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Testing of the system code designed for simulation of hypothetical beyond design-basis accident on fast breeder reactor

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Abstract

The purpose of the present study was to develop integrated system of codes (ISC) for performing continuous self-consistent calculation of the whole life cycle of fission products (FP) in the nuclear power plant with nuclear reactor unit (RU) equipped with fast reactor (BN) starting from accumulation of FPs in the reactor core to their exit in the environment and migration outside the NPP site territory. SOKRAT-BN integrated code was used in the calculations in combination with KUPOL-BN and NOSTRADAMUS codes.

Practical importance of the study is attributed to the development of the ISC for substantiation of BN RU safety. Simulation of hypothetical beyond design-basis accident at the NPP equipped with BN RU accompanied with escape of radioactive isotopes in the reactor premises was performed as the test task. Results of solution of the test problem confirm practical applicability of the developed ISC. Development of the methodology for simulation of migration and precipitation of radioactive impurities in the sodium coolant represents scientific novelty. Software module TRANS-FR designed for simulation of transport of radioactive fission products (RFP) and radioactive corrosion products (RCP) in the primary cooling circuit and in the gas system of the RU was developed and integrated in the SOKRAT-BN ISC taking into account the main physical processes taking place during transport and accumulation of RFP and RCP.

Software interface modules for data exchange between the SOKRAT-BN, KUPOL-BN and NOSTRADAMUS codes were developed and tested.

Keywords: Severe beyond design-basis accident; KUPOL-BN code; NOSTRADAMUS code; SOKRAT-BN integrated code; Fast reactor; Fission products; Substantiation of BN reactor safety.

Introduction

Russia has a long waiting list of foreign orders for construction of nuclear power generation facilities – more the fifty countries would like nuclear power reactors to be built on their territories, and fast reactors are also found in this list. A complex of studies needed to substantiate safety of the nuclear power unit has to be performed for obtaining the license to implement construction works, commissioning, operation and decommissioning of the power unit of nuclear power installation on the specific site. In order to prove safety of the RU and to obtain permits from oversight agencies supporting calculations must be made using licensed computer codes.

Dedicated codes were developed at the SSC RF-IPE in the process of design and substantiation of Russian fast reactors BN-350, BN-600 and BN-800. Review of some of these codes is contained in the monograph [1]. BOS-TWC codes calculating in two-dimensional approximation sodium

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boiling in reactor fuel assemblies, COREMELT code for calculation of melting of reactor fuel assemblies and melt drainage, TWOCOM code for calculation of yield of gaseous fission products from fuel pins, BRUT code for simulation of molten matter retention inside the reactor vessel and ANPEX code for investigation of reactor runaway on prompt fission neutrons in case of secondary criticality were developed. KUPOL code [2] for analysis of hydrogen safety and its subsequent version KUPOL-BN intended for fast reactor simulation were developed as well.

At present requirements imposed on reactor safety and, consequently, requirements imposed on computer codes are steadily becoming more and more stringent. Increasing computer capacities allowed practically addressing the task of development of multi-physics codes. Integrated software tool which in the field of nuclear engineering combines neutron physics calculation module, thermal hydraulics loop module, as well as modules for calculation of yield of fission products (FP), activation of corrosion products (CP), containment module for analysis of hydrogen safety, module describing transfer and deposition of radioactivity in the environment surrounding the NPP and analysis of exposure loads, etc. is understood under the term multi-physics code. In order to satisfy international safety requirements pertaining to reactor safety substantiation it is necessary to perform analysis of consequences of initiating events (IE) leading to severe accidents (SA) or to beyond design-basis accidents (BDBA). Experimental simulation of such processes is usually extremely expensive, hazardous and hardly feasible in practical terms.

In pursuance with Paragraph 6.2, RB-044-09 Standard [5]: “Studies of beyond design-basis accident are recommended to be implemented using system software describing in an integrated manner the development of different processes (from the initiating event to the emergency radioactivity release) and the events during the design-basis accident”. The above listed domestically developed software instruments, although they are capable to calculate the processes in the BN-type reactor facility in emergency operation mode, do not allow implementing the integrated once-through calculation of severe accidents in reactor facilities of BN type. Integrated software tool (IST) SOKRAT-B1 [3] designed to calculate VVER reactors developed at the Nuclear Safety Institute of the Russia Academy of Sciences was used in the substantiation of safety of VVER-100 reactors in China, India, etc.

At present the base version of the SOKRAT-BN IST developed on the basis of the SOKRAT-B1 IST and intended for calculation of physical processes taking place during the phases of heavy and severe accidents at the liquid-metal cooled nuclear power installations is available. For ensuring the unified self-consistent calculation of the behavior of fission products (FP) in sodium-cooled reactor facilities (BN RF), release of FPs in the environment for different BN RF operation conditions and calculation of FP behavior in the environment the SOKRAT-BN IST in combination with KUPOL-BN software complex and NOSTRADAMUS software complex are expected to simulate the following chain of events:

It is necessary to calculate generation and accumulation of FP isotopes in fuel pins in the reactor core, release of FPs in the fuel pin gas gap, destruction of the fuel pin cladding, release of FPs in the primary cooling circuit, transport of FPs along the RF circulation loops, deposition and washing away of the FPs, release of FPs inside the NPP buildings, transport of FPs to the places in the buildings and structures with loss of containment tightness formed as the result of the accident, and FP migration in the environment. Results of testing the SOKRAT-BN IST modules ensuring simulation of the whole chain of development of events described above are presented in the present study.

Simulation of migration of radioactive impurities in the SOKRAT-BN IST

The task of simulation of penetration of radionuclides in the coolant, their transfers and behavior within the circulation loops of the RF is allowed to be implemented by the SOKRAT-BN IST. In particular, the TRANS-FP computation module is included in the above IST designed to calculate the transfer and behavior of radioactive fission products (RFP) and radioactive corrosion products (RCP) in the primary circuit and in the reactor facility gas system.

The original version of the TRANS-FP module included empirical models of transfer and behavior of RCP and gaseous RFP (described in [6]) and simplified models of behavior of volatile and non-volatile FPs. Transfer equation is applied for description of FP and CP transport in the following form:

$$\frac{dC_i(t)}{dt} = S_i(t) - C_i(t)(R_i(t)+\lambda) - C_i(t)\sum_{\substack{j=1\\ j\neq 1}}^{Ncomp} Q_{ij}(t) + \frac{1}{V_i}\sum_{\substack{j=1\\ j\neq 1}}^{Ncomp} C_{ij}(t)Q_{ij}(t)V_j + \frac{S_i}{V_i}C_i^p(t)R_i^p(t).$$

The equation is solved jointly with equation describing the variation of concentration.

$$\frac{dC_i^p(t)}{dt} = \frac{V_i}{S_i}C_i(t)\ (R_i(t) - C_i^p(t)R_i^p(t+\lambda)$$

Here, j and i are the indices corresponding to channels; R_i is the rate of deposition; N_{comp} is the number of channels; $S_i(t)$ is the source of impurities in the channel; S_i is the total surface area of channel walls; R_i^p is the rate of re-suspension of impurities; Ci is the concentration of impurity in the suspended state; C_i^p is the concentration of impurity in the deposited state; and Q_{ij} is the rate of gas (liquid) exchange between the channels i and j.

Existing models are suggested to be supplemented with improved model of behavior of cesium isotopes in sodium coolant. Cesium may be present in the form of easily dissolvable metal impurity, it can be accumulated in depositions on the walls of elements of the first cooling loop of the RF, as well as released in the gas system [7]. Results of experimental studies of cesium behavior in sodium loops [8–10]