THERMAL HYDRAULIC CALCULATION OF THE IRT-200 REACTOR WITH LEU FUEL IRT-4M, SOFIA

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ABSTRACT

Neutronic calculations of the **IRT-200** research reactor with fuel assemblies (FA) of the **IRT-4M** type, containing low enriched uranium **(19.75 %),** performed by the Institute for Nuclear Research and Nuclear Energy (**INRNE**) jointly with the **RERTR** Program at Argonne National Laboratory (**ANL**), confirmed justness of selection of its initial core configuration. On the base of neutronic calculation results thermal hydraulic calculations were done by **PLTEMP** code. Three possible operational regimes have been considered and for each one of them the margin coefficient of the water onset of nucleate boiling (**ONB**) on the fuel element surface for the maximum power density fuel assembly has been determined. The calculations have been carried out for the core water inlet temperature of **45 ^{0}C**. The thermal hydraulic calculations demonstrated satisfaction of thermal hydraulic safety margins requirements even at **1 MW** power level.

INTRODUCTION

Selection of the initial core configuration of the **IRT-200** research reactor was carried out in close collaboration with scientists from RRC "Kurchatov Institute. The initial configuration for the **LEU** core was selected on the base of calculation results comparison for neutron flux value at the experimental channels and initial core excess reactivity value important for safety ensuring **[1]**. Calculation results for the control rods worth demonstrates that shutdown margins safety requirements are satisfied for the selected initial core configuration. The results of burn-up calculation indicated that the selected core would last about four years for continuous operation. **FA** power levels in the initial core configuration of the **IRT-200** reactor were calculated using the diffusion theory by the **REBUS** code. All control rods were fully withdrawn in the **REBUS** calculation model **[1]**.

More detailed and reliable data about **FA** power values were obtained by using **MCNP** calculation **[2]**. **MCNP** calculations were carried out for the model with shim rods located in positions providing critical state of the core. The results of the **MCNP** calculation were used for thermal hydraulic calculation of the initial core of the **IRT-200** reactor.

FUEL ASSEMBLIES AND THE IRT-200 REACTOR INITIAL CORE CONFIGURATION

Table 1. The main parameters of the **IRT-4M** type **FA**

Parameter	Value	Parameter	Value
Fuel element thickness (mm)	1,6	Clad material	alloy CAB-1
Meat material	UO_2-Al	Clad thickness (mm)	0.45
Meat length (mm)	600	Heat exchange surface of 8-tube FA (m^2)	1.58
Meat thickness (mm)	0.7	Heat exchange surface of 6-tube FA (m^2)	1.38
Uranium enrichment (%)	19.75	The ^{235}U mass (nominal) in 8-tube FA (g)	300
Uranium density in meat (g/cm^3)	3.0	The ^{235}U mass (nominal) in 6-tube FA (g)	265

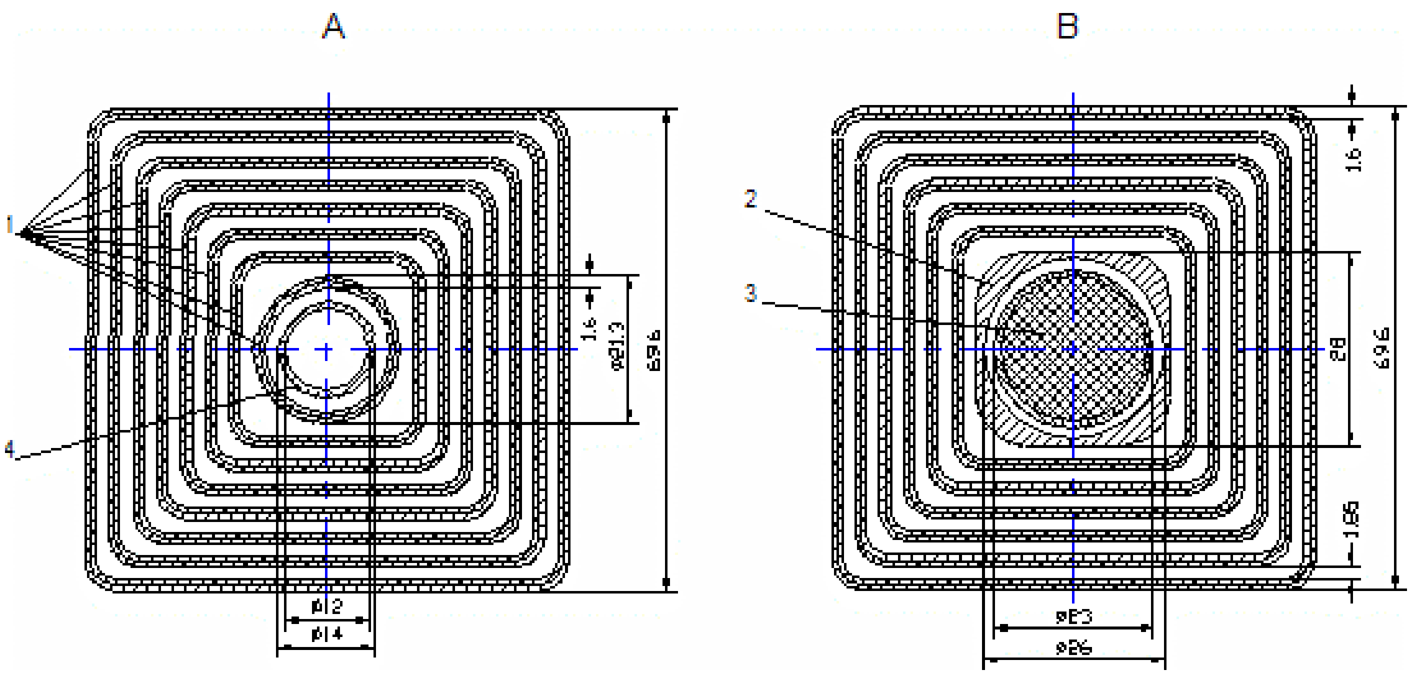
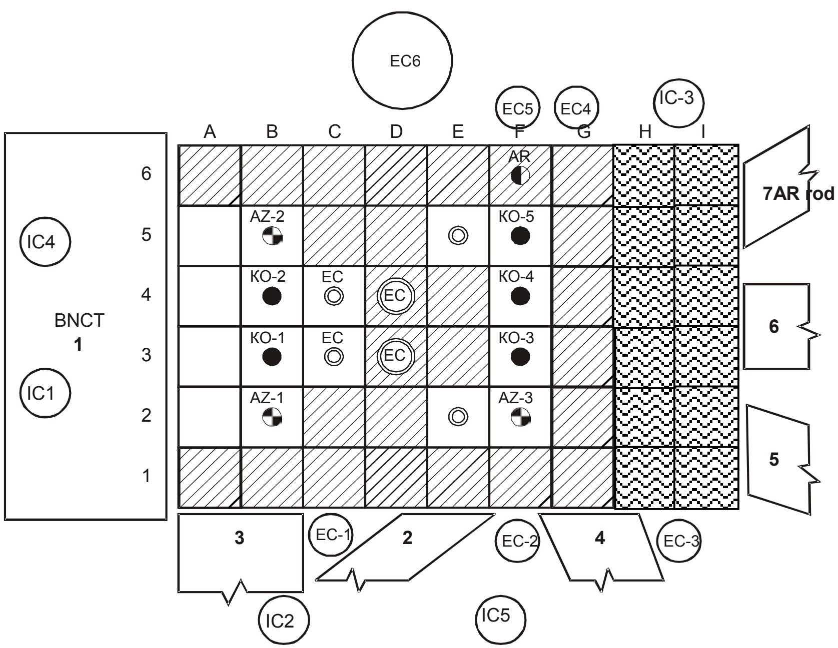


Figure 1. The cross sections of the 8-tube (A) and 6-tube (B) IRT-4M type FA
1 –fuel elements, 2 –channel, 3 – control rod, 4 – displacer tube

Table 2. Geometrical characteristics of the fuel elements of IRT-4M type FA

FE No	Dimension (square side), mm		Radii of curvature of corners, mm	
	Outer	Inner	Outer	Inner
1	69,6	66,4	9,3	7,7
2	62,7	59,5	8,5	6,9
3	55,8	52,6	7,7	6,1
4	48,9	45,7	6,9	5,3
5	42,0	38,8	6,1	4,5
6	35,1	31,9	5,3	3,7
7	28,2	25,0	4,5	2,9
8	21,3	18,1	-	-

Figure 2. Initial core configuration of the IRT-200 reactor



— 6-tube IRT-4M type FA

— 8-tube IRT-4M type FA

— 6-tube FA with shim rod

— 6-tube FA with safety rod

— beryllium block

— beryllium block with hole Ø48 mm

— beryllium block with AR rod

— water displacer with air cavity

THERMAL HYDRAULIC CALCULATIONS

Reactor thermal hydraulic parameters have been determined by means of the **PLTEMP/ANL V2.0 CODE,** developed in the Argonne National Laboratory, Chicago, USA, jointly with the Kyoto University, Japan [3].

The mathematical model permits to consider all factors, having influence on the thermal hydraulic parameters of reactor core and to calculate all values, characterizing the reactor core cooling process. Division of power in the reactor core volume into 15 sectors along the axial axis is set in the model. The calculations have been carried out not only for each fuel element, but also for the separate fuel assemblies. Temperatures of both sides of tube surfaces have been determined. Calculations have been carried out for reactor operation regimes at power level of **200, 500 и 1000 kW**.

Non-admission of water boiling at the fuel elements surfaces has been accepted as a safety margin criteria in the reactor operation. The calculated value of the onset boiling temperature at the fuel element surface ($\mathbf{t_{ob}}$) is a function of the local pressure of water inside the assembly channel, the water saturation temperature and the local thermal flux density.

The $\mathbf{t_{ob}}$ value is determined according to the Forster and Greif correlation:

$$\mathbf{t_{ob} = t_s + 2.04\ q\ 0.35\ p\ -0.23}$$

where:

$\mathbf{t_s}$ – saturation temperature at pressure **p** , ^{0}**C**

q – local heat flux, $\mathbf{kW/m^2}$

p – coolant local pressure, **bar**

The margin coefficient of the water onset boiling on the surface of the maximum power density fuel element (**k**) is determined by:

$\mathbf{k = (t_{ob} – t_{in})/ (t_{max} – t_{in})}$ **,**

where $\mathbf{t_{in}}$ **–** water temperature at core inlet.

- In the selection of safe thermal reactor regime (admissible power level) the regime of the maximum power density fuel assembly in the reactor core cell E2 has been analyzed.

- The minimum permissible margin coefficient of the onset boiling on the fuel element surface (ONB), is accepted equal to **1.45**, the water temperature at the reactor core inlet is **45 ^{0}C**.

- Each fuel assembly power as part of the total reactor power, as well as the volume power peaking factors according to the results from neutron calculations of the reactor initial core configuration (see Figure 2) are presented in Table 3 [2]. Data for the maximum power density fuel assembly in E2 cell are presented in Table 4 and Table 5.

Table 3. FA power in the reactor initial core and volume power peaking factors (kv)

FA positions	FA power [%]	k_v
F2	6.91	1.60
F3	7.87	2.03
F4	7.78	2.04
F5	6.46	1.63
E2	9.76	2.28
E5	9.53	2.32
C3	8.99	2.29
C4	8.99	2.29
B2	5.67	2.27
B3	3.86	2.13
B4	3.87	2.13
B5	5.77	2.27
A2	3.68	2.07
A3	3.56	1.65
A4	3.57	1.65
A5	3.76	2.06

Table 4. Distribution of power density along axial axes of FA in E2 cell

Layer № (from top to bottom)	Power density, [%]
1	3.69
2	4.39
3	5.66
4	6.59
5	7.43
6	8.06
7	8.49
8	8.62
9	8.52
10	8.23
11	7.75
12	7.10
13	6.17
14	4.97
15	4.35

Table 5. Power of fuel elements (FE) of the fuel assembly in E2 cell

Fuel element	1	2	3	4	5	6
FE power, %	25.4	20.3	16.8	14.2	12.3	10.9

The cooling of the reactor core is realized by a water flowing through the reactor core from top downwards. The necessary flow through the reactor core is ensured by means of an ejector and circulation pump.

In according with results of hydraulic calculation of first cooling circuit in case of operation of one pump(84 m3/h) the water flow rate through reactor core and reflector is 226 m3/h.

The water pressure drop in the reactor core by this - 3.55 kPa.

Table 6. Water flow rate through the reactor and its element at water pressure drop in the reactor core of 3.55 kPa

Element name	Flow rate through 1 element [m^3/h]	Number of elements	Wate rflow rate[m^3/h]
8-tube FA	9.80	4	39.12
6-tube FA	8.66	12	104.64
Beryllium block without hole and hole of 48 mm and cap	2.50	25	62.50
Beryllium block with AR	3.87	1	3.87
Displacer block with air cavity	1.85	3	5.55
Border between displacer blocks and vessel	10.43	-	10.43
Total flow	-	-	226.11

Table 7. Water velocity in IRT-4M type FA (m/s) at water pressure drop in the reactor core of 3.55 kPa

Water channel (gap)	8-tube FA	6-tube FA
1. between FA	0.91	0.91
2	0.84	0.84
3	0.93	0.93
4	0.96	0.96
5	0.92	0.92
6	1.07	1.07
7	1.05	1.05
8	0.97	-
9	1.05	-
Displacer tube	0.20	-
Between channel and control rod	-	0.48
Water flow rate through cell with FA [m^3/h]	9.78	8.72

Table 8. Thermal hydraulic calculation results for the IRT-200 reactor initial core configuration

Parameter	Reactor power, kW		
	200	**500**	**1000**
Maximum calculated power density in the FE meat, W/cm^3	86.6	216.4	432.9
Maximum temperature of fuel element meat, ^{0}C	50.57	58.46	70.88
Maximum heat flux (outer/inner surface of fuel element), kW/м^2	30.86/26.06	77.12/65.17	154.45/130.14
Maximum temperature (outer/inner surface of fuel element), ^{0}C	50.44/50.47	58.14/58.20	70.23/70.36
Water onset boiling temperature at fuel element surface (according to Forster-Greif), ^{0}C	119.81	122.04	124.33
Onset boiling margin coefficient at fuel element surface	13.741	5.863	3.144
Safety margin of flux instability	20.370	8.076	3.943
Critical heat flux according to Mirshak, kW/m^2	2238	2239	2239
Temperature at reactor core outlet, ^{0}C	45.91	47.263	49.52

CONCLUSION

Results of the thermal hydraulic calculations of the initial core configuration of reactor IRT-200 show that reactor operation at power of 1000 kW is safety in case of operation of one pump of first cooling circuit.

REFERENCES

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3. A.P.Olson. “A users guide to the PLTEMP/ANL V2.1 CODE”, October 13, 2003