction in cost of in-vessel shielding;

(ix) the use of single-shell type steam generators;

(x) a thorough, economical and cost-effective arrangement of the reactor and its circuits, reduction in construction and structural materials.

As the starting point for this analysis, the existing design of the reactor will be taken. Therefore, R&D on the BN-1600 reactor is continuing.

Some specialists are now showing interest in developing a modular fast-reactor concept of the sodium advanced fast reactor (SAFR) type and the power reactor innovative safe module (PRISM) type (see J. D. Griffith, this Symposium), with a view to adapting such a concept to the specific conditions of this country. Development work is also envisaged to analyse the advantages of the modular concept.

5. Summary

The scientific and technological experience gained in development of sodium-cooled fast reactors is assessed as satisfactory. The ability of these reactors to work reliably and safely has been proved in practice. Good performance of equipment, control and monitoring systems has been successfully demonstrated. There are well-trained personnel for the adequate and safe operation of reactors.

It is now necessary to seek and find new ways to make fast reactors cheaper, with regard to both capital costs and costs of electricity.

It is also necessary to develop more approaches based on inherent safety and self-protection, which are intrinsic features of fast-reactor physics, to ensure the high level of safety.

The best way to achieve these aims is a progressive development through experience and practice.

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Discussion

G. Vendryes (*CEA, Paris, France*). The construction of the BN-800 plant is proceeding at a slow pace. Why is this so? What is the present state of construction, both on site and in factories? And when is the plant expected to be complete?

M. F. Troyanov. The present situation is complicated. Orders have been placed and some parts of the vessel are being manufactured. A new regulatory body covering environmental matters has now issued further requirements and we need to gain clearance through them. There are also financial constraints but we are doing all we can to overcome these difficulties.

J. M. Cassels, F.R.S. (*Norwich, U.K.*). May I first congratulate the Soviet school on their achievements, which reflect the Royal Society's motto 'nullius in verba', rendered into common speech as 'put your money where your mouth is'. Secondly, the Soviet submarine which recently sank was alleged to have a reactor using liquid bismuth coolant. Has this coolant, with its favourable neutron capture cross section and benign chemistry, been considered for fast reactors?

M. F. Troyanov. Perhaps two years ago we considered improved safety concepts for fast reactors and some specialists proposed different coolants such as lead or bismuth. There was a presentation of the work to a scientific congress in 1987. However, our understanding of sodium is based on much experience, whereas the alternative coolants are still at the generic stage. In addition, most of the problems with fast-reactor coolants concern engineering rather than nuclear properties.

B. Saitcevsky (*Unipede, France*). Professor Troyanov referred to single-shell steam generators for BN-1600. Does this mean that he intends to change from the modular type to the large single-shell type (as in Superphénix) and are the reasons technical or economic?

M. F. Troyanov. There is no simple answer to this question. We are satisfied with the concept of modular steam generators as used in BN-600 and would like to retain them for BN-1600. However, if we are to reduce the capital cost we have to change to single-shell steam generators.

C. V. Gregory (*UKAEA, Dounreay, U.K.*). Professor Troyanov referred to two recent sodium–water reaction incidents in the modular steam generators in BN-350. Could he tell us the outage time for the plant following these incidents?

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M. F. Troyanov. The outage times were not significant for the plant. The relevant loops were simply isolated and the plant returned to operation almost immediately using four loops.

J. Walker (*Birmingham, U.K.*). Could Professor Troyanov indicate the steps being taken to minimize the sodium void effect in BN-1600?

M. F. Troyanov. It is not worth altering the value of the sodium void effect in isolation, one needs to look at all the appropriate effects, including the Doppler coefficient, for example. We are still trying to reduce sodium void effects and the axially heterogeneous core design is one possibility.

H. Bailly (*CEA, Cadarache, France*). A number of different types of fuel have been developed: vibropack MOX, nitride and carbide for example. Is this for economic or technical reasons? What will be the best fuel in the future?

M. F. Troyanov. The impression Mr Bailly has of an extensive range of advanced fuels being studied is not quite correct. For example, in BR-10 we use nitride fuel to obtain the highest possible neutron flux level and to avoid use of plutonium. Generally, we have an experimental programme on advanced fuels which is on a scale very typical of those in other countries.

M. Y. H. Bangash (*Middlesex Polytechnic, U.K.*). Professor Troyanov mentioned the stability analysis for these reactors. What criteria does he use for any specific site regarding the evaluation of safety margins for operational and overload conditions? What analytical methods are available? Does he select a particular seismic spectrum and if so can he list the following data for me: reactor type, site, seismic load combination, acceleration, displacements of plant? In view of the Chernobyl accident, has his criterion changed on blast load phenomenon? What measures are taken to evaluate protective safety factors? Are any reports available?

M. F. Troyanov. Some regulations covering seismic events have recently been revised but are now again under review following a series of earthquakes in our country. We currently apply the old standards but will have to revise the analysis for old sites to the new standards.