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Effect of U-9Mo/Al Fuel Densities on Neutronic and Steady State Thermal Hydraulic Parameters of MTR Type Research Reactor

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Abstract

The objectives of this research work are to carry out a detailed neutronic and steady state thermal hydraulics analysis for a MTR research reactor fuelled with the low enrichment U-9Mo/Al dispersion fuels of various uranium densities. The high density uranium fuel will increase the cycle length of the reactor operation and the heat flux in the reactor core. The increasing heat flux at the fuel will causing increase the temperature of the fuel and cladding so that the coolant velocity has to be increased. However, the coolant velocity in the fuel element has a limit value due to the thermal hydraulic stability considerations in the core. Therefore, the neutronic and the steady state thermal hydraulic analysis are important in the design and operation of nuclear reactor safety. The calculations were performed using WIMS-D5 and MTRDYN codes. The WIMS-D5 code used for generating the group constants of all core materials as well as the neutronic and steady state thermal hydraulic parameters were determined by using the MTRDYN code. The calculation results showed that the excess reactivity increases as the uranium density increases since the mass of fuel in the reactor core is increased. Using the critical velocity concept, the maximum cool ant velocity at fuel channel is 11.497 m/s. The maximum temperatures of the cool ant, cladding and fuel meat with the uranium density of 3,66 g/cc are 70.85°C, 150.79°C and 153.24°C, respectively. The maximum temperatures are fulfilled the design limit so reactor has a safe operation at the nominal power.

Keywords: U-9Mo/Al dispersion fuel, MTR research reactor, critical velocity, high uranium density, MTRDYN

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1. Introduction

Conceptual design of innovative research reactor (RRI reactor) has been completed from aspect of neutronic and kinetic parameters [1-4]. Previous design reactor power is 20 MW (thermal), because the flux is too small so the reactor power increased to 50 MW (thermal). At the power of 50 MW with high uranium density, core power density becomes high so the active height of fuel is increased from 60 cm to 70 cm while the other dimensions were remain fixed. RRI is a tank-in-pool type research reactor, a material testing reactor (MTR) with plate type of fuel elements and has a core grid position of neutron trap. The light water is used as the coolant and heavy water as reflector. The maximum thermal neutron flux at the reflector will be not less than $5{\cdot}10^{14}$n/cm^{2}s.

RRI reactor is designed using compact core so that the heat transfer area and the amount of fuel are small, but heat flux at the fuel plate are high. Heat flux must be compensated by setting the coolant flow rate so that the reactor continued to operate safely. The flow rate of the cooling water through the channel of fuel plate is a very important parameter because instability and vibration can occur in the fuel [5]. Design criteria commonly used in determining the maximum flow rate of coolant flow is 2/3 of the critical velocity which is determined by the IAEA [6]. In this paper, the mass flow rate in the range of 750 – 900 kg/s will be analyzed.

The enhancement of cycle length of RRI reactor operation using high uranium density with the low enrichment uranium (LEU) affects neutronic and thermal hudraulic parameters. Therefore, the optimum parameter has to be obtained by varying the uranium densities of U-9Mo/Al fuel. In this work, the uranium density of the U-9Mo/Al fuel is varied for 3.66, 5.34 and 6.52 gU/cc.

The calculated neutronic parameters are excess reactivity, the maximum thermal flux, control rods worth, power peaking factor and power density. Those parameters are used for analysis of the steady state thermal hydraulic of RRI reactor. In the steady state thermal hydraulic, the temperatures of fuel, cladding and coolant are calculated. The saturation temperature of water, melting temperature of cladding as well as fuel meat are used as a limit value in optimize the parameters [8].

The WIMSD-5B code [9] is used for calculating group constants for different regions at the reactor core. Using the data from WIMSD-5B it will calculated the neutronic and steady state thermal hydraulic parameters using MTR-DYN code [9].

2. Methodology

RRI reactor has 25 core grid positions with a 5 × 5 core configuration. As seen in Figure 1, there are 16 standard fuel elements, 4 control rods and a central neutron flux trap position. A standard fuel contains 21 fuel plates while a follower fuel contains 15 fuel plates. The control rods are of follower type using material of AgInCd with composition of 80% Ag, 15% In, and 5% Cd. The shape of standard fuel, follower fuel and control rod are shown in Figures 2-4. In the design, the reactor might be added two safety rods if the shutdown margin of one stuck rod is less than 0.5% $\Delta k/k$. Previous research work found that the core using fuels with uranium densities of 5.34 gU/cc and 6.52 gU/cc must be added by the safety rods.

The cell calculations are carried out by using WIMS-D5 code with 69 neutron energy group of ENDF/B-VI library The four energy groups formed are fast neutron region 0,821 MeV <E ≤ 10 MeV, slowing down region 5.53 keV <E ≤ 0.821, 0.625 resonance region <E ≤ 5.53 keV and thermal0.625 <E ≤ 0.0 eV. The WIMS-D5 will generate macroscopic absorption cross section (Σa), the ν-fission cross section ($\nu\Sigma f$), the diffusion coefficient (D), the scattering matrix (Σs, g→g) and the fission spectrum for all groups of core materials, which are used as input data to MTR-DYN code.

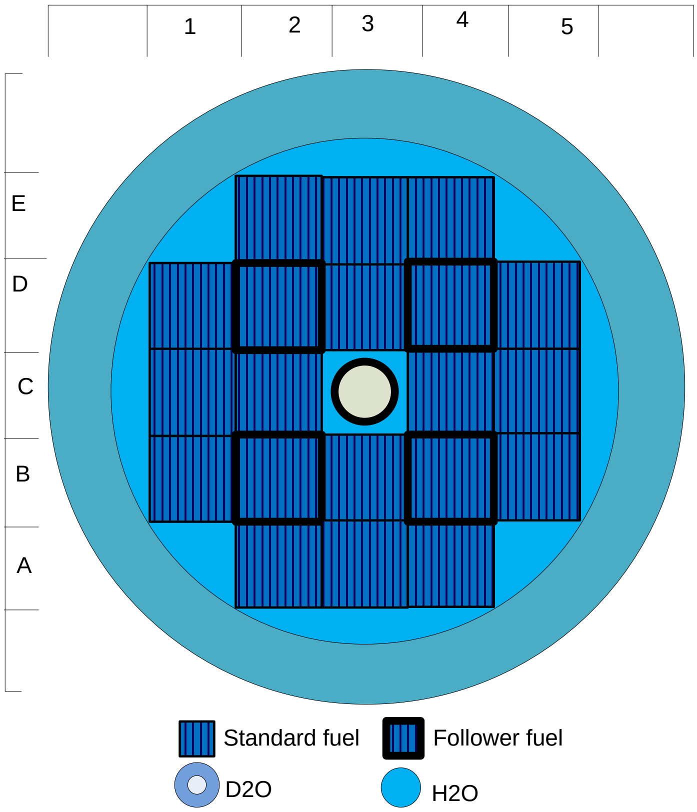

Figure 1: Core configurations of the RRI reactor [1]

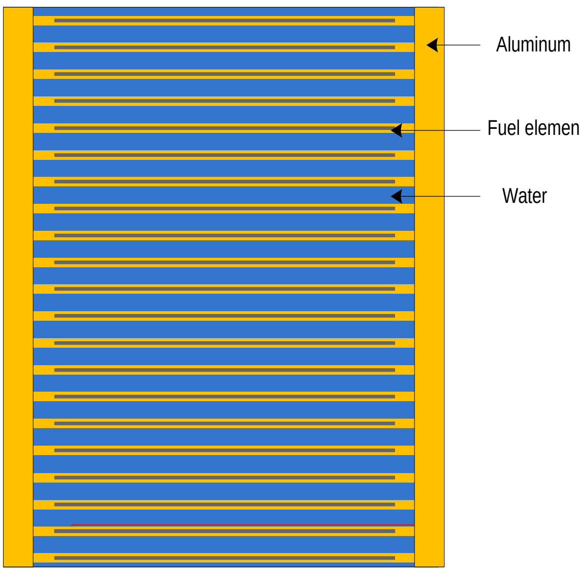

Figure 2: Standard fuel element [1]

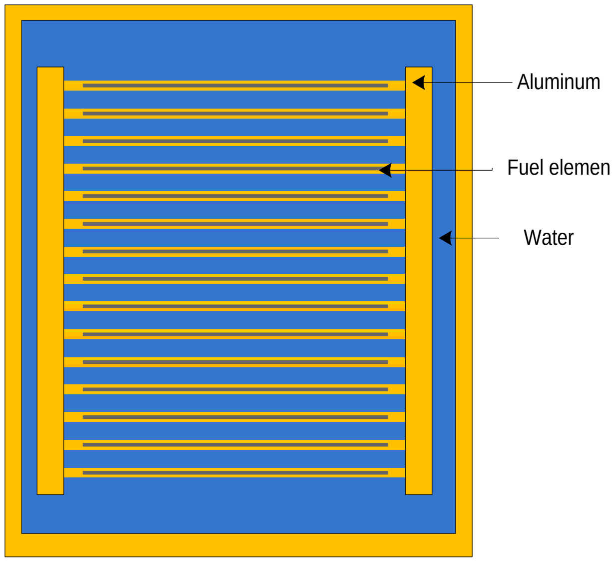

Figure 3: Fuel follower element [1]

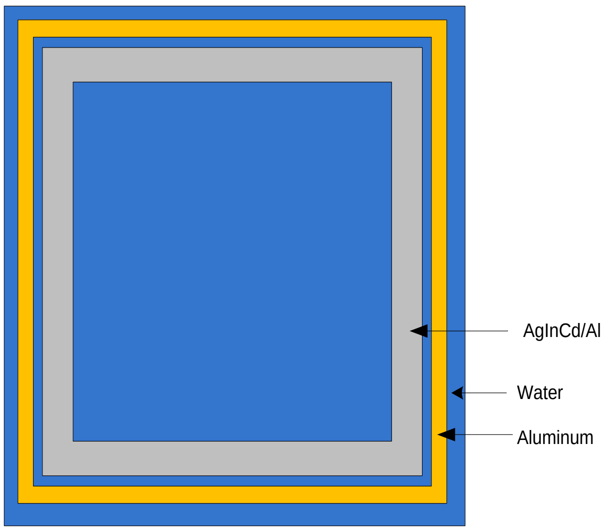

Figure 4: Control rod element

The MTR-DYN code is a coupled neutronic (N) and thermal-hydraulic (T/H) code for the MTR research reactor type. This code is developed by 3-dimensional multigroup neutron diffusion method by finite difference method. Flowchart of N and steady state T/H calculations are shown in the Figure 5.

The neutronic parameters are determined by using adiabatic models, time dependent multi group neutron diffusion problem is solved by flux factorization approach, the spatial, time-dependent neutron flux is split into time-dependent amplitude and shape function. Heat conduction equation in the fuel rods are discretized in space and time using the finite-difference method. Heat conduction is considered only in the radial direction. Fluid dynamics is modeled in a single-phase flow conditions. Mass flow rate in each cooling channel is to be determined.

Analysis of the neutronic and steady state thermal hydraulic parameter are performed at the thermal power of 50 MW. Core cooled under forced convection with an in let temperature is 44.5 °C. The coolant pressure is 10 atm at under surface of the water.

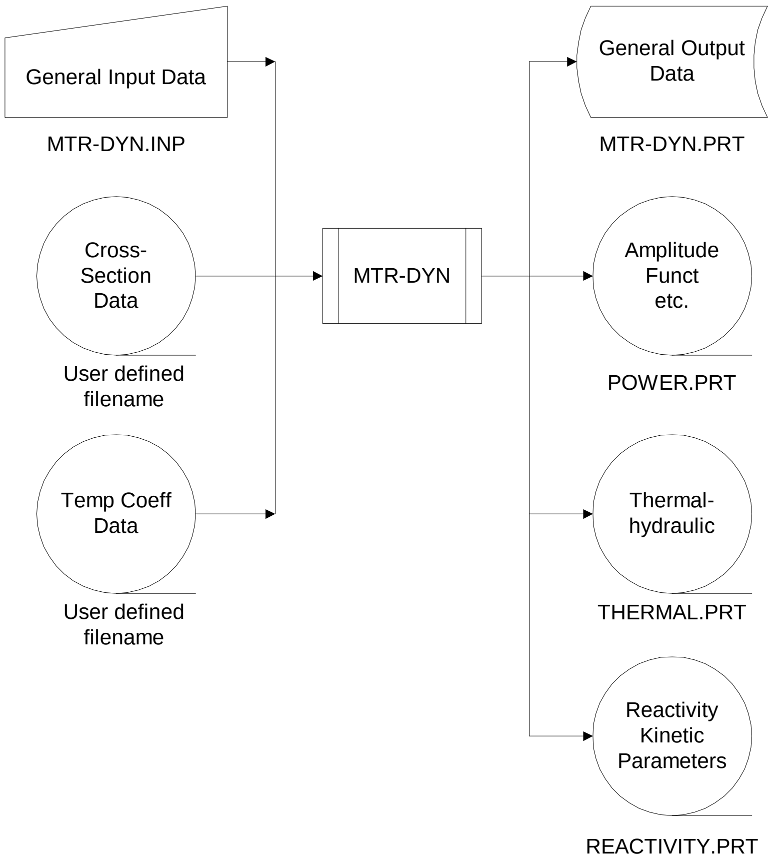

Figure 5: MTR-DYN code input/output file structure

3. Results and Discussions

The calculated neutronic parameters, such as excess reactivity, control rods worth, power peaking factor, the thermal neutron flux and power density are shown in Table 1. The excess reactivity depends on the fuel uranium density, so it is needed to selectan optimum fuel uranium density in order to operate the reactor at 50 MW. Axial power peaking factor is dependent on the neutron flux distribution and effected by control rod position during operation reactor. The maximum radial and axial power peaking factor is less than the limit values of 1.4 and 1.8, respectively.

Heat flux or power density in the core depends on the reactor power operation, the burn up and the number of fuel elements in the core. Heat flux is not uniform and depends on the axial and radial position in the core reactor. Figure 6 shows axial power peaking factor for hottest channels resulting from the calculation at power of 50 MW.

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No.	Parameters	Uranium density (gU/cc)		
		3.66	5.34	6.52
1.	Excess reactivity (%Δ*k*/ *k*)	12.75	14.01	15.74
2.	Contol rod worth (%Δ*k*/ *k*)	21.69	23.22	21.80
3.	Maximum radialpower peaking factor	1.1530	1.2028	1.2012
4.	Cycle length reactor operation (days)	21	35	48
5.	Maximum axial power peaking factor	1.7435	1.7435	1.7323
6.	Local power peaking factor	2.1397	1.9532	2.1080
7.	Maximum thermal neutron flux (E15 n/cm^{2}s)	1.47	1.46	1.41
8.	Power density (W/cc)	637.26	639.01	639.01

TABLE 1: Neutronic parameters with various uranium density

Heat flux or power density in the core depends on the reactor power operation, the burn up and the number of fuel elements in the core. Heat flux is not uniform and depends on the axial and radial position in the core reactor. Figure 6 shows axial power peaking factor for hottest channels resulting from the calculation at power of 50 MW.

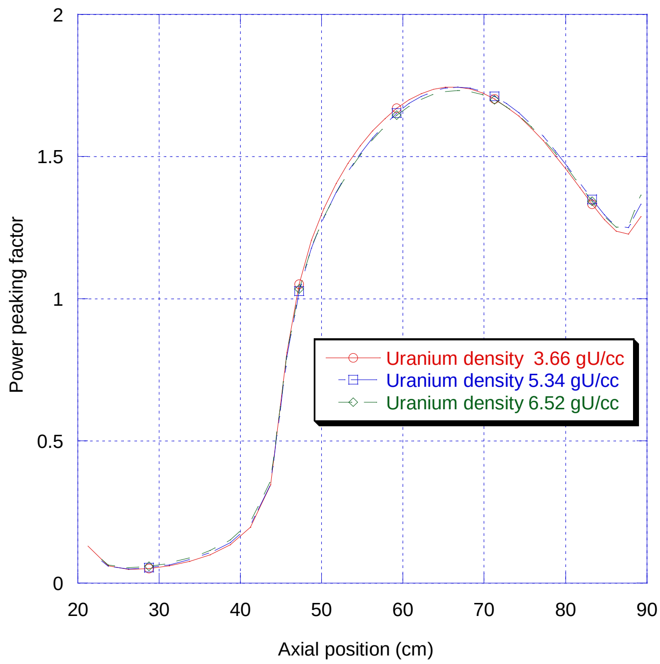

Figure 6: Axial heat flux profile of the core hot channel

The coolant mass flow rate in the RRI reactors is limited by the flow instability phenomenon. Flow instability can be happened in the reactor core with high thermal power and characterized by a flow excursion. When the flow rate and the heat flux are relative high, a small increase in heat flux causes a sudden large decrease in flow rate. For higher uranium density, the thickness of the plate or the channel width is increased so that the reactor is stable at a high flow rate. In addition to reducing the heat flux in the fuel, it can be done increasing the height of fuel so the maximum temperature of fuel and cladding will be reduced. Calculation results of coolant velocity with various mass flows are shown in Table 2. Based on the calculation, obtained the critical velocity through the fuel channel is 17.899 m/s. By using the design criteria where the maximum coolant speed is2/3 of criticality velocity so that maximum mass flow on the core is 800 kg/s and a maximum coolant speed atfuel channel is10.92 m/s.

Parameters	Mass flow rate (kg/s)			
	750	800	850	900
Coolant flow rate (kg/m²s)	6709.65	7156.96	7604.27	8051.58
Coolant velocity (m/s)	10.24	10.92	11.61	12.29

TABLE 2: Coolant velocity with various mass flow rate

Figure 6 shows that the axial hot channel conditions is resulted from the calculations for 50 MW with various fuel uranium densities. From Figure 7 shows the axial hot channel where if the uranium density increases, the temperature of the fuel also increases.

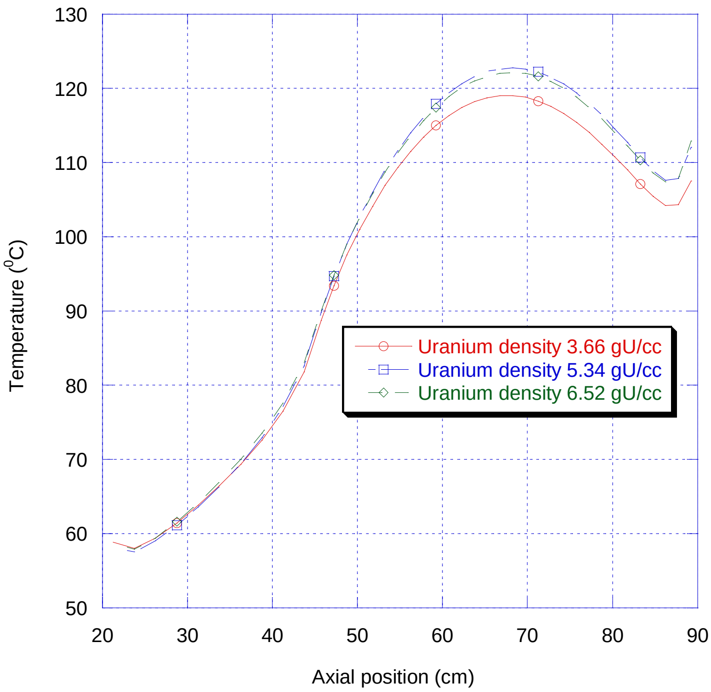

Figure 7: Fuel temperature along the axial position in the hot channel

The temperature distribution in the fuel, cladding and coolant are performed at mass flow rate of 800 kg/s. If the uranium density increasing, it makes temperature of the fuel, cladding and coolant increasing because the thermal conductivity decreases. The temperature of the fuel, coolant and cladding should be limited to maintain the integrity of the fuel. Calculation results at a steady state thermal hydraulic with various uranium densities are shown in Table 3. For the uranium densities of 3.66 gU/cc, 5.34 gU/cc and 6.52 gU/cc, with the mass flow rate of 800kg/s, the fuel and cladding temperature are less than 170 oC so still within safe limits.

Parameters	Uranium density (gU/cc)		
	3.66	5.34	6.52
Coolant temperature at core inlet (°C)	44.5	44.5	44.5
Coolant temperature at core outlet (°C)	59.49	59.49	59.49
Maximum temperature at coolant (°C)	70.85	73.06	73.92
Maximum temperature at cladding (°C)	150.79	161.20	163.70
Maximum temperature at fuel meat (°C)	153.24	163.95	166.52

TABLE 3: Maximum temperature with various uranium density at fuel elements

Neutronic and thermodynamic calculation results indicate that the effect of fuel densityin the design of the MTR type research reactor is very important for the safety of the operation and efficiency of fuel used. The higher density of fuel reactors is more efficient because the longer operating cycle but the power density will be high so that the temperature of the fuel increase. Both of these factors should be taken into account in the design of the research reactor core.

4. Conclusions

Effect of uranium densities in the neutronic and thermal-hydraulic parameters of RRI reactor have been carried out.Based on the calculation of MTR-DYN code, it is clear that all uranium densities of U-9Mo/Al fuel that are surveyed in this research work can be utilized as a candidate fuel for the RRI reactor core with thermal power of 50 MW, since the maximum temperature of cladding is less than 170 °C with the mass flow rate of 800 kg/s. However, as a future work, the transient analysis has to be carried out to obtain an optimum uranium density of U-9Mo/Al fuel for RRI reactor.

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