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SODIUM HANDLING TECHNOLOGIES FOR FAST REACTOR DECOMMISSIONING: A REVIEW

K. O. Toleubekov*, N. E. Mukhamedov, A. S. Surayev, S. A. Dolzhikov

Institute of Atomic Energy NNC RK, 10 Beibyt atom st., 180010 Kurchatov, Kazakhstan

** E-mail for contacts: toleubekov@nnc.kz*

Sodium fast neutron reactors currently demonstrate the highest technological readiness for industrial operation and transition to a closed nuclear fuel cycle. At the same time, the expansion of the fleet of nuclear power plants with fast sodium reactors in the future will inevitably lead to the need to solve the problems of their decommissioning. One of the key challenges of this stage is the management of sodium as a coolant, which is due to its physicochemical properties. These physical and chemical features significantly complicate the processes of dismantling and decontamination of reactor plant equipment. In this regard, the accumulated experience in management of sodium during the decommissioning of experimental demonstration fast sodium reactors is of particular value. Thus, this article provides an overview of current and promising sodium management technologies at the stage of decommissioning sodium fast neutron reactors.

Keywords: *fast sodium reactors, SFR, decommissioning, sodium processing, closed nuclear fuel cycle*

1. Introduction

Fast neutron reactors are a key element in the implementation of a closed nuclear fuel cycle, which can significantly reduce the need for natural uranium and involve significant reserves of depleted uranium in the fuel cycle [1-2].

The first fast neutron reactors were commissioned in the USA (Clementine, 1946) and the Soviet Union (BR-2, 1956). Mercury was used as a coolant, but subsequently its use was abandoned due to unfavorable chemical, thermophysical and corrosive properties [3-5].

Currently, sodium and lead [6-7], as well as sodium-potassium [8] and lead-bismuth alloys [9] are used as coolant in fast reactors. Sodium is the most common coolant for fast reactors today. Sodium fast reactors (SFR) have gone from experimental installations to pilot demonstration nuclear power plants [10-11] and are considered in the near future as the basis for serial industrial implementation [12].

The development of the concept of decommissioning SFR is getting particular relevance in preparation for their industrial introduction. This stage is the final and one of the most important in the life cycle of any nuclear power plant, which necessitates its consideration already at the design stage.

In general, the SFR decommissioning strategy is based on the accumulated experience of thermal neutron reactor decommissioning [13]. At the same time, the use of sodium as a coolant significantly complicates the dismantling of the main and auxiliary equipment due to its high chemical activity in contact with water and air [14].

Under these conditions, the accumulated experience of decommissioning pilot demonstration SFR serves as a technological basis for effective and safe disposal of sodium [15-19]. Analysis of the sodium management technologies used in closed pilot experimental plants allows us to form a reliable basis for developing an effective strategy for decommissioning serial industrial SFR. In this regard, this article provides an overview of current sodium management technologies at the stage of SFR decommissioning.

2. Main sodium management technologies during SFR decommissioning

Sodium is the most commonly used coolant in fast reactors due to its thermophysical and neutron properties, such as high thermal conductivity, low saturated vapor pressure and weak neutron deceleration. These features ensure efficient operation of the fast neutron reactor [20].

At the same time, the use of sodium leads to a number of serious technological difficulties, including at the stage of SFR decommissioning. The high chemical activity of sodium, as well as the difficulty of cleaning equipment from sodium, significantly complicate the dismantling of main and auxiliary equipment. In this regard, the completion of SFR decommissioning requires the development of special technical solutions [21].

In general, the concept of sodium management during SFR decommissioning is as follows [22-23]:

1) Purification of sodium from radioactive contamination;

2) Sodium drainage from main and auxiliary equipment;

3) Treatment of internal surfaces of equipment from sodium residues;

4) Reuse or processing of sodium for safe long-term storage as radioactive waste.

2.1. Purification of sodium from radioactive contamination

The SFR use a three-loop coolant circulation system. Liquid sodium is used in the first (primary sodium) and second (secondary sodium) loops, while water is used in the third loop. Consequently, the sodium in the first loop is exposed to the highest level of radiation due to the direct contact with the reactor core.

During SFR operation, neutron irradiation leads to the formation of activation products in the primary loop sodium and its impurities. The main contributors to radioactivity are the isotopes Na-22 and Na-24 [24]. Additionally, during long-term operation of the SFR, radioactive corrosion products (Mn-54, Co-58, Co-60) and fission products from fuel rod leaks (Cs-134 and Cs-137) accumulate in the sodium [25–26].

The experience of SFR operation shows that the main radionuclides determining the primary sodium activity are Cs-134 ($T_{1/2}$ – 2.1 years) Na-22 ($T_{1/2}$ – 2.6 years), Co-60 ($T_{1/2}$ – 5.3 years) and Cs-137 ($T_{1/2}$ – 30 years). The radioactivity of the secondary sodium is low and rapidly decreases to a safe level after reactor shutdown. At the same time, the activity of long-lived radionuclides depends on the reactor's operational duration and power, the amount of sodium in the system, and previously implemented measures to reduce activity [27–28].

Thus, the residual radioactivity of the primary sodium after reactor shutdown poses several challenges for ensuring radiation safety. Accordingly, the first step in managing primary sodium during SFR decommissioning is to reduce its radioactivity prior to drainage.

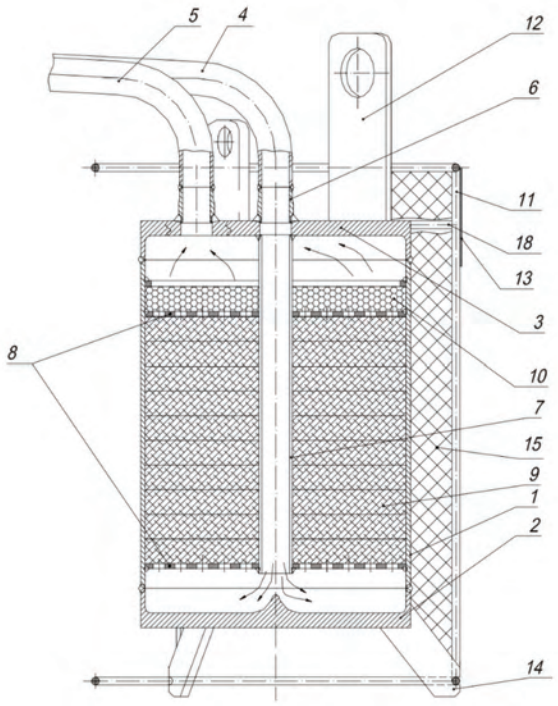
Initial storage of sodium reduces activity due to decay of radionuclides such as Cs-134, Na-22, and Co-60. For example, in 2001, the specific activity of radionuclides in the primary sodium of the BN-350 reactor was studied following its shutdown in 1999. The analysis showed that the specific activity of radionuclides in the primary sodium was less than 3% of the activity of Cs-137. The specific activity of the primary sodium in 2003 was approximately 7.25×10^8 Bq/kg, with Cs-137 contributing more than 97% of the activity of the primary sodium [29].

Thus, as a result of initial storage, the main radionuclide contributing to nearly all of the primary sodium activity is Cs-137. However, the specific activity of the primary sodium remains high (shown in the example of the BN-350 reactor). This poses risks to personnel and the environment. Therefore, after initial storage, the primary sodium must be purified from cesium before drainage.

Currently, the most commonly used and technically proven method for removing radioactive cesium from sodium is based on the principle of cesium sorption using graphite materials [30]. The use of graphite materials to remove radioactive cesium from sodium began in the 1970s. The need for such purification arose due to the increase in the specific activity of cesium in sodium. This led to an increase in the radiation background in the reactor facilities. Sorption enabled the extraction of radioactive cesium directly from sodium, thereby reducing the accumulation of radiation on the surfaces of equipment [31].

The BN-350 reactor primary sodium decontamination system during it decommissioning is an example of the use of this technology. It was developed on the basis of a design that was previously successfully used in cleaning the primary sodium of the EBR-II reactor [32].

The sodium purification system included seven sequentially connected cesium traps. The cesium trap was a steel vessel, inside which there was a graphite sorbent in the form of stacked disks. Gridded vitreous carbon of RVC grade was used as a sorbent. Figure 1 shows the scheme of the cesium trap of the BN-350 reactor [33–34].



1 – shell; 2 – bottom plate; 3 – top plate; 4 – inlet pipe; 5 – outlet pipe; 6 – inlet nozzle; 7 – central inlet pipe; 8 – perforated plate; 9 – graphite; 10 – mesh filter; 11 – electrical support frame; 12 – electrical conduit support; 13 – label; 14 – support leg; 15 – thermal insulation.

Figure 1. The scheme of the cesium trap of the BN-350 reactor

The principle of operation of the primary sodium purification system is based on the forced circulation of sodium through cesium traps. Sodium enters the trap through the central inlet pipe (pos. 7), after which, by changing the flow direction and reducing the speed, it passes through the sorbent layer (pos. 9). After passing through the sorbent, sodium passes through a mesh filter (pos. 10). The filter is designed to retain possible sorbent particles in sodium. The purified sodium is then passed to the next trap for further purification and finally returned to the primary system.

The principle of purification of sodium from radioactive cesium used in the decommissioning of the British PFR reactor is largely similar to that previously described. At the same time, unlike the technologies used in the BN-350 and EBR-II reactors, the purification of sodium from radioactive cesium was carried out after its drainage and chemical neutralization [35].

The method is based on draining sodium from the reactor vessel and other systems. Subsequently, the sodium reacts with a large amount of water. A controlled reaction produces an aqueous solution of sodium hydroxide. This reaction is known as the NOAH process (*see Section 2.3 of this article*). The aqueous sodium hydroxide solution is then neutralized with hydrochloric acid to form an aqueous sodium chloride solution.

The resulting sodium chloride solution was purified of Cs-137 using an ion exchange method with selective sorbents. The central component of this method is special ion-exchange columns through which the sodium chloride solution is passed. The columns are filled with specialized resins that selectively capture cesium ions.

2.2. Sodium drainage

Sodium draining is the next important stage in the decommissioning of SFR. It is necessary to enable the subsequent dismantling of the main and auxiliary equipment. The sodium draining methodologies applied in different SFR decommissioning projects are largely identical in most aspects, taking into account only the design features of specific facilities [36].

Creation of conditions for sodium draining

Sodium draining is a complex engineering task. The complexity of sodium draining is due to its high chemical activity, as well as the need to maintain sodium in a liquid state. Maintaining the sodium temperature above its melting point is necessary to ensure that the metal remains liquid and flowable (typically 120–150 °C). This enables the safe and efficient draining of sodium from pipelines, heat exchangers, and equipment. Draining of highly chemically active sodium is made possible by establishing an inert atmosphere (most commonly using argon). The free volumes above the sodium are filled with argon, which prevents contact with air and moisture and thereby reduces the risk of ignition or explosion.

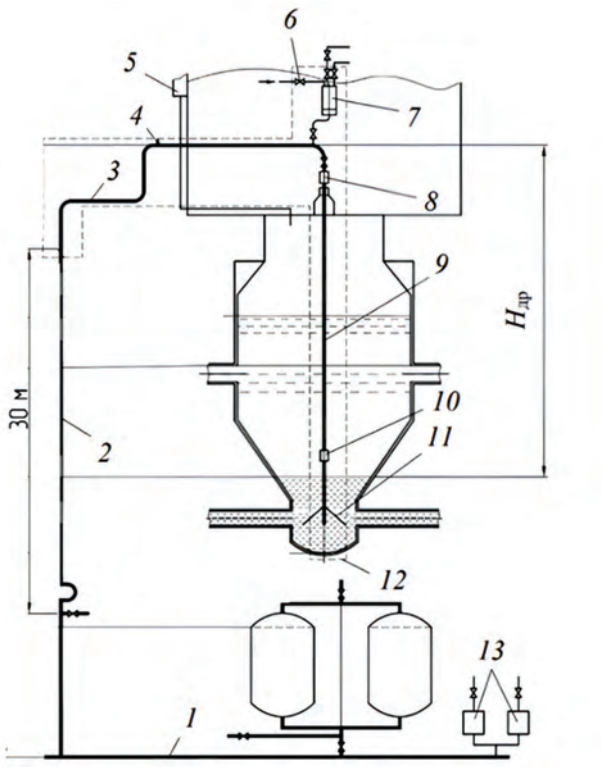
Draining of the main volume of sodium

Drainage of sodium from the reactor vessel, the primary and secondary circuit systems is carried out through standard or specially created drainage systems. The sodium is drained either by gravity or using drainage pumps. To increase the efficiency of drainage, additional purging with an inert gas is applied. This allows displacing additional volumes of sodium from pipelines and equipment.

Draining sodium from the primary and secondary circuits is generally similar to each other. The design of the primary and secondary circuits allows for almost complete removal of sodium through the combined use of inert gas purging and maintaining the equipment temperature above the melting point of sodium. At the same time, the complex geometry of the reactor vessel's internal area leads to the formation of sodium pockets that cannot be drained or displaced by purging.

A complicating factor is that the sodium in the reactor vessel is radioactive, although its activity has been reduced as a result of purification from Cs-137. This significantly limits the possibility of physical access and makes the complete removal of sodium potentially hazardous for personnel. Therefore, ensuring the full removal of sodium from reactor systems requires the use of specialized methods for handling and removing residual sodium, taking into account both the radiological conditions and the high chemical activity of sodium.

In this regard, additional methods such as drilling and laser cutting of the internal components of the reactor vessel have been developed to facilitate the drainage of residual sodium [37]. The decommissioning of the BN-350 [38], PFR [39], and Superphénix [40] reactors serves as an example of the application of these methods.



1– drainage collector; 2, 3 – pipeline; 4 - pressure sensor; 5 – reactor vessel water seal; 6 – valve; 7 – drainage tank; 8 – flow meter; 9 – drainage channel; 10 – throttle sleeve; 11 – reflector; 12 – boundary of the designed drainage system; 13 – induction pumps

Figure 2. The scheme of sodium drainage in the reactor vessel of the BN-350 reactor

A distinctive feature of sodium drainage in the BN-350 reactor was the presence of a large volume of sodium below the level of the pressure pipelines. To achieve complete drainage, a drilling device was developed that could reach the bottom of the reactor. The system included a drainage pipe in the central channel, a drainage tank, and pipelines connecting the device to storage containers. A key challenge of the process was drilling inside the reactor containing liquid sodium at ~ 300 °C at a depth of more than 13 m. Figure 2 shows the scheme of the drainage system of the BN-350 reactor.

In the case of decommissioning the British PFR, during sodium removal a significant portion of the sodium remained inside the reactor vessel, the primary circuit, and the core support plate. As a result, a special drilling machine was developed to drill through the upper plate of the core support structure and the thermal insulation in order to release the trapped sodium, as well as equalize the sodium level between the inner and outer pools. A specific feature of the process was that the drilling was carried out inside the reactor filled with liquid sodium at a depth of 14 m.

A pipe-piercing machine was also developed. It was used to facilitate the removal of residual sodium from the core support plate and from the primary sodium pump discharge lines. The machine destroyed sodium pipelines inside the reactor using a hydraulic press. At the same time, it could be installed only after the sodium level in the reactor dropped enough to open the pump outlet channels and the operator could control the direction of the press. Figure 3 shows the general scheme and appearance of the pipe-piercing machine.

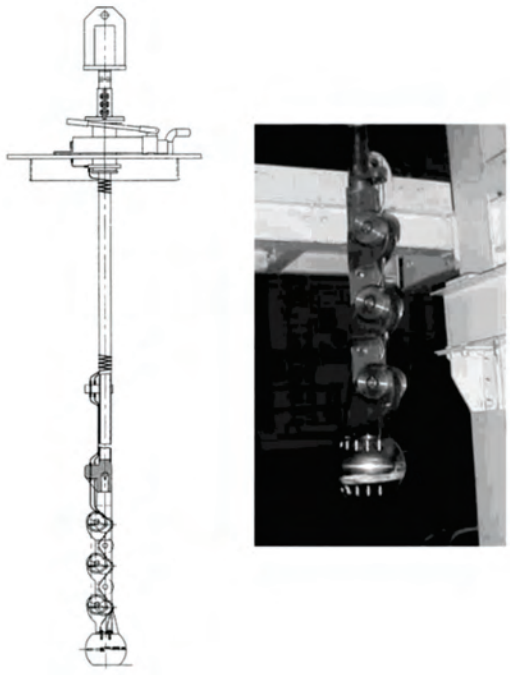


Figure 3. The scheme and appearance of the pipe-piercing machine

During the decommissioning of the Superphénix reactor, the presence of significant volumes of residual sodium retained in the pipelines of the primary circuit was revealed. The «Charli» remotely operated robotic system was developed to solve this issue. The robot was equipped with a manipulator with a laser cutting head.

It was introduced into the reactor vessel and moved along the internal structures to the areas where sodium was retained. Laser cutting of the elements that retained residual sodium allowed it to flow into the lower part of the reactor vessel. Thus, sodium could be removed and processed. Figure 4 shows the appearance of the «Charli» robot [41].

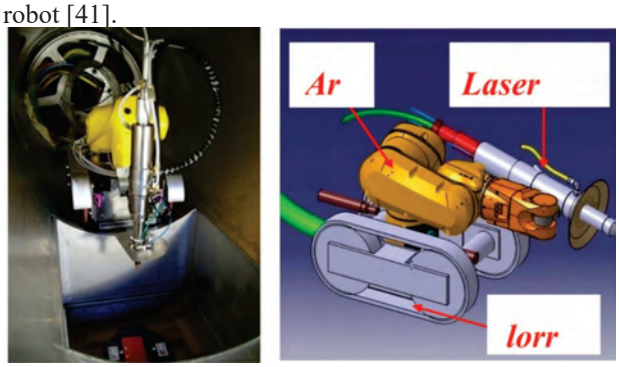


Figure 4. The appearance of the «Charli» robotic system

Chemical neutralization of sodium residues

As a result of draining and the application of additional technical solutions, the bulk of the sodium can be removed from the reactor systems. At the same time, part of the sodium inevitably remains in the systems in the form of films and local accumulations. In this regard, prior to the start of subsequent stages of SFR decommissioning, it is necessary to neutralize the residual sodium to ensure safety.

The most well-developed method for handling residual sodium is chemical neutralization. Chemical neutralization methods are based on converting residual metallic sodium into a chemically more stable and safer-to-handle form. This makes it possible to ensure its safe removal [42].

At present, several methods of chemical neutralization of residual sodium are known, which have been applied during the decommissioning of SFRs:

1. Hydrocarbonation method

In the context of residual sodium passivation during the decommissioning of SFRs, the hydrocarbonation (carbonation, passivation) method is considered one of the most widely used and preferred approaches. It has been applied during the decommissioning of the EBR-II [43], BN-350 [44], Phénix, and Superphénix reactors [45].

The physico-chemical basis of the hydrocarbonation method is the interaction of humidified carbon dioxide (CO_2) with sodium. The presence of water vapor in CO_2 leads to the formation of a sodium carbonate (Na_2CO_3) layer and the release of hydrogen (H_2):

$2Na + 2CO_2 + 2H_2O \rightarrow 2NaHCO_3 + H_2$

$2Na+2H_2O \rightarrow 2NaOH+H_2$

$NaHCO_3 + NaOH \rightarrow Na_2CO_3 + H_2O$

$2Na + CO_2 + H_2O \rightarrow Na_2CO_3 + H_2$

It has been established that the hydrocarbonation process using humidified CO_2 proceeds effectively in sodium layers up to 20 cm thick. If the sodium layer exceeds this thickness, the unreacted sodium becomes covered with a dry carbonate layer, forming a protective film. The concentration of the generated hydrogen can be maintained at a safe level by controlling the humidity of the gas mixture [46].

Thus, the reaction results in the formation of a solid, chemically inert Na_2CO_3 layer, which is safe for storage and prevents uncontrolled chemical reactions. Depending on the chosen sodium management strategy, the carbonate can either be removed from the reactor systems (e.g., by gas purging or water washing) or left in situ inside the reactor, as it is already chemically safe [47].

2. Steam-Nitrogen and WVN Process

Alternative methods for removing residual sodium involve the use of a nitrogen-steam mixture known as the Steam-Nitrogen and WVN (water vapor-nitrogen) processes. The Steam-Nitrogen process was applied during the decommissioning of the Fermi-1 reactor [48]. The WVN process was used during the decommissioning of the EBR-II [49] and PFR [50] reactors.

In the Steam-Nitrogen process, superheated steam mixed with nitrogen (in a proportion of 20 to 80%) is introduced into the system in order to convert the residual sodium into sodium hydroxide (NaOH):

$Na + H_2O \rightarrow NaOH + 1/2\ H_2$

$NaOH + H_2O \rightarrow NaOH \cdot nH_2O$ (n = 1, 2 и т.д.)

In the Steam-Nitrogen process, metallic sodium reacts with steam to form sodium hydroxide and hydrogen. It has been established that sodium hydroxide is hygroscopic and absorbs water to form hydrates of NaOH. Pure NaOH melts at 318 °C, however sodium hydroxide hydrates with n > 2 are liquid at room temperature. This leads to difficulties in controlling the reactions to ensure safety.

In the WVN process, a mixture of nitrogen and water vapor (in a proportion of 1 to 60%) is introduced into the system. Sodium reacts with a mixture of nitrogen and water vapor to form NaOH, NaOH hydrates, and H2.

Unlike the Steam-Nitrogen process, the treatment of residual sodium is carried out below the boiling point of water and the melting point of sodium (typically in the temperature range of 20–90 °C). A small amount of water vapor provides a more controlled reaction between sodium and steam. However, the rate of residual sodium treatment becomes significantly lower compared to the Steam-Nitrogen process.

Thus, as a result of treating residual sodium with a nitrogen-steam mixture in these two processes, water-soluble NaOH and NaOH hydrates are formed. They can be removed from the sodium systems after the completion of the process by washing with liquid water.

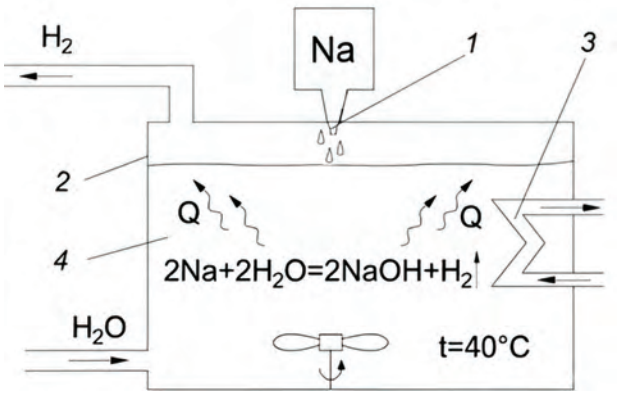
2.3. Processing of Sodium After Draining

After draining, sodium is stored in special drain tanks. In the context of sodium management during the decommissioning of SFRs, it must be either processed or reused. Sodium can be reused as a coolant in other SFRs, in research programs, as well as in industry.

An example of this approach is the project for transporting the secondary sodium from the BN-350 reactor to the Ulba Metallurgical Plant for use in tantalum production [51]. Another example is the canceled project for the reuse of primary sodium from the Fermi-1 reactor in the Clinch River experimental fast reactor [52].

However, opportunities for reuse are often limited due to contamination of sodium with impurities, corrosion and activation products, as well as the lack of end users capable of accepting radioactive sodium. As a result, the primary method of managing sodium after draining is its processing followed by storage.

The widely used sodium processing technology is the NOAH process, which is based on the continuous conversion of sodium into sodium hydroxide. This technology was developed for the treatment of primary sodium during the decommissioning of the Rapsodie research reactor [53]. At present, technology of the NOAH process was chosen as the main method for sodium processing in almost all reactors under the decommissioning, including Fermi-1, KNK-II, EBR-II, PFR, BN-350, Superphénix and Phénix reactors [54].



1 – nozzle; 2 – reaction tank; 3 – heat exchanger; 4 – water-alkaline solution

Figure 5. The general scheme of sodium processing into an alkaline solution

The principle of the NOAH process is based on the interaction of a small amount of liquid sodium with an aqueous sodium hydroxide (NaOH) solution. The concentration of NaOH is controlled by adding water. Since the chemical reaction is exothermic, continuous cooling through a heat exchanger is required. The hydrogen generated during the reaction and other impurities, including radioactive ones, are filtered, dried, and diluted with nitrogen before being released into the environment. Figure 5 shows the overall scheme of sodium processing into an alkaline solution [55].

The resulting NaOH alkaline solution is immobilized in solid form by cementation. The choice of cementation technology depends on the approved decommissioning strategy of the specific SFR. The solid cement blocks are classified as low-level radioactive waste and are suitable for long-term storage [56–57]. Figure 6 shows cement blocks produced from processed sodium from the Superphénix reactor [58].

3. Conclusion

The decommissioning of SFR is a complex technological process. Sodium management during SFR decommissioning is one of the most challenging and important stages. The high chemical reactivity of sodium, the presence of residual radioactivity, and the design features of SFRs require a comprehensive and step-by-step approach.

An analysis of dismantling experience from actual SFR decommissioning projects shows that an effective sodium management strategy includes several interrelated stages: reduction of sodium radioactivity, draining of the main volume, neutralization of residual sodium, and its processing or reuse.

This review has shown that a number of sodium management technologies have been developed and successfully demonstrated, including:

Removal of radionuclides from sodium using sorbents;

Use of robotic systems and specialized devices for draining operations;

Chemical neutralization of residual sodium by hydrocarbonation methods and by treatment with a mixture of water vapor and nitrogen;

Processing of sodium into an aqueous alkaline solution followed by cementation.

One of the key challenges in sodium management is the generation of large volumes of secondary solid and liquid radioactive waste in the form of precipitates, solutions, and contaminated structural materials. This necessitates additional processing and storage stages [59–60].

In this context, further improvement of sodium management during SFR decommissioning remains a relevant task. This may involve both optimization of existing technologies and the development of alternative approaches [61–62].

Figure 6. Appearance of concrete blocks produced from processed sodium from the Superphénix reactor

Improving reactor operating conditions will help minimize the accumulation of radionuclides. This will reduce the need for sodium purification from radionuclides, sodium processing, and its storage, and consequently lead to a decrease in the generation of radioactive waste [63–64].

Nevertheless, the accumulated experience and existing sodium management technologies provide a reliable foundation for developing strategies for the safe and efficient decommissioning of future commercial SFR.

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Reference

1. Taylor R., Bodel W., Stamford L. and et.al. A Review of Environmental and Economic Implications of Closing the Nuclear Fuel Cycle—Part One: Wastes and Environmental Impacts. Energies. 2022, 15, 1433. https://doi.org/10.3390/en15041433
2. Kiegiel K., Smoliński T., Herdzik-Koniecko I. Advanced Nuclear Reactors—Challenges Related to the Reprocessing of Spent Nuclear Fuel. Energies. 2025, 18, 4080. https://doi.org/10.3390/en18154080
3. Patenaude H. K., Freibert, F. J., 2023. Oh, My Darling Clementine: A Detailed History and Data Repository of the Los Alamos Plutonium Fast Reactor. Nucl. Technol., 209(7), 963–1007. https://doi.org/10.1080/00295450.2023.2176686
4. Kazachkovskii O.D., Bagdasarov Y.E., Nikolaev M.N., 2006. 50 years after the start-up of BR-2 — The first fast reactor in Europe. At Energy, 100, 176–180. https://doi.org/10.1007/s10512-006-0070-z
5. Fleitman A.H., Weeks J.R., 1971. Mercury as a nuclear coolant. Nucl. Eng. Des., 16 (3), 266-278. https://doi.org/10.1016/0029-5493(71)90015-X
6. Tuček, K.; Carlsson, J.; Wider, H. Comparison of Sodium and Lead-Cooled Fast Reactors Regarding Reactor Physics Aspects, Severe Safety and Economical Issues. Nuclear

Engineering and Design 2006, 236, 1589–1598. https://doi.org/10.1016/j.nucengdes.2006.04.019.
7. Dong, W.; Chen, J.; Lan, X.; Chen, D. Latest Progress on the Thermal-Hydraulics Research in Lead-Cooled Fast Reactors: A Review. Applied Thermal Engineering 2025, 273, 126479. https://doi.org/10.1016/j.applthermaleng.2025.126479
8. Dubinin, N.E. Heat Capacities and Thermal Coefficients of Sodium's and Eutectic Sodium–Potassium's Coolants for Nuclear Reactors. Appl. Sci. 2025, 15, 7566. https://doi.org/10.3390/app15137566
9. Xing, M.; Fan, J.; Shen, F.; Lu, D.; Li, L.; Yu, H.; Fan, J. Comparative Analysis on the Characteristics of Liquid Lead and Lead–Bismuth Eutectic as Coolants for Fast Reactors. Energies 2025, 18, 596. https://doi.org/10.3390/en18030596
10. Aoto K., Dufour P., Hongyi Ya., et.al., 2014. A summary of sodium-cooled fast reactor development. Progress in Nuclear Energy, 77, 247-265. https://doi.org/10.1016/j.pnucene.2014.05.008
11. Wang, J.; Talabi, S.; Bilbao y León, S. Sodium Fast Reactors. In Nuclear Power Reactor Designs: From History to Advances; Wang, J., Talabi, S., Bilbao y León, S., Eds.; Elsevier: Amsterdam, The Netherlands, 2023; https://doi.org/10.1016/B978-0-32-399880-2.00010-2
12. Vasyaev A.V., Kerekesha A.V., Kryukov A.N., et al., 2025. Current status of the BN-1200M project: a review. At Energy, 138, 121-132. https://doi.org/10.1007/s10512-025-01236-6
13. Park, K.; Son, S.; Oh, J.; Kim, S. Sustainable Decommissioning Strategies for Nuclear Power Plants: A Systematic Literature Review. Sustainability 2022, 14, 5947. https://doi.org/10.3390/su14105947
14. Carnevali, S.; Proust, C.; Soucille, M. Unsteady Aspects of Sodium–Water-Air Reaction. Chemical Engineering Research and Design 2013, 91(4), 633–639. https://doi.org/10.1016/j.cherd.2013.02.005
15. Michelbacher, J.A.; Henslee, S.P.; Poland, H.P.; et al. Planning for Closure and Deactivation of the EBR-II Complex. In Proceedings of the International Conference on Nuclear Engineering (ICONE); Nice, France, 1997; p. 341.
16. Artz, D.; Engelhardt, G.; Kathol, W.; et al. Decommissioning of the Sodium-Cooled Fast Breeder Reactor Plant KNK. In Proceedings of the 2nd ASME–JSME Nuclear Engineering Joint Conference; San Francisco, CA, USA, 1993; pp. 789–794.
17. Nelson, R. Future of Dounreay—Challenges and Opportunities. Nuclear Energy 1999, 38(4), 233–238.
18. Mukhamedov, N.; Toleubekov, K.; Vityuk, G.; et al. Decommissioning of the BN-350 Fast Neutron Reactor: History Review and Current Status. Energies 2025, 18, 3486. https://doi.org/10.3390/en18133486
19. Guidez, J.; Prêle, G. The Dismantling. In Superphenix: Technical and Scientific Achievements; Atlantis Press: Paris, France, 2017; pp. 311–323. https://doi.org/10.2991/978-94-6239-246-5_24.
20. Poplavskii, V.M.; Efanov, A.D.; Kozlov, F.A.; et al. Sodium as a Coolant for Fast Reactors. Atomic Energy 2010, 108, 281–288. https://doi.org/10.1007/s10512-010-9290-3
21. Kaskov, D.O.; Volozheninov, T.P.; Tashlykov, O.L. Problems of Decommissioning Fast Neutron Reactors with Sodium Coolant. In Proceedings of the Energy and Resource Saving. Power Supply. Non-traditional and Renewable Energy Sources. Nuclear Energy. Danilov Readings; Yekaterinburg, Russia, 2013; pp. 504–513. (In Russ).
22. OECD Nuclear Energy Agency (NEA). Selecting Strategies for the Decommissioning of Nuclear Facilities: A Status Report (NEA No. 6038); OECD Publishing: Paris, France, 2006.
23. Skvortsov, A.I.; Safutin, V.D.; Yamov, V.Y.; et al. Conditioning a BN-350 power-generating unit for safe storage. At Energy 2005, 98, 65–68. https://doi.org/10.1007/s10512-005-0170-1
24. International Atomic Energy Agency. Comparative Assessment of Thermophysical and Thermohydraulic Characteristics of Lead, Lead-Bismuth and Sodium Coolants for Fast Reactors. IAEA-TECDOC-1289, Vienna, 2002.
25. Sidorov, I.I.; Tashlykov, O.L.; Shcheklein S.E.; et al. Disposal of Coolant, Equipment and Metal after the End of Service Life of NPP Power Units. Izvestiya vuzov. Yadernaya Energetika. 2025, 3, 186–201. https://doi.org/10.26583/npe.2025.3.13 (in Russ).
26. Polley, M.V.; Jose, J.; et al. Radioactive contamination measurements of the primary sodium pipes in FBTR by gamma spectrometry. Annals of Nuclear Energy 2009, 36(5), 641–649. https://doi.org/10.1016/j.anucene.2008.12.030
27. International Atomic Energy Agency. Liquid Metal Coolants for Fast Reactors: Cooled by Sodium, Lead and Lead-Bismuth Eutectic. NP-T-1.6; International Atomic Energy Agency, Vienna, 2012.
28. Bessonov, I.A.; Chalpanov, S.V.; Tashlykov O.L. Purification of fast reactors primary sodium coolant from cesium radionuclides. In Proceedings of the Energy and Resource Saving. Power Supply. Non-traditional and Renewable Energy Sources. Nuclear Energy. Danilov Readings; Yekaterinburg, Russia, 2021; pp. 437–440. (In Russ).
29. Tazhibaeva, I.; Pustobaev, S.; Zhantikin, T.; et al. The Handling with Sodium Coolant at Fast Reactor BN-350; Glory K Ltd. Publ.: 2010; p. 320. (In Russian).
30. Kozlov, F.A.; Sorokin, A.P.; Konovalov, M.A. Sodium purification systems for NPP with fast reactors (retrospective and perspective views). Izvestiya vuzov. Yadernaya Energetika 2015, 3, 5-19. https://doi.org/10.26583/npe.2015.3.01
31. Hechetkin, Yu.V.; Kizin, V.D.; Polyakov, V.I. Radiation safety of NPP with fast reactor and sodium coolan; Energoatomizdat: Moscow, 1983 (in Russian).
32. Michelbacher, J.A.; Baily, C.E.; Baird, D.K.; et al. Shutdown and Closure of the Experimental Breeder Reactor-II. In Proceedings of the International Conference on Nuclear Engineering (ICONE 10); Arlington, 2002; pp. 857–865. https://doi.org/10.1115/ICONE10-22462
33. Romanenko, O.G.; Allen, K.J.; Wachs, D.M.; et al. Cleaning Cesium Radionuclides from BN-350 Primary Sodium. Nuclear Technology 2005, 150(1), 79–99. https://doi.org/10.13182/NT05-A3607
34. Polyakov, V.I.; Shtynda, Yu.E. Investigations to validate technology for fine cleaning of BN reactor sodium coolant from cesium isotopes. NNC RK Bulletin 2001, 1, 48–54. (in Russian)
35. McIntyre, A.W. PFR liquid metals disposal at Dounreay. In Proceedings of the Technical Committee Meeting on Sodium Removal and Disposal from LMFRs in Normal

Operation and in the Framework of Decommissioning; Aix-en-Provence, France, 3–7 Nov 1997; pp. 152–159.
36. Rovneiko, A.V.; Tashlykov, O.L.; Shcheklein, S.E. The features of the fast neutron reactors decommissioning. In Proceedings of the International Scientific and Practical Conference «Promising Energy Technologies. Ecology, Economics, Safety and Personnel Training», Yekaterinburg, Russian Federation, 11 October 2016; pp. 61–66. (In Russian)
37. International Atomic Energy Agency. Treatment of Residual Sodium and Sodium Potassium from Fast Reactors: Review of Recent Accomplishments, Challenges and Technologies. IAEA-TECDOC-1769, IAEA, Vienna, 2015.
38. Merisov, G.S.; Klimenko, A.S.; Gorbanenko, O.A. Temperature and flow parameters control system for BN-350 reactor sodium draining device. NNC RK Bull. 2003, 1, 18–22. (In Russian)
39. McCulloch, R.; Rees, G. The Dounreay PFR Liquid-Metal Disposal Project. Nuclear Technology 2005, 151, 15–27. https://doi.org/10.13182/NT05-A3604
40. Baque F., Haudiquet C., Major P. Superphénix LMFR Decommissioning Sodium Retention Treatment by Mechanical Drilling of Internal Structures. In Proceedings of the 8th International Conference on Radioactive Waste Management and Environmental Remediation, ASME 2001, Bruges, 2001; pp. 1235-1239. https://doi.org/10.1115/ICEM2001-1223
41. Charli cleans up at Creys-Malville. Available online: https://www.world-nuclear-news.org/Articles/Charli-cleans-up-at-Creys-Malville (accessed on 26 January 2026).
42. International Atomic Energy Agency. Radioactive Sodium Waste Treatment and Conditioning; IAEA-TECDOC-1534; IAEA: Vienna, Austria, 2007.
43. Sherman, S. R.; Knight, C. J. Post-Operational Treatment of Residual Na Coolant in EBR-II Using Carbonation. In Nuclear Power - Deployment, Operation and Sustainability; Tsvetkov, P., Ed.; IntechOpen: London, UK, 2011. https://doi.org/10.5772/19095
44. Romanenko, O.G.; Pugachev, G.; Rovneyko, A.; et.al. Processing of residual sodium in the BN-350 reactor coolant circuits. In Proceedings of Topical Meeting on Decommissioning, Decontamination and Reutilization; Idaho Falls, USA, August 2010; pp. 268 – 271.
45. Rodriguez, G.; Gastaldi, O. Sodium Carbonatation Process Development in a View of Treatment of the Primary Circuit of Liquid Metal Fast Reactor (LMFR) in Decommissioning Phases. In Proceedings of the 8th International Conference on Radioactive Waste Management and Environmental Remediation (ICEM 2001), Bruges, Belgium; ASME: New York, NY, USA, 2001; pp. 1717–1721. https://doi.org/10.1115/ICEM2001-1304
46. Michelbacher, J.A.; Henslee, S.P.; Knight, C.J.; et al. Decommissioning of Experimental Breeder Reactor-II Complex, Post Sodium Draining. In Proceedings of the Technical Meeting on Decommissioning of Fast Reactors After Sodium Draining; Cadarache, France, 26–30 Sep 2005; pp. 334–365.
47. International Atomic Energy Agency. Treatment of Residual Sodium and Sodium Potassium from Fast Reactors: Review of Recent Accomplishments, Challenges and Technologies (IAEA-TECDOC-1769); International Atomic Energy Agency: Vienna, Austria, 2015.
48. Goodman, L. Fermi 1 Sodium Residue Cleanup. In Decommissioning of Fast Reactors After Sodium Draining (IAEA-TECDOC-1633); International Atomic Energy Agency: Vienna, Austria, 2009; pp. 49–54.
49. International Atomic Energy Agency. Radioactive Sodium Waste Treatment and Conditioning: Review of Main Aspects (IAEA-TECDOC-1534); International Atomic Energy Agency: Vienna, Austria, 2007.
50. Gunn, J.; Mason, L.; Husband, W.; et al. Development and Application of Water Vapor Nitrogen (WVN) for Sodium Residues Removal at the Prototype Fast Reactor, Dounreay. In Decommissioning of Fast Reactors After Sodium Draining (IAEA-TECDOC-1633); International Atomic Energy Agency: Vienna, Austria, 2009; pp. 123–134.
51. Nosov, Y.V.; Rovneiko, A.V.; Tashlykov, O.L.; Shcheklein, S.E. Decommissioning Features of BN-350, -600 Fast Reactors. Atomic Energy 2019, 125, 219–223. https://doi.org/10.1007/s10512-019-00470-z
52. Goodman, L. Summary of FERMI 1 Sodium Residue Cleanup. In Proceedings of the Technical Meeting on Decommissioning of Fast Reactors after Sodium Draining; Working Material, Cadarache, France, 2005.
53. Rodriguez, G.; Gastaldi, O.; Baqué, F. Recent sodium technology development for the decommissioning of the Rapsodie and Superphenix reactors and the management of sodium wastes. Nuclear Technology 2005, 150, 100–110. https://doi.org/10.13182/NT05-A3608
54. Toleubekov, K.O., Mukhamedov, N.E, Surayev, A.S.; et al. Decommissioning of a Nuclear Facility with Sodium-Cooled Fast Reactor as a Final Stage of Its Life Cycle: Overview of Current Projects. Science and Technology of Nuclear Installations 2026, 2026 (1), 4174549. https://doi.org/10.1155/stni/4174549
55. Kaskov, D.O.; Volozheninov, T.P.; Tashlykov, O.L. Problems of Decommissioning Fast Neutron Reactors with Sodium Coolant. In Proceedings of the International Scientific and Practical Conference «Energy and Resource Conservation. Energy Supply. Non-traditional and Renewable Energy Sources. Nuclear Power Industry»; 2023; pp. 504–513. (in Russian)
56. Rodriguez, G.; Thomine, G.; Serpantié, J.P.; Demoisy, Y. ATENA: Project of Storage and Disposal Plant for Radioactive Sodium Wastes. In Proceedings of the 8th International Conference on Radioactive Waste Management and Environmental Remediation (ICEM2001), Bruges, Belgium, 30 September–4 October 2001. https://doi.org/10.1115/ICEM2001-1231
57. Tazhibayeva, I.; Pugachev, G.; Herrick, A. Decommissioning of Kazakhstan's BN-350 Fast Reactor. Nuclear Future 2008, 4, 281–286.
58. Bouilly, V.; Velon, M.; Rauber, J.-C.; et al. SUPERPHENIX Dismantling – Status and Lessons Learned. In Proceedings of the International Conference on Fast Reactors and Related Fuel Cycles: Next Generation Nuclear Systems for Sustainable Development (FR17); Yekaterinburg, Russian Federation, 26–29 June 2017.
59. Romanenko, O.G.; Tazhibaeva, I.L.; Yakovlev, I.L.; et al. Immobilization of Cesium Traps from the BN-350 Fast Reactor (Aktau, Kazakhstan). In Proceedings of the

WM2011 Conference; Phoenix, AZ, USA, 27 February–3 March 2011.
60. Dounreay Site Restoration Ltd. Dounreay Interim Integrated Waste Strategy (WSU/STRATEGY/P033(08) ISSUE 6); Scottish Environment Protection Agency: Dounreay, United Kingdom, March 2010.
61. Legkikh, K.G.; Smykov, V.B. Innovative technologies for immobilizing the sodium primary coolant of fast neutron reactors and processing liquid radioactive waste generated during reactor operation. Atomic Energy 2024, 137, 129–135. https://doi.org/10.1007/s10512-025-01186-z
62. Smykov, V.B.; Zhurin, A.V.; Legkikh, K.G.; et al. Reprocessing of Primary and Secondary Coolants During the BN-350 Reactor Decommissioning. Izvestiya vuzov. Yadernaya Energetika 2023, 3, 164–169. https://doi.org/10.26583/npe.2023.3.16 (in Russian)
63. Gao, J.; Lv, S.; Ruan, Z.; et al. Fuel cladding chemical interactions for MOX fuel in fast reactors. Nuclear Engineering and Design 2024, 424, 113208. https://doi.org/10.1016/j.nucengdes.2024.113208
64. Miao, Y.; Oaks, A.; Mo, K.; et al. Metallic fuel cladding degradation model development and evaluation for BISON. Nuclear Engineering and Design 2021, 385, 111531. https://doi.org/10.1016/j.nucengdes.2021.111531

ШАПШАҢ НЕЙТРОНДЫ РЕАКТОРДЫ ПАЙДАЛАНУДАН ШЫҒАРУ КЕЗІНДЕ НАТРИЙМЕН ЖҰМЫС ІСТЕУ ТЕХНОЛОГИЯЛАРЫ: ШОЛУ

К. О. Толеубеков*, Н. Е.Мухамедов, А. С. Сураев, С. А. Должиков

Атом энергиясы институтының филиалы, Курчатов, Қазақстан

** Байланыс үшін E-mail: toleubekov@nnc.kz*

Қазіргі уақытта натриймен салқындатылатын шапшаң нейтронды реакторлар өнеркәсіптік эксплуатацияға ең жоғары технологиялық дайындықты көрсетеді және жабық ядролық отын цикліне өтуге мүмкіндік береді. Сол уақытта, болашақта шапшаң натрий реакторлары бар атом электр станцияларының паркін кеңейту олардың пайдаланудан шығарылу мәселелерін шешу қажеттілігін туындатады. Осы кезеңдегі басты мәселелердің бірі – салқындатқыш ретінде натрийді қолдану, себебі оның физика-химиялық қасиеттері бұл процестерді күрделендіреді. Физика-химиялық ерекшеліктер реакторлық қондырғылардың демонтажы мен дезактивациясын елеулі түрде қиындатады. Осыған байланысты, тәжірибелік демонстрациялық шапшаң натрий реакторларын пайдануланудан шығару кезінде натриймен жұмыс істеу тәжірибесі ерекше мәнге ие. Сондықтан, бұл мақалада шапшаң нейтронды реакторларды пайдаланудан шығару кезеңінде натрийді өңдеудің қазіргі және перспективалық технологияларының шолуы берілген.

Түйін сөздер*: шапшаң нейтронды реактор, SFR, пайдаланудан шығару, натрийді өңдеу, жабық ядролық отын циклі.*

ТЕХНОЛОГИИ ОБРАЩЕНИЯ С НАТРИЕМ ПРИ ВЫВОДЕ ИЗ ЭКСПЛУАТАЦИИ РЕАКТОРА НА БЫСТРЫХ НЕЙТРОНАХ: ОБЗОР

К. О. Толеубеков*, Н. Е. Мухамедов, А. С. Сураев, С. А. Должиков

Филиал «Институт атомной энергии» РГП НЯЦ РК, Курчатов, Казахстан

** E-mail для контактов: toleubekov@nnc.kz*

Реакторы на быстрых нейтронах с натриевым теплоносителем в настоящее время демонстрируют наибольшую технологическую готовность к промышленной эксплуатации и переходу к замкнутому ядерному топливному циклу. В то же время расширение парка атомных станций с быстрыми натриевыми реакторами в перспективе неизбежно приведёт к необходимости решения задач их вывода из эксплуатации. Одной из ключевых проблем данного этапа является использование натрия в качестве теплоносителя, что обусловлено его физико-химическими свойствами. Эти физико-химические особенности существенно усложняют процессы демонтажа и дезактивации оборудования реакторных установок. В связи с этим особую ценность представляет накопленный опыт обращения с натрием при выводе из эксплуатации опытно-демонстрационных быстрых натриевых реакторов. Таким образом, в настоящей статье представлен обзор современных и перспективных технологий обращения с натрием на этапе вывода из эксплуатации реакторов на быстрых нейтронах.

Ключевые слова*: быстрые натриевые реакторы, SFR, вывод из эксплуатации, переработка натрия, замкнутый ядерный топливный цикл.*