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Codes for NPP severe accident simulation: development, validation and applications

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Abstract: The software tools that describe various safety aspects of NPP with VVER reactor have been developed at the Nuclear Safety Institute of the Russian Academy of Sciences (IBRAE RAN). Functionally, the codes can be divided into two groups: the calculation codes that describe separate elements of NPP equipment and/or a group of processes and integrated software systems that allow solving the tasks of the NPP safety assessment in coupled formulation. In particular, IBRAE RAN in cooperation with the nuclear industry organizations has developed the integrated software package SOCRAT designed to analyze the behavior of NPP with VVER at various stages of beyond-design-basis accidents, including the stages of reactor core degradation and long-term melt retention in a core catcher. The general information about development, validation and applications of SOCRAT code is presented and discussed in the paper.

Keywords: *nuclear power plant, safety, calculation codes, severe accident*

A long-term IBRAE RAN experience in developing software has allowed formulating the methodological approach that includes the following basic directions:

- Working out the models based on the equations of mathematical physics and modern knowledge of processes and the phenomena that occur at various operating modes of reactor installations;

- Based on these models, development and validation of the deterministic computer codes for nuclear power plants safety assessment;

- Calculation-based and theoretical works to support the experimental programs;

- Application of the developed software complexes for safety analysis of nuclear power plants.

Using the physical approaches to develop models allows considerable improving of the process and phenomenon modelling quality and reducing uncertainty of calculation results. The software efficiency is tested through validation against experimental data. While doing this, the assessment of the existing knowledge base on physical processes and phenomena is being conducted that allows formulating the tasks for experimental studies more accurately. Participation of IBRAE RAS in the integral experiments is of a special significance since it allows verifying the new physical and numerical models in self-consistent way. While developing models and codes for severe accident analysis, the basic uncertainties of the used physical models have been revealed; estimations of applicability of the existing codes to the safety analysis of NPP with various reactors have been made.

Most of IBRAE RAS codes are developed within the frameworks of joint projects with the Russian and international nuclear stakeholders.

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Table I. Codes developed at IBRAE RIAN in cooperation with Russian and foreign institutions

Emergency stage	*Basic IBRAE RAN codes*	*Partners*
Early stage of reactor core degradation	SVECHA, QUENCH, MFPR	NRC/IPSN/EC FZK/RIAR
Late stage of reactor core degradation	CONV2D&3D LOHEY	OECD/RRC KI/ IPSN
Interaction of melt with concrete and catcher	RASPLAV/SPREAD	SPbAEP, NITI, AEP
Containment mechanics	CONT	REA/AEP, NRC/DOE

Thus, in the late 1990s, the works on development of the Russian code for safety analysis of new designs of NPP with VVER in conditions of severe accidents were started upon the initiative of JSC "SPbAEP" in cooperation of expert teams from IBRAE RAN, Russian Federal Nuclear Center “All-Russian Research Institute of Experimental Physics” (FSUE RFNC VNIIEF) and National Research Center «Kurchatov Institute» (NRC KI). Later, this code, which received the name SOCRAT, started to be applied also for safety assessment of the VVER projects operated or constructed in Russia. In 2010, the basic version of the code SOCRAT was certified by the Russian regulator (Rostehnadzor). Since 2011, the work has been conducted on developing and validation of the advanced version of the code that allows assessments of the radiological consequences of severe accidents. The quality of the models and validation allow considering the SOCRAT as a best-estimate code. Integration of numerous physical models into one code provides end-to-end modelling of all essential stages of severe accidents and obtaining of the entire picture of the accident evolution from a moment of its occurrence (initiating event) up to release of radioactive fission products out of the NPP containment into the environment.

Thermohydraulic models of the integrated code SOCRAT describe the behavior of the two-phase coolant with non-condensable gases in the core, primary and secondary circuits of a reactor installation at all stages of severe accident including stage of total core un-cover and stage of in-vessel melt retention. They include the various modes of coolant flow, interphase interactions, various modes of heat exchange with walls of hydraulic channels, friction at channel walls, presence of non-condensable gases, coolant ejection under containment. Also, the models of the SOCRAT code allow describing the operation of pumps, valves, hydraulic reservoirs and other elements of reactor installation equipment. The set of the basic elements used to model the input deck of the primary and secondary circuits, allows describing the tracing of any hydraulic loops with the accuracy that is sufficient for modern calculations of severe accidents.

Thermohydraulic processes in a system of communicating containment rooms in are modelled self-consistently using the integrated in SOCRAT containment codes KUPOL-M and ANGAR, representing the certified codes with lumped parameters.

Physical mutually-consistent models describing the processes of fuel cladding oxidation by steam, thermomechanical behavior of fuel rods and absorbers, melting of reactor core and other in-vessel materials, melt relocation are used for numerical analysis of severe accidents at a stage of reactor core degradation. While doing this, the real material composition of the reactor core is being taken into account.

Code SOCRAT allows modelling the processes of melt interaction with water at a stage of melt retention in the lower plenum, formation and distribution of a corium liquid phase, stratification of metal and oxide components, reactor pressure vessel degradation and melt release into containment.

The basic NPP objects that are modelled by the code SOCRAT in the advanced version are presented in Fig. 1. They are as follows:

- Fuel;
- Fuel assemblies;
- Reactor core and in-vessel structures;
- Reactor coolant system including safety systems;
- Steam generator and main steam line;
- Containment.

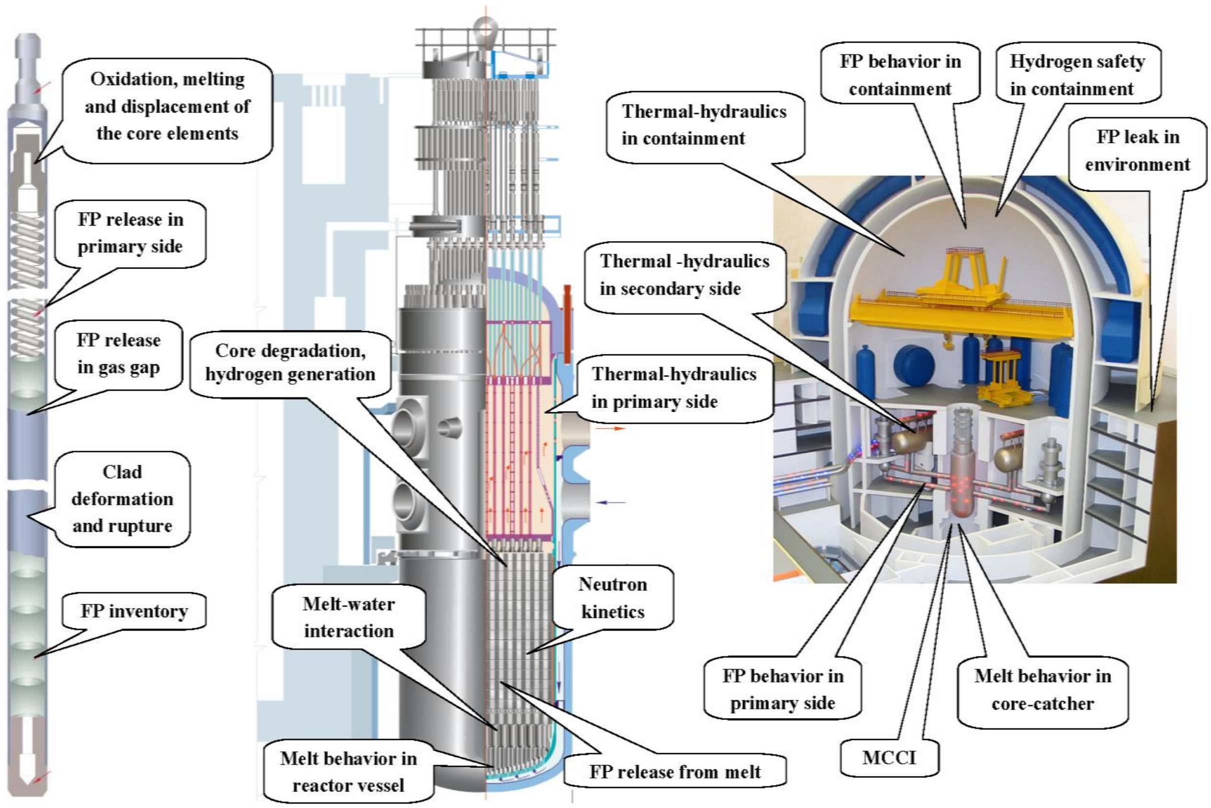


Fig. 1. Phenomena modeled in SOCRAT

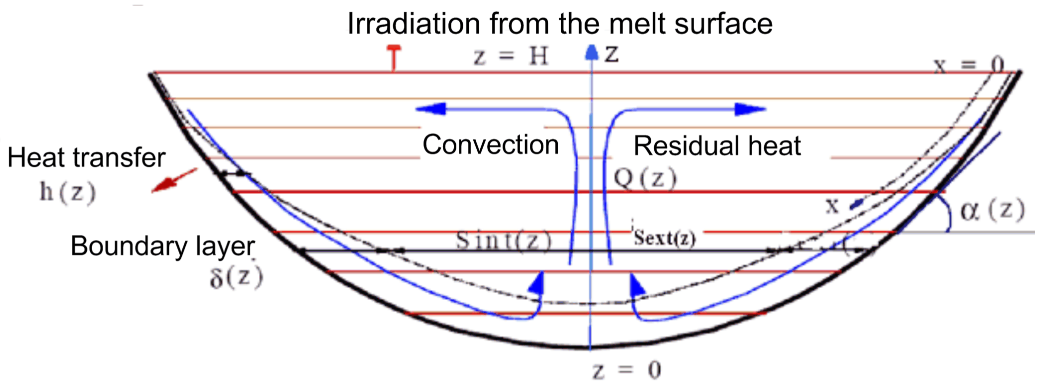


Fig. 2. Main processes simulated in the layer of homogeneous melt

The basic physical models of the integrated code SOCRAT at in-vessel stage of accident are presented below:

The advanced version of the integrated code SOCRAT allows carrying out calculations of parameters required for assessing the radiological consequences of severe beyond design-basis accidents at NPP with VVER reactors and, in addition to the basic version, describes in details the following processes:

- Buid-up of radioactive fission products (FP) in fuel and their release into the fuel rod's gas gap;

- Transport and sedimentation of radioactive fission products in various physical and chemical forms in the reactor primary circuit and in the containment;

- Release of radioactive fission products into environment.

Permanent validation of the SOCRAT code as well as of its physical models is one of the most important stages of the development and application. Models and algorithms of the SOCRAT code have passed all-round assessment against large data set, received in separate effect tests and integral experiments performed in Russia and abroad.

The experimental programs that were used for the code validation are as follows: CORA, QUENCH (Germany), PHEBUS (France), RASPLAV, MASCA (Russia - OECD), ISTC/PARAMETER, ERCOSAM-SAMARA (joint Rosatom-Euroatom project), LOFT, PBF, international standard problem ICSP MASLWR, international benchmark BSAF (analysis of the accident at the Fukushima Daiichi NPP).

Fig. 3 shows the calculated and measured temperatures of the surface of the fuel assembly simulator in the PARAMETER/SF1 experiment. The PARAMETER program investigates phenomena associated with re-flooding of a degrading VVER like core under postulated severe accident conditions, in an early phase when the geometry is still mainly intact. The figure confirms that the SOCRAT code correctly reflects the dynamics of the fuel assembly temperature behavior at all stages of the experiment (heating up, oxidation, and overheated core re-flood) under conditions of the presence of chemical power sources and convection and radiation heat exchange. This results from a sufficiently large set of models of SOCRAT code and their validation in a wide range of initial data.

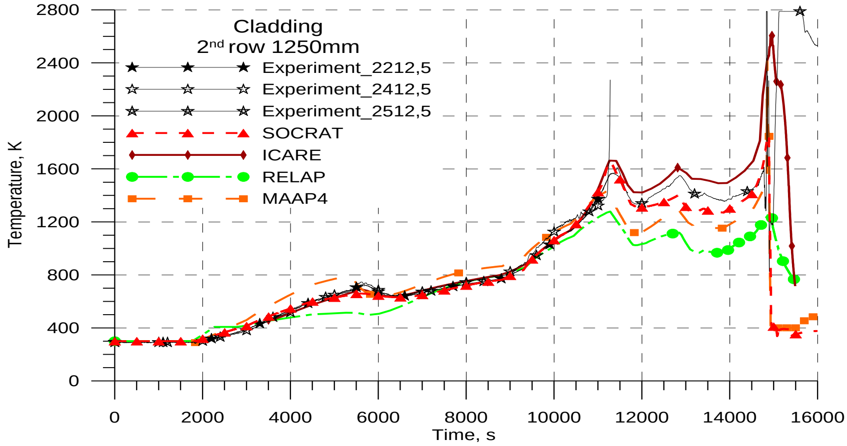
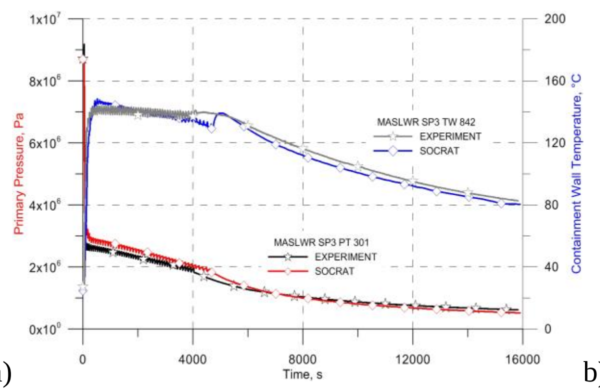
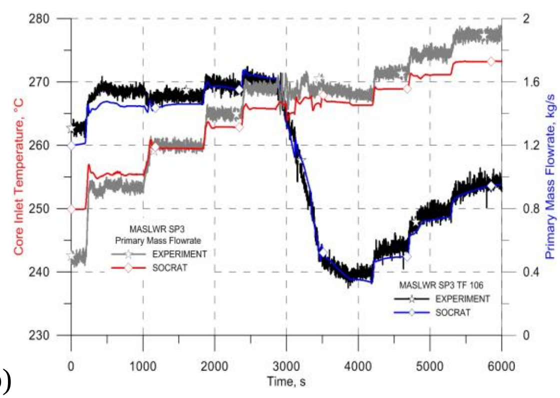


Fig. 3. Modeling of fuel assembly temperature behavior in the PARAMETER/SF1 experiment /2/

Other example of SOCRAT validation is participation in cooperation with JSC «OKB "GIDROPRESS"» at all stages of the international standard problem (ISP) «Evaluation of Advanced Thermohydraulic System Codes for Design and Safety Analysis of Integral Type Reactors». In this exercise, an accident with a feed-water loss in the secondary circuit (test SP2) and a maneuvering mode of the reactor operation (test SP3) were investigated in a series of two integrated experiments on scale model of perspective reactor MASLWR with passive safety systems. Comparison of the calculated and measured pressures of the primary circuit and temperatures in containment for the SP2 test is demonstrated in Fig. 4a. The close agreement between the experimental and calculated data testifies the correct and consistent work of models of coolant flow and heat exchange in the presence of non-condensable gases that is of special importance for a reactor installation with passive safety systems. Fig. 4b presents the coolant temperature at the entrance to reactor core and its flow rate for the SP3 test. Modelling of this test resulted in a good agreement with the results of measurements of not only temperature, but also flow rate parameters of two-phase coolant of the primary circuit in the natural circulation mode. As a whole, the SOCRAT code is capable to simulate the thermohydraulic behavior of a reactor installation even prior to the beginning of essential reactor core degradation.



a)



b)

Fig. 4. Modelling of the primary circuit pressure and containment temperature in the SP-2 test (a) and the primary circuit coolant temperature and its flow rate in the SP-3 test (b)

The code went through the practical test and approbation in 2011, when a severe accident stroke the Japanese NPP "Fukushima-Daiichi" on March 11. The express analysis was conducted at IBRAE RAN using the integral SOCRAT code /3/. The possible consequences of the accident, the forecast and characteristic times of emergency process development in reactor cores and SNF pools for the power units 1-4 were estimated /Fig.5/.

Unit #	SOCRAT Timing	Real timing
1	12.03 16:25	12.03 15:36
2 (peak of pressure in containment after water ingression in the core)	15.03 05:45	15.03 06:14
3	14.03 08:00	14.03 11:01
4	15.03 21:00	15.03 06:00

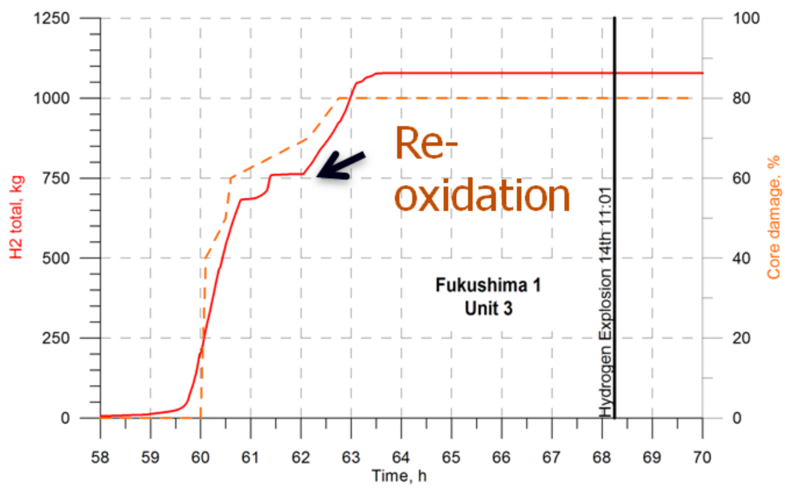


Fig. 5. Estimated amount of hydrogen generated at Unit 3 by the time of explosion

The following sequence of processes was analyzed:

- Decrease of the coolant level in reactor core;

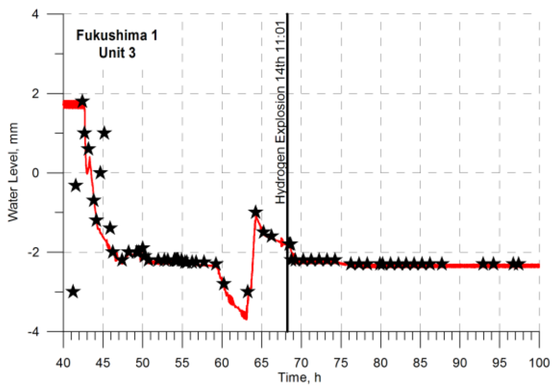
- Increase of containment pressure;

- Temperature increase and hydrogen generation;

- Release of hydrogen and fission products;

- Further degradation of reactor core.

Fig. 5 shows the calculated and real moments of hydrogen explosion at different units. Comparison of the calculated and measured data for the mass level of coolant in the reactor core of the Unit 2 and for the pressure in the primary circuit for the Unit 3 is shown in Fig. 6. The figure demonstrates that the SOCRAT code qualitatively describes the processes of heating, degradation and reflooding of reactor core. The calculations were based on the assumptions that the power of decay heat release corresponded to the typical BWR project, and data on water injection in the reactor coolant system and on safety system operation corresponded to TEPCO evidences that were available at the time of the accident.

a) 

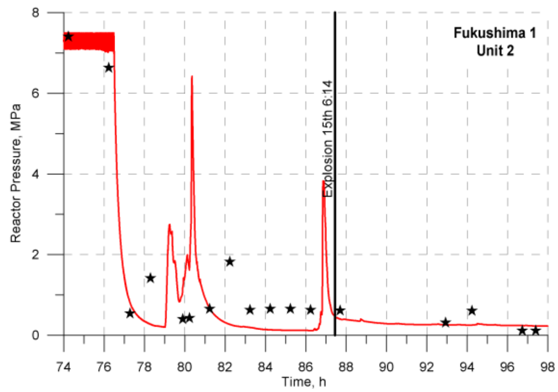
b) 

Fig. 6. Comparison of the operational characteristics of the BWR-4 reactor installation measured during the accident at the NPP Fukushima Daiichi with those calculated using the SOCRAT code: (a) Change of a water level in the reactor core of 3rd power unit; (b) Change of pressure in the primary circuit of 2nd power unit

IBRAE RAN has continued this work by joining the OECD-NEA Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (BSAF) Project conducted by Tokyo Engineering Power Company (TEPCO) and the Nuclear Energy Agency of the Organization for Economic Co-operation and Development (NEA/OECD).

Today SOCRAT code is widely used by the leading Russian design and scientific organizations for analysis of beyond design-basis severe accidents at NPP with reactors on thermal neutrons with water coolant, for assessment of hydrogen safety, efficiency of melt retention systems, and for analysis of efficiency of NPP passive safety systems.

The typical thermohydraulic model of the primary circuit of VVER-1000/B-320 reactor is presented in Fig. 7. It allows a quite detailed modeling of beyond design-basis accidents with loss of coolant in the primary circuit in a wide range of locations and diameters of leaks.

The full list of the SOCRAT code applications is quite wide. It can be noted that it is used for the following units with VVER reactor: VVER-440/230 (Kola NPP), VVER-1000/B320 (the Balakovo NPP), VVER-1000/B428 (China), VVER-1000/B412 (India), VVER-1500/B448, VVER-1200/B392м (NVNPP-2), VVER-1200/B491 (LNPP-2).

Presently, the code is used at SPbAEP, AEP, OKB GP, NRC KI, IPPE, and is transferred to MPEI as a tool of training of students and post-graduate students.

In 2012, IBRAE RAN experts prepared and conducted a course of lectures for the Vietnamese specialists that were trained at the Central Institute for Advanced Training (TsIPK) Obninsk, Russia: Training course: “Application of computer codes for safety analysis of NPPs. Deterministic Safety Analysis and code SOCRAT”. This course included 2 weeks of 96 hrs training. Of them, the lectures took about 55 hrs, practical work - 41 hrs, and one day was devoted to testing.

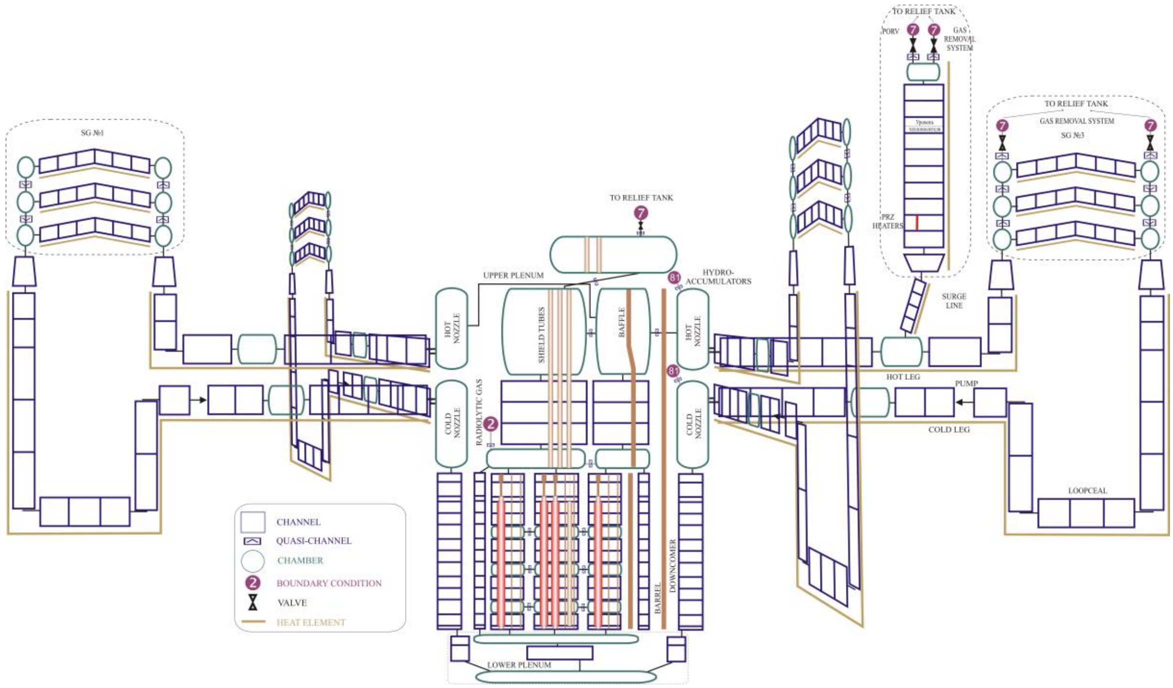


Fig. 7. Typical nodalization scheme of VVER-1000 reactor installation with passive safety systems used in the SOCRAT code

The further development of the SOCRAT code includes the following:

1. Improvement of the current version of the integrated code SOCRAT, participation in international benchmarks in order to verify the code, adaptation of physical models and computing algorithms for various designs of reactors with thermal neutrons and water coolant, preparation and training of new users.

2. Development of the new version of the integrated code SOCRAT-BN for modelling of physical processes in reactors with fast neutrons and sodium coolant, that is being done based upon practical experience received by the IBRAE RAN.

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