SOCRAT-BN SIMULATION OF SIENA LOSS-OF-FLOW EXPERIMENTS

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ABSTRACT

To solve actual problems in the area of safety analyses of sodium-cooled fast reactors (SFRs) IBRAE RAS develops SOCRAT-BN code system [1]. The code system is based on the “water” SOCRAT code system, which was developed for the PWR safety analysis. The SOCRAT-BN code system allows to carry out coupled analysis of design and beyond design basis accidents at SFR using thermal hydraulic, neutron kinetic, strain stress, etc. simulation. For complex thermal hydraulic simulation of sodium behavior one- and two-dimensional, non-homogenous, non-equilibrium two-fluid models were developed and included in the code system. One-dimensional model can be used to simulate loop pipes, since there is no need to know spatial distribution of physical parameters. Two-dimensional approach could be used to simulate reactor core and heat exchangers where we need to simulate spatial behavior of sodium coolant.

The paper describes the validation of the two-dimensional thermal hydraulic model on the experiments with fuel rod imitators in the triangular geometry. The approach of the simulation and obtained results is presented in the paper. Siena loss-of-flow experiments (Japan) with 19- and 37-pin rod bundles were simulated using two-dimensional approach and conventional closure relations for bundle geometry. Simulation shows the capability of the two-dimensional thermal hydraulic model to predict sodium boiling behavior, sodium and pin surface temperature evolution during pump rundown, etc. Although the numerical results are in good agreement with the experimental data, the two-dimensional thermal hydraulic model should be further improved. Other experiments are planned to be simulated, in order to further develop and validate the two-phase sodium flow model.

KEYWORDS
sodium-cooled fast reactor, sodium boiling simulation, SOCRAT-BN

1. INTRODUCTION

To simulate physical phenomena and processes in structural elements of the reactor during emergency processes and transients it is necessary to take into account spatial effects. Using the one-dimensional approximation in this case can lead to significant errors. For example, for beyond design basis accidents considering spatial effects required for modeling of sodium boiling in the fuel assemblies of SFR under conditions of loss of coolant flow without scram. In this case, the nonuniformity of cooling of the fuel rods is increasing and loss of coolant finally may lead to seal failure and melting of some of them. To account for these effects modernization of the thermal-hydraulic module of code SOCRAT-BN was performed through the introduction of a two-dimensional model that takes into account the of mass, energy and momentum transfer in both directions.

Validation of the implemented model was carried out on the experiments with pump rundown.

2. VALIDATION OF SOCRAT-BN CODE

Simulation of the experiments with loss of flow was carried out to validate the two-dimensional thermal hydraulics model of the SOCRAT-BN code. Experiments were performed in the SIENA sodium loops in Japan. The stand was a circulating sodium loop with various sets of assemblies with fuel rod simulators installed on the lifting section. Experiments were carried out with the following method. At first stand was kept under steady-state conditions (nominal flow rate of the coolant and constant heating). On the outer wall of the hexagonal duct compensating heater worked to keep assembly under adiabatic boundary conditions. Then, according to the postulated dependence pump flow rate decreased by reducing the supplied voltage. When dryout was detected electric heating was turned off to prevent failure of fuel rod simulators with tantalum core.

Electrical simulators were installed in a triangular package and distanced by a wire winding. The outer diameter of simulators 6.5 mm. With a 1.3 mm thick wire winding triangular lattice step was 7.9 mm. To simulate electrically heated rods the model that took into account the change of tantalum core resistance according to the temperature change were used. It became clear in a preliminary analysis that there was a DC voltage source used in the experiments, and therefore the heating decreased with temperature rise. In the calculations of the heating in one axial core cell was calculated by a simple formula.

$$P_i = \frac{P \cdot R_i}{R_{tot}} \qquad (1)$$

Where R_i - resistance in the axial layer, R_{tot} – total core resistance and P - total power at a constant voltage and current resistance of the core.

2.1. Experiment with 19-Pin Bundle

In one of the experiments [2] there was 19-pin assembly of fuel rod simulators installed at the SIENA on the lifting section. The assembly was enclosed in a hexagonal casing size 36.7 mm between opposite walls. Although the authors suggest that the power profile can be considered uniform height and radius, we take into account possible heterogeneity of the power in the core as was mentioned above. The average heat flux was 150 W / cm^2. Figure 1 shows a schematic representation of the test section. Sodium enters assembly from the bottom, flows through unheated input section. Temperature of the inlet sodium is equal to 366°C. Then sodium enters the heating section of the assembly. The heated coolant flows to the output section that does not have electric heating. Finally sodium goes into the expansion tank. In the expansion tank hot sodium mixes with cold sodium from the by-pass channel. Then it goes to the heat exchanger and back to the pump. Coming out of the flow of sodium assembly turned in expansion tank where it is mixed with sodium from the bypass channel. From the surge tank to the heat exchanger, sodium, and then the pump suction.

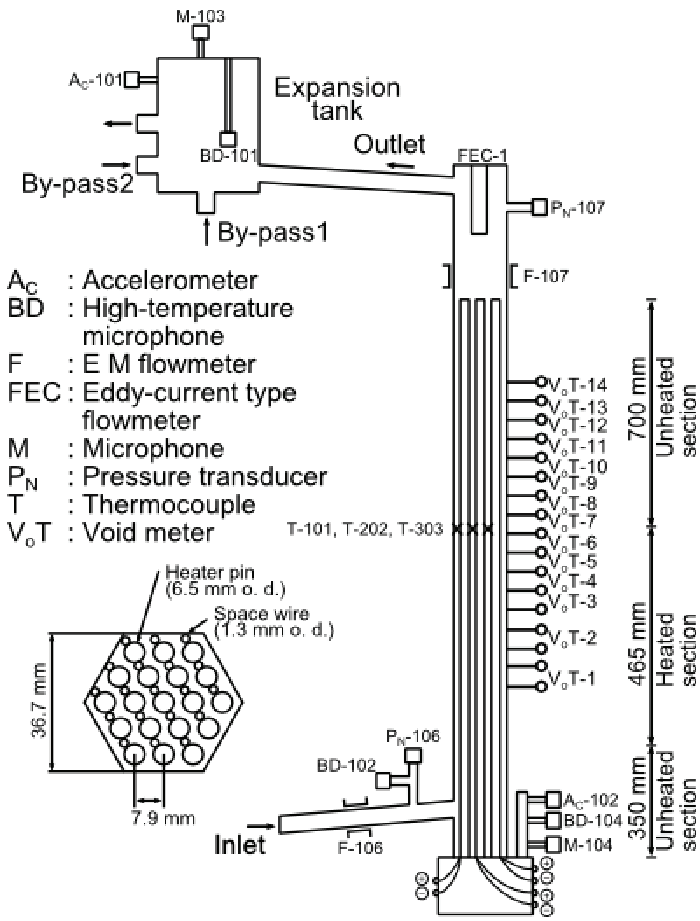

Figure 1. Test section with 19-pin subassembly.

Geometric parameters of the test assembly are shown in Table I. Fuel rod simulators can be divided in to three sections: inlet unheated section, heated section and outlet unheated section.

Table I. Geometric parameters of the 19-pin assembly

Total rod length, mm	1515
Inlet section length, mm	350
Heating length, mm	465
Outlet section length, mm	700
Rod diameter, mm	6.5
Triangular lattice step, mm	7.9
Hexagonal duct distance between opposite walls, mm	36.7

To measure the coolant temperature in the assembly three thermocouples were installed at the outlet of the heated section, figure 2.

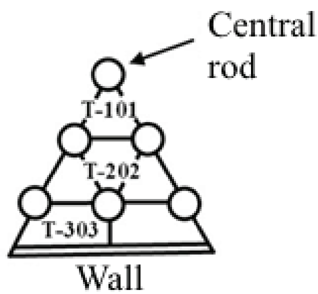

Figure 2. Thermocouple locations in 19-pin subassembly.

To simulate the experiment the nodalization scheme was developed. It includes the one-dimensional and two-dimensional hydraulic models. The input and output sections were simulated in one-dimensional approximation, and heated section and part of the adjoining outlet section were modeled in two-dimensional approximation. This is because the temperature in the experiment was measured only at the heated section at different distances from the center of the assembly. Pump and expansion vessel were modeled as boundary conditions for pressure. Nodalization scheme is shown in Figure 3. In the radial direction, the assembly was divided into 3 layers.

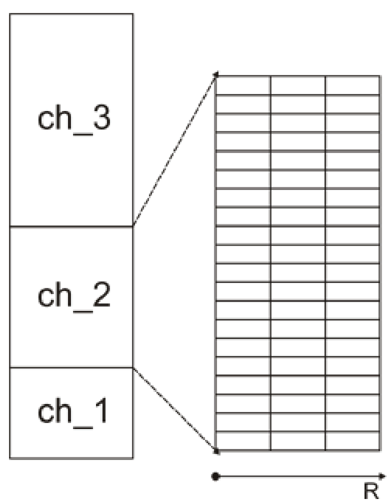

Figure 3. Nodalization scheme of 19-pin bundle.

Pump flow rate was decreased linearly; the graph of experimentally measured sodium inlet velocity and calculated with the SOCRAT-BN code velocity are plotted in Figure 4. The nominal value of velocity is 2.44 m/s.

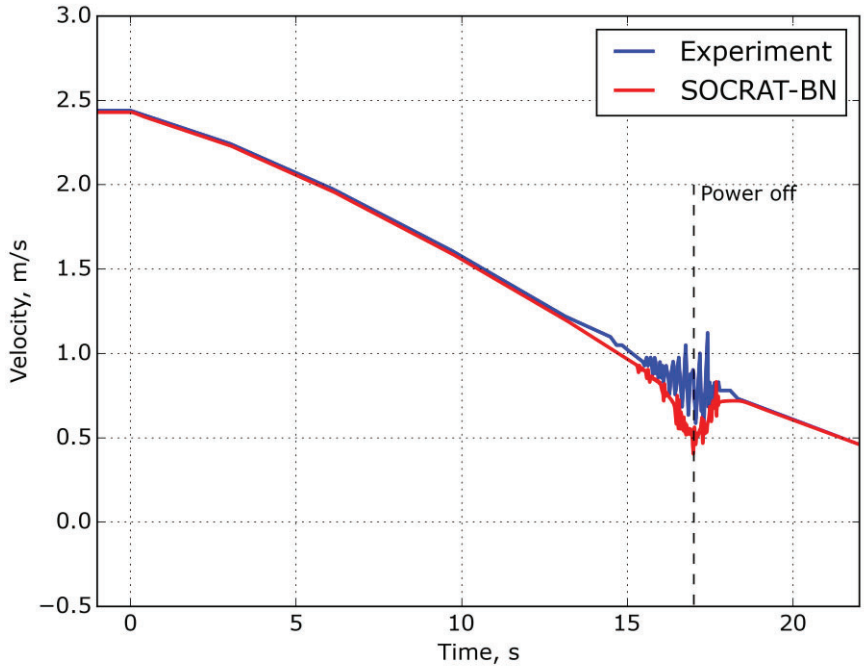

Figure 4. Sodium inlet velocity.

According to the velocity graph when the coolant starts boil at 15.4 second from the beginning of the pump rundown the resistance of the assembly increases and flow rate begins to decline rapidly. 17 seconds after the beginning of pump rundown heaters power was cut off, and the void fraction began to decrease and coolant velocity began to return to a linear dependence.

Figure 5 shows graphs of coolant temperature at the outlet of the heated section.

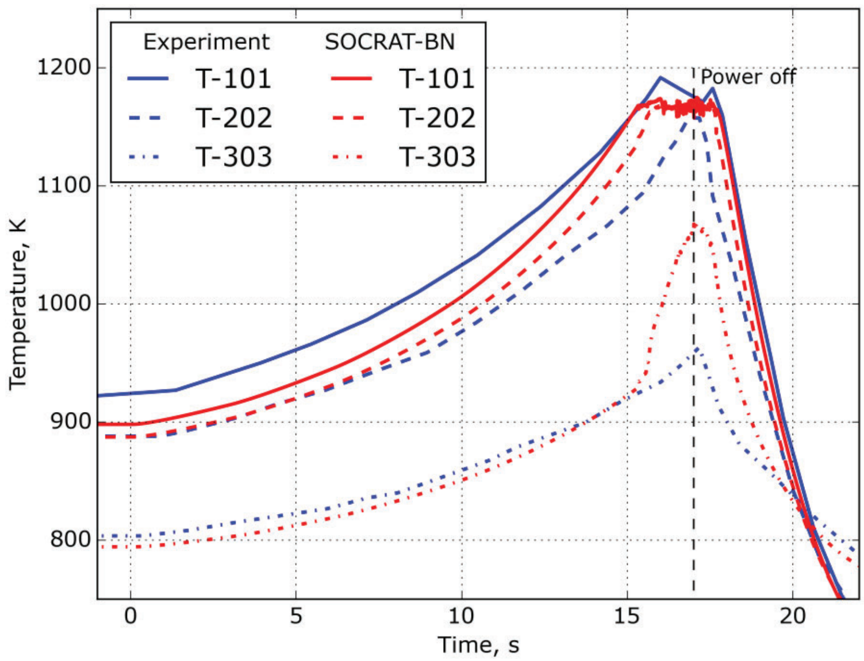

Figure 5. Coolant temperature at the end of heated section.

The graph shows that calculated temperature of the coolant increase faster than it was observed in the experiment. Calculated average heat flux to the coolant before boiling inception equals to 112 W/cm^2, compared to 150 W/cm^2 at a nominal flow rate of the coolant. Such reduction of heat flux is not fully explain the difference in temperature rates, based on the comparison with experiment. Perhaps compensating electric heating and thermal insulation were not providing adiabatic conditions on the assembly housing.

Figure 6 shows graphs of void fraction at the exit of the heated area. The solid red line represents the calculated void fraction in the central radial layer. The dotted red line shows the calculated void fraction reduced to the total cross section of the assembly. In the experiment, void fraction was measured by the resistance between the electrodes mounted on opposite walls of the housing.

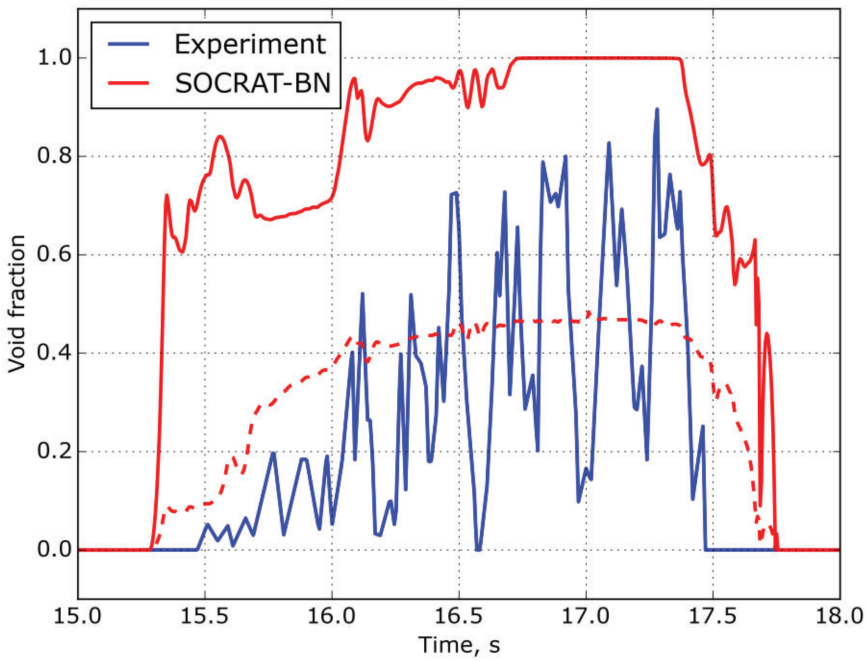

Figure 6. Void fraction at the end of heated section.

The graphs shows that the difference in boiling points of the coolant is 0.2 second. The calculation showed that steam was concentrated in the central part of the assembly.

2.2. Experiment with 37-Pin Bundle

The other experiment was performed with 37-pin bundle [2]. The average steady-state heat flux was 50.8 W/cm^2 and before boiling it equals to 42.2 W/cm^2. Figure 7 shows a schematic representation of the test section. Sodium enters assembly from the bottom, flows through unheated input section. Temperature of the inlet sodium is equal to 480°C.

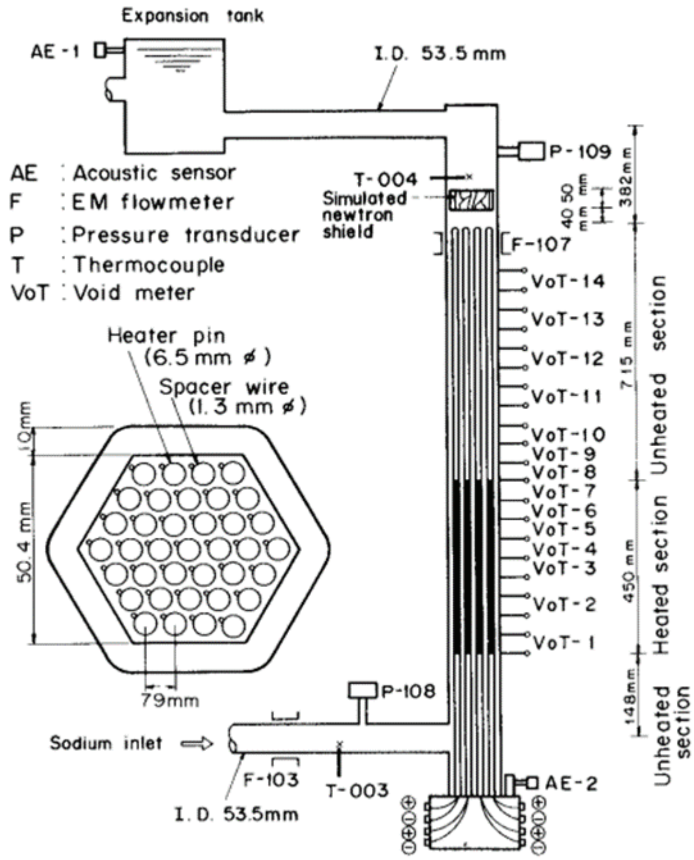

Figure 7. Test section with 37-pin subassembly.

Geometric parameters of the test section are shown in the table II.

Table II. Geometric parameters of the 37-pin assembly

Total rod length, mm	1313
Inlet section length, mm	148
Heating length, mm	450
Outlet section length, mm	715
Rod diameter, mm	6.5
Triangular lattice step, mm	7.9
Hexagonal duct distance between opposite walls, mm	50.4

Heaters wall temperature was recorded during pump rundown. Thermocouples were placed at the outer wall of the fuel rod simulators, figure 8. Shaded rod was broken and the experiment was conducted with 36 rods heated.

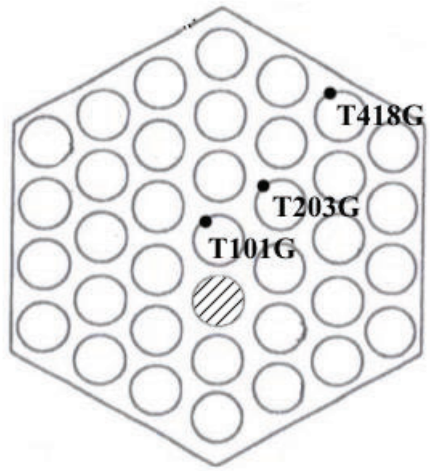

Figure 8. Thermocouple locations in 37-pin subassembly.

This experiment was modeled with the similar nodalization scheme. There were four layers in radial direction.

At the initial time the coolant inlet coolant velocity was 5.10 m/s. Figure 9 shows calculated inlet coolant velocity compared with measured one.

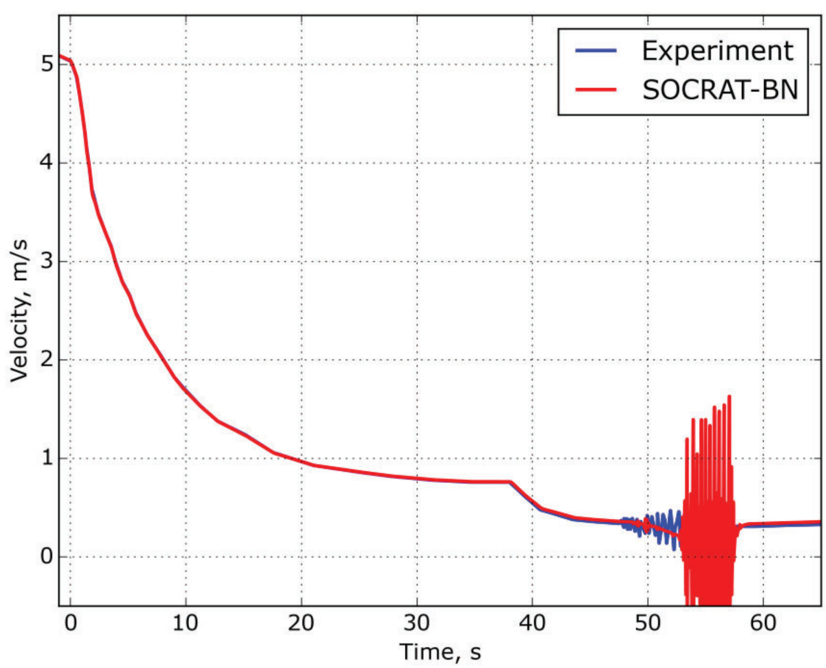

Figure 9. Sodium inlet velocity.

Figure 10 shows graphs of rods temperature at the outlet of the heated section.

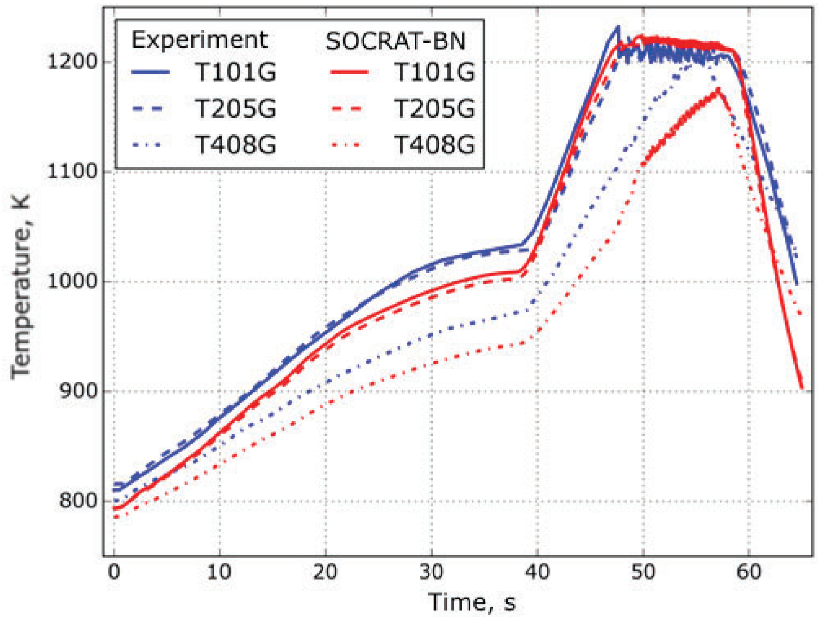

Figure 10. Thermocouple locations in 37-pin subassembly.

Calculated average heat flux to the coolant before boiling inception equals to 39.8 W/cm^2, while in the experiment it was 42.2 W/cm^2.

3. CONCLUSIONS

The results of the verification calculations show that the integral code SOCRAT-BN truthfully describes the hydraulic behavior of two-phase sodium flow. Used in the code SOCRAT-BN two-dimensional two-phase fluid model take allows to calculate transients under the rod geometry. Therefore, the code can be used to calculate nuclear core of fast reactors with liquid sodium as coolant.

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