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THERMAL-HYDRAULIC CALCULATIONS OF THE WWR-SM RESEARCH REACTOR

The paper presents calculations of the thermal power distribution in the reactor core (RC) of the WWR-SM research nuclear reactor of the Institute of Nuclear Physics of the Academy of Sciences of the Republic of Uzbekistan, settlement Ulugbek, Tashkent, both for all fuel assemblies loaded into the core and for each fuel element of a separate fuel assembly. These calculations were carried out for RC configurations with a different number of fuel assemblies – 18, 20, and 24. The power distribution reactor core was performed using the IRT-2D code. A detailed simulation of the power distribution in the fuel element was performed using the MCNP4C code, while the fuel elements were modeled as square pipes with straight angles without rounding. The power distribution was calculated for each side of each fuel tube and divided into 15 axial nodes. The results of modeling of the thermal-hydraulic state were obtained using the PLTEMP code for various RC configurations. In the calculations, it was assumed, that the inlet water temperature is 45 and 48 °C for all RC configurations, the heat hydraulic parameters were taken from the calculation of the flow rate of the first circuit through the core 1250 m^3/h. An analysis of the thermal power distribution of nuclear fuel in the reactor core of the WWR-SM research reactor showed that even with a conservative approach, permissible operating modes are not exceeded. During the operation of the three main circulation pumps, which provide the coolant flow through the core at the level of 1250 m^3/h, the heat exchange crisis does not occur in the most energy-stressed fuel assemblies, namely, the temperature of the fuel rod clad and coolant remains below the permissible limits.

Keywords: reactor core, fuel assembly, fuel element, heat flux, thermal power.

1. Introduction

The WWR-SM research reactor is 10 MW power light water-cooled, light water moderated tank-type reactor. The reactor operates since 1959. IRT-3M type highly enriched uranium (HEU, 36 %) and IRT-4M type low enriched uranium (LEU, 19.7 %) fuel assemblies are used in the reactor core. In 2008 - 2009 reactor core was converted to the use of LEU fuel. More information about the WWR-SM reactor can be found at [1].

During the normal reactor operation on power with any reactor core configuration (18, 20, 24 fuel assemblies) heat removal is provided by three main circulation pumps, which provide the coolant flow through the reactor core at the flow rate of 1250 m^3/h.

The task of the thermal calculation of the reactor is to determine the main thermal technical parameters with known design and specified power. The main goal of the thermal-hydraulic calculation of reactors is to establish the distribution of heat fluxes and temperatures across the reactor core (RC), find the maximum temperature of the fuel to confirm the impossibility of its melting in the fuel elements (FE) with the maximum heat load, and determine the reserve before the heat transfer crisis.

All thermal-hydraulic parameters – heat fluxes, temperatures, and coolant parameters – were determined for the average and maximum loaded FE. The calculation was carried out for 15 nodes along with the height of the core with coordinates, starting from the bottom point of the fuel assembly (FA) - 0 cm, to the top point – 60 cm for every 4 cm.

2. Analysis of the thermal power distribution of nuclear fuel in the reactor core of the WWR-SM research reactor

Calculations of the distribution of thermal power in the RC were carried out both for all FAs loaded into the core and for each element of an individual FA. Knowledge of the thermal load experienced by the FA and its elements is a prerequisite for the safe operation of a research nuclear reactor. These calculations were carried out for RC configurations with a different number of FAs – 18, 20, and 24.

The distribution of thermal power for the RC configuration of 18 FAs of the IRT-3M type and 20, 24 FAs of the IRT-4M type calculated according to using IRT-2D code [2] is presented in Tables 1 - 4 (the energy distribution is normalized to 1).

From Table 1 it is seen, that the most heat-stressed is a FA placed in cell 3 - 4, but at the same time, it has a significant margin of heat load.

Table 1. **Distribution of thermal power in the RC for 18 fuel assemblies of the IRT-3M type**

Y	Thermal power, %					
6	0.000	5.790	5.970	6.380	5.570	0.000
5	5.710	4.350	5.530	6.010	3.730	0.000
4	0.000	3.720	6.170	5.590	4.350	5.960
3	0.000	6.180	6.580	6.050	6.360	0.000
X	2	3	4	5	6	7

Table 2. **Distribution of thermal power in the RC for 20 fuel assemblies of IRT-4M type**

Y	Thermal power, %					
6	0.000	5.21	4.891	4.785	4.137	0.000
5	4.205	5.438	5.985	5.978	4.974	4.636
4	4.897	5.018	5.863	5.849	5.283	4.042
3	0.000	4.387	4.532	4.596	5.297	0.000
X	2	3	4	5	6	7

Table 3. **Distribution of thermal power in the RC for 24 fuel assemblies of the IRT-4M type**

Y	Thermal power, %					
6	3.918	4.314	4.207	4.038	3.577	3.247
5	3.368	4.562	5.727	5.648	4.287	2.995
4	3.300	4.471	5.615	5.568	4.332	3.109
3	3.815	4.098	3.921	3.983	4.147	3.754
X	2	3	4	5	6	7

In the RC, there is axial symmetry of the location of the fuel assemblies in the cells of the RC. However, due to the fact, that fuel assemblies are located symmetrically and have different burnups of fuel, the symmetry is slightly smeared.

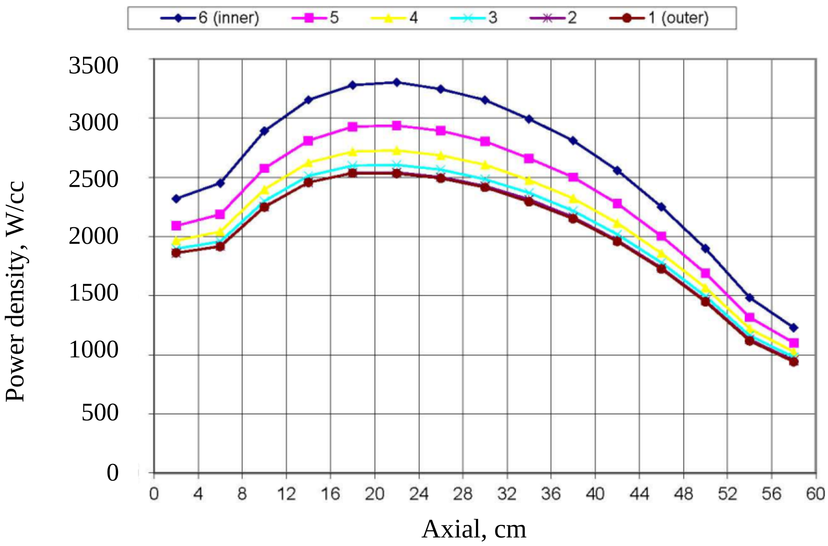
When comparing the results shown in Tables 1 - 3, the most heat-stressed FA is in the RC with 18 fuel assemblies loading and located in cell 3-4.

Table 4 shows the values of the thermal load distribution for each FE in FA located in cell 3 - 4 calculated using the MCNP4C program [3 - 5] (results of Table 1). Uncertainties in calculations in the MCNP4C model – 3 %.

Table 4. **Distribution of thermal power on FE for FA located in cell 3 - 4**

FE number	Mass of fuel element, g	Power of fuel element, kW
1 (inner)	33.30	72.41
2	39.20	84.57
3	46.00	98.86
4	56.60	121.56
5	62.70	134.49
6 (outer)	67.90	145.80
Total	305.70	657.70

Using the MCNP4C program [3 - 5], the distribution of thermal power along the length of each FE of a FA placed in cell 3 - 4 was determined (results of Table 1). The results of these calculations are presented in Figure.



Distribution of thermal power along the length of the fuel elements for fuel assemblies in cell 3 - 4.
(See color Figure on the journal website.)

The Figure shows that the maximum heat output is in the central (on height) regions of the FEs, and the center of the outer FE is the most heat-stressed place. The remaining 5 fuel elements have approximately the same thermal load. It is appropriate to note here that when monitoring the tightness of the claddings of FAs it is necessary to pay special attention to the tightness of external FEs.

3. Thermal physical parameters of the reactor core of the WWR-SM research reactor

The thermal model of the RC operation (i.e., the water and FE temperature, heat flux density, etc.) for a specific load of research nuclear reactor is determined by four independent parameters: the thermal power of the research nuclear reactor, water pressure drop in RC – Δp, and the water temperature at the RC inlet – T_{in} and water pressure at the RC inlet – P_{in}.

The last parameter is related to the water level in the basin, which varies slightly (except for emergencies with leaks). Therefore, the pressure during normal operation is almost constant. Since the normal water level in the central tank of the reactor is 3.5 m above the top of the RC, the inlet pressure to it is 1.35 bar.

As a limit to the safe operation of the research nuclear reactor, the value of the research nuclear reactor thermal power is accepted at which surface boiling occurs. This value (for a specific load of research nuclear reactor) is a function of the water pressure drop in the RC, as well as the temperature and pressure of the water at its inlet.

The nominal value of the water temperature at the inlet to the RC was taken equal to 45 °C, and the limit of normal operation was 48 °C.

Under these conditions, the limit of normal operation was determined by thermal power. The possible deviations of the values of the thermal power of the research nuclear reactor, the pressure drop in the RC, and the water pressure at the inlet from it from the calculated values were taken into account due to the inaccuracy of their measurement.

The choice of the safe thermal regime of the research nuclear reactor was based on the analysis of the most heat-stressed FAs in the RC (where the FE's maximum temperature is reached).

According to technical specifications, for low-enriched fuel/highly enriched fuel assemblies, the maximum cladding surface temperature should be less than 98 °C for IRT-4M and less than 100 °C for IRT-3M, while the catalog description of fuel assemblies determines the onset nucleate boiling factor on the surface of a fuel element by temperature. The boiling factor is assumed to equal to 1.3 when using the Bergles and Rohsenow correlation for IRT-4M and equal to 1.4 when using the Forster Greif correlation for IRT-3M. Since the surface temperature of the FE should not exceed the boiling point of the coolant, the maximum cladding temperature of the most heat-stressed FE in the RC was determined using the Forster Greif formula for FAs of the IRT-3M type

$$tnk = ts + 2.04g^{0.35}\ p^{-0.23}, \qquad (1)$$

where ts – the saturation temperature at pressure p, °C; g – the local heat flux density, kW/m^2; p – the local pressure, bar.

The results of modeling the steady-state of the thermal-hydraulic state, performed using the PLTEMP code [6] for RCs with highly-enriched fuel assemblies, all mixed RCs and RCs with low-enriched fuel assemblies, are presented in Table 5. Uncertainties in calculations in the PLTEMP model – 5 %. Here it was assumed that the inlet water temperature is 45 °C for research nuclear reactor and 10 MW full power for RCs with 18 IRT-3M and 11 MW for all other RC configurations. Table 5 clearly shows that the catalog limits for the maximum cladding temperature and the minimum boiling factor are kept for all cores of the cycles when switching from highly-enriched fuel to low-enriched fuel assemblies.

A detailed simulation of the energy density distribution in the fuel elements was performed using the MCNP4C code, while the fuel elements were modeled as square pipes with right angles without rounding. The energy density was calculated for each side of each fuel pipe and divided into fifteen axial nodes.

Table 5. **Parameters of steady-state thermal-hydraulic states for reactor RC at a 10 MW power for RC with 18 IRT-3M and other mixed RCs (uncertainty does not exceed 5 %)**

RC configuration	IRT-3M fuel assemblies				IRT-4M fuel assemblies			
	Boiling factor	Maximum cladding temperature, ºC	Peak power density, kW/cm^3	FA location	Boiling factor	Maximum cladding temperature, ºC	Peak power density, kW/cm^3	FA location
18 IRT-3M	1.54	94.9	3.14/3.28	3-4/4-4				
16 IRT-3M/4 IRT-4M	1.44	99.4	3.5	3-5	1.59	87.1	2.05/1.59	5-5/4-4
14 IRT-3M/6 IRT-4M	1.47	98.1	3.41/2.61	3-5/3-3	1.64	85.5	1.94/1.53	4-4/5-4
12 IRT-3M/8 IRT-4M	1.52	96.0	3.25/2.55	3-5/3-3	1.59	86.9	2.02/1.60	4-4/4-4
10 IRT-3M/10 IRT-4M	1.57	94.0	3.11/2.47	3-5/3-3	1.36	94.5	2.13/1.92	3-4/4-4
8 IRT-3M/12 IRT-4M	1.67	90.6	2.84/2.40	3-3/3-3	1.41	92.6	2.02/1.98	3-4/5-5
6 IRT-3M/14 IRT-4M	1.71	89.3	2.76/2.34	3-3/3-3	1.44	91.3	1.95/1.94	3-4/4-4
4 IRT-3M/16 IRT-4M	1.99	82.4	2.26/2.03	5-2/4-2	1.56	87.4	2.02/1.92	4-4/3-3
2 IRT-3M/18IRT-4M	2.02	81.8	2.21/1.99	5-7/4-2	1.34	95.1	2.12/2.00	3-4/4-4
20 IRT-4M					1.36	94.5	2.10/2.08	3-4/4-4

Table 6. **Parameters of the steady thermal-hydraulic state of RCs from 24 IRT-4M type FAs**

RC configuration	6-tube IRT-4M				8-tube IRT-4M			
	Boiling factor	Maximum cladding temperature, °C	Peak power density, kW/cm^3	FA location	Boiling factor	Maximum cladding temperature, °C	Peak power density, kW/cm^3	FA location
24 6-tube FAs	1.54	88.4	2.044	3-5				
22 6-tube FAs / 2 8-tube FAs	1.46	90.7	2.175	3-5	1.68	84.1	1.751	4-4

Table 7. **Thermal hydraulic parameters of the RC for the inlet water temperature of 48 °C**

Loading	Maximum heat flux, MW/m^2	Boiling factor	Cladding surface temperature, °C	Maximum allowed power, MW
IRT-3M				
18 IRT-3M	0.91	1.50	97.1	11.0
16 IRT-3M/4 IRT-4M	0.97	1.44	99.8	10.7
14 IRT-3M/6 IRT-4M	0.97	1.44	99.9	10.9
12 IRT-3M/8 IRT-4M	0.95	1.46	99.0	10.8
10 IRT-3M/10 IRT-4M	0.89	1.53	96.2	11.0
8 IRT-3M/12 IRT-4M	0.82	1.63	92.9	11.0
6 IRT-3M/14 IRT-4M	0.80	1.67	91.6	11.0
4 IRT-3M/16 IRT-4M	0.66	1.94	84.8	11.0
2 IRT-3M/18IRT-4M	0.65	1.97	84.1	11.0
IRT-4M				
16 IRT-3M/4 IRT-4M	0.77	1.58	87.8	10.7
14 IRT-3M/6 IRT-4M	0.75	1.6	87.6	10.9
12 IRT-3M/8 IRT-4M	0.78	1.53	89.1	10.8
10 IRT-3M/10 IRT-4M	0.86	1.32	96.9	11.0
8 IRT-3M/12 IRT-4M	0.82	1.36	94.9	11.0
6 IRT-3M/14 IRT-4M	0.79	1.40	93.6	11.0
4 IRT-3M/16 IRT-4M	0.79	1.51	89.9	11.0
2 IRT-3M/18IRT-4M	0.86	1.30	97.4	11.0
20 IRT-4M	0.86	1.32	96.7	11.0
24 IRT-4M	0.63	1.54	88.4	11.0

Table 6 shows the results of the analysis of the steady-state of the thermal-hydraulic state for RCs from 24 IRT-4M type FAs (8- and 6-pipe) at a temperature of 45 °C.

We have also determined the thermal power operational limits for higher inlet water temperatures. The results of the analysis of the thermal-hydraulic states of the RC using the PLTEMP code for an inlet water temperature of 48 °C for all RC configurations are presented in Table 7. Here are the maximum allowable operating power values to keep the catalog recommendations of the fuel assemblies used.

The cladding temperature and the limits of the boiling factor on the surface of the FE with respect to temperature are satisfied for all RCs.

4. Conclusion

An analysis of the distribution of the thermal power of nuclear fuel in the reactor core of the WWR-SM research reactor showed that even with a conservative approach, permissible operating modes are not exceeded. During the operation of the three main circulation pumps, which provide the coolant flow through the reactor core at the level of 1250 m^3/h, heat exchange crises do not occur in the most energy-stressed fuel assemblies, namely, the temperature of the fuel elements cladding and coolant remains below the permissible limits.

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ТЕПЛОГІДРАВЛІЧНІ РОЗРАХУНКИ ДОСЛІДНОГО РЕАКТОРА ВВР-СМ

Наведено розрахунки розподілу теплової потужності в активній зоні (АЗ) дослідницького ядерного реактора як для всіх тепловиділяючих збірок (ТВЗ), завантажених у зону, так і кожного елемента окремої ТВЗ. Ці розрахунки проводилися для конфігурацій АЗ з різною кількістю ТВЗ – 18, 20 і 24. Детальне моделювання розподілу щільності енерговиділення в твелах було виконано з використанням програми MCNP4C, при цьому твели моделювалися як квадратні труби з прямими кутами без заокруглень. Щільність енерговиділення була обчислена для кожної зі сторін кожної паливної труби і розділена на 15 осьових вузлів. Результати моделювання теплогідравлічного стану виконано з використанням програми PLTEMP для різних АЗ реактора. При розрахунках приймалося, що температура води на вході дорівнює 45 і 48 °С для всіх конфігурацій АЗ, при цьому теплогідравлічні параметри було взято з розрахунку витрати води 1-го контуру через АЗ 1250 м^3/год. Аналіз розподілу теплової потужності ядерного палива в АЗ дослідного реактора ВВР-СМ показав, що навіть при консервативному підході перевищення допустимих режимів роботи не відбувається. При роботі трьох головних циркуляційних насосів, які забезпечують витрату теплоносія через АЗ на рівні 1250 м^3/год, криз теплообміну в найбільш енергонапружених ТВЗ не виникає, а саме температури стінок твелів і теплоносія залишаються нижче допустимих меж.

Ключові слова: активна зона реактора, тепловиділяючі збірки, паливний елемент, тепловий потік, теплова потужність.

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ТЕПЛОГИДРАВЛИЧЕСКИЕ РАСЧЕТЫ ИССЛЕДОВАТЕЛЬСКОГО РЕАКТОРА ВВР-СМ

Приведены расчеты распределения тепловой мощности в активной зоне (АЗ) исследовательского ядерного реактора как для всех тепловыделяющих сборок (ТВС), загруженных в зону, так и каждого элемента отдельной ТВС. Эти расчеты проводились для конфигураций АЗ с различным количеством ТВС - 18, 20 и 24. Детальное моделирование распределения плотности энерговыделения в твэлах было выполнено с использованием программы MCNP4C, при этом твэлы моделировались как квадратные трубы с прямыми углами без закруглений. Плотность энерговыделения была вычислена для каждой из сторон каждой топливной трубы и разделена на 15 осевых узлов. Результаты моделирования теплогидравлического состояния выполнены с использованием программы PLTEMP для различных АЗ реактора. При расчетах принималось, что температура воды на входе равна 45 и 48 °С для всех конфигураций АЗ, при этом теплогидравлические параметры были взяты из расчета расхода воды 1-го контура через АЗ 1250 м^3/ч. Анализ распределения тепловой мощности ядерного топлива в АЗ исследовательского реактора ВВР-СМ показал, что даже при консервативном подходе превышения допустимых режимов работы не происходит. При работе трех главных циркуляционных насосов, которые обеспечивают расход теплоносителя через АЗ на уровне 1250 м^3/ч, кризисов теплообмена в самых энергонапряженных ТВС не возникает, а именно температуры стенок твэлов и теплоносителя остаются ниже допустимых пределов.

Ключевые слова: активная зона реактора, тепловыделяющие сборки, топливный элемент, тепловой поток, тепловая мощность.

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