# VERIFICATION OF CALCULATION TECHNIQUE OF CRITICAL HEAT FLUX IN RELAP5/MOD3.2 CODE APPLIED TO THE MIR RESEARCH REACTOR ON EXPERIMENTAL DATA OBTAINED FOR THE ANNULUS AT AVERAGE WATER PRESSURES

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## INTRODUCTION

The problem of calculating the critical heat flux is very important for determining the ultimate power of the MIR research reactor working FA. Good knowledge of the critical heat flux parameters is an integral part of safety analysis.

The thermal-hydraulics model of the MIR reactor facility primary circuit being currently developed on the basis of the RELAP5/MOD3.2 code [1] allows a rather wide range of heat removal modes to be covered within the scope of a unified approach. Nevertheless, in spite of the universality and verification of this code, the problem of verifying the critical heat flux calculation technique for the research reactor conditions remains urgent.

The MIR reactor is a research, thermal, heterogeneous, channel-type reactor with the core immersed into the water pool. The core is formed of beryllium units with the "turnkey" size of 150 mm. Along their axis there are single-flow zirconium channels to place working FAs and the Fild-type loop channels to place experimental FAs. The working FA contains 4 coaxially located annular fuel elements with the column made of U dioxide-Al mixture and with an aluminium alloy cladding.

The main thermal-hydraulic characteristics of the reactor are presented in Tab.1.

Table 1

Main thermal-hydraulic characteristics of the MIR reactor

| Name of parameter | Dimensions | Value |
|---|---|---|
| Maximum reactor design heat power | MW | 100 |
| Maximum heat load of the working FA | $MW/m^2$ | 4.2 |
| Primary circuit coolant flow rate | $m^3/h$ | 2000÷3300 |
| Maximum coolant rate in the working FA | m/s | 10 |
| Coolant temperature at the core inlet | °C | up to 70 |
| Coolant temperature at the core outlet | °C | up to 98 |
| Maximum temperature of the fuel element cladding of the working FA | °C | 170 |
| Coolant pressure in the input collector | MPa | 1.05÷1.25 |
| Coolant pressure in the output collector | MPa | 0.65÷0.85 |
| FA hydraulic diameter | M | $5\cdot10^{-3}$ |
| Coolant flow in FA | | From top to bottom |

Performance of the experiments on departure from nuclear boiling investigation directly on the MIR RF working FA seems problematic. Therefore, the references data are involved to verify the calculation technique of critical heat flux in the RELAP5/MOD3.2 code for the MIR RF conditions.

## 1. PROBLEM STATEMENT

The RELAP5/MOD3.2 code uses the 1986 AECL-UO Critical Heat Flux Lookup Table method by Dr. Groeneveld and co-workers [1]. The data bank includes 15000 tube data points. The range of the Table:

- mass flux: 0÷7500 kg/s×m$^2$;
- pressure: 0.1÷20.0 MPa;
- quality: (- 0.5)÷1.0.

The MIR operating modes are characterized by the following change range of the coolant basic parameters:

- mass flux up to 10000 kg/s×m$^2$;
- core inlet pressure up to 1.25 MPa;
- coolant relative enthalpy up to -0.2;

In terms of mass fluxes this range is beyond the Groeneveld Table. Thus, verification of the calculation technique of critical heat flux in the code is necessary as applied to the MIR research reactor.

## 2. REVIEW OF EXPERIMENTAL DATA OBTAINED FOR ANNULUSES

Ref.[2] presents the analysis of experimental data obtained in the annuluses. As follows from analyzing the range of the empirical correlations, they developed predominantly for the pressures of 2-20 MPa, mass fluxes of 200-5000 kg/s×m$^2$. It is shown that the effect of a heating type (external, internal and bilateral), heat-dispersing surfaces curvature, pressure (with different gaps of the annular slot, mass void fractions and rate) on the critical heat flux has not been sufficiently studied yet. Few data have been obtained at small (less than 0.1 m) and large (more than 1 m) lengths, at small (less than 500 kg/s×m$^2$) and large (more than 3000 kg/s×m$^2$) mass fluxes of the coolant and at small pressures (2-5 MPa).

Ref.[3] presents the Table for describing the critical heat fluxes in different annuluses in a wide range of geometrical and regime parameters. The maximum usage range of the Table is as follows:

- Pressure: 0.1 - 20 MPa;
- Mass flux: 50 - 6000 kg/s×m$^2$;
- Balance void fraction: –0.5 – 0.9;
- Thermal diameter: 3.8 - 97 mm;
- Heated length: 0.04 – 2.8 m;

– Radius ratio: 0 - 1 – for external heating and 0.3 - 1 for internal heating.

The analysis of the Table shows that the critical heat flux values for average water pressures (up to 2 MPa) and coolant subcooling (x<0) are not provided by the experimental data. The available data are extrapolated to the indicated region. Besides, the Table does not cover the region of mass fluxes (up to 10000 kg/s×m$^2$) typical for research reactors.

Ref.[4] presents the data on investigation of critical heat flux in the annulus for average water pressures (0.5÷1.7 MPa), high mass fluxes (2000÷10000 kg/s×m$^2$) and coolant subcooling. A good description of the experimental facility, working section, departure from nuclear boiling attainment technique allows modeling of the experiment by the RELAP5/MOD3.2 thermal-hydraulic code.

## 3. EXPERIMENTAL FACILITY AND EXPERIMENTS PERFORMANCE TECHNIQUE FOR CRITICAL HEAT FLUX DETERMINATION

Ref.[4] describes the experimental facility, presents the measurement technique and results of critical heat fluxes in the annulus corresponding to the geometry of fuel elements in the MARIA research reactor. The experiments were performed in 1977 at the WIW stand in the Institute of Atomic Research in Swierk (Poland).

The experimental data were obtained for the conditions presented in Tab.2.

Table 2

Geometrical and regime parameters in the experiment

| Name of parameter | dimensions | Parameter value |
|---|---|---|
| Gap type | - | Annular |
| Heating type | - | Internal |
| Gap width | mm | 1.8 |
| Diameter of the heated surface | mm | 10 |
| Coolant flow direction | - | From bottom upwards |
| Length of the heated section | m | 0.45 |
| Coolant temperature at the working section inlet | °C | 40÷120 |
| Coolant mass flux, | kg/s×m$^2$ | 2000÷10000 |
| Outlet coolant pressure | MPa | 0.5÷1.7 |

The heated element was a small tube made of stainless steel, placed in the aluminium alloy pipe. On the surface of the internal pipe there were spacing ribs designed for concentric layout of the pipes. Surface roughness of the small tube is at most 0.01 mm.

The measurement errors of the thermal-hydraulic parameters in the experiment are presented in Tab.3.

Table 3

Measurement errors of the thermal-hydraulic parameters
in the experiment

| Name of parameter | Measurement error |
|---|---|
| Inlet pressure, pressure in the middle of the experimental section and at the annular gap outlet | 2% |
| Pressure drop at the experimental section | 0.025% |
| Inlet, outlet water temperature | 0.25°C |
| Water heating at the experimental section | 0.25°C |
| Water flow rate | 1% |
| Input power | 2% |

The departure from nuclear boiling was attained by smooth increase of the heated pipe power at the constant coolant flow rate and was recorded by a special detector operating on the principle of a balanced bridge.

Upon the performed calculations 120 values of the critical heat flux were obtained versus the coolant regime parameters.

## 4. RELAP5/MOD3.2 NODALIZATION SCHEME FOR THE EXPERIMENTAL SECTION

The RELAP5/MOD3.2 nodalization scheme of the experimental section is presented in Fig.1. It included 13 volumes, 12 junctions, 9 heat structures.

The hydraulic path of the experimental section was described by the ANNULUS 105 component consisting of 9 sections with equal height. Components TV101 and TJ102 of the nodalization scheme determined the coolant parameters at the experimental section inlet (temperature and pressure; coolant mass flux, respectively). Component TV706 was used as a boundary condition (temperature and pressure) for the coolant at the experimental section outlet. Components SV103, SV107 modeled the hydraulic path sections at the experimental section inlet and outlet, respectively.

Uniform axial distribution of energy release at the working section was incorporated in the design.

The departure from nuclear boiling was modeled by increase of the heated pipe power at the constant coolant mass flux through it.

The moment of the departure from nuclear boiling origin was determined by abrupt decrease of the heat transfer coefficient at one of the high-altitude sections of the heated element. In this case the following was recorded: an elevation of the section, where the departure from nuclear boiling arose, heat flux at this elevation.

To assess the sensitivity of the nodalization scheme to the nodalization degree, the calculations were made with the increased number of design sections in component A105 and relevant heat structure (n=18) and with the decreased number of sections (n=5). The critical heat flux was changed by ≈ by 3%, i.e. negligibly. In this event there were no changes in identification of the coolant flow regime.

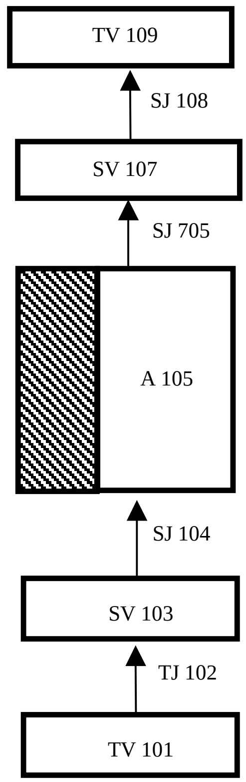


Fig. 1. Nodalization scheme of the experimental section:
TV - time - dependent volume; SJ – single-junction; SV – single- volume; A - annulus; TJ - time - dependent junction;  - heat structure

## 5. CALCULATION RESULTS OF CRITICAL HEAT FLUX BY THE RELAP5/MOD3.2 CODE COMPARED TO EXPERIMENTAL DATA

As the result of performed calculations 120 values of critical heat flux were obtained versus the coolant regime parameters. The place of the departure from nuclear boiling origin was an output element of the working section model.

Fig.2-3 show the calculated and experimental values of the critical heat flux as typical ones versus the equilibrium qualities at the working section outlet for mass fluxes of 10000, 6000 and 2000 kg/s×m$^2$ and the coolant pressure of 0.5 MPa.

It was established that the RELAP5/MOD3.2 code gives the underrated (conservative) value of critical heat flux (~ by 20÷30% as compared to the experimental one) for mass fluxes of 8000 and 10000 kg/s×m$^2$ in the whole investigated range of pressures

(0.5÷1.7MPa) and temperatures. In this case the departure from nuclear boiling occurs under the bubble flow regime (mode 4 in the regime chart for vertical volumes). The conclusion can be made of the fact that extrapolation of the critical heat flux values beyond the range of the Groeneveld table by the mass flux (over 7500 kg/s×m$^2$) in the code is performed conservatively.

The code satisfactorily calculates the critical heat flux values for the mass flux of 6000 kg/s×m$^2$. Deviation of the calculated values from the experimental ones makes up ~ 6÷15%.

For the mass flux of 4000 and 2000 kg/s×m$^2$ the RELAP5/MOD3.2 code systematically overrates the critical heat flux value (~ by 150÷200%) for the whole investigated range of pressures and temperatures. The departure from nuclear boiling occurs at this mass flux under the slug flow regime (mode 5 in the regime chart for vertical volumes). As it was stated above, the conditions of departure from nuclear boiling origin depend not only on the flux regime parameters – pressure, mass flux, relative enthalpy, but also on the heating type (external, internal and bilateral) as well as on geometrical parameters – width of the annular gap, heat-dispersing surfaces curvature, and at small mass fluxes – the coolant flow direction. At small annular gaps comparable with the fly-off bubble diameter (the fly-off bubble diameter makes up ≈ 3.9 mm for the pressure of 1.7 MPa and 4.4 mm for the pressure of 0.5 MPa) there is decrease of the critical heat flux, as the close position of the walls in the narrow crevice channels impedes the vapor removal from the heat-dispersing surface, which facilitates the formation of vapor locks and the decrease of the critical heat flux [2].

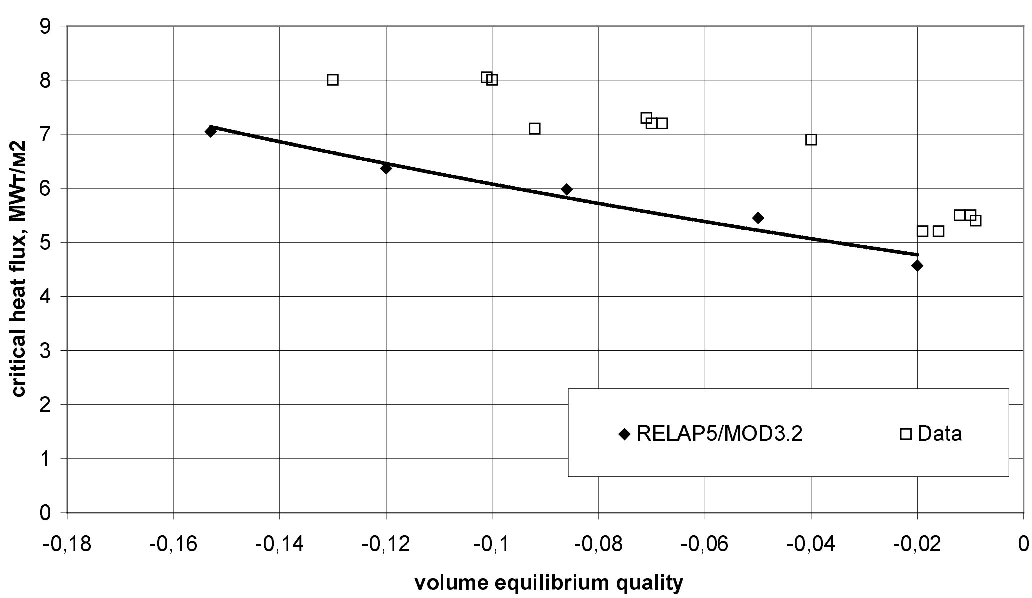


Fig. 2. Calculated and experimental CHF data for pressure 0.5 MPa and mass flux of 10000 kg/s×m$^2$

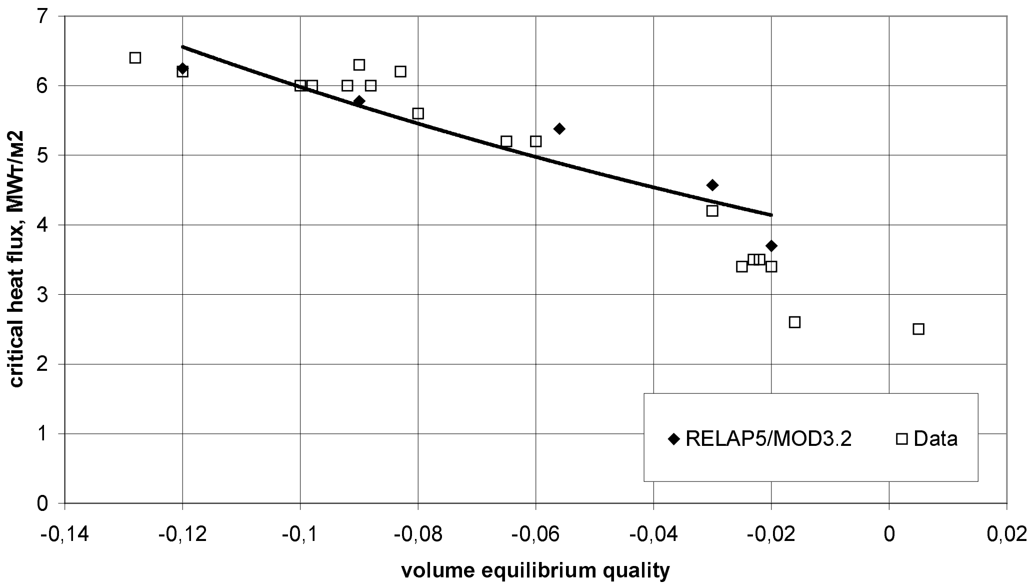


Fig. 3. Calculated and experimental CHF data for pressure 0.5 MPa and mass flux of 6000 kg/s×m2

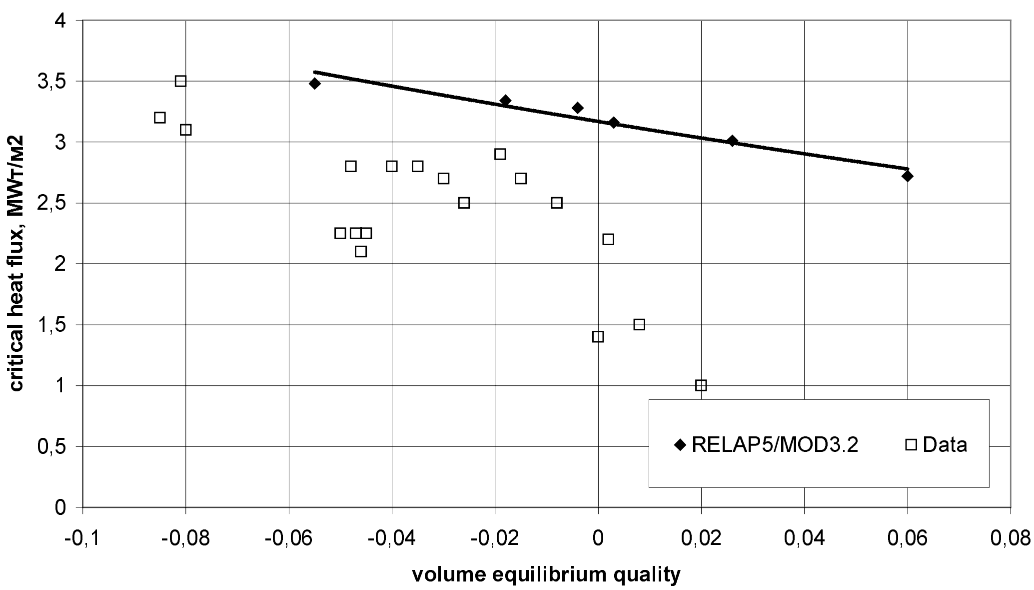


Fig. 4. Calculated and experimental CHF data for pressure 0.5 MPa and mass flux of 2000 kg/s×m$^2$

## CONCLUSIONS

Verification of the calculation technique of critical heat flux in the RELAP5/MOD3.2 code for average water pressures (0.5-1.7 MPa) and a wide range of mass flux (up to 10000 $kg/s\times m^2$) by the annulus experiments showed the following.

- Extrapolation of the critical heat flux values beyond the range of the Groeneveld table by mass flux (over 7500 $kg/s\times m^2$) in the code is performed conservatively. The code gives the underrated critical heat flux (~ by 20÷30% as compared to the experimental one) for mass fluxes of 8000 and 10000 $kg/s\times m^2$ in the whole investigated range of pressures (0.5-1.7 MPa) and temperatures.
- The code satisfactorily calculates the critical heat flux values for the mass flux of 6000 $kg/s\times m^2$. Deviation of the calculated values from the experimental ones makes up ~ 6÷15%.
- For the mass flux of 4000 and 2000 $kg/s\times m^2$ the RELAP5/MOD3.2 code systematically overrates the critical heat flux value (~ by 150÷200%) for the whole investigated range of pressures and temperatures. Probably, that is the result of the narrow annular gap in this case. At small annular gaps comparable with the fly-off bubble diameter there is decrease of the critical heat flux, as the close position of the walls in the narrow crevice channels impedes the vapor removal from the heat-dispersing surface, which facilitates the formation of vapor locks and the decrease of the critical heat flux.

## REFERENCES

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