Development of Fast Reactors in the USSR and the Russian Federation; Malfunctions and Incidents in the Course of their Operation and Solution of Problems

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Abstract. Development of fast reactors in the USSR was approved in the Government Decree of 1950. It was followed by development and construction of the critical assemblies, research reactors, the pilot and demonstration power reactor BN-350, power reactor BN-600 and power reactor the BN-800. The operation of research and power reactors highlights, including abnormal and emergency situations, their causes and ways of overcoming them are presented.

Key Words: Sodium leaks and fires; Reactivity increase; Untight fuel rod.

1. Introduction

The initial idea of potential nuclear fuel breeding originated in the USA and the first success in development of fast reactors designed for implementation of this idea was achieved there. The significant contribution in this area was made by Enrico Fermi. With a very small delay, similar studies started in the USSR; however, that was the place where fast reactor development reached its peak. The chief scientific supervisor of these research studies in the USSR was A.I. Leypunsky. Great achievements in this area were made by scientists and engineers from France and the UK.

On April 2, 1944, at the scientific workshop in the Metallurgical Laboratory, E. Fermi gave his presentation dedicated to evaluation of plutonium breeding in the fast reactor. For the reactor infinite in size, with the 5%-concentration of plutonium in the mixture of uranium-238 and plutonium-239, the breeding value equal to 1.37–1.57 was indicated, and for the critical sphere with a radius of 100 cm this value was equal to 1.23–1.43 [1]. Then, it was followed by the construction of the pilot reactor "Clementine" with the capacity of 20 kW and mercury coolant in 1946, the research fast reactor EBR-1 with the capacity of 1.4 MW and sodium-potassium coolant in 1951, the research fast reactor EBR-2 with the capacity of 62 MW and sodium coolant in 1964, the fast power reactor ENRICO FERMI with the capacity of 300 MW(t) in 1963, the unique research reactor FFTF of a loop type, with the capacity of 400 MW, in 1980. Before its construction, the project of a large power reactor called CRBR was developed, but it was not implemented, as the President of the country signed the order that prohibited construction of fast reactors.

On 07.12.1946, I.V. Kurchatov and N.N. Semenov sent to the Chairman of the Science and Technology Council (STC) under the First Chief Directorate of the USSR Council of Ministers, B.L. Vannikov, a letter in which they proposed that the work on fast neutron reactors should be undertaken. On 26.05.1947, at the same Science and Technology Council meeting I.V. Kurchatov already reported on the results of the performed calculations pertaining to plutonium breeding in fast reactors. In the STC Resolution, the proposal was stated to develop a pilot fast reactor "Zvezda." However, this intention was not implemented in view of involvement with atomic bomb development and testing.

After completion of nuclear weapon tests, A.I. Leypunsky sent a Position Paper to the First Chief Directorate, where he stated the principal physical ideas and the high-priority tasks on fast reactors. These proposals were approved in the Government Decree of 1950. It was followed by development and construction of the critical facility BR-1 in 1955, research reactor BR-2 with the capacity of 100 kW and mercury coolant in 1956, research reactor BR-5 with the capacity of 5 MW, with sodium coolant in the primary circuit and NaK coolant in the secondary circuit in 1959. In 1973, the reactor capacity was increased to 8 MW, and the NaK coolant was replaced with the sodium one. Then in 1961 the critical facility BFS-1 was commissioned, in 1969 BOR-60 research reactor was constructed with the capacity of 60 MW, in 1970 - the critical facility BFS-2. In 1973 the pilot and demonstration power reactor BN-350 was commissioned with the capacity of 250 MW(e) and parallel production of 120 000 m^3/day of distilled water, in 1980 the BN-600 power reactor with the capacity of 600MW(e) and in 2015 the BN-800 power reactor of 880 MW (e) were constructed and commissioned.

The accumulated experience of their operation has shown a number of important advantages of fast reactors as compared to thermal reactors, including low pressure of coolant and gas in the reactor vessel, high breeding of nuclear fuel and low reactivity margin in favor of refueling interval (as low as below β), no formation of local critical masses, good stability of power density fields in the core, high thermodynamic parameters of steam generated and thus, high efficiency. The main disadvantage of sodium-cooled fast reactors still consists in their cost effectiveness that is worse as compared to thermal reactors. This problem is to be solved in the BN-1200 reactor project.

2. Research reactors

In the course of operation of research and power reactors, lots of deficiencies and problems effecting their reliability and safety were revealed. The research reactor (RR) BR-2, constructed in 1956, was dismantled in 1957; the design of this fast reactor with mercury coolant was found unpromising primarily for the reason of poor compatibility of the mercury coolant with structural materials and its high toxicity level. The BR-5 (BR-10) reactor operation did not avoid certain difficulties and problems either [2].

One of the early problems in the beginning of BR-5 RR operation was gas in the core; its presence in the core reduced reactivity, and its removal from the core resulted in reactivity increase. One of the sources of gas ingress into the core was its presence in the circuit after the circuit had been filled with sodium. Filling the preliminarily evacuated circuit with sodium, with vent valves open in the course of filling, made it possible to avoid partial underfilling of the circuit with sodium. Another source of the gas present in the core turned out to be the pump that entrapped gas from the pump tank. That fact required modification of the pump design. However, these measures proved insufficient to solve the so-called “gas problem.” When the circuit was filled with sodium, the gas was still present under the fuel element spacing wire in such amount that, in the course of its removal from the core, its reactivity effect could exceed β. That problem was eliminated by increasing sodium temperature in the circuit up to 300°C and by washing-out the gas bubbles in the subcritical state with a high sodium flow rate.

One more problem in the early period of BR-5 RR operation consisted in burn-through of the sodium circuits by short circuit earth current of electric heating systems. Once, because of a burn-through in the sodium level meter case in the reactor, sodium started to leak outside through the hole that was formed, finding its way onto the reactor control and protection system (CPS) rods. The leakage stopped as a result of “self-healing”, but the solidified sodium oxides ruled out any further operation of the reactor CPS. The reactor vessel together with the CPS was dismantled, washed off and only after that placed back. Then, in the course of another short circuit, a burn-through in the drain pipe occurred, and as at that time the circuit was under vacuum, a lot of air found its way into it and resulted in fuel assemblies plugging with oxides. The fuel assemblies had to be withdrawn from the entire core, washed off and only after that placed back to the reactor. After that, isolation transformers were installed in all the sodium circuit heating systems with the aim to isolate neutral circuits, thus preventing ground short circuit burn-through in the pipelines. Moreover, the electric heating system sections came to be used to inform the personnel about the fact and location of the sodium leak onset. Afterwards, these modifications were used in the BN-350, BOR-60, BN-600 and other projects. Coolant leakages in sodium and sodium-potassium circuits at the early stages of BR-5 operation were caused by a poor quality of mounting work and erroneous actions of the personnel. For instance, once, in the course of installation of an oxide detector, as a result of a certain error in the procedure of its heat exchanger heating-up (with its defrosting), its tube ruptured, followed by sodium leak (∼10 l) through the heat exchanger shell gasket into the room and sodium fire.

At the BR-5 reactor the first experience was gained and the first incidents were observed pertaining to the steam generator installed in one of the loops. Its heat exchanging surface was in the form of two coaxial tubes of 21x1.5 mm and 16x1.5 mm, with the gap in between filled with mercury. After five months of its operation the first water leak to the mercury layer was detected, with water leaking through the 40-60 mm long crack in the inner tube, formed as a result of intergranular corrosion. For that reason, all in all 8 tubes were plugged. Then the outer tube lost its tightness in the point of its welding to the tube-sheet. So, ∼35 litres of mercury went into the NaK circuit. After that the steam generator was dismantled and replaced with the air-cooled heat exchanger.

An unexpected incident happened in the course of recovery of 1000-litre cold trap containing ∼50 litres of the mixture of sodium oxides, hydroxides, hydrides and sodium. The accumulated sodium impurities were removed from the cold trap following technology tested at a 200-litre cold trap: first, pumping the steam-gas mixture through the cold trap for 7 hours with the output hydrogen control, and then, with the hydrogen concentration in the exhaust gas reduced virtually to zero, filling it with water. During 7 hours ∼2000 kg of steam passed through the cold trap. Fifteen hours later, the cold trap was heated with steam again up to 100^{o}C and then filled with water. After that, pressure started to grow very fast in the cold trap, the personnel left the room, and ∼30 minutes after the cold trap body burst with a detonating gas explosion in the room. The room was destroyed, but the personnel were not injured. That accident was caused by incomplete neutralization of the residual sodium whose surface was “protected” against interaction with steam by hydroxide crust. Later on, that technology was improved and successfully used at the IPPE, i.e. the procedure of cleaning with water was replaced with cleaning with alkali solution, and at the RIAR they used small amount of water passing through cold traps at intervals for a long period of time [2].

The BR-5 and BR-10 research reactors were in operation with a UO_2 core, two UN cores and one UC core. Besides, experimental fuel assemblies with other types of fuel were installed into the reactor, including mixed nuclear fuel. In this case, the incidents with fuel cladding failure were inevitable. However, they were reliably detected by the specifically developed fuel element clad integrity control (CIC) system; the reactor was shut down and the leaky fuel assembly was withdrawn. In the course of the leaky fuel assembly withdrawal, there was a certain contamination of the central hall, but it was sufficiently easy to decontaminate it. However, when it was necessary to withdraw a leaky fuel assembly with UC fuel, the central hall was seriously contaminated, and it took a lot of time to clean it. The reason consisted in pyrophorosity of this fuel. The withdrawal procedure for this type of leaky fuel assemblies requires air-tight conditions.

At the end of BR-10 research reactor operation, while loading a fuel assembly with a new neutron source to the reactor, the worker did not check the completion of the procedure of the fuel assembly installation into the pressure header, and for this reason, when the rotating plug started to move, the fuel assembly top fitting was broken and a few neighboring assemblies were damaged. It required a lot of time and efforts to withdraw the damaged fuel assemblies. Afterwards the similar incidents happened once at the BOR-60 and once at the BN-350 reactors. Thus, it is required to the control the completion of the procedure of the fuel assembly installation into the pressure header. The BN-600 provides for a variant of this system

development. The French specialists developed a system of video-monitoring under the sodium layer in the Phenix reactor.
Development, construction and operation of the BR-2, BR-5, and BR-10 research reactors made it possible to determine the main solutions for future high-power fast reactors:

- three loop cycle of the reactor thermal energy;
- sodium coolant;
- oxide fuel;
- barrier vessel of the reactor.

3. BN-350

The main problems and incidents encountered during the construction and operation of the BN-350 reactor were related to the steam generators and structural materials that were used for core components fabrication (fuel elements, FA, CPS control rods). There were other types of incidents: sodium leaks and fires, stuck spent FA drums, loss of electricity supply [3].
In accordance with the BN-350 design, it had 6 loops and thus, 6 steam generators; each of them had two vertical evaporators with heat-exchange tubes of the Field type. During the post-installation testing, at the points where the tubes had been welded to the tube-sheets and where the bottom caps had been welded to the outer Field tube, the leaking welds were detected and fixed. And during the early period of its operation 9 cases of interloop leaks in 5 steam generators were detected in the places of caps welding to the tube. These leaks were caused by certain violations in the technology of caps production and control over cap-tube welded joints. By decision of the Minister (E.P. Slavsky), the steam generators were repaired "in-situ", with replacement of all the Field tubes and thorough examination of all the welds. During 1974 and early 1975 the first three steam generators were repaired, and the reactor reached the power of 350 MW (th). After nine days of operation of steam generator No.5, a large interloop leak was detected in it. The reactor was shut down, the valves in the lines of feed water, superheated steam and continuous blowdown were closed; the drainage of water from the damaged evaporator started. However, they failed to drain sodium, as the drainage pipeline got plugged with reaction products. Five hours later, a sodium leak was detected in the pipeline weld, at the point where the pipeline was welded to the output evaporator nozzle, in the form of a hole of ∼15mm in diameter. And hydrogen was leaking through the same hole. Sodium and hydrogen ignited and that saved the room from possible, even more dangerous consequences. According to some estimations, ~800 kg of water entered the evaporator, evidently through the valves not tightly closed, or from the neighboring evaporator through the continuous blowdown pipeline that was integrated with the similar line of the damaged evaporator. The majority of the evaporator heat-transferring tubes were destroyed under the impact of alkali solution; the evaporator housing was also partially damaged. About 300 kg of sodium went into the steam generator room.
Failure of fuel element cladding and FA wrapper tube bend represented another incident at the BN-350 because of the radiation exposure of stainless steel OH18N10T used in the fuel rods' structure with the change of its mechanical properties and swelling. During the BN-350 project development, no relevant research of radiation effects on structural materials was carried out. An extensive program had to be developed, and a great number of steels had to be investigated on determination of radiation effects on their swelling, embrittlement, creep characteristics resulting in the selection of optimum steels for fuel elements claddings – ЧС-68 (CW 316 Ti), for the external fuel assembly shroud – ferritic-martensitic steel EP-450. It allowed the fuel burnup to be increased from 5% h.a. to 10% h.a.
In 1976 ingress of water (water steam) into the spent fuel subassemblies pool (SFSAP) and the tank of spend fuel subassemblies (SFSAT) located in it, containing 5.6 m^3 of sodium-potassium alloy was initiated. As a result of accumulation of water - SFSAT alloy interaction products the

SFSAT was blocked. By that time there were 37 assemblies in the SFSAT. The source of water ingress into the SFSAT was the concrete of removable displacers – the upper radiation shielding, the water from which was released under irradiation. All in all ~500 kg of water entered the alloy. By the technology offered by the operator, the SFSAP was being washed off with water-oil emulsion from 1979 through 1984; at times the washing-off was accompanied by large release of hydrogen into the SFSAP gas plenum with the growth of gas pressure. The washing-off allowed the displacers and the tank of spend fuel subassemblies to be withdrawn from the SFSAP. 4000–4500 kg of alloy were left in the unknown state in the SFSAP. In order to avoid a potential formation and explosion of potassium and sodium peroxide, at the suggestion of IPPE they were turned into soda through their interaction with carbon dioxide.

The decay heat removal system inspection demonstrated its immaturity due to reversing sodium coolant circulation direction in the secondary circuit. In order to improve the decay heat removal system reliability, the new secondary coolant flow circulation through the Czechoslovakian steam generator "Nadezhnost" was developed and implemented (the steam generator was mounted in the course of replacing the damaged fifth-loop steam generator), that made it possible to develop natural circulation without reversing coolant circulation direction under normal conditions. Concurrently, the steam generator was transferring heat to the air circulating through the steam generator "Nadezhnost" duct. In addition, the backup power supply system was upgraded.

4. BN-600

The BN-600 pilot and demonstration fast reactor was brought to power on 08.04.1980. Conceptually, it had to become a predecessor and a prototype of series power reactors with a sufficiently high breeding ratio and competitive economy.

The BN-600 design solutions were much different from the BN-350 ones: integral primary circuit layout with the pumps and intermediate heat exchangers located in the reactor vessel, higher temperature level of the primary and secondary circuit coolant and "operating heat" in the tertiary circuit, new designs of the basic facility including steam generators, heat exchangers, and pumps; the new structural materials of the core elements, improved interloop leaks control systems. The number of sections was reduced to three, each with its turbo generator. The new sodium fabrication and transportation process was developed.

In the initial period of operation the incidents were related to the main circulation pumps in the primary circuit and again – with steam generators [4]. As the pumps were brought to high speed, the pump shaft vibration was detected. It resulted from torsion oscillations of pump shafts and motor shafts interconnected with a coupling sleeve. This called for refurbishment and the replacement of pump shafts, the Chief Designer (OKBM) together with the operator made short work of.

During the first years of plant operation 12 events involving interloop leaks occurred in the superheaters of the once-through steam generator, consisting of 8 sections, each including an evaporator module, a superheater module and a reheater module. Each section could be shut-off by the secondary and tertiary circuit valves. The evaporator modules were made of ferritic-martensitic steel (1X2M), superheater modules– of stainless steel (OH18N10T). The capacity of one section was not high (61 MW), therefore its disconnection did not force the reduction of reactor power.

All cases of interloop leaks occurred in the point of tube-to-steam generator tube plate welding; there were no leakages in the evaporator modules. Cracking was caused not by chlorine corrosion, but by deficiencies of welding procedure and subsequent control. After improvement of the technology since 1985, only one case of interloop leak was observed. In the first three cases during the initial period of operation due to imprompt decision–making, a fairly large amount of water went into the secondary circuit - 40 kg, 40 kg, 20 kg and a considerable

quantity of sodium entered the tertiary circuit – 2500 kg, 600 kg, 200 kg. What is more, hydrogen leakage and burning was observed in the first of them through the packing gland of the shutoff valves.
At the initial period of operation, numerous events of fuel rod untightness were observed. In the course of investigations in the IPPE and RIAR hot laboratories it was found that, the cracking was caused by cladding corrosion from the inner side that was aided by fuel assembly shuffling in the core with a 180^{o} turn, which resulted in the increase of linear heat rate that was already high enough. The latter was decreased from 540W/cm to 480W/cm by eliminating in-vessel refueling and core height increase from 750 mm to 1000 mm over the period of the first core retrofitting (1986–1987).
Since the time of commissioning works there were 27 events of sodium leaks from the circuits into the rooms. In most cases the integrated leak was within 1–10l, and only in 4 cases the leak was significant – approximately several hundred litres. One of the most disturbing leaks – that from the defective pipeline (fatigue crack) to the primary cold trap in the area of pipeline heat extension compensation. Release of radioactive sodium fire products through the ventilation system did not cause environmental contamination above the pollution standard. In all cases of sodium leaks no personnel were injured.
In 1995 an increase of resistance modulus was observed as the central rotation column (CRC) was rotating during fuel assemblies and CPS rods' reloading operations; at a later stage the situation worsened [4]. As a result of the repair operation elaborated at OKBM, the CRC was lifted 2 m up, and in the course of its examination, column shell deformation was found, which occurred in the area of its loss-of-integrity resulting from swelling of graphite located inside the column in the course of its interaction with sodium that found its way deep inside the CRC. The unique repair works performed by the operations personnel together with OKBM allowed normal CRC rotation to be restored.
In January 1987 another unique event happened: in the course of reactor operation at power an insignificant short-time increase in reactivity was observed, which was compensated for by an automated controller. The subsequent analysis of the events that also included the results of cover gas composition change analysis (presence of methane and hydrogen in it) made it possible to suppose, that an "icicle-shaped formation" resulting from long-time condensation of sodium vapors on it and the oil vapors of primary main pumps' oil system fell into the sodium from the rotating plug. With the aim of avoiding such an incident in the future, the oil system of pumps was improved, and temperature of rotating plugs was raised.

4. BN-800 and Future

In 2015 the installation was and commissioning works were completed at the new fast power reactor BN-800 with the capacity of 880 MW(e). In August 2016 the reactor reached 100% power. The principal difference of BN-800 from BN-600 in addition to increased capacity consists in essential design safety enhancement. However, concerning this project, everything is still ahead.
Over the past 60 years in our country the vast and most important experience in fast neutron sodium-cooled reactors has been gained, that ensures safe and reliable operation of these reactors. The following factors contributed to it:

- development of research reactors and their operation at the early stage;
- planning and conducting major experimental studies at thermophysical material test facilities, critical test facilities and research reactors;
- highly-skilled personnel training;
- continued support on the part of the industry senior management;
- joint activities of IPPE, OKBM, LOTEP, GSPI, VNIINM, OKB "GIDROPRESS" and operators;

- sharing of best practices with counterparts abroad.

In the future, commissioning of the closed fuel cycle, NPPs with new BN-1200 reactors and developments of technologies for the BN competitive price are in project.

References

[1] FERMI, E., Transactions, v. 2 1939-1954, USA. Publishing Company “Nauka”, Moscow, 1972, p.p. 220-224

[2] LEYPUNSKY, A., et al., L. “To the Background of Development and Operation of Fast Neutron Research Reactor BR-5 (BR-10)”, Articles, Memoirs, Photographic Documents. Obninsk, 2009, SSC RF – IPPE.

[3] TROYANOV, M., “BN-350 Reactor: Development, Design, Operation Experience.” IAEA Conference “Nuclear Power and its Fuel Cycle”, 1977, IAEA-CN-36/594

[4] OSHKANOV, N., et al., “BN-600 Liquid Metal Reactor: its Main Specific Features and Operation Experience.” Bulletin of Higher Educational Institutions – “Yadernaya Energetika”, No.2, 2009.