Workshop on Operational and Safety Aspects of
Sodium Cooled Fast Reactors,
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РОСАТОМ

STATE ATOMIC ENERGY CORPORATION «ROSATOM»

SODIUM COOLED FAST REACTORS:

VITAL SAFETY ISSUES AND APPROACHES TO THEIR RESOLUTION

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OUTLINE

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3. Proposals on cooperation

1. Background

Studies on sodium cooled fast reactors (SFR) have been carried out in the USSR and in Russia for over half a century. During this period the following reactors have been designed, constructed and put into operation:

- BR-5/10 research fast reactor, which was in operation at the IPPE, Obninsk from 1959 to 2002;
- BOR-60 experimental fast reactor, which has been in operation at the RIAR, Dimitrovgrad from 1969 till now;
- BN-350 prototype reactor, which was in operation in Shevchenko/Aktau from 1972 to 1999;
- Commercial BN-600 reactor having been operated at Beloyarsk NPP, Zarechny over 30 years from 1980 till now;
- Commercial BN-800 reactor is now under construction on Beloyarsk NPP site (its start-up is planned for 2014).

Sodium leaks in fast reactors

Traditionally, when considering SFR safety, much attention is paid to the issues concerning high chemical activity of sodium. In Russia we have gained considerable experience on sodium leaks from the circuits and leaks in the steam generators (SG).

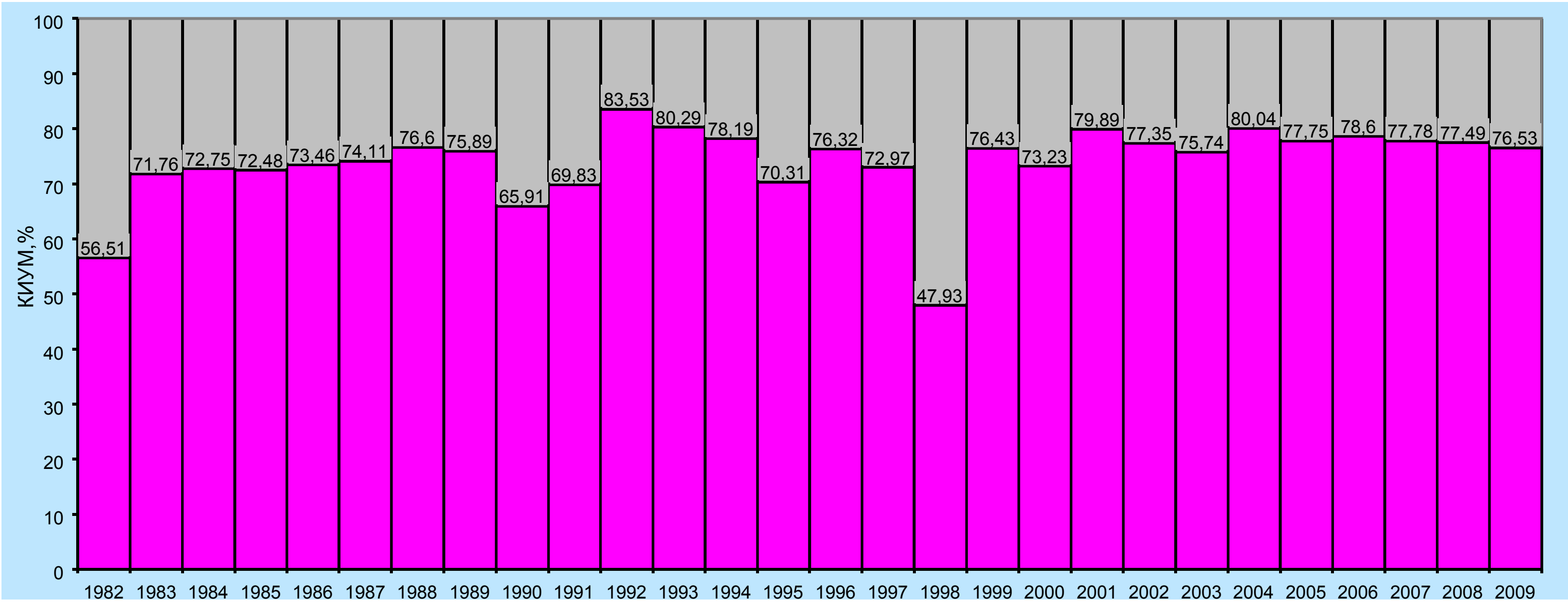
Reactors	Number of leaks from the circuits	Number of SG leaks
BR-5/BR-10	19	-
BOR-60	-	1
BN-350	15	14
BN-600	27	12
Total	61	27

All sodium leaks in BN-600 took place in the early stage of reactor operation and were timely detected and confined by the relevant safety systems and personnel.

The last sodium leak from the circuit occurred more than 16 years ago. During over 19 years steam generators operated without any inter-circuit leaks in spite of multiple replacements of SG modules carried out in this time period.

None of occurred sodium leaks caused exceeding of safety limits radiation effect on the inhabitants and the environment.

Load factor of the BN-600 reactor power unit for the whole period of its operation



Sodium fast reactor operating experience gained in Russia makes it possible to not only draw a conclusion about commercial development of SFR technology, but also demonstrates the possibility of reliable and safe operation of such reactors.

From BN-600 to BN-800

The design of BN-800 reactor which is under construction is a logical advancement of the BN-600 design with enhanced safety features meeting modern requirements, such as:

- application of decay heat removal system independent from the third circuit;
- presence of additional passive reactor safety system using safety rods suspended in sodium flow;
- arrangement of core debris catcher in the lower section of the reactor vessel.

Future plans

This January Federal Target-oriented Program (FTP) “The New Generation Nuclear Power Technologies for 2010-2015 Period and with Outlook to 2020” was approved by the RF Government.

Within the framework of this FTP advanced SFR of the next generation (BN-1200) with significantly improved technical and economical characteristics will be designed.

BN-1200 would be basis in achieving the goal of commercialization of SFR and closed nuclear fuel cycle.

2. Safety approaches for new designs

The main feature of the next stage of SFR development is simultaneous achievement of the goals of improvement of economical and safety characteristics.

In our vision the main path for this is max use of inherent safety characteristics, as well as safety systems based on passive operating principles.

This approach was used in optimization of characteristics of BN-1200 in the stage of its conceptual design. The main characteristics of the BN-1200 reactor taken for design development are presented in the following table.

The main characteristics of the BN-1200

Parameters	Values
Reactor thermal power, MW	2900
Reactor electric power, MW	1220
NPP efficiency, % − gross − net	 42.0 39.0
Number of heat removal loops	4
Reactor lifetime, years	60
Primary sodium temperature (IHX outlet/inlet), ^{0}C	410/550
Secondary coolant temperature (SG outlet/inlet), ^{0}C	355/527
Parameters of the third circuit: − pressure (at the SG outlet), MPa − temperature (at the SG outlet), ^{0}C − feed water temperature, ^{0}C − reheated steam temperature in the steam-steam reheater, ^{0}C	 14.0 510 240 250

BN-1200 - new decisions

Main decisions adopted in the BN-1200 reactor design in order to improve its economic and safety features are as follows:

- integral arrangement of the primary circuit in the reactor vessel this practically eliminating the possibility of radioactive sodium release from the primary circuit;
- simplification of the reactor refueling system owing to elimination of the intermediate storage drums for fuel subassemblies and arrangement of high-capacity in-vessel storage (IVS);
- change-over from section-module steam generator design to integral SG design;
- use of passive reactor safety systems based on diverse operating principles as well as of decay heat removal system independent from the main heat removal loops.

Closed fuel cycle safety

Change-over to closed fuel cycle of nuclear power is critical and conceptual choice of Russia. Within the framework of FTP it is planned:

- creation of semi-industrial plant for BN MOX fuel fabrication;
- development and tests of high-density fuels;
- development of the new generation structural materials;
- development of dry methods of reprocessing of BN SNF;
- experimental studies on radwaste final disposal.

In this stage, the key safety aspects are as follows:

- assurance of nuclear and radiation safety in all CNFC stages;
- overall decrease of energy system environmental impact.

Safety approaches in the area of fuel cycle

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Some thoughts.

There is no doubt in order to get public support it is desirable to eliminate long live radwaste toxicity in repository (for example through transmutation of all minor actinides and long lived fission products), however we should not forget that the main our goal is nuclear power competitiveness (with all the additional transmutation costs). This lead us to the idea of the need for finding of optimum transmutation level.

This optimization have to be done by taking into account that assurance of radiation safety associated with on ground fuel cycle facilities are not less, but might be even more important.

Obviously, it is interested to consider the possibility and expedience of unification of the national and international standards on releases concerning reactor plants and nuclear fuel cycle facilities.

In particular, defining requirements to the possibility and extent of discharge of liquid radioactive waste of various categories is a relevant objective.

Development of research infrastructure

In the institutes and enterprises of the Rosatom there is an extensive research and trial infrastructure, which has been and can be further used for carrying out necessary R&D work on SFR safety analysis, as well as verification of computer codes used for these purposes.

Irradiation of experimental devices with various fuel compositions and structural materials can be also performed in BOR-60 experimental reactor, as well as in the BN-600 reactor.

It is planned to use in the future MBIR experimental reactor for this purpose.

Experimental facilities of the IPPE

IPPE possesses several dozens of experimental facilities related to various aspects of sodium cooled fast reactors. All these facilities can be divided into the categories according to their purpose:

- SFR core neutronics;
- studies of fuel and materials under radiation;
- SFR safety;
- thermal hydraulics;
- sodium coolant technology.

Experimental facilities of the IPPE for neutronics and safety studies

- **BFS-1 critical facility (studies on neutronics of the core, blankets, in-vessel fuel storage and in-vessel shielding on the full-scale models of research and power fast reactors up to 1000 MWth);**
- **BFS-2 critical facility (studies on full-scale models of the core, blankets, in-vessel fuel storage and in-vessel shielding).**
- **PLUTON test facility (studies on the processes of materials movement caused by interaction of uranium-containing corium simulators with sodium);**
- **AR-1 test facility (studies on thermal-physics processes; studies of flow stability and heat transfer characteristics in case of coolant boiling);**
- **Integral model of fast reactor, facility for testing decay heat removal system;**
- **IK-MT liquid metal test facility (carrying out tests of elements of automatic safety systems of sodium-water steam generators for SFR and studies on the small leaks in sodium-water steam generators);**
- **SAZ facility for testing safety system of NPP with SFR).**

Experimental facilities of the Rosatom institutions for safety studies

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RIAR - BOR-60 experimental reactor;

OKBM - water facility for full-scale tests of the main pumps for SFR;test facility for studies on the core debris catcher for the case of beyond design accident in SFR with core meltdown;

OKB Gidropress - complex of test facilities designed for carrying out studies on thermal hydraulics of sodium-water steam generators and structural and vibration tests of the main elements of the steam generators.

Experimental facilities for new designs

FTP implies creation of the new experimental facilities and equipment, upgrading and development of existing test facilities required for carrying out R&D work for justification of SFR.

The R&D work programs include upgrading of the complex of the large critical facilities BFS and designing and construction of the multi-purpose research fast reactor MBIR with sodium coolant, which is planned to use for tests of the fuel and structural materials.

Now the preparation work is carried out on terms of reference for designing research reactor MBIR.

Start-up of MBIR reactor is planned for 2019.

The main parameters of MBIR reactor

Parameters	Values
Thermal power, MW	~ 150
Electric power, MW	~ 40
Max neutron flux, n/cm^2·s	~ 6.0·10^{15}
Standard fuel	Vipac-MOX, (PuN+UN)
Experimental fuel	Innovative types of fuel, fuel with MA
Core height, mm	600
Max power density, kW/L	1100
Max annual fluence, n/cm^2	~ 1·10^{23} (up to 45 dpa)
Lifetime, years	50
Number of independent loops with various coolants	Up to 4
Total number of experimental subassemblies and irradiation devices for radioisotopes production	Up to 12 (core) Up to 5 (radial shielding)
Number of experimental channels	Up to 3 (core)
Number of horizontal experimental channels	Up to 6 (outside reactor vessel)
Number of vertical experimental channels	Up to 8 (outside reactor vessel)

MBIR design and experimental capabilities

Experimental devices:

- 1 test loop in the core central area
  D~130 mm, Fn~$5 \cdot 10^{15}$ n/(cm^2·s)

- 2 test loops in the reflector
  D~130 mm, Fn~$1.8 \cdot 10^{15}$ n/(cm^2·s)

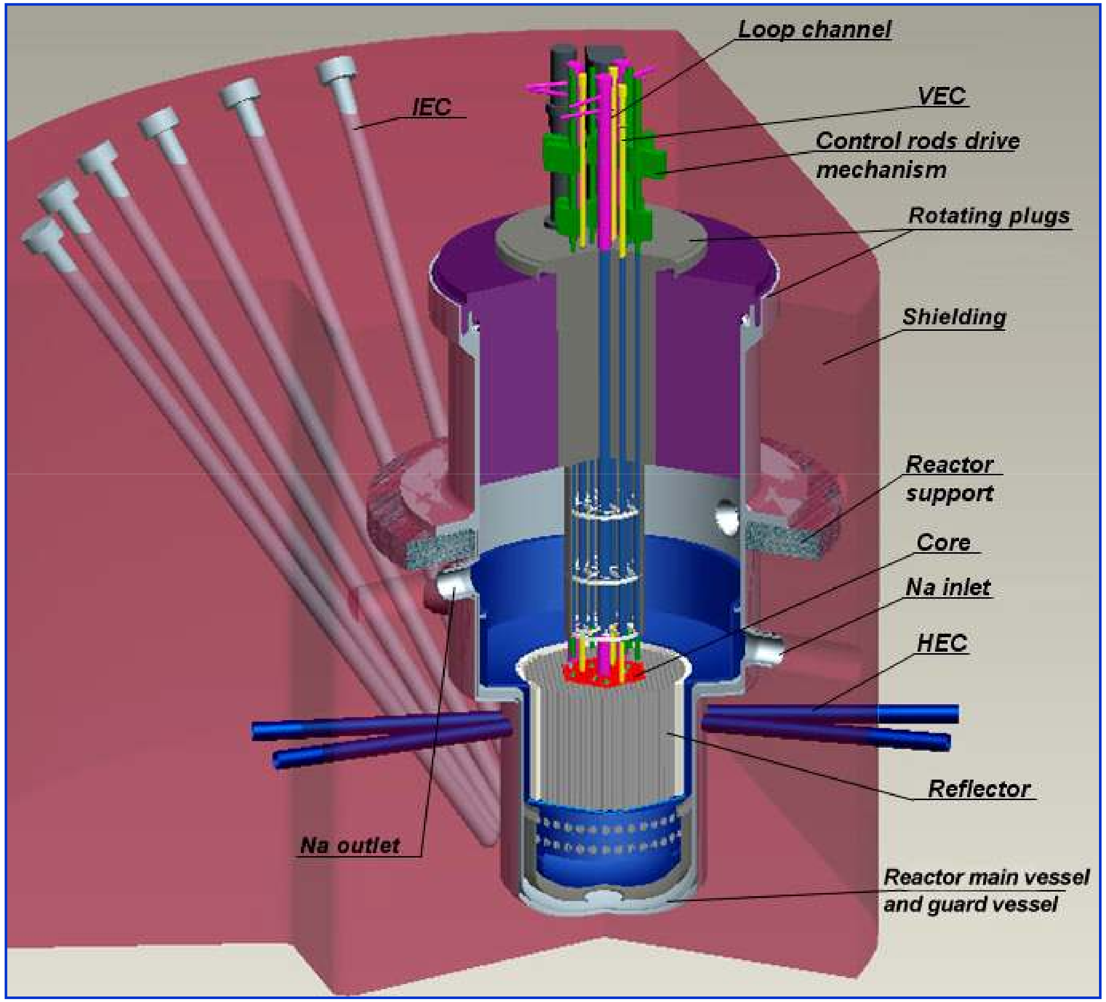
Loop channels coolants: - liquid metals
- gas
- molten salts

Vertical experimental channels (VEC):
3 channels in the core
D~60 mm, Φn~(3.2-5)·10^{15} 1/(cm^2·s)

Material test subassemblies:
up to 15 channels in the core and reflector
D~60 mm, Φn ~ $5 \cdot 10^{15}$ 1/(cm^2·s)

Horizontal experimental channels (HEC):
up to 6 channels outside reactor vessel
D~150-200 mm, Φn ~ $0.5 \cdot 10^{14}$ 1/(cm^2·s)

Inclined experimental channels (IEC):
up to 7 channels outside reactor vessel
D~150-300 mm, Φn ~ $0.5 \cdot 10^{14}$ 1/(cm^2·s)



Development of computer codes

For the purpose of safety analysis over 20 computer codes are used in Russia allowing to simulate the whole range of transients, abnormal operating conditions and design basis and beyond design accidents typical for SFR.

According to FTP, the new generation integrated code systems will be designed for the analytical studies on safety of advanced NPP with nuclear fuel cycle, including BN-1200.

There is a goal to make certification of all computer codes used for analysis of safety and other characteristics of advanced SFR. Besides, FTP implies upgrading of existing codes, including development on their basis of the new generation integrated code systems capable of designing reactor components and systems of advanced NPP with SFR, making their comprehensive safety analysis, optimization of NPP characteristics and information support of NPP operation.

3. Potential cooperation areas

The following promising topics can be proposed for scientific and technical cooperation in the area of SFR operation and safety:

- specification of safety criteria and requirements to new generation of fast reactors;
- upgrading of the computer codes and their verification by means of carrying out benchmarks. (Benchmarks on the analysis of severe beyond design accidents are of special interest, in particular, those on the accidents with the core meltdown, as well as on estimation of degree of influence of some parameters on the course of certain beyond design accident and its consequences, for instance, reactivity feedback coefficients);
- exchange of available experimental data for verification of computer codes;
- use of the available experimental facilities for arranging and carrying out experiments aimed at obtaining of necessary data for verification of computer codes;
- information exchange on current status of activities in SFR area including operation and safety issues.

Proposals for joint studies

- Joint activities on development of MBIR design, construction and implementation of experimental studies
- inherent safety characteristics of advanced SFR and their contribution to reactor safety assurance;
- role of active and passive safety systems in their optimal combination from the standpoint of safety and technical and economical characteristics;
- various approaches to the mitigation of consequences of severe beyond design accidents and their effectiveness;
- effect of various approaches to safety assurance on the economical characteristics of SFR, including evaluation of potentiality of decreasing SFR cost by means of application of passive safety systems;
- effect of the basic conceptual approaches (rated power level, NPP configuration, concepts of the main components and systems of the NPP, etc.) on SFR economical characteristics, etc.