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Thermal analysis of IRT-T reactor fuel elements

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Abstract. The article describes the method and results of thermo-physical calculations of IRT-T reactor core. Heat fluxes, temperatures of cladding, fuel meat and coolant were calculated for height of core, azimuth directions of FA and each fuel elements in FA. Average calculated values of uniformity factor of energy release distribution for height of fuel assemblies were shown in this research. Onset nucleate boiling temperature and ONB-ratio were calculated. Shows that temperature regimes of fuel elements at rated power of the reactor keep within limit values and meet safety requirements. Obtained results could be applied at feasibility study of increasing thermal power of IRT-T research reactor to 12 MW level.

1. Introduction

IRT-T research reactor uses IRT-3M type fuel assemblies in the reactor core [1]. The technical report of IRT-3M safety provision gives the values of the following parameters:

- inlet coolant temperature — 45 °C;
- outlet pressure — 133 kPa;
- pressure drop — 37 kPa.

The values of nonuniformity of energy release at ~220 kW/m^2 average heat flux density can reach ~2 times higher than the maximum values due to sufficiently high heterogeneity of materials arrangement in the reactor core. Therefore, in order to establish correspondence of thermal parameters of fuel elements to passport values it is needed to determine values of nonuniformity of energy release over fuel assemblies and reactor core. For this purpose, it was used MCU-PTR software designed for precision neutron-physical calculations [2]. Distributions of energy release were obtained as a result of three-dimensional calculations, also, the most stressed fuel assemblies and most stressed areas in them (directions) were selected.

2. The method of calculation of thermal-physical parameters

Thermal analysis was carried out for steady state of reactor core at rated power (6 MW) in accordance with specially designed method [3].

The distributions of energy release over calculation cells, height and azimuth of all fuel assemblies of reactor core were obtained as a result of neutron-physical calculation. 8-tube and 6-tube fuel assemblies with the most stressed parameters were used for the calculation. These distributions were input data to carry out thermal analysis.

Coolant temperature distribution and temperature distribution of fuel element's wall over the height of the most stressed fuel assemblies were calculated within approximation of one-dimensional cylindrical channel with coaxial fuel elements. Inlet coolant temperature, pressure and pressure drop in reactor core were set constant [4].

The results of calculation over height areas of the most stressed azimuth and height direction of fuel assemblies determined:

- coolant temperature between fuel elements;
- fuel element's wall temperature;
- temperature of cladding-core border of fuel elements;
- maximum temperature of fuel element's core;
- surface boiling point;
- surface boiling safety margin;
- heat fluxes from fuel element's surface;
- coolant heating over each ring area between fuel elements.

3. Calculation models for fuel assemblies

Thermal regime of fuel assemblies was calculated by using ASTRA software [5], designed for calculation of fuel assemblies with tubular coaxial fuel elements. The software makes it possible to calculate distribution of heat fluxes from fuel element's surface, distribution of coolant temperatures in the gaps between fuel elements, distribution of outer surface temperatures of fuel elements, distribution of temperatures of a contact between cladding and core of the fuel elements, as well as distributions of maximum temperatures of fuel elements' cores over the height of reactor core. The software also allows carrying out calculation for direct flow as well as for Field scheme flow of the coolant.

Distributions of temperature over the volume of cylindrical models for the most stressed fuel assemblies in (r,z)-geometry were obtained by using ASTRA software. The influence of azimuth unevenness of energy release in these fuel assemblies was estimated by using distributions of energy release of the most stressed directions in the model.

The calculation of three-dimensional fuel assembly was carried out by using two-dimensional ASTRA software with the following conditions:

- Perimeters of cross-section, cladding thickness and core thickness should be preserved since heat flux is moving across fuel elements' walls. This condition cannot be fulfilled for all the gaps and fuel elements. However, since in the most stressed conditions generally works the outer fuel element, its perimeters of cross-section were preserved along with cladding thickness, core thickness and thickness of the surrounding gaps with the coolant.
- Thicknesses of areas were preserved for the rest fuel elements and gaps.
- Azimuth unevenness of energy release distribution was taken into account as follows: radial distributions of energy release over eight azimuths with 45° step (see Figure 2, 3) was obtained; averaged radial distribution of energy release was used to calculate coolant heating; the distribution with maximum irregularity factor was used for calculation of the most stressed parameters [6].

4. Calculation results

During the reactor operation, the fuel burns up and distributions of energy release over reactor height and over cross section of reactor core are evened. Therefore, maximum values of heat flux densities from fuel elements' surface, maximum values of fuel and coolant temperature should be observed in "fresh" fuel assembly [7].

Averaged calculation values of irregularity factors of energy release distribution over the fuel assembly height are:

- for eight-tube — 1,28;
- for six-tube — 1,27.

Average value of specific energy release in fuel element's core for the most stressed eight-tube fuel assembly is 1.093 kW/cm^3 (for six-tube fuel assembly – 0.974 kW/cm^3).

The boiling point (T_b) of coolant is 114.5 °C at the outlet coolant pressure of 1.33 bar. Maximum value of coolant temperature does not exceed 63.4 °C when the coolant of eight-tube fuel assembly is heated up to average 9.4 °C. Maximum values of fuel temperature (82.5 °C), cladding temperature (81.0 °C) and maximum values of heat flux density from fuel elements to coolant — 487 kW/m^2 (on the surface of the outer fuel element) are achieved in the outer fuel element, which has the highest specific power in fuel assembly. This value of heat flux density is much lower than designed value, which is 800 kW/m^2.

Figure 1 shows the ONB ratio for the outer fuel element of eight-tube fuel assembly at inlet coolant temperature 45 °C.

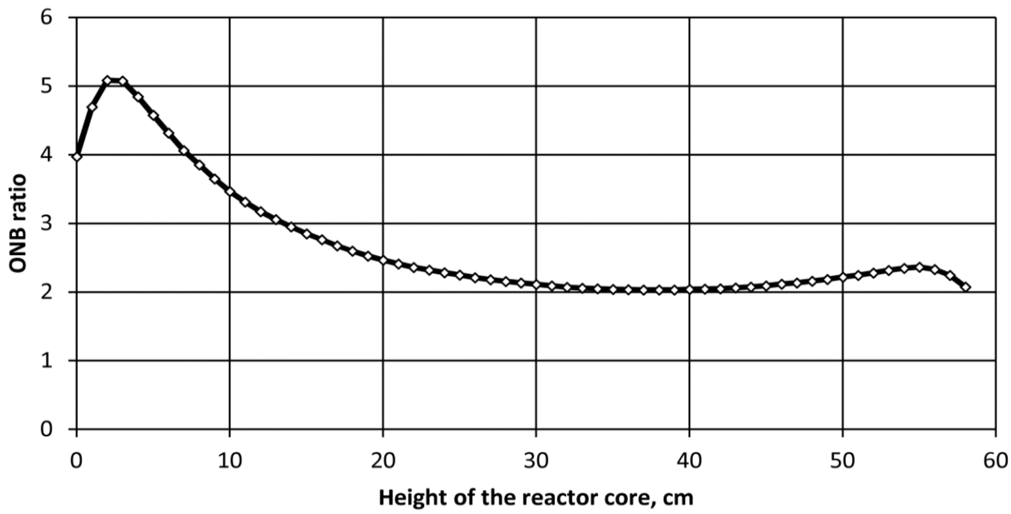

Figure 1. ONB ratio for the outer fuel element of the most stressed eight-tube fuel assembly during the first cycle at inlet coolant temperature 45 °C.

Unlike energy release distribution over “fresh” reactor core, in the real cartograms of fuel loading for IRT-T reactor the amount of “fresh” fuel assemblies generally does not exceed 1-2 pcs. In the irradiated fuel assemblies the burnup lies in the range 5–60 %, thereby, there is not only reduced power in these assemblies but also significantly reduced irregularity factors of energy release. Therefore, in the real cartograms of loading maximum values of heat flux density and fuel temperature are found in “fresh” fuel assemblies with zero burnup.

As an example the cycle was considered which has irradiated fuel assemblies as well as 2 “fresh” ones. Figure 2 shows burnup distribution over reactor cells and designed energy release distribution obtained by means of MCU-PTR software.

Average ^{235}U burnup over reactor cells (% of initial loading)			
10.07	53.31	54.53	11.20
0.00	60.09	54.03	60.22
51.90			56.98
53.16			56.08
0.00	60.93	55.63	58.90
10.37	54.19	54.36	11.01

Distribution of the density of number of fission over reactor core cells (normalized to the average value), rel. u.			
1.21	0.89	0.86	1.21
1.26	0.80	0.90	0.74
1.08			1.08
1.08			1.08
1.31	0.81	0.87	0.74
1.24	0.90	0.86	1.20

Figure 2. Burnup and energy release (normalized to the average value) distributions over reactor core cells at the beginning of the third cycle.

As in the case of "fresh" reactor core, the maximum values of relative power are ~1.3 and found in "fresh" fuel assemblies with zero burnup, which, in this case, are six-tube assemblies. Azimuth distribution of energy release for the most stressed six-tube fuel assembly is shown in Figure 3.

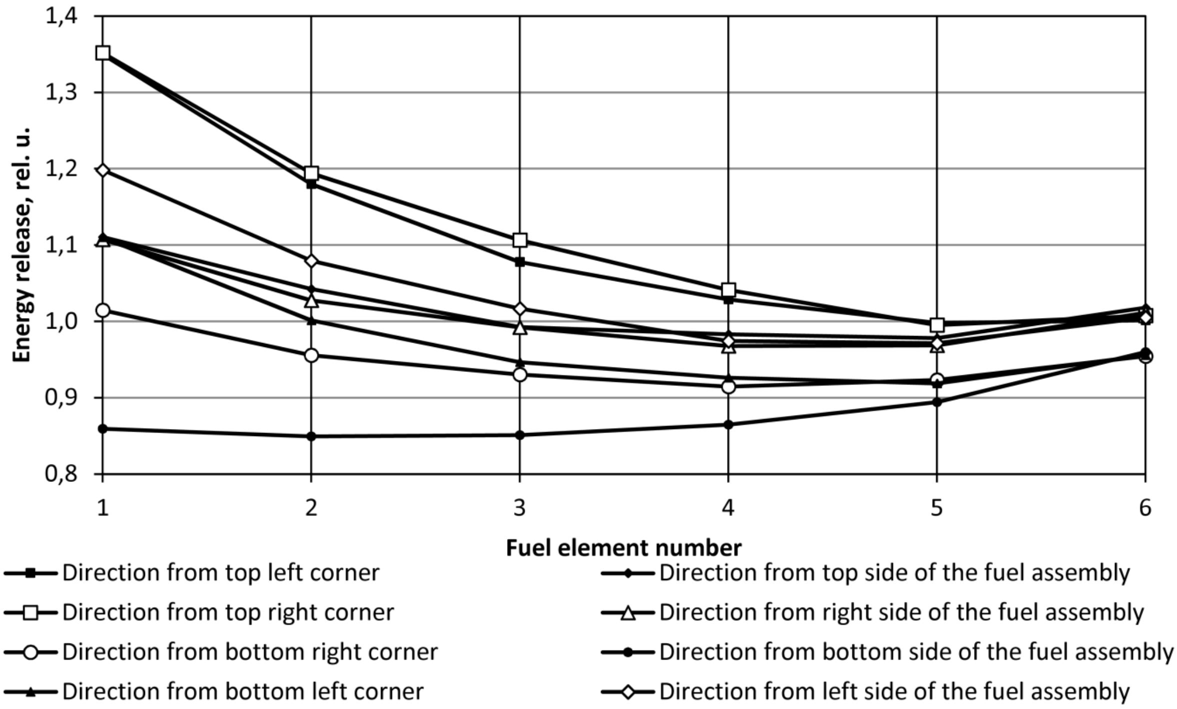

Figure 3. Radial distributions of energy release for six-tube fuel assembly over different directions (normalized to average value over fuel assembly, count from inner to outer fuel element).

Figure 4 shows distribution of ONB ratio for the outer fuel element of six-tube fuel assembly at inlet coolant temperature 45 °C.

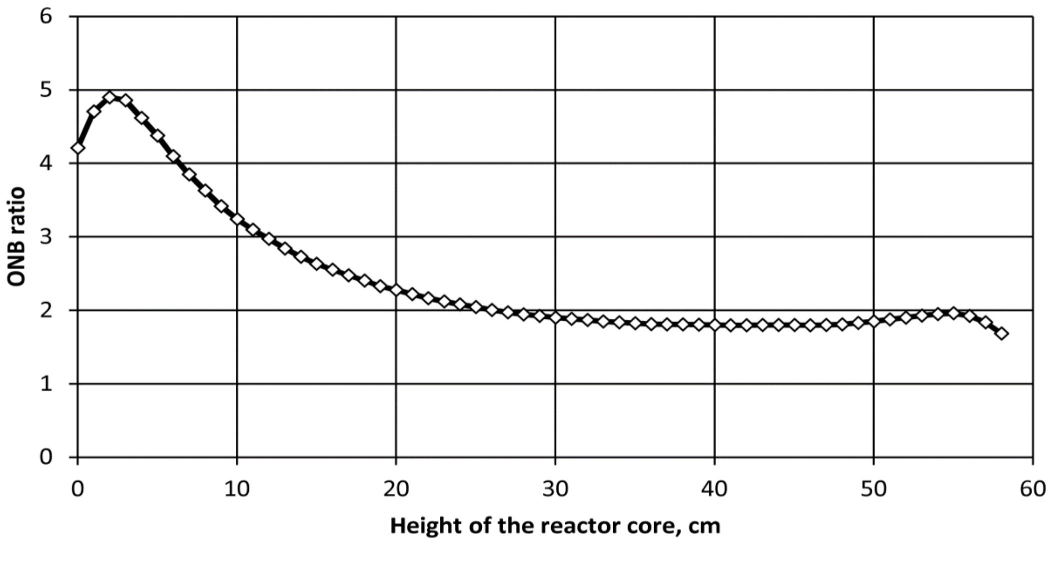

Figure 4. ONB ratio at inlet coolant temperature 45 °C for the outer fuel element of the most stressed six-tube fuel assembly during the 3rd cycle.

The maximum value of specific energy releases in fuel element's core for six-tube fuel assembly is 2.69 kW/cm^3. At inlet coolant temperature 45 °C the maximum value of coolant temperature in given azimuth direction does not exceed 67.2 °C. The outer fuel element has the highest specific power in fuel assembly, therefore, maximum values of fuel temperature (87.4 °C), cladding temperature (85.6 °C) and maximum values of heat flux density from fuel elements to coolant — 538.3 kW/m^2 are achieved in this outer fuel element

Coolant ONB ratio is 1.8.

5. Conclusion

This paper describes the method of determination of thermal-physical and energy parameters for IRT-T reactor. The calculation of thermal-physical regimes of fuel elements was carried out during different cartograms of fuel loading. It was shown that at IRT-T reactor rated power (6 MW) temperature regimes of fuel elements keep within the limit values and meet safety requirements of reactor operation. Maximum values of fuel elements' surface temperature and fuel elements' cores at 6 MW reactor power does not exceed 86 °C. Maximum coolant heating in the gaps between fuel elements is 15 °C and maximum heat flux density from fuel element's surface does not exceed 540 kW/m^2.

One of the reactor safety criteria is that in case of not exceeding the designed value of relative power of the fuel assembly (which is 1.3 in "fresh" fuel assembly) heat flux density in its most stressed area does not exceed 540 kW/m^2 and ONB ratio is not lower than 1.8.

Acknowledgment

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References

[1] Varlachev V A, Glukhov G G and Skuridin V S 2011 *Research Nuclear Reactor IRT-T*, Tomsk Polytechnic University (Tomsk:TPU Press)

[2] Alekseev N I and Gomin E A, 2011 MCU-PTR Program for High-precision Calculations of Pool and Tank Type Research Reactors, (Atomic En. Vol 109/3) ed N N. Ponomarev-Stepnoi (Moscov: Atomnaya Energiya Press) pp 149–158

[3] Naymushin A G, Chertkov Yu B, Lebedev I I and Anikin M N, 2013 Method and Results of Thermo-physical Calculations of IRT-T Fuel Elements (Proc. of Institution of Higher Education, Physics vol 11/2), (Tomsk:TSU Press) pp 360–366

[4] Nagy, Mohamed E., and Amgad M. Shokr. Hot channel factors evaluation for thermalhydraulic analysis of MTR reactors. Nukleonika 49.2 (2004): 61-67.

[5] Taliev A V, 2006 Modernized ASTRA code for thermal calculation of research reactor FAs consisting from tubular coaxial fuel elements (Preprint IAE-6405/5, Moscow)

[6] Zagar, Tomaz, M. Ravnik, and A. Peršic. "Analysis of TRIGA reactor thermal power calibration method." International Conference Nuclear Energy in Central Europe. 1999. pp. 91-98

[7] Naymushin A G, Chertkov Yu B, Lebedev I I and Elpashev S N, 2012 Feasibility Study of Energy Realise Distributions in IRT-T Reactor for Different Control Rods Positions (Proc. of Institution of Higher Education, Physics vol 11/2), (Tomsk:TSU Press) pp 353–359