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Thermal-hydraulic calculations for a fuel assembly in a European Pressurized Reactor using the RELAP5 code

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Abstract. The main object of interest was a typical fuel assembly, which constitutes a core of the nuclear reactor. The aim of the paper is to describe the phenomena and calculate thermal-hydraulic characteristic parameters in the fuel assembly for a European Pressurized Reactor (EPR). To perform thermal-hydraulic calculations, the RELAP5 code was used. This code allows to simulate steady and transient states for reactor applications. It is also an appropriate calculation tool in the event of a loss-of-coolant accident in light water reactors. The fuel assembly model with nodalization in the RELAP5 (Reactor Excursion and Leak Analysis Program) code was presented. The calculations of two steady states for the fuel assembly were performed: the nominal steady-state conditions and the coolant flow rate decreased to 60% of the nominal EPR flow rate. The calculation for one transient state for a linearly decreasing flow rate of coolant was simulated until a new level was stabilized and SCRAM occurred. To check the correctness of the obtained results, the authors compared them against the reactor technical documentation available in the bibliography. The obtained results concerning steady states nearly match the design data. The hypothetical transient showed the importance of the need for correct cooling in the reactor during occurrences exceeding normal operation. The performed analysis indicated consequences of the coolant flow rate limitations during the reactor operation.

Key words: fuel assembly • pressurized water reactor (PWR) • safety analysis • RELAP5

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European Pressurized Reactor

The European Pressurized Reactor (EPR) is an light water reactor in which light water H_2O is used as moderator (for slowing-down neutrons) and core coolant (for heat removal). The EPR was designed by a French company AREVA on the basis of experience gained during many years of operation of N4 and KONVOI reactors. Electric power generation in a nuclear power plant with the EPR is similar to that of a conventional power plant with ranking cycle. A significant difference between these two systems is visible during the system shutdown. After the reactor shutdown in a nuclear power plant, the so-called decay heat is generated, as a result of decay of fission products and actinides. Although it decreases relatively fast, as shown in Fig. 1, it is a major concern for the power plant operators. The uncontrolled decay heat and disability of the core structures cooling are the main reasons for core damages and severe accidents. The problems created by nonremoval of decay heat were visible during the events in the Fukushima Daichii Nuclear Power Plant in 2011, when the damage to the reactor cores and the release of the radioactivity necessitated evacuation. For

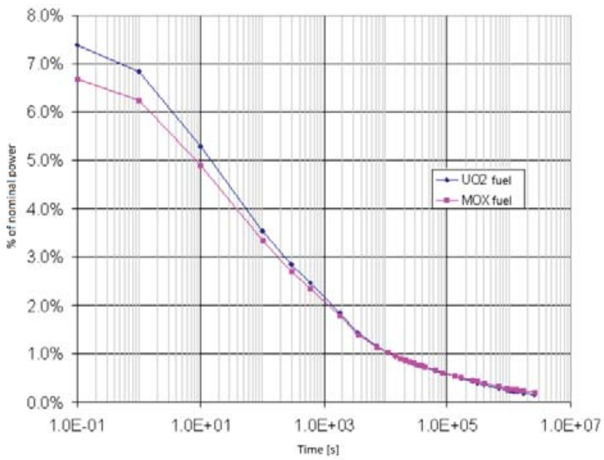

Fig. 1. Decay heat in an EPR [2].

safety reasons, mentioned heat must be removed [1]. After the reactor shutdown, that is, after neutron absorber rods are inserted, over 7% of the reactor nominal power (decay heat) is still generated, which in the case of the EPR equals about 315 MW [2]. Because the shutdown reactor and spent fuel pools need to be cooled constantly, all necessary efforts shall be made to provide cooling water even if electric power for driving pumps is no longer supplied.

The EPR belongs to a group of pressurized water reactors (PWR), which have two circuits: primary and secondary ones. Its distinctive features are modern detection devices, safety systems and special core catcher, that is designed to keep the molten corium outside of the reactor pressure vessel, but in control over its temperature and high decrease of the radioisotopes (in the form of volatile species) releases to the containment atmosphere. The reactor core itself consists of 241 fuel assemblies, being at

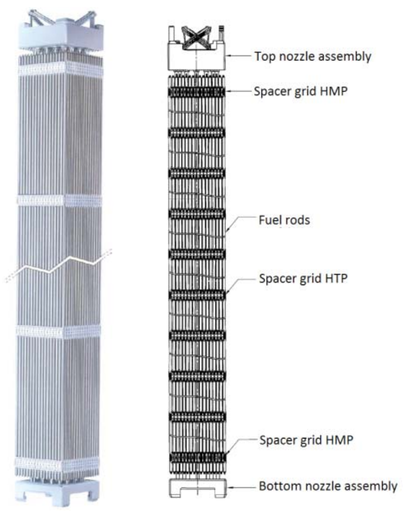

Fig. 2. Fuel assembly [4, 5].

different power levels, due to their location in the core and enrichment.

Safety analyses concerning the EPR fuel assembly are allowing to investigate safety margins during normal operation and operation with lower coolant flow rate or a hypothetical failure of pumps, leading to decreased coolant flow rate. Calculations were performed using RELAP5, a thermal-hydraulic calculation code [3].

Fuel assembly

The EPR fuel assembly is built as a square grid made of 17 × 17 rods, including 265 fuel rods and 24 guide rods for control rods or measuring apparatus. As fuel, the EPR uses uranium dioxide UO_2 enriched to 2.25–3.25 wt% of the entire rod. Some fuel rods also contain gadolinium oxide Gd_2O_3, which will 'burn out', that is, absorb neutrons to enable more stable operation of the reactor during the first fuel loading. The gadolinium content varies from 2 to 8% depending on the location of the fuel assembly in the reactor core.

Apart from fuel rods and guide rods, connected with spacer grids, the fuel assembly also contains inlet and outlet connections and springs for fixing the component in the core (Fig. 2).

The fuel rod consists of fuel pellets kept in a thin-walled cladding made of zirconium alloy. An innovative M5 alloy, which apart from zirconium also consists of niobium (1%), oxygen, and iron, was used by the reactor manufacturer as the cladding material. Since the alloy contains no tin as an alloy-forming element, it is highly corrosion proof, as M5 features lower hydrogen production resulting from zirconium oxidation. A cross section of the fuel rod is pictured in Fig. 3.

The fuel rods and guide rods are connected with each other in the form of a 17 × 17 matrix by 10 spacer grids, part of which is shown in Fig. 4. The spacer grids also have another function: as water flows between them, it is mixed better and put into whirling motion, which helps remove the heat more efficiently.

While coolant flows through the fuel assembly, its pressure decreases with subsequential increase in temperature. Pressure drop results from the resistance of the coolant flow through the fuel assembly. An average pressure drop in the EPR fuel assembly is about 0.188 MPa. As the fission energy is absorbed, the temperature of the coolant flowing through the fuel assembly increases by about 36 K. The cooling water flow rate through one fuel assembly is about 96.097 kg/s. At standard rating conditions, about 18.672 MW of power is generated, on an average, in one fuel assembly [4].

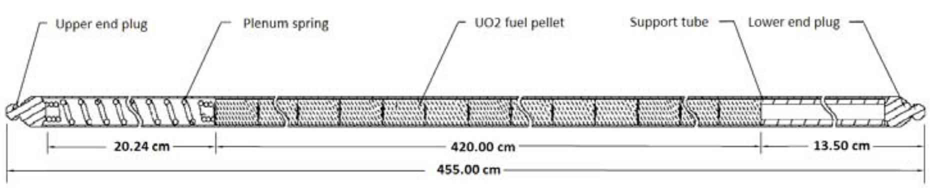

Fig. 3. Fuel rod [4].

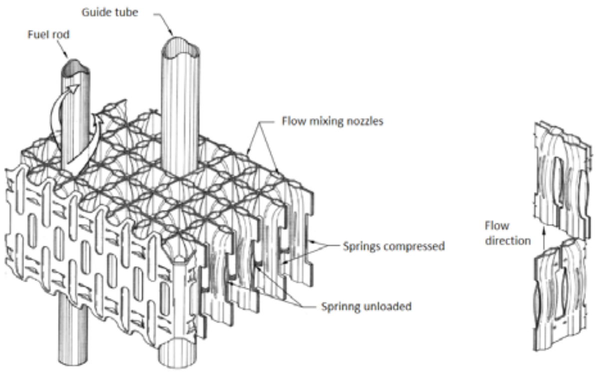

Fig. 4. Part of spacer grids [4].

RELAP5 code

RELAP5 is a code designed to perform thermal--hydraulic calculations concerning light water reactors (LWR), that is, for fluids such as water, steam and water mixture, noncondensible gases, and nonvolatile matter (boron). The code includes modules dedicated to reactors, particularly a point kinetics model, pumps (including a jet pump typical of boiling water reactors), valves, pipes, heat structures, turbine, separator, water accumulator, and logical elements for system control. The code provides applications typical of LWRs, simulating small coolant loss, anticipated transients without SCRAM, power outage, and loss of flow. The code was developed at the U.S. Idaho National Laboratory [3].

Fuel assembly nodalization

In order to input geometrical data, the fuel assembly geometry has to be discretized into control volumes; the connected control volumes constitute a calculation model. A sample division into control volumes for a pipe can be found in Fig. 5. RELAP5, being a system code, can make the impression that the number of components and control volumes is small (when compared to techniques used in computational fluid dynamics, or CFD software); nevertheless, such an approximation in calculations concerning large components is adequate, and the calculations are made in relatively short time. The time in the case of the safety analysis is crucial, because during this kind of process, large amount of calculation is needed. Nowadays, safety analyses done with the use of the system codes are beginning to be always done with the uncertainty and sensitivity study. Apart from the calculations of the accident scenarios, results are evaluated in terms of safety margins by their susceptibility to various parameter changes. That is why the RELAP5, with its low computational time, is a great tool for performance of numerous calculations. In addition, RELAP5 code

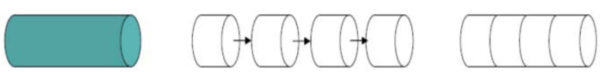

Fig. 5. Pipe discretization represented by control volumes.

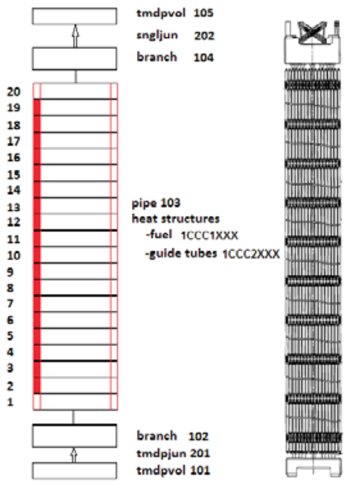

Fig. 6. Nodalization and the end view of a fuel assembly [4].

is a commercial code that was verified and validated for nuclear applications in the past 30 years.

The calculations are performed separately for each control volume, as the tool solves equations relating to mass, momentum, and energy balances for each phase of the fluid. When creating control volumes, one has to adhere to certain guidelines and rely on one's experience in modeling complex thermal-hydraulic systems. In defining the sizes of the control volumes, the geometry complexity and the rate of changes in basic parameters within the geometry need to be taken into account. Nodalization (a division into control volumes) of a fuel assembly as a whole is demonstrated in Fig. 6.

Once created, the fuel assembly nodalization consists of a lower and upper source, a time-dependent volume, where the pressure, temperature, and void fraction are declared. Other components required to model the fuel assembly is a time-dependent junction, branch, and pipe. To model the process of heat exchange and generation, one has to introduce heat structures. To make the tool differentiate between fuel rods and control rods, two heat structures, both marked in red in Fig. 6, were added to the pipe; the red-filled area on the left-hand side means that the given volume is a heating component (an active structure).

Results

Steady-state parameters

In order to obtain the results for the steady state of the fuel assembly in the EPR operating at nominal power, first, one has to check whether and when the steady state is reached in the model. To this end, a number of variables are examined: temperature within fuel at half the height of the assembly, temperature of the coolant at the outlet of some rods and guide rods, and the pressure of the coolant flowing out of the assembly. If these parameters vary by less than 1%, the steady state is assumed to have been reached. Table 1 lists steady-state parameters

Table 1. Design and calculated steady-state parameters

Item	Parameter	Simulation result	Design data	Unit
1	Pressure at the fuel assembly inlet (lower connection)	15.6875	15.688	MPa
2	Pressure drop between the lower and upper connections	186.926	188.0	kPa
3	Overall water mass in the fuel assembly	76.165	75.0	kg
4	Height where steam appeared	–	–	m
5	Rate of the coolant flow through the fuel assembly	96.097	92.26	kg/s
6	Temperature of the liquid phase of the coolant at the upper connection of the fuel assembly	601.84	604.85	K
7	Mean flux density of the heat given off by the fuel cladding	562.40	547.0	kW/m^2
8	Mean velocity of the flow along the fuel assembly	5.72	5.0	m/s
9	Temperature in the middle of the fuel at half of the height of the assembly	1172.8	n/a	K

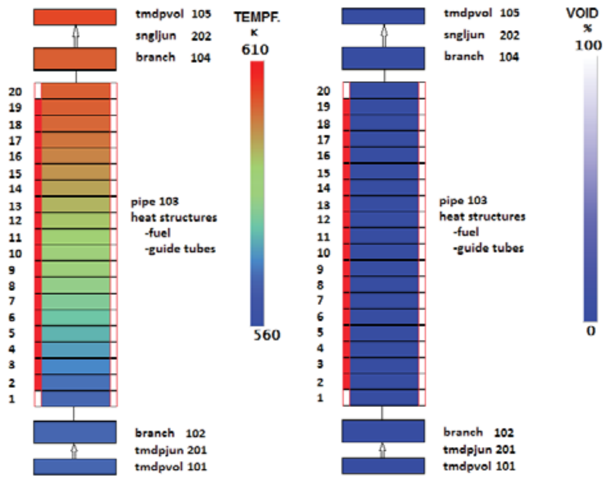

Fig. 7. Temperature and void fraction of coolant.

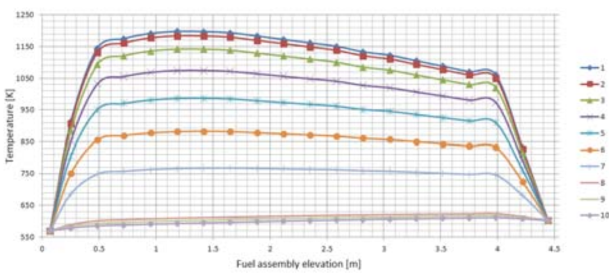

Fig. 8. Axial temperature distribution in the fuel rod.

obtained from the EPR fuel assembly model and the data provided by the manufacturer.

When the results are compared with data provided by the manufacturer, they seem highly similar. The most important thermodynamic parameters of the coolant, that is, temperature and pressure, are nearly equal to the values found in the technical specifications (Table 1). The right temperature obtained at the assembly outlet means that the power generated by the fuel rods and the heat transfer surface area are properly calculated. The correctness of the pressure drop calculations is proved by the pressure at the assembly outlet. As for the assembly, the inlet pressure was declared, while the outlet one was calculated based on the right selection of roughness of the assembly materials and on the local losses in the connections and spacer grids. A detailed analysis of the steady-state parameters along the fuel assembly

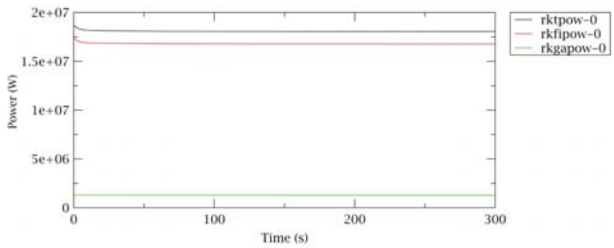

Fig. 9. Power generated in the fuel assembly.

height is shown in Figs. 7–10. Figure 7 illustrates the change in temperature of the coolant along the fuel assembly, in Kelvins, and the void fraction on the right-hand side.

Figure 7 clearly shows that the flowing coolant warms up uniformly as it passes and cools down the fuel rods. The fuel assembly power and flow rate were chosen by designers so that no two-phase flow occurs during operation at nominal power; this is evident on right-hand side of Fig. 7, where the void fraction is zero.

Figure 8 depicts a temperature distribution within the fuel structure; 1 is the middle part of the fuel, while 10 is the outer wall of the fuel cladding. Lowest temperatures (items 8–10) represent the temperature in the cladding, 7 is the helium gap in the fuel rod, and the rest are the temperature of the fuel pellet itself. Owing to the relatively low thermal conductivity of uranium dioxide (~3.5 W/mK), the fuel is characterized by a large temperature gradient of about 550 K. However, the maximum value of 1172.8 K is very different from UO_2 melting point of 2820 K [4]. The inner (item 8) and outer (item 10) temperatures of the fuel cladding are close to each other, since M5 is a good thermal conductor.

Figure 9 illustrates the power generated in the fuel assembly, broken down into characteristic components. The steady state occurs after 10 s, when the changes in the parameters in the current step are less than 1% over the previous step.

The black line marks the overall power generated in the fuel assembly; the red line indicates the power from fission reactions; and the green line is the power generated as a result of decay of fission products and actinides.

Despite the nominal power of the fuel assembly (18.677 MW), the flow is single-phased, and there is no coolant vaporization. This is a consequence

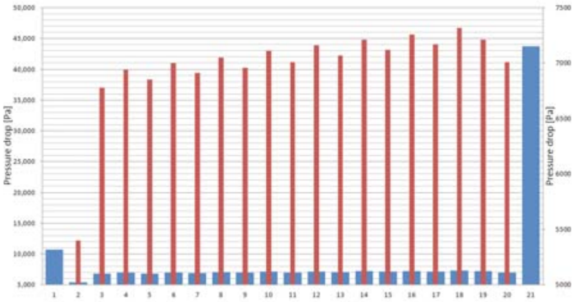

Fig. 10. Pressure drop in each volume in the fuel assembly (blue bars, left axis; red bars, right axis, only for the active fuel part).

of a very high pressure which at the fuel assembly inlet amounts to 15.688 MPa. However, the pressure decreases along the assembly by values indicated in Fig. 10. The largest pressure drop occurs in the connections (blue bars at the far right and left, 11 and 43 kPa, respectively) and results from the change in the reduction of the flow surface area. The overall pressure drop across the fuel assembly is 188 kPa. Owing to the large difference between the first and last control volumes and those in the middle, a second axis was added to facilitate reading the pressure drops in the active part of the fuel (volumes 2–20, red bars). Pressure drops in the middle part correspond with 5–7.5 kPa (right axis). Differences between neighbor bars come from location of spacer grids (volumes 4 and 6–18), which introduce additional pressure drops.

Change in the flow rate of coolant down to 60% of the nominal value

In this case, the analysis concerned a situation where the coolant flow rate is immediately decreased to 60% of the nominal value without SCRAM (reactor trip). Such a situation is very unlikely, but it should be analyzed with respect to safety. The consequences of such scenario for the fuel assembly parameters should be investigated. The general understanding of the phenomena present during this scenario can be explained by the sum of the two neutronic effects – the Doppler effect and the negative void fraction coefficient for the EPR assembly. The effects influence the safety-related parameters such as cladding temperature, which is discussed in this section. To facilitate the observation of parameter changes, the calculations were performed after a 100-second-long steady-state period; then, within 1 s, the coolant flow rate was decreased.

The drop in the coolant flow rate led to a rise in temperature at the fuel assembly outlet and the saturation temperature was reached (Fig. 11). Line numbers 01–20 were assigned according to the nodalization (01, inlet; 20, outlet of the fuel assembly).

We can see that the coolant temperature rose faster than during operation at nominal parameters and is nonuniform. The nominal outlet temperature was already reached in the ninth control volume (component number 103090000). In the five following volumes, heating occurred up to the saturation temperature; from the 16th volume on, steam appeared in the flow (Fig. 12).

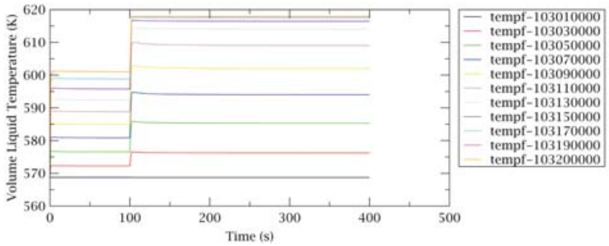

Fig. 11. Fluid temperature in the fuel assembly compared to the nominal parameters at 100% mass flow rate.

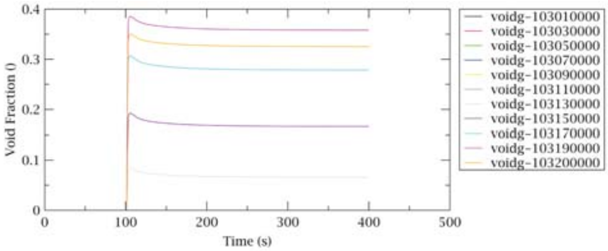

Fig. 12. Void fraction.

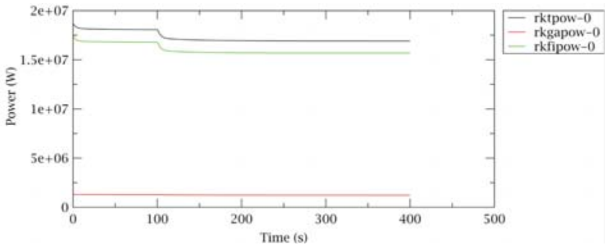

Fig. 13. Power generated in the fuel assembly.

The amount of void is so small, however, that the heat is still properly removed from the fuel rods, and damaging the fuel cladding and the fuel itself is impossible.

As the moderator temperature increased, its density drops and neutrons are slowed down less efficiently. This effect is defined by a moderator temperature coefficient, and with decreasing density, it leads to negative reactivity. The drop in the flow rate of the coolant leads to the increase in the fuel temperature; this, as a consequence of the Doppler effect, also results in negative reactivity and reduces immediately the power generated in the assembly during the fission reaction (Fig. 13).

As in Fig. 9, the black line marks the overall power generated in the fuel assembly; the red line indicates the power resulting from fission reactions; and the green line is the power generated as a result of decay of fission products and actinides.

The way in which the pressure changes in such a scenario should also be examined. Pressure drops in each control volume along the assembly are shown in Fig. 14.

It can be seen that the bars on the right-hand side are clearly longer, which is associated with the appearance of steam. It has greater velocity, which leads to increased pressure drop that can be determined from the relation [6]

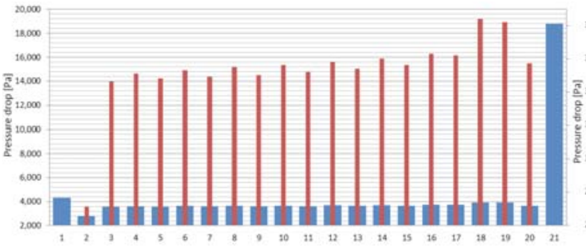

Fig. 14. Pressure drop along the fuel elements.

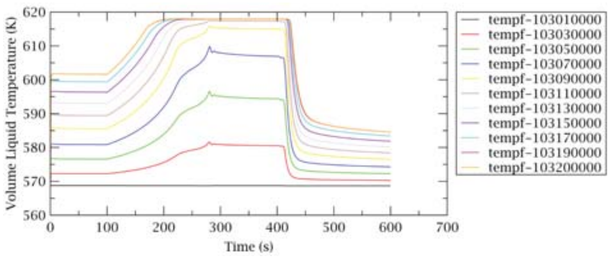

Fig. 15. Coolant fluid temperature.

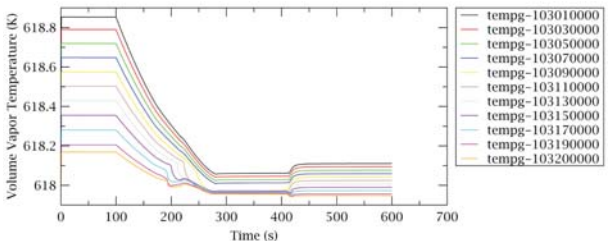

Fig. 16. Coolant gas temperature.

$$(1)\ -\Delta p = C_{f,lo}\frac{2}{D_h}\frac{G^2}{\rho_l}\int_0^L \varphi_{lo}^2 dz + g\sin\varphi\int_0^L[\alpha\rho_v + (1-\alpha)\rho_l] + G^2\int_0^L \frac{d}{dz}\left[\frac{x^2}{\alpha\rho_v} + \frac{(1-x)^2}{(1-\alpha)\rho_l}\right]dz + \left(\sum_{i=1}^{N}\varphi_{lo,d,is_i}^2\right)\frac{G^2}{2\rho_l}$$

The first, second, and third terms in the relation (1) are the pressure drops resulting from friction, gravity, and steam acceleration, respectively (a non-homogeneous model); the last one represents local losses. The symbol φ_{lo}^2 is the loss factor regarding the two-phase flow and is greater than one.

The transient state: the drop in the coolant flow rate and the SCRAM

The transient state is described by the following scenario. The calculations start at the steady state at nominal conditions of the reactor operation (100 s). Then, the rate of flow of the coolant through the fuel assembly starts to decrease linearly by 0.5 kg/s for a period of 180 s. After this period, the flow rate stabilizes at 6.08 kg/s, which is about 6.26% of the nominal flow rate. From the beginning, the fuel assembly is heated by the nominal power but changes in moderator density and fuel temperature lead to inherent power change. This results from reactivity effects related to the aforementioned safety factors (particularly the higher temperature and lower density of the moderator, meaning poorer moderation and larger proportion of fast neutrons that do not contribute to fission reactions, and therefore, to the power drop). Then, within 400 s, the SCRAM (reactor emergency shutdown) signal is given, forcing the fission reaction to stop by inserting safety rods (the negative reactivity), and the reactor power decreases. However, the flow rate of the decay heat from the decay of fission products and actinides remains (Fig. 1). Such a state in the nuclear power plant could result from the decrease in the rotational speed of a primary-circuit pump, followed by the insertion of the control rods initiated by the SCRAM signal. In the EPR, such a drop in the flow rate would be immediately signaled to an operator and control and automation system would use stand-by pumps to supply the missing coolant, or the SCRAM signal would be initiated earlier. Despite many safety measures provided in the nuclear reactor, running such a scenario is required to examine how the core would react if enough coolant is not supplied. Characteristic parameters concerning the fuel assembly during a transient are listed in Table 2.

During a transient, physical quantities change in time. To help analyze the changes in detail, they were presented in diagrams for certain characteristic time points and periods. For us, the characteristic time points and periods are the following: 171.5 s, steam appears; 171.5–280 s, steam continues to appear and the power drops; 280–400 s, cooling with steam and water mixture; 400–450 s, restoring water cooling. Figure 15 shows the temperature of the liquid phase. As the flow rate drops, the temperature rises until it reaches the value (about 618 K) matching the saturation pressure in the channel. This value is reached first at the end of the heating part. Then, the value is reached in the preceding volumes. During calculations, a temperature that is lower than the value matching the saturation pressure remains in the ninth volume. The gaseous phase temperature (Fig. 16) decreases as the pressure drops. The amount of the heat supplied does not lead to superheating the steam and drying the fuel cladding, which would result in a sudden rise in the temperature of gas and, eventually, of the fuel cladding.

From the 171st second, the coolant flow becomes two-phased at the end of the fuel active part. Over time, the two-phase flow propagates toward the first part of the assembly. The changes in the void fraction are clearly depicted in Fig. 17. The lower the flow rate (100–280 s), the larger is the void fraction in

Table 2. Characteristic parameters during a transient

Item	Parameter	Result	Time [s]
1	Peak fuel temperature	1167.4 K	223.0
2	Peak fuel cladding temperature	626.52 K	227.0
3	Minimum coolant density	250.1 kg/m^3	281.0
4	Coolant mass flow rate when steam appears at the end of the active part	60.68 kg/s	171.5

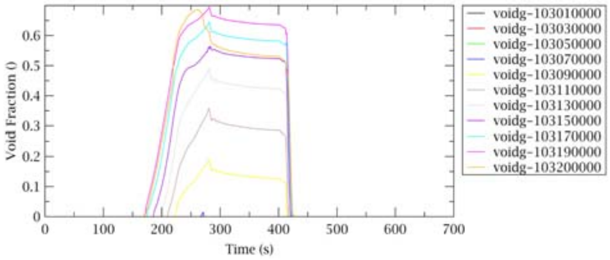

Fig. 17. Void fraction.

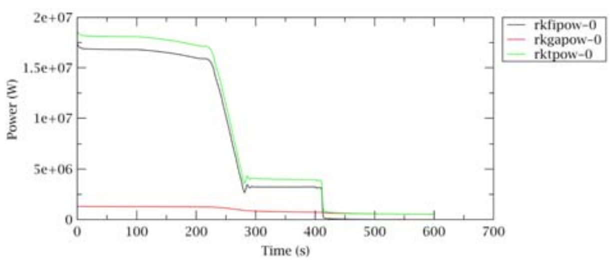

Fig. 21. Power generated in the fuel assembly.

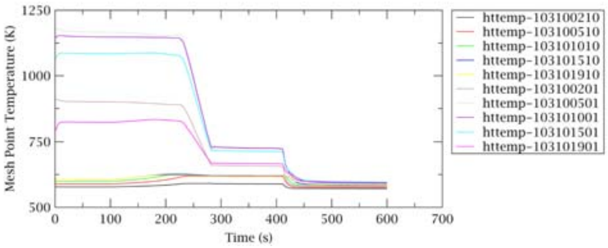

Fig. 18. Fuel and cladding temperature.

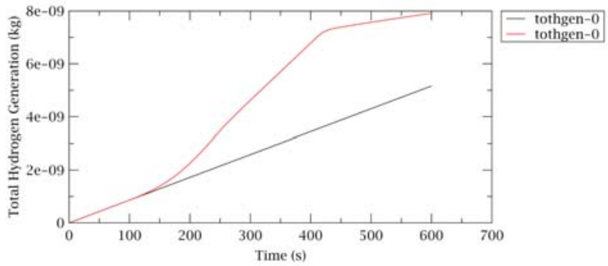

Fig. 22. Hydrogen production.

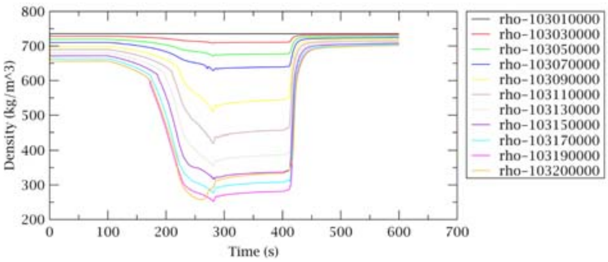

Fig. 19. Moderator density.

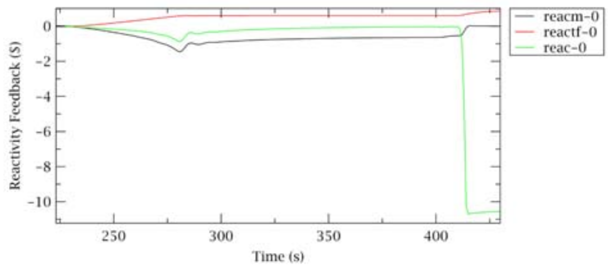

Fig. 20. Reactivity feedbacks.

the control volumes. Initiating the SCRAM (400 s) stops the fission reaction, and the power in the assembly is lowered more than 10 times, which results in the appearance of water along the whole height of the assembly.

The melting points of the materials of which the fuel assembly is made are the following: the fuel pellet (UO_2), 2820 K; the fuel cladding (the M5 alloy), 1450 K; the connections (stainless steel), 1454 K. To find out whether any of these points were exceeded in the material, we should analyze the temperature diagram concerning the fuel element (Fig. 18). According to the legend in the diagram, the parameters ending with one represent temperatures in the middle of the fuel in the 2nd, 5th, 10th, 15th, and 19th control volumes. Temperatures of the fuel cladding were assigned to parameters ending with zero. As can be seen, the temperature of the fuel decreased, while that of the fuel cladding increased by more than 10 K. This happens due to the feedback of the moderator temperature coefficient, which with decreasing moderator density (Fig. 19) makes reactivity to attain negative values (the black line in Fig. 20). The increase in temperature of the cladding material did not exceed its melting point at 2100 K, but the increased rate of cladding oxidation was reached. This effect is presented in Fig. 22.

Owing to the poorer moderation, the reactor power (Fig. 21) and the fuel temperature dropped. The fuel temperature drop leads to positive reactivity (the red line), but after summing up the temperature coefficients of the moderator and fuel, the reactivity is negative (0–400 s, the green line). Then, by inserting the safety rods into the assembly, the initiated SCRAM signal provides the negative reactivity (410 s, the green line in Fig. 20).

It should be noted that in the calculations, the RELAP5 code takes no account of the heat generated during the fuel cladding oxidation, which is why the actual value is greater. The oxidation of the zirconium fuel cladding is accompanied by the production of hydrogen, which in certain concentrations is a highly explosive substance. The fuel cladding oxidation occurs according to the following relation [7]:

$$Zr + H_2O = 2H_2 + ZrO_2 + 6500 \text{ [kJ/kg]} \tag{2}$$

By including a specific module in the RELAP5 code, we can calculate the amount of the hydrogen produced. The integrated hydrogen production for a single assembly is shown in Fig. 22. The black line represents the hydrogen production during operation at nominal parameters, while the red line shows the production in the scenario under consideration. It is evident that only about 5×10^{-9} kg H_2 was produced, and the relating amount of heat (0.33 J) can be omitted. If the fuel claddings were exposed, the oxidation would be much more intense and the amount of the heat generated should be included in the balance, as this would result in the rise of the fuel cladding temperature.

Summary

A fuel assembly model for the EPR was created using the RELAP5 code. In the model, the fuel assembly was nodalized (divided into control volumes). Based on technical documentation, the fuel assembly geometry, local pressure drops, and material qualities were assumed.

Two steady-state simulations were performed: the first one at nominal parameters, in order to verify the correctness of the steady-state model, and the second one at the coolant flow rate decreased to 60%. The data obtained from the two steady-state simulations were close to those provided by the manufacturer in the technical documentation.

An analysis was performed for one transient. However, taking into account the number of safety systems installed in the EPR, the transient under consideration can be described as hypothetical, for it was assumed that no safety systems were in place to ensure the coolant flow and that the SCRAM signal was delayed. Nevertheless, such an analysis proves the importance of these systems, and the safety of nuclear reactors, which, in the event of the drop in the coolant flow rate inherently decreases their power. Safety margins, which are broad during normal operation, were also within appropriate limits during the transient in question.

The results are very satisfactory and form an excellent basis for carrying further studies on the fuel assembly behavior during the reactor operation. Transients relating to the loss-of-coolant accidents (LOCA) and the loss-of-offsite power (LOOP) can be particularly interesting.

The system RELAP5 code allows to investigate thermal-hydraulic phenomena in detail, but for analyzing severe accidents, codes dedicated to such phenomena should be used. The RELAP/SCADAP extension would enable the tool also to calculate the phenomena associated with processes such as fuel cladding oxidation or hydrogen production. More professional severe accident codes, such as ASTEC (Accident Source Term Evaluation Code) and MELCOR (Methods for Estimation of Leakages and Consequences of Releases), also make it possible to consider the core movement, as well as the behavior of aerosols and fission products in the containment, and to evaluate the amount of their release.

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