UDK 621.039.553

EXPERIMENTAL STUDIES IN SUBSTANTIATION OF SODIUM COOLED FAST REACTORS SAFETY

[1] E.G. Batyrbekov, [1] M.K. Skakov, [1] V.A. Vityuk, [2] V.V. Baklanov,
[2] A.D. Vurim, [2] A.V. Pakhnits, [3] K. Kamiyama, [3] K. Matsuba

[1] RSE "National Nuclear Center of the Republic of Kazakhstan", Kurchatov, Kazakhstan
[2] Branch "Institute of Atomic Energy" RSE NNC RK, Kurchatov, Kazakhstan
[3] Fast Reactor Cycle System Research and Development Center, JAEA, O-arai, Ibaraki, Japan

A large volume of scientific and research activities is currently carried out in substantiation of new projects on nuclear reactor safety, in which high emphasis is placed on "beyond-design basis" accidents, in spite of the fact that applied technical solutions reduce the probability of such accidents occurrence.
To study parameters of severe accidents and develop measures to mitigate consequences an experimental base is necessary, which enables emergency situation modelling maximally close to real conditions.
The National Nuclear Center (NNC) – the leading research organization in the atomic sphere in the Republic of Kazakhstan – possesses such an experimental base, providing scientific and technical support for peaceful uses of nuclear power, involving studies oriented towards the improvement of nuclear reactor safety [1].
The article presents the concept and general results of researches carried out on NNC's experimental base in substantiation of nuclear reactors safety, realized in collaboration with JAEA, which is a key foreign partner in this area, as well as current plans on new experimental program implementation.
Key words: *fast neutron reactor, severe accident, impulse graphite reactor, experimental program.*

INTRODUCTION

The leading nuclear power countries are studying six innovative next-generation nuclear systems [2], in which the major factor of their development is confidence in their safety for the population and the environment. In modern projects special attention is paid to severe accidents, in spite of the fact that these accidents are of low probability. To study parameters of severe accidents and develop measures for the mitigation of consequences an experimental base is necessary, which enables emergency situation modelling that is maximally close to real conditions. Although representative data on core behavior in severe accident conditions can be acquired during tests at research reactors, there are only several reactors in the world which provide for the required parameters and safe conditions for experiment conduction [3–6]. One of such reactors is the impulse graphite reactor (IGR) of NNC [7]. Reactor experiments, as a rule, involve tests at specially constructed non-reactor test benches and facilities.

From the beginning of the 1990s an experimental program has been carried out at the facilities of the National Nuclear Center jointly with Japanese colleagues from JAEA in support of fast sodium cooled nuclear reactor development [8, 9]. A series of in-pile and out-of-pile experiments [10, 11] were conducted, associated with the possibility of proving the controlled removal of core melt materials to safe areas inside the reactor vessel during severe accident progression. As a result, the efficiency of special-device application in core was demonstrated, providing directed (controlled) movement of melt fuel, as well as the possibility of melt movement via control rod channels. Such an approach enables a considerable reduction in the occurrence of the "recriticality" phenomenon [12, 13].

JUSTIFICATION OF EXPERIMENTAL FACILITIES CHOICE FOR RESEARCHES IN SUPPORT OF INNOVATIVE REACTOR PROJECTS

The IGR reactor (Figure 1) was put into exploitation in 1961 and as of today remains one of the best impulse research reactors in the world. Maximum density of thermal neutron flux composes 0.7×10^{17} cm^{-2} s^{-1}, maximum neutron fluence – 3.7×10^{17} cm^{-2}.

Figure 1 – IGR (Impulse Graphite Reactor)

In the more than half a century of IGR operation history, a significant complex of experimental operations in the area of nuclear reactor safety has been implemented for different purposes – space, transport, power and research. Methodological and technological experience was accumulated, and new technical decisions and methodological approaches were developed to carry out reactor tests and analysis of results [14–18].

A special bench with the EAGLE facility was constructed in NNC RK designed for conducting research in support of the construction of a new sodium cooled fast reactor, where experiments have been conducted to study different stages of severe fast reactors accidents development.

The EAGLE (Figure 2) facility provides the possibility of obtaining a melt simulator with temperatures of up to 3300 K due to working mixture induction heating, consisting of UO_2+SS or aluminum oxide (Al_2O_3) in masses up to 26 kg and 15 kg correspondingly. Sodium mass in the facility – up to 140 kg.

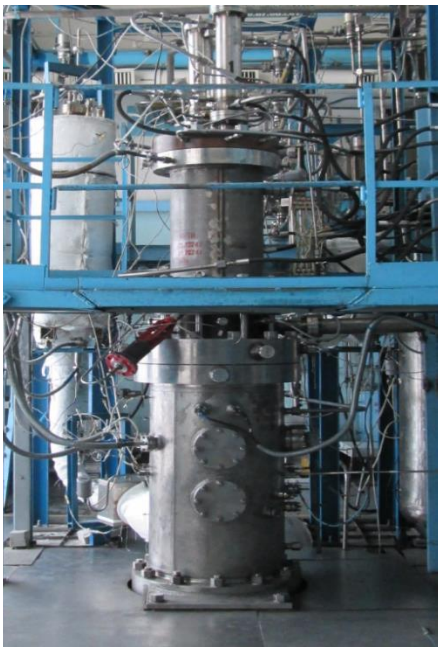


Figure 2 – EAGLE Facility

EXPERIMENT CONCEPT AND RESULTS

Researches, conducted at IGR and out-of-pile facilities address the necessity of constructing an extremely safe fast reactor, the design of which contains decisions which enable the exclusion of recriticality and the mitigation of severe accident consequences with core melting. Recriticality is a typical phenomenon that may occur during the production of melted fuel in pool relative to high enrichment in critical conditions of neutron multiplication. Its occurrence may lead to a damaged reactor vessel overheating and radioactivity escaping beyond it.

The general methodology for studying melting law, core melt movement inside a fast reactor, and its interaction with elements of structural materials and coolant consists of the following: at the first stage, a number of experiments are conducted at facility on processes studied through out-of-pile modeling, resulting in the elaboration of the general testing scheme and the choice of reactor experiments conditions and parameters. At the second stage, the information, obtained during out-of-pile studies is analyzed, requirements are developed on the scheme and conditions for conducting reactor experiments, during which maximal approximation of processes to the real operation conditions of core elements at nuclear power reactors is attempted. Test modes at reactor facilities are elaborated during irradiation experiments at relatively low power of the research reactor using special mock-ups, that are practically similar upon design and material composition.

In total, eleven integral and over one hundred supporting methodological experiments will have been carried out at the IGR reactor through the implementation of the experimental program on different aspects of severe accidents study by 2020 (Figure 3).

In demonstration experiments under conditions of reactor irradiation, melting was provided as well as controlled melt movement of model fuel assemblies (FA), containing up to 8.6 kg of uranium dioxide of 17 % enrichment. Different types of FA were used depending on experimental purpose with fuel elements number from 12 to 88. In practically all experiments, sodium was used of up to 10 kg in mass.

Experimental devices are equipped for measurement using more than 100 detectors, enabling the registration of such parameters as temperature (including temperature of fuel), pressure, pressure pulse, empty space in sodium, acoustic events, and neutron flux density. Non-destructive and destructive studies are carried out after reactor tests.

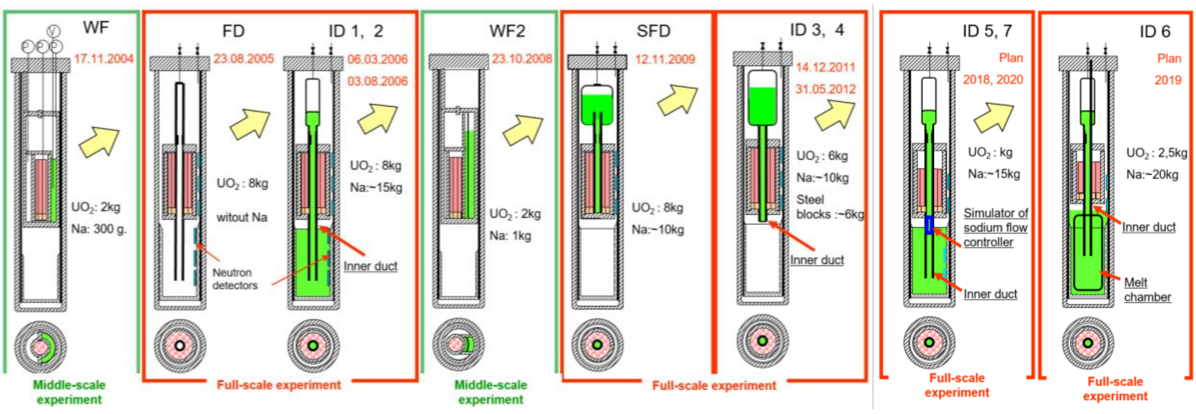


Figure 3 – Scheme of Integral Reactor Experiments

At the same time, at the non-reactor EAGLE test bench more than 50 modeling experiments using uranium dioxide and corium simulator as aluminum oxide were carried out. Processes with high temperature melt use, similar to real fuel melt upon thermophysical and hydrodynamic properties, were studied.

As a result of the conducted experiments, unique data were obtained on core structural elements disrupt-tive parameters in severe accident conditions at fast reactors, the possibility of controlling core melt material removal via special channels was demonstrated, as well as removal via control rods guide tube towards safe areas inside the reactor vessel.

Currently, preparation for demonstration reactor experiments is being carried out, oriented to study the impact of the structural elements of the control rods guide tube on the melt movement process and the melt cooling process, removed from the core in limited volume of sodium plenum. During preparation for demonstration reactor experiments at the non-reactor EAGLE facility, a series of experiments were realized on corium simulator movement along a simulated guide tube, modeling its main structural peculiarities (Figure 4, a). Al_2O_3 aluminum oxide was used as corium simulator, which was melted in an electric-furnace facility and moved to melt receiving capacity. Experiments were conducted to study the efficiency of corium cooling in sodium media (Figure 4, b).

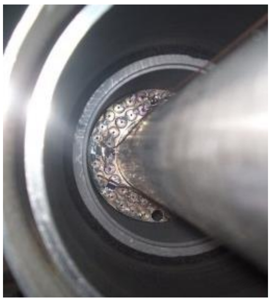
a) b)

Figure 4 – Experimental Equipment for Researches: Rods Guide Tube Mock-Up (a) and Device Designed for Corium Cooling Efficiency Study (b)

Upon research results, it was experimentally determined, that the possibility was provided to control melt materials removal from the core to safe areas inside the prospective fast reactor vessel in case of a severe core melt accident, in particular in the lower plenum. New experimental data were obtained on the cooling effects of corium simulator that was moved to the lower plenum after sodium reflooding.

Irradiation experiments with IGR reactor devices were carried out within the experimental verification preparation based on research reactor conditions to study the possibility of core (corium) materials melt controlled movement along the control rod guide tubes (Figure 5).

Assembly of Model Fuel Elements with Uranium Dioxide Pellets



Assembly in the Case

Ready Assembly

Figure 5 – Preparation for Reactor Experiment

The results of this work allowed us to work out proposals and recommendations on measures improving the safety and efficiency of work in case of severe accidents involving fast reactors, which are defined by the nuclear community as one of the most promising areas for the further development of nuclear power systems.

The achieved results and the competences of NNC developed in the course of carrying out these experiments are currently in demand for new experimental programs preparation. For example, French Atomic Energy Commission and Alternative Energy Sources (CEA) expressed its interest in implementing a joint fuel testing program for the ASTRID Advanced Demonstration Reactor based on IGR [19].

NNC and CEA conducted preliminary studies to determine the possibility and conditions for the testing of ASTRID reactor assemblies [20, 21]. The results of the work performed showed the principal feasibility of tests' requirements. Thus, for example, during the experiment the circulation of liquid metal coolant through FA at a given flow rate will be ensured by the creation of a sodium loop. The creation of such a circuit will also significantly expand the available experimental possibilities for future research.

CONCLUSION

The operational capabilities of the IGR reactor and the EAGLE test-bench allow the implementation of studies for obtaining of experimental information on physical and thermal processes in nuclear reactors and the behavior of fuel and core structural materials of nuclear power facilities in transient and accident modes. The results of the experiments conducted are particularly important for the industrial and scientific communities. NNC and JAEA are ready to maintain and further develop cooperation in this regard.

ACKNOWLEDGEMENT

The authors express their gratitude to the EAGLE project staff from both NNC and JAEA for long-term fruitful work.

REFERENCES

1. N.A. Nazarbayev, V.S. Shkol'nik, E.G. Batyrbekov, etc. Carrying Out Scientific, Technical and Engineering Works to Bring the Former Semipalatinsk Test Site to a Safe Conditions (in 3 volumes) // RSE "National Nuclear Center of RK", Ministry of Energy RK, Kurchatov – 2016. – Volume 1 – p. 320. Volume 2 – p. 448. Volume 3 – p. 596.
2. GEN IV International Forum, "GIF R&D Outlook for Generation IV Nuclear Energy System", 21 August 2009.
3. Reactor facilities for fuel pins and FA testing in emergency and transient operation modes / V. P. Burukin, A.V. Klinov, Yu.G. Toropov // Nuclear technology abroad. – 1988. No.6. p. 7–15.
4. Overseas programs of reactor research in emergency and transient modes of NPF fuel pins operation / V. P. Burukin, A. V. Klinov, Yu.G. Toporov // Nuclear technology abroad. – 1988. No.5. p. 3–7.
5. M.N. Devisheva. Overseas research and development programs on NPP safety based on water-cooled reactors: review – M.: CSRIatominform, 1989. – p. 44.
6. Nuclear Fuel Behavior Under Reactivity-Initiated Accident (RIA) Conditions: State-of-the-art Report / Nuclear Energy Agency, OECD. – Paris, 2010. – 210 p. – ISBN 978-92-64-99113-2, NEA/CSNI/R (2010).
7. I. V. Kurchatov, S.M. Feinberg, N.A. Dollezhal. Impulse Graphite Reactor IGR. Nuclear Power. – 1964. – V. 17. – No 6. – p. 463-474.
8. Kotake S, Sakamoto Y, Mihara T, Kubo S, Uto N, Kamishima Y, Aoto K, Toda M. Development of advanced loop-type fast reactor in Japan. Nucl Technol. 2010; 170:133–147.
9. Hishida M, Kubo S, Konomura M, Toda M. Progress on the plant design concept of sodium-cooled fast reactor. J Nucl Sci Technol. 2007; 44:303–308.
10. Kubo, Sh. Et al. Main outcomes and future plan of the EAGLE project / NNC RK Bulletin, issue 1, March 2016, p. 13-18.
11. Konishi, K. et al. The Eagle project to eliminate the recriticality issue of fast reactors; Progress and results of in-pile tests / Proceedings of 5th Korea-Japan Symposium on Nuclear Thermal Hydraulics and Safety (NTHAS-5), Jeju, Korea Nov. 2006, pp. 465–471.
12. Kamiyama K, Saito M, Matsuba K, Isozaki M, Sato I, Konishi K, Zuyev VA, Kolodeshnikov AA, Vassiliev YS. Experimental study on fuel-discharge behavior through in-core coolant channels. J Nucl Sci Technol. 2013;50(6):629–644.
13. Kamiyama, K. et al. Experimental studies on the upward fuel discharge for elimination of severe recriticality during core-disruptive accidents in sodium-cooled fast reactors. J Nucl Sci Technol. 2014; 51(9):1114-1124.
14. A.D. Vurim, V.A. Gaidaichuk, Yu.L. Istomin, Yu.V. Aleinikov, Zh.R. Zhotabayev. Experimental studies at the IGR reactor in support of the method determining the spatial position of fuel in experimental devices under conditions simulating disruptive severe accidents and fuel melting. NNC RK Bulletin, No.4, 2010, p. 33–40.
15. A.D. Vurim, Yu.A. Popov, V.A. Vityuk, Zh.R. Zhotabayev. Investigations in support of fuel mass indirect determination technique in the central experimental channel of IGR reactor on the parameters of thermal neutrons field. NNC RK Bulletin, No. 4, 2010, p. 41–49.
16. V.A. Vityuk, A.D. Vurim, V.M. Kotov, and A.V. Pakhnits. Determination of the parameters for fuel assembly tests in a pulsed graphite reactor - Atomic Energy, Vol. 120, No. 5, September, 2016, pp. 323–327, DOI:10.1007/s10512-016-0138-3.
17. Skakov, M., Mukhamedov, N., Deryavko, I. On the issue of substantiation of test modes for ampoule experimental device in research reactor. Bulletin of the Tomsk Polytechnic University-Geo Assets Engineering, vol. 328, issue 7, pp. 51–58.
18. Skakov, M., Mukhamedov, N., Deryavko, I., Wieleba, W., Vurim, A. Research of structural-phase state of natural corium in fast power reactors. Vacuum. 2017;141: 216–221.
19. The ASTRID technological demonstrator. 4th-Generation sodium-cooled fast reactors. –Tome 3, December, 2012., 96 pages.
20. F. Serre, F. Payot, C. Suteau, L. Trotignon, E. Batyrbekov, A. Vurim, A. Pakhnits, V. Vityuk, S. Kubo, A. Katoh, Y. Tobita, K. Kamiyama, K. Matsuba, J. Toyooka. R&D and Experimental Programs to support the ASTRID Core Assessment in Severe Accidents Conditions. – Proceedings of International Congress on Advances in Nuclear Power plants (ICAPP 2016), April 17-20, 2016 – San Francisco (CA, USA), Vol. 3, 2016, pp. 2173–2182.
21. F. Payot, F. Serre, A. Bassi, C. Suteau, E. Batyrbekov, A. Vurim, A. Pakhnits, V. Vityuk. The SAIGA experimental program to support the ASTRID Core Assessment in Severe Accident Conditions. – Proceedings of International Conference on Fast Reactors and Related Fuel Cycles: Next Generation Nuclear Systems for Sustainable Development (FR17), Yekaterinburg, June 26–29, 2017 (Paper ID IAEA-CN245-067).

НАТРИЙ ЖЫЛУ ТАСЫҒЫШЫНДАҒЫ ШАПШАҢ НЕЙТРОНДАРДАҒЫ РЕАКТОРЛАРДЫҢ ҚАУІПСІЗДІГІНЕ ҚОЛДАУ КӨРСЕТУДЕГІ ЭКСПЕРИМЕНТТІК ЗЕРТТЕУЛЕР

[1] Э.Ғ. Батырбеков, [1] М.Қ. Сқақов, [1] В.А. Витюк, [2] В.В. Бакланов, [2] А.Д. Вурим, [2] А.В. Пахниц, [3] К. Камияма, [3] К. Мацуба

[1] «Қазақстан Республикасының Ұлттық ядролық орталығы» РМК, Курчатов, Қазақстан
[2] ҚР ҰЯО РМК «Атом энергиясы институты» филиалы, Курчатов, Казахстан
[3] Шапшаң нейтрондардағы реакторлар жүйелерін зерттеу және әзірлеу орталығы, Жапонияның атом энергиясы агенттігі, О-арай, Ибараки, Жапония

Қазіргі уақытта ядролық реактордың қауіпсіздігі жөніндегі жаңа жобаларды негіздеуде ғылыми-зерттеу жұмыстарының ауқымды көлемі жүргізілуде, бұл ретте қолданылатын техникалық шешімдер жобадан тыс авариялардың туындау мүмкіндігін азайтатындығына қарамастан, мұндай аварияларға ерекше көңіл бөлінеді.
Ауыр авариялардың параметрлерін зерделеу және олардың салдарын жою шараларын әзірлеу үшін авариялық жағдайды нақты жағдайға барынша жақын модельдеуге мүмкіндік беретін эксперименттік база талап етіледі.
Ұлттық ядролық орталық (ҰЯО) – ядролық энергияны бейбіт мақсатта пайдалануды ғылыми және техникалық қолдауды қамтамасыз ететін, ядролық реактордың қауіпсіздігін күшейтуге бағытталған зерттеулер жүргізетін эксперименттік базасы бар Қазақстан Республикасының атом саласындағы озық зерттеу ұйымы [1].
Бұл мақалада ядролық реакторлардың қауіпсіздігін негіздеу үшін ҰЯО-ның эксперименттік базасында аталған салада басты шетелдік әріптес болып табылатын JAEA-мен ынтымақтастықта жүргізілетін зерттеулердің тұжырымдамасы мен негізгі нәтижелері, сондай-ақ эксперименттердің жаңа бағдарламаларын іске асыру жөніндегі ағымдағы жоспарлар ұсынылған.
Түйінді сөздер: *шапшаң нейтрондардағы реактор, ауыр авария, импульстік графитті реактор, экспериментік бағдарлама.*

ЭКСПЕРИМЕНТАЛЬНЫЕ ИССЛЕДОВАНИЯ В ПОДДЕРЖКУ БЕЗОПАСНОСТИ РЕАКТОРОВ НА БЫСТРЫХ НЕЙТРОНАХ С НАТРИЕВЫМ ТЕПЛОНОСИТЕЛЕМ

[1] Батырбеков Э.Г., [1] Скаков М.К., [1] Витюк В.А., [2] Бакланов В.В., [2] Вурим А.Д., [2] Пахниц А.В., [3] Камияма К., [3] Мацуба К.

[1] РГП «Национальный ядерный центр Республики Казахстан», Курчатов, Казахстан
[2] Филиал «Институт атомной энергии» РГП НЯЦ РК, Курчатов, Казахстан
[3] Центр исследований и разработки систем реакторов на быстрых нейтронах, Японское агентство по атомной энергии, О-арай, Ибараки, Япония

В настоящее время проводится большой объем научно-исследовательских работ в обоснование новых проектов по безопасности ядерного реактора, в которых особое внимание уделяется запроектным авариям, несмотря на тот факт, что применяемые технические решения сокращают вероятность возникновения таких аварий.
Для изучения параметров тяжелой аварии и разработки мер ликвидации ее последствий требуется экспериментальная база, которая позволяет моделировать аварийную ситуацию максимально близко к реальным условиям.
Национальный ядерный центр (НЯЦ) – лидирующая исследовательская организация в атомной сфере в Республике Казахстан – обладает такой экспериментальной базой, обеспечивая научную и техническую поддержку мирному использованию ядерной энергии, проводя исследования, направленные на усиление безопасности ядерного реактора [1].
В данной статье представлены концепция и основные результаты исследований, проводимых на экспериментальной базе НЯЦ в обоснование безопасности ядерных реакторов, в сотрудничестве с JAEA (Японское агент¬ст¬во по атомной энергии), которое является главным иностранным партнером в данной сфере, а также текущие планы по реализации новой программы экспериментов.
Ключевые слова: *реактор на быстрых нейтронах, тяжелая авария, импульсный графитовый реактор, экспериментальная программа.*