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Experimental studies into the thermal-hydraulic performance of the VK-300 reactor based on a draft tube model

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Abstract

The paper presents an experimental study into the thermal-hydraulic performance of the VK-300 reactor based on a model of a single draft tube at a pressure of 3.4 MPa, various flow rates and the model inlet relative enthalpies of –0.05 to 0.2. The experimental procedures include generation of a steam-water mixture circulation with a preset flow rate and a relative enthalpy through the test section at a pressure of 3.3 to 3.4 MPa, and measurement of thermal-hydraulic parameters within the circuit's representative upflow and downflow lengths of practical interest. There have been confirmed the designs used to support the reactor facility serviceability and the assumptions concerning the thermal-hydraulic performance of a natural circulation circuit used in the analysis thereof. It has been shown that, across the analyzed range of the relative enthalpy values, the draft tube has an annular-dispersed or an annular flow of the steam-water mixture, both providing for the significant separation of the steam-water mixture ($K_{sep} = 0.4$) at the draft tube edges and in the mixing chamber. The perforation in the upper part of the draft tubes allows the separation coefficient to be increased at the first stage and creates more favorable conditions for the second-stage separation. The measured values of the void fraction in the mixing chamber and in the draft tube are in a satisfactory agreement with calculations based on Z.L. Miropolskiy's method and the RELAP code and may be used to verify the VK-300 thermal-hydraulic codes. It has been shown that steam may enter the ring slit that simulates the annular space and reach the reactor core inlet. Further investigations need to be conducted to study this effect for its guaranteed exclusion and for the development of emergency response procedures.

Keywords: Nuclear reactor; Natural circulation; Separation of steam-water mixture; Steam-water flow pattern; Void fraction.

Problem definition

VK-300 is a more powerful version of the VK-50 reactor attractive as a heat and electricity source for the construction of district nuclear cogeneration plants. This is an integral vessel-type boiling-water single-circuit reactor with in-vessel separation of steam. The reactor circuit comprises an active region, draft tubes and separating devices. In the upflow direction, the circuit consists of an active region and a riser section. Water evaporates in part within the active region and passes through the outlet grid to enter the mixing chamber which is designed to level off the void fraction coming in from the fuel assemblies having different heat release. After flowing through the riser section, the mixture enters the separation chamber which forms the first stage of phase separation. A part of the flow goes up to enter the separating devices where most of the mixture separation into phases takes place, namely into saturated steam with a humidity of up to 0.1% (by mass) and water with a small steam content. The rest of the flow (water from the separation chamber) enters the natural circulation circuit's downcomer where it mixes with the water separated in the separating devices and with feedwater.

The reactor has been designed with the first-stage separation coefficient of $K = 0.3$. The reason is that the K value at the second separation stage shall not exceed 0.6 to 0.7. Eddy-current centrifugal separators have the capacity sufficient to provide for a still greater separation coefficient but this requires respective pressure losses the dynamic pressure of natural circulation cannot make up for. However, $K = 0.3$ across the tube and in the separation chamber may be achieved only with an annular-dispersed or an annular flow of the two-phase mixture, and both flows are achievable

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with a relative enthalpy of more than 0.19. One objective of this study is to find out if it is possible to achieve the steam-water mixture separation with such K values in the given geometrical and thermal-hydraulic conditions.

Following the first-stage separation, water can entrap steam and carry it up to the reactor core thus affecting its thermal-hydraulic and neutronic performance. To demonstrate the possibility of this is the second objective of the study.

Reactor model. Experimental setup. Investigation paramaters

A single draft tube was used as the reactor model to study the thermal-hydraulic parameters [1]. The test section dimensions were selected with regard for the equivalent flow sections of the draft tubes, the mixing chamber and the separation chamber per one draft tube. In the initial approximation, the volume and power scaling factor, equaling the ratio of the flow section areas of the above full-scale and simulated circuit components, was equal to zero. This model was selected based on the following considerations. The reactor's riser section is a system of draft tubes with identical thermal-hydraulic processes taking place therein. Therefore, the values of thermal-hydraulic parameters obtained for one tube may be extended to the rest of the tubes, while the thermal-hydraulic upset for different draft tubes may be allowed for by measurements at different relative enthalpy and flow rate values.

The heights of the full-scale and simulated components from the active region's outlet grid to the axial separator inlets (to the submerged plate) were also identical.

The power required to achieve the nominal flow rate of the steam-water mixture per one draft tube ($G = 80$ t/h) at a relative enthalpy value of up to $X = 0.2$ is about 30 MW. No electrically heated test facilities of such a power were available to the authors, so a decision was made to use a cogeneration plant as the energy source, and to prepare the steam-water mixture with the required composition and flow rate in a large-volume mixer from the boiler feedwater and separated steam from drum separators.

Based on design considerations, a tube of Ø219×10 mm was used for making the test section body, which, with the draft tube model diameter being Ø136×3 mm, made it possible to provide for the volume and power scaling factor of $K_{mod} = 2.37$ for all circulation circuit components. The draft tube model diameter was reduced from 200 to 130 mm, and the width of the ring slit that simulated the flow section length for one nuclear plant draft tube was about 31 mm. It is assumed that the reduction of the draft tube diameter to 130 mm will not affect the flow structure. The thermal-hydraulic correspondence between the prototype and the model (ΔP, φ and others) requires the condition $K_{mod} = G_{prot}/G_{mod} = 2.37$ to be observed for all components of the natural circulation circuit.

The test section design is shown in Fig. 1. The basis of the test section is the body (1) made of a Ø219×10 mm tube which accommodates all structural elements. From the mixer, the coolant (water or a steam-water mixture) flows through the outlet grid (2) to enter the mixing chamber (4) limited by a solid plate (3) from above. The draft tube (6) rests against the outlet grid. The outlet grid has three holes, so a part of the coolant flows directly into the draft tube, while the rest of it enters the annular space of the mixing chamber (4) and flows further into the draft tube through windows (5) in its side surface. This provides for the complete simulation of the prototype outlet assembly design. The length of the draft tube is 2.054 m. After flowing through the draft tube (6), the steam-water mixture enters the separation chamber (11) and is separated in part, while the separated water, as the heavier phase, flows down into the annular space (the ring slit) (7) and leaves the test section through a nozzle (10). The rest of the steam-water mixture is fed into the other axial separator (9), and the steam enters the steam space through a splitter to leave the test section through a nozzle (13). The separated water enters the space between the separators (14) and leaves the tube section through a nozzle (12). A water level of 200 to 300 mm (the height of the axial separator is ≈600 mm) was maintained on the submerged plate (8) during the experiments. The flowchart's components 15, 16 and 17 are pressure takeoff chambers for measuring the pressure drop (loss) within respective circuit lengths.

Therefore, the test section components in a length from the outlet grid of the active region (2) to the submerged plate (8) are adequate or, structurally, are very close to prototype components and have the same height. To this extent, the test section is a fully benchmark single-tube model (as far as the riser flow is concerned) of the VK-300 reactor. At the given stage of the study, the inlet part of the test section (from the submerged plate (8) to the nozzle (13)) is not fully adequate to the prototype since this does not affect the nature of thermal-hydraulic processes.

A simplified hydraulic diagram of the experimental bench (Fig. 2) consists of a water circuit and steam lines.

The water from the discharge end of the NTs circulation pump is fed to the SM mixer through the G_w metering orifice and the VR_1 control valve. It is also there that steam with a pressure of $P \approx 3.2$-3.4 MPa and a temperature of 400 to 430°C is fed through the VZ_2 shutoff valve (installed at the fixed steam branch), the G_s metering orifice and the VR_4 control valve. The steam-water mixture formed in the mixer flows into the MO test section. The water separated in the draft tube and in the mixing chamber is fed at the saturation temperature to the TO cooler via a nozzle and to the US mixing unit through the G_{sep} metering orifice and the VR_2 control valve and further to the circulation pump inlet. The water separated in the axial separators is fed to the US mixing unit through a nozzle and the BU level metering tank, mixed with the water separated at the first stage and supplied to the circulation pump inlet. The separated steam is fed either for the release into the atmosphere or to cover the cogeneration plant's auxiliary needs through a nozzle, the VR_3 control valve and the VZ_3 shutoff valve. It should be noted that the water line (ring slit – outlet water nozzle – TO cooler and further up to the mixing unit) is a model of the reactor's downflow circuit up to the first separation stage. The second water line (water nozzle – BU and further up to