CHARACTERISTIC THERMAL-HYDRAULIC PROBLEMS IN NHRs: OVERVIEW OF EXPERIMENTAL INVESTIGATIONS AND COMPUTER CODES

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Abstract

The paper briefly reviews the specific thermal-hydraulic problems for AST-type NHRs, the experimental investigations that have been carried out in the RF, and the design procedures and computer codes used for AST-500 thermohydraulic characteristics and safety validation.

1. INTRODUCTION

In the course of the AST-500 design development a large scope of experimental investigations on reactor plant thermal-hydraulics has been carried out to select and verify the design decisions and the parameters, and to validate safety [1,2,3]. A complex of test facilities has been created, including electrically heated models of the reactor with the guard vessel, and simulators of various scale for in-depth study of reactor thermal-hydraulics, both under normal and emergency conditions. Table 1 gives the characteristics of the main experimental facilities.

The peculiarities of thermohydraulics in NHRs under normal operation and emergency conditions are associated with natural coolant circulation, reduced parameters of the plant and with specific features of the integral reactor, such as availability of built-in steam-gas pressurizer (SGP), in-reactor heat exchangers for the emergency residual heat removal, and a guard vessel (GV). Proceeding from these factors the main attention in performing the experimental work was paid to studying the following effects and phenomena:

- natural circulation flow stability;
- ultimate values for core heat loads;
- thermohydraulics of the built-in steam-gas pressurizer;
- behavior of non-condensable gases (gas transfer, gas distribution, dissolved gas in water);
- natural circulation modes under loss of coolant conditions;
- operation of the emergency heat removal HXs at steam condensation from a steam-gas mixture;
- reactor-guard vessel system behavior at LOCAs;
- passive safety systems initiation and functioning, and others.

Together with the tasks of studying the thermohydraulic processes running in the reactor plant under normal operation and emergency conditions, and validation of the operability and efficiency of the safety systems provided, obtaining the representative data for the verification of computer codes and calculation methods represents the most important objective for the experiments.

Table 1: Experimental facilities used for AST-500 thermal-hydraulics investigation

Test rig	Circul-ation	Max. power, MW	Max. pressure, MPa	Volume, m^3	Scaling	
					Volume	Height
L-186 Studying of CHF	FC	2	18.0	19 rods 13.6 mm		1:1
37-rod bundle Partial core uncovery	NC	0.1	5.0	37 rods, 10mm, $T^{max}_{clad.} = 700°C$		1:1
L-800 Large-scale models of steam-gas pressurizer	FC	0.7	18	2	1:14	1:2
KMR-2 AST-500 reactor large scale model	NC	2.5	7.0	1.3	1:170	1:1
1385-MONOBLOCK Integral reactor model	NC	1.8	18	0.3	1:700	1:2

Abbreviations: FC - forced circulation, NC - natural convection.

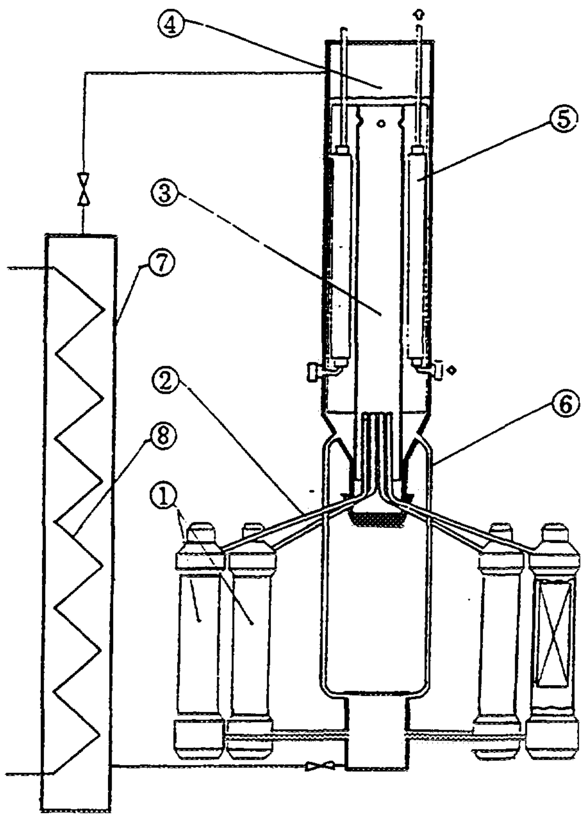
The paper reviews the experimental investigations that have been performed for the AST-500 reactor plant thermal-hydraulic design as concerns the aforementioned problems. The characteristics of the computer codes for the analysis of AST thermal-hydraulics which have been validated experimentally are presented as well. The investigation results on the AST static thermal-hydraulic characteristics are presented in [3].

2. NATURAL CIRCULATION FLOW STABILITY INVESTIGATION

Investigations of AST-500 primary circuit thermohydraulic characteristics, including the problem of natural circulation flow stability, were performed on the thermophysical test facilities 1385 and KMR-2 [9,10]. These facilities represent electrically heated models of the AST-500 integral reactor, different in scale, with coolant natural circulation and built-in steam-gas pressurizer, containing thus all main components of the AST circulation circuit.

The 1385 facility (Fig.1) includes a four-assembly heating zone. The modelling scale in respect to natural circulation circuit height is 1:2. Maximum power is 1.8 MW.

The KMR-2 facility (Fig.2) is a large-scale model of the AST-500 circuit, the height of which is close to the real one. The scaling factor for the volume and the flow areas in the circuit components was 1:170. Maximum power is 2.5 MW. The core simulator consist of two 37-rod heater assemblies with practically full-scale fuel rod simulators.



1.- Heating columns
2.- Chimney tubes
3.- Riser
4.- Steam-gas pressurizer
5.- Heat exchanger
6.- Downcomer
7.- Guard vessel simulator
8.- GV heat exchanger

Fig.1. 1385 test facility. Model of integral reactor

The influence of circuit pressure, coolant subcooling and steam content at the fuel assemblies outlet, coolant subcooling at the fuel assemblies inlet, inlet hydraulic resistance, fuel assemblies power non-identity upon parallel-channel and whole-circuit flowrate oscillations were studied. Flow instability zone boundaries versus coolant subcooling value at the outlet, or outlet steam quality, and coolant inlet temperature were obtained. Fig.3 shows the dependence of the whole-circuit flowrate oscillations amplitude versus relative enthalpy at the fuel assembly outlet.

Using the experimental data obtained, the development of computer codes for flow instability analysis based on linear and nonlinear models has been carried out.

Resulting from the calculations and experiments performed, the natural circulation stability boundaries for the AST-500 reactor were determined. Its operation mode parameters were selected assuring whole-circuit coolant circulation stability with a necessary margin. With such an approach the possible parallel-channel flow oscillations in individual fuel assemblies are limited in amplitude and do not reduce the core heat-engineering reliability and equipment operability.

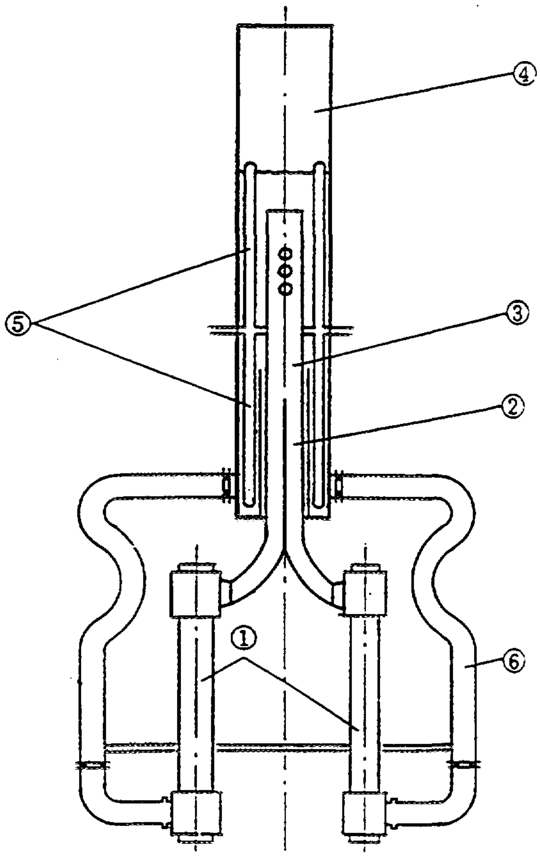


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Fig.2. Natural circulation circuit of KMR-2 test facility

An additional potential for enhancement of parallel-channel flow stability is provided by the design margins available in the thermal-hydraulic characteristics of the primary circuit which allow to increase the coolant subcooling to saturation temperature at fuel assembly outlet.

3. BOILING CRISIS INVESTIGATION

To define the ultimate admissible heat loads for the AST-500, experimental investigations of boiling crisis were performed on core fuel assembly models under conditions corresponding to the AST parameter range [7]. The six experimental assemblies used for the investigation represented 7 and 19-rod bundles of electrically heated fuel rod simulators having the actual diameter, length and pitch.

The main part of the experimental data on the boiling crisis were obtained for the following parameter range:

pressure	1.0-6.0 MPa
mass flow velocity	0-1400 kg/(m s)
inlet temperature	130-220°C

The influence of radial and axial power distribution non-uniformity in a rod bundle on the ultimate heat loads for real peaking factors of $Kr = 1.22$ and $Kz=1.8$ was studied as well.

On the basis of the obtained data, a correlation for critical power calculations was developed specifically for the AST reactor operating conditions, describing the whole data base with a root mean square error less than 6%. The AST-500 reactor core heat-engineering reliability analysis performed with this correlation showed that the minimum critical power ratio for fuel assemblies at nominal parameters amounts to 2.5.

4. TWO-PHASE FLOW CHARACTERISTICS INVESTIGATION

A knowledge of the two-phase flow parameters in the riser section of the primary coolant flow path is necessary for natural circulation flowrate determination and for hydrodynamic and neutron-physical stability analyses etc.

Non-equilibrium and equilibrium two-phase flow characteristics were investigated using a 7-rod bundle model of the AST-500 fuel assembly, equipped with a simulator of the draught section (chimney) in form of a circular tube under conditions corresponding to the range of AST-500 operational parameters: P=1.5-2.5 MPa, w=320-790 kg/(m s), q=0.3-1 MW/m^2. Void fraction variation along the height of the draught section was determined in experiments using acoustic transducers.

Resulting from the investigations performed, the data on steam condensation lengths in non-equilibrium two-phase flow were obtained, and the influence of flow parameters on condensation length was studied. Dependences of void fraction in the draught section on the value of balance mass steam quality (x) at a fuel assembly model outlet were obtained for x ranging from - 0.01 to 0.05.

The influence of the scale factor on two-phase flow pattern in the riser was also studied. The void fraction profile spectra for steam bubbles velocities and sizes in the model of the reactor draught section of 450 mm diameter and 2 m height in a parameter range: P=0.6-1.1 MPa, w=200 kg(m s), x=-0.3-0.65% described in [8] were obtained experimentally. The statistically average size of the steam bubbles in the experiments was 6 mm. The experiments showed an appreciable non-uniformity of steam bubble distribution over the cross section and a complex pattern of flow. The average values of void fraction in a large diameter draught section were 10-20% lower than corresponding values for small-diameter tubes. This is explained by the non-onedimensional structure of two-phase flow.

5. INVESTIGATION OF STEAM-GAS PRESSURIZER CHARACTERISTICS

The presence of non-condensable gases in integral reactors with steam-gas pressurizer influences both the thermal-hydraulics of the primary circuit and the pressurizer characteristics under steady-state conditions and in transients, as well as in accident sequences. Reactor parameter variation within the operational range of power, the dynamics of the pressurizer, and the efficiency of heat removal from the reactor through in-reactor heat exchangers during LOCAs all depend on the gas quantity in the pressurizer.

During the development of the AST-500 design, calculational-theoretical analysis of this problem and related experimental investigations were performed. In the experiments the build-in steam-gas pressurizer characteristics were studied, as well as processes of gas transfer and gas distribution in the primary circuit. The experiments were carried out on a large-scale model of a pressurizer - the L-800 test rig (Fig.4) and on the reactor models 1385 and KMR-2 rigs (Fig.1,2). Fig.5 shows the obtained loading characteristic of the 1385 test rig pressurizer that represents a dependence of the pressure rise in the pressurizer on gas and gas concentrations in the circuit water versus the volume-average partial gas pressure in the pressurizer.

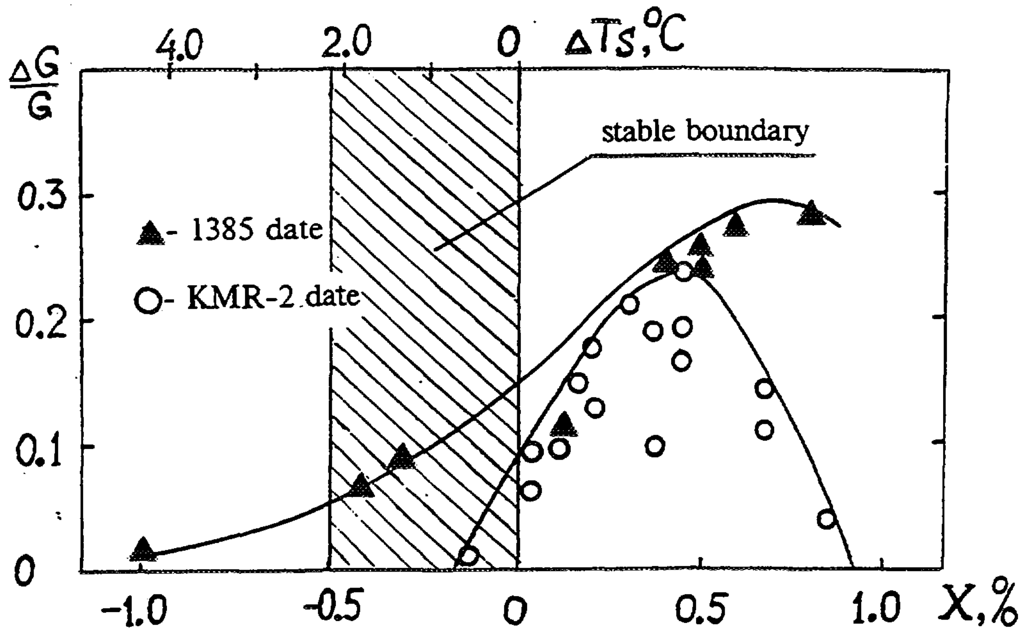


Fig.3. Effect of outlet relative enthalpy on ampl-itude of mass flow oscillation of natural circulation

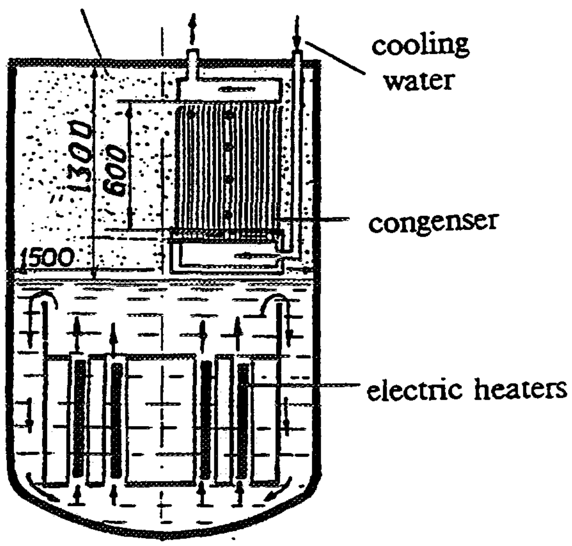


Fig.4. L-800 rig. Large scale model of steam-gas pressurizer with condenser located in steam-gas volume

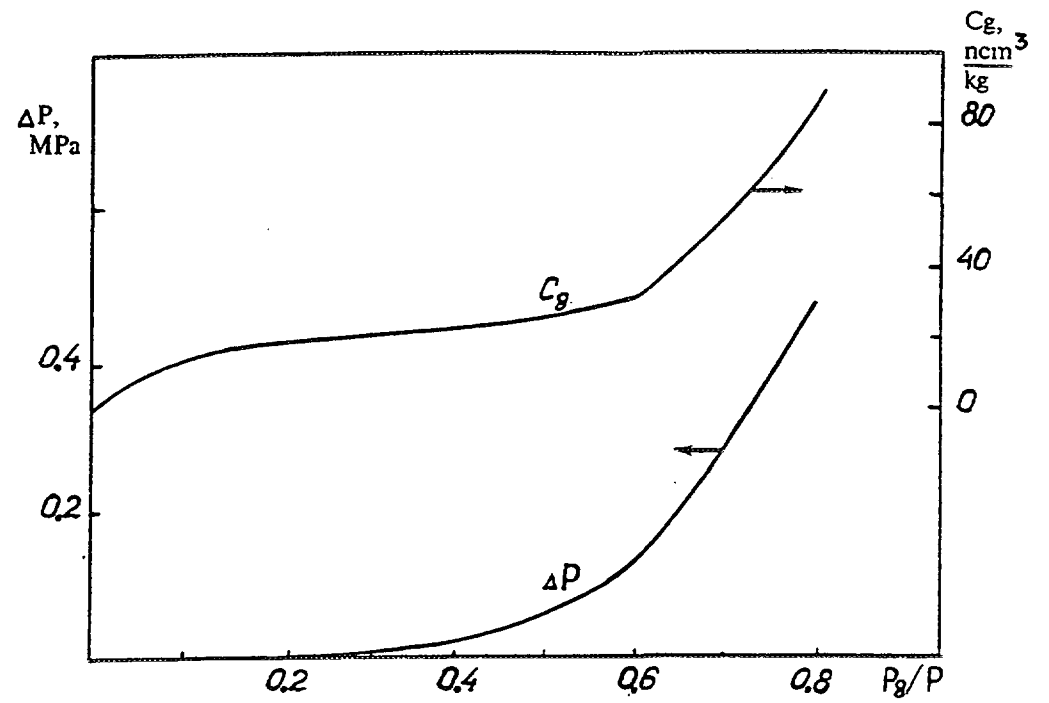


Fig.5 Change of pressurizer pressure and gas concentration in water depending on gas content in steam-gas volume

The procedure for the calculational analysis of the static characteristics of the steam-gas pressurizer and the gas distribution in the primary circuit of an integral reactor were developed and the computer code GARRIC was created. Calculation models of the code were verified based on the results of the experimental investigations.

6. RESIDUAL HEAT REMOVAL EFFICIENCY UNDER EMERGENCY CONDITIONS

Experimental investigations of the intensity of steam condensation from a steam-gas mixture were performed with tube bundle models for LOCA conditions accompanied by emergency residual heat removal heat-exchanger dry-out. The experiments were carried out on the L-800 test rig (Fig.4) using multi-row models of the AST-500 heat-exchanger tube bundle [6]. Investigations of gas influence on the intensity of steam condensation on the tube bundle of the 1385 test rig heat exchanger were performed in a wider range of parameters. Data obtained with the rig L-800 on the influence of gas content in the steam-gas plenum of the rig upon thermal power of the condenser are presented in Fig.6.

The procedure for calculational analysis of heat transfer intensity at steam condensation on a dried-out tube bundle of in-reactor ERHR heat exchangers was developed based on experimental results. The procedure is included in the UROVEN/MB-3 computer code which is used for the analysis of loss-of-coolant accidents in AST reactor plants.

7. RESIDUAL HEAT REMOVAL

The emergency residual heat removal channel with a steam condenser located on the reactor represents a simple and passive secondary circuit independent residual heat removal

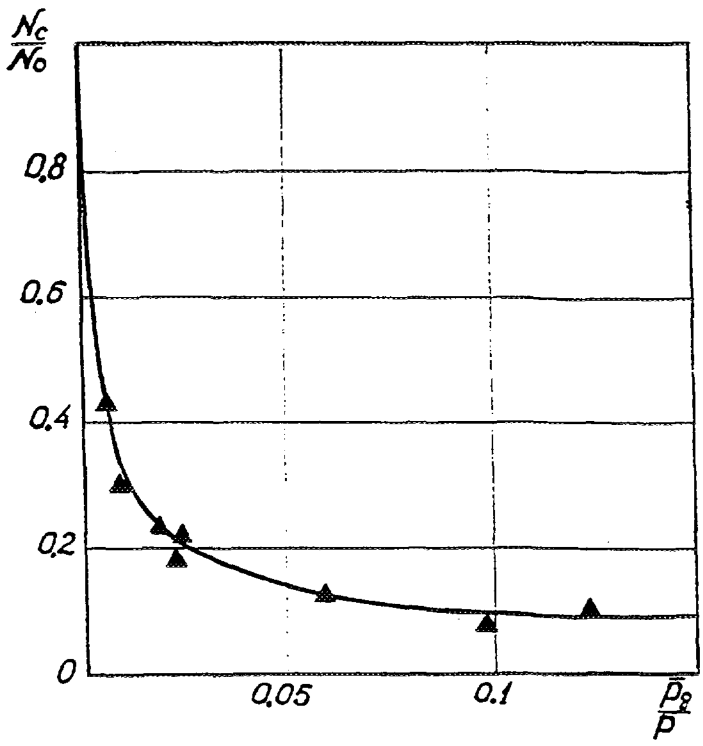


Fig.6. Effect of gas content in pressurizer on relative condenser thermal power

path for utilization in advanced NHRs. The condenser interior communicates with a pressurizer. At normal operation it is filled with a steam-gas mixture having a high content of gas which blocks heat removal from the reactor. The ERHR channel is actuated by gas blowdown from its tubing, followed by the condenser transition to a steam-condensing mode of operation with returning the condensate to the primary circuit.

The channel operation efficiency within the real range of working parameters and thermal loads is corroborated by the results of the experimental investigations carried out on a large-scale model of the condenser in the L-800 test rig [5]. In the course of the experiments, the modes of condenser actuation were investigated, and the influence of pressure, gas content and cooling water parameters on the heat removal intensity was studied. Optimal characteristics of the condenser were determined resulting from the provision of maximum heat removal capacity and the elimination of condensate entrainment into the dump line.

8. EXPERIMENTS WITH FUEL ASSEMBLY MODEL UNCOVERY

Within the framework of experimental validation of the safety required for the AST and other integral PWR-type reactors, investigations were performed on the dynamics of the thermohydraulic characteristics of 37-rod and 7-rod models of fuel assemblies under conditions with partial uncovery of heater rods in cases of small-break LOCAs. Experiments on coolant boil-off from the fuel assembly, on rod bundle dry-out and subsequent slow reflooding (velocity of 0.002-0.01 m/s), as well as experiments on the cooling of completely uncovered rod bundles by steam-gas mixture natural convection were carried out. The studied

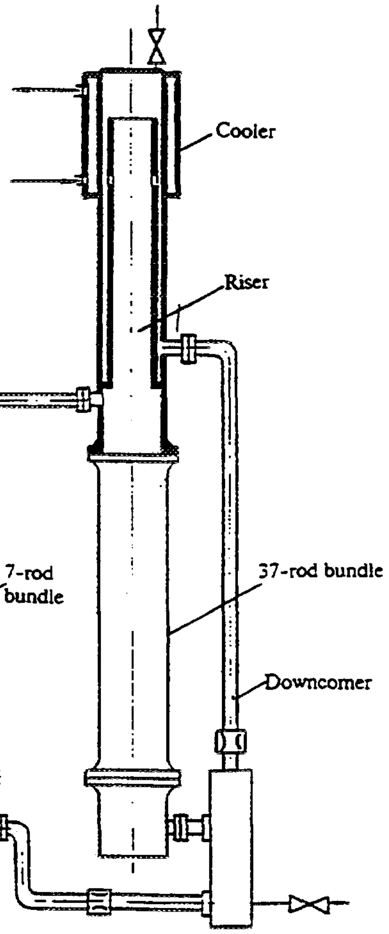


Fig.7. Primary circuit model with two test bundles for investigation of thermohydraulics during core uncovery

range of pressure variables was 0.85-5.0 MPa, that of power 15-40 kW, and that of heat loads non-uniformity over the assemblies 1.0-1.5. The maximum temperature of the fuel rod simulators was 750°C.

The test facility (Fig.7) represents a natural circulation loop consisting of two fuel assemblies (FA) models: a 37 rod bundle (d=10 mm, l=14 mm, H=3.5m) and a 7-rod one (d=14mm, l=18.7 mm, H=3 m) connected in parallel, and of a riser and a downcomer section. There is a cooler in the upper part of the model above the downcomer. The FA models include bundles of rods made of steel tubes which are electrically heated uniformily along the height, and spaced in a triangular lattice. The temperature of the heater rods, the temperatures both of the steam and the vessel metal, and the pressure difference along the height of the sections were measured.

Partial Uncovery of Rod Bundles

In the course of the experiments data on void fraction in rod bundles and on the quench front dynamics were obtained. The cooling of an uncovered part of the bundle by single-phase flow of superheated steam was also studied. The experimental data showed good agreement of a quench front coordinate with a two-phase swell level in the rod bundle under small-break LOCA conditions. The difference between rod temperature rise coordinate and the level position was 0.1-0.2 m which is within the limits of the level determination error in the experiments. This difference may be caused by heater rod cooling near the interphase surface due to entrained droplets. Correlations for the calculation of the intensity of heat transfer to superheated steam in a fuel rod bundle for the laminar and transition range of Re numbers were recommended based on the analysis of experimental data.

The effects of interaction between FA regarding parallel channel and loop oscillations at fuel assembly dry out and reflooding, including gravitation reflooding conditions have been investigated. The influence of heat load non-uniformity over the test assemblies upon the thermal-hydraulic characteristics of a system consisting of two FA models was studied as well.

Cooling of Completely Uncovered Rod Bundle

An investigation of cooling completely uncovered heater assemblies by natural convection of gas and steam-gas mixture was performed for the conditions of severe accidents. The experiments have confirmed that there exists a mechanism of heat transfer from a dried fuel rod bundle heated up to a high temperature (Tclad=600-700°C) by gas natural convection vortexes through a riser to a heat exchanger located in the upper part of the circuit. A considerable intensification of heat transfer at a pressure increase was noted. It should be expected that in the integral reactor with the operation of the built-in ERHR heat exchangers, the cooling of a dried core by natural convection will be more intensive due to the weak interaction of downstream and upstream flows because of the large cross section of the riser. This mechanism of fuel rod cooling should be taken into account in the analysis of accidents with core dry-out.

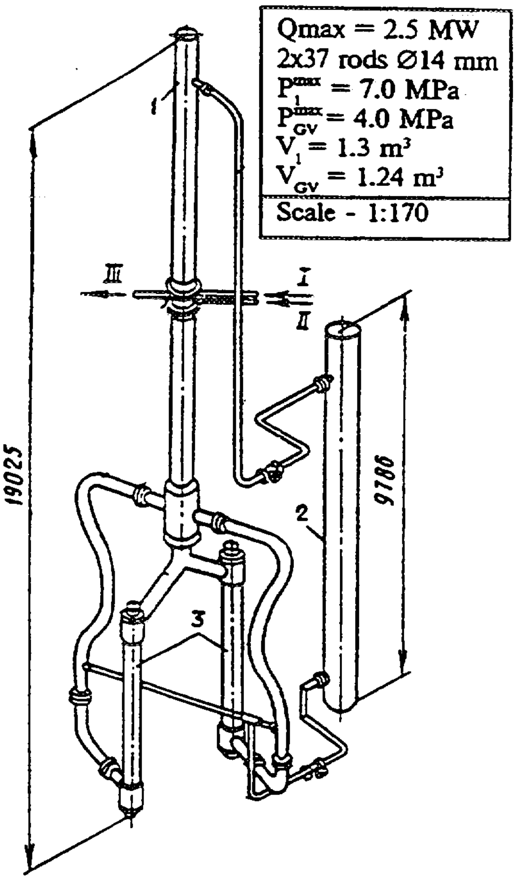
9. EXPERIMENTS AT LARGE-SCALE REACTOR MODELS

To validate the AST-500 safety, a complex of investigations on thermohydraulic processes at loss-of-coolant and loss-of-heat removal accidents was performed at the KMR-2 test facility (Fig.8). The facility includes a large-scale model of the AST-500 integral reactor and its guard vessel simulator serving for confinement of leaks at loss-of-coolant accidents.

Proceeding from the AST design features, the following processes important for a correct description of emergency conditions were studied in the experiments:

- dynamics of the built-in steam-gas pressurizer;
- flow circulation and heat transfer in the primary circuit under loss-of-coolant conditions;
- reactor-guard vessel system thermohydraulic behavior;
- heat-and mass transfer and gas distribution inside the guard vessel [2,3].

The test program included experiments with coolant outflow into the guard vessel model at different sizes and locations of openings imitating leaks in the primary circuit at ruptures both in the pressurizer area and in the reactor vessel bottom, as well as imitation of an unscrammed loss-of-heat removal accident. The range of break sizes studied in the facility was 4-30 mm which corresponds to 50-400 mm for the AST-500 primary circuit. Similar investigations were carried out on the smaller-scale model of an integral reactor in the 1385 test rig (Fig.1) for an expanded range of parameters.



1. Primary circuit model of integral reactor
2. Guard vessel simulator
3. Heating columns

Fig.8. KMR-2 Test facility

Experimental data on the reactor-guard vessel system behavior at primary circuit loss-of-integrity accidents were obtained that showed the efficiency of the AST-type guard vessel as a passive confinement system, providing for keeping the core under coolant level. Normal fuel rod cooling was shown under the "waterfall" and steam-condensate circulation condition, taking place after considerable coolant losses from the primary circuit. Coolant subcooling values and flow circulation mode variations at water outflow from the test rig are shown in Fig.9.

Issues such as non-condensable gas effects, heat-and mass transfer and gas distribution in the guard vessel model, and temperature stratification in a water volume of the guard vessel model were investigated. Data on heat transfer coefficients for steam condensation on the guard vessel walls were obtained for a wide range of steam-air mixture parameters (P=0.4-3.0 MPa; Pg/P=0.18-0.8). They showed a considerable rise of steam condensation intensity at a pressure increase. The experimental data are well described by the calculation model based on the heat and mass transfer analogy using correlations for natural-convective heat transfer on a vertical wall. The known Uchida correlations, used for the description of containment thermal-hydraulics in some Western codes (MARCH 3 and the others) does not take into account the influence of the parameters and underestimates the value of heat transfer coefficients at increased pressures of P > 0.3 MPa.

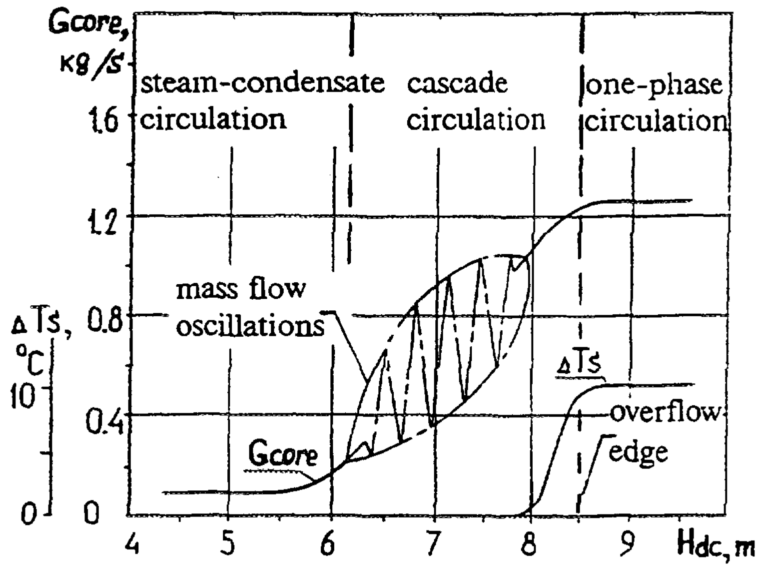


Fig 9. KMR-2. Natural circulation modes during small LOCA depending on water level in downcomer

10. CODES FOR THERMAL-HYDRAULICS ANALYSIS

The complex of codes used at OKBM for thermal-hydraulic calculations of AST-type reactor plants under stationary conditions, for the analysis of natural circulation stability and for transients and accidents is presented in Fig.10.

OKBM-developed Codes

The codes that have been developed at OKBM describe the details of thermal-hydraulic processes in the reactor during normal operation and in emergency situations. They can handle the distinctive design features of the integral reactor, such as built-in pressurizer with a large amount of non-condensable gas, in-reactor heat exchangers, guard vessel, and passive safety systems. The calculation models used in the codes rely upon the results of experimental investigations of thermal-hydraulics conforming to integral reactors of the AST type.

Empirical relationships are used for the calculation of heat and mass transfer, hydraulic resistance, steam slipping and leakage flow rates. The VERESK-M code for the calculation of stationary thermohydraulic characteristics describes non-equilibrium boiling in the core and in the riser section using the models given in [11,12]. In the codes DKAST and UROVEN/MB-3 for the analysis of thermal-hydraulics in transients, the equilibrium model of two-phase flow with steam slipping is used.

The procedures for calculating the steam condensation from a steam-gas mixture on a tube bundle of a heat exchanger are based on the results of experimental investigations on the in-reactor heat exchanger models. An empirical dependence is used for the calculation of heat transfer coefficients for steam condensation on the guard vessel walls. The procedure for the calculation of the steam-gas pressurizer characteristics and for the distribution of gas in the primary circuit have been developed based on the results of the experiments in the large-scale model of pressurizer (L-800 test rig) and in integral reactor models.

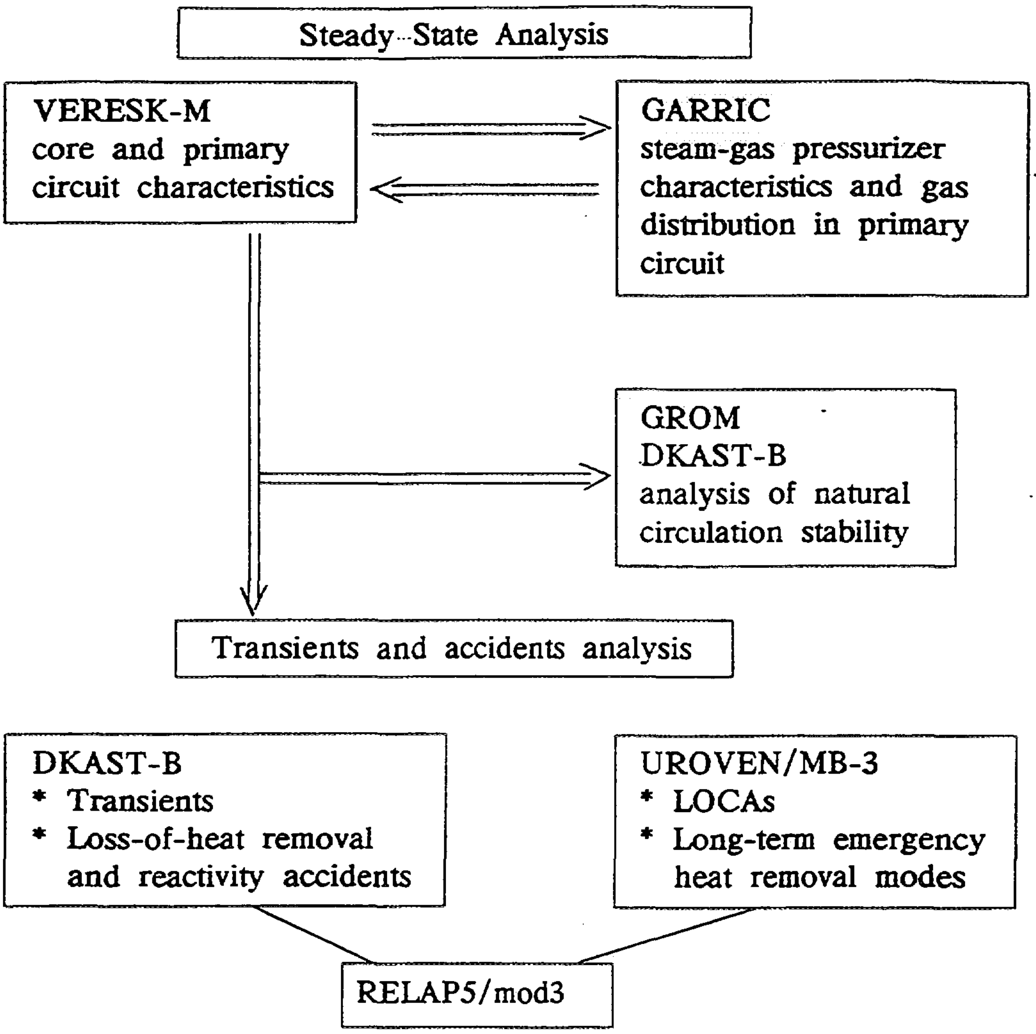


Fig.10 Code package used for thermohydraulic analysis of AST reactor plants

The reliability of calculational prediction of thermal-hydraulic characteristics of the AST under normal and emergency conditions has been confirmed by verification of the codes performed on the basis of the results of experimental investigations of separate effects and integral phenomena, and on integral data obtained from different scale models of the integral reactor in the KMR-2 and 1385 test facilities.

Use of RELAP5/mod3 code

The RELAP5/mod3 code is used in OKBM for alternative analysis of the thermal-hydraulics of AST-type reactors under emergency conditions. A verification study is under way to estimate the code applicability to integral reactors. Calculation analysis of experimental modes realized on the integral reactor models (KMR-2 and 1385 test rigs) has shown a satisfactory accuracy of the plant behavior representation in the RELAP5/mod3 code. The class of transients and accidents corresponding to the range of the RELAP5/mod3 code applicability for AST-type integral reactors has been determined based on the results of verification studies.

11. CONCLUSIONS

Comprehensive experimental investigations of the AST reactor thermal-hydraulic problems for normal and emergency conditions have been performed. These investigations included the study of primary circuit natural circulation characteristics, natural circulation stability issues, boiling crisis, two-phase flow characteristics, steam-gas pressurizer and gas distribution characteristics, intensity of steam condensation from steam-gas mixtures, as well as emergency processes studies using different-scale models of the reactor, etc.

The calculation models and computer codes for the analysis of AST-reactor plant thermal-hydraulics under normal operation and emergency conditions have been developed and validated experimentally. The code verification confirmed the reliability of calculational prediction of the AST reactor thermal-hydraulics [13].

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