Dynamic Analysis on the Safety Criteria of the Conceptual Core Design in MTR Type Research Reactor

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ABSTRACT

A research activity with high priority in Batan is designing a new research reactor MTR type with a new fuel. The core is compact core model that consists of 16 fuels and 4 control rods. The high-density uranium fuel will increase the cycle length of the reactor operation but the compact core model will also increase the heat flux in the reactor core. The increasing heat flux at the fuel will cause the temperature of the fuel and cladding increase so that the coolant flow rate could be increased. However, the coolant flow rate in the fuel element has a limit value due to the thermal hydraulic stability in the core. Therefore, the dynamic analysis is important in evaluating design and operation of nuclear reactor safety. The objective of this research work is to carry out a dynamic analysis for a conceptual core of MTR research reactor fuelled with the low enrichment U9Mo-Al dispersion with 390 uranium fuel loading (3.96 gram/cc). The calculations were performed using WIMSD-5B, Batan-2DIFF, Batan-3DIFF, POKDYN, and MTRDYN codes. The WIMSD-5B code was used for generating the group constants of all core materials. The neutronics and kinetics parameter such as keff, PPF, the core reactivity balance, coefficient reactivity were calculated by Batan-2DIFF code and control rod reactivity worth was calculated by Batan-3DIFF code. POKDYN code is used to determine the maximum control rod speed. Steady state thermal hydraulic parameters and dynamic analysis were determined using the MTRDYN code. The calculation results showed that the core reactivity balance, reactivity coefficient and also control rod reactivity worth meet the safety criteria. Using the dynamic analysis, the maximum control rod speed is achieved as 0,0846 cm/s. The maximum temperatures of the coolant, cladding and fuel meat with the uranium density of 3,96 g/cc are 76.01°C, 192.02°C and 196.24°C, respectively. The maximum value of fuel meat temperature for safety limit is 2100C, it means that the maximum temperatures fulfilled the design limit, therefore, the reactor has a safe operation at the nominal power. The dynamic analysis showed that inherently safety can save the operation reactor when insertion of reactivity occurs in the core.

INTRODUCTION

One of research activity with high priority in the Center for Nuclear Reactor Technology and Safety-Batan is to design new research reactor using uranium-molybdenum fuel (U-9Mo/Al). High loading of fissile material in the U-9Mo/Al fuel is expected to increase the operation cycle; hence, higher reactor availability and utilization can be achieved while the fuel cost can be reduced[1].

An early conceptual design of Material Testing Reactor (MTR) type from neutronics aspect has been derived by T. Surbakti., et al.[2] with main characteristics of core configuration as specified as follow. The RRI research reactor has a nominal power of 20 MW, uses a uranium-molybdenum alloy, U-9Mo/Al fuel with geometry adopting from RSG-GAS' fuel. The core configuration in 5 x 5 lattice is made up of 20 fuel elements and 5 irradiation positions and produces thermal neutron flux in the order of 2.87×10^{14} neutron cm^{-2} s^{-1}. This neutron flux value is still lower than one stated as acceptance criteria of MTR type reactor in which the reactor should have a maximum neutron flux in irradiation positions and in the reflector region with high thermal neutron flux at least in the order of 1.0×10^{15} and 0.5×10^{15} neutron cm^{-2} s^{-1}, respectively.

The second conceptual design was proposed by I. Kuntoro. et al.[3] where the grid core is the same but the height of the fuel and power are

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different. To fulfill that acceptance criterion, the core was designed with the fuel elements 70 cm of height and the power level of 50 MW and the others size are the same with the lattices of reactors which previously design. The core configuration is specified by the number and positions of fuel elements and irradiation positions. After finding the optimum core configuration met the acceptance criteria that the value of neutron flux in the center of the core is not less than $1.0x10^{15}$ n/cm^2 s, it is necessary to analyze the dynamic of the core to understand the characteristic of the core from safety feature. However, dynamic analysis has not been done for this MTR conceptual core design. For this purpose, the conceptual core design is considered in a transient condition. The calculation concern for rapid transient initiated by positive reactivity induced accident during a control rod withdrawal at the power of 1 MW. Many researches have performed neutronic and thermal-hydraulic calculations to understanding the safety features for research reactor using PARET, RELAP, COOLOD-N2 and EUREKA codes.[4,5,6,7,8] but for this research using MTR-DYN code.

This research is a subsequent evaluation design for conceptual core design of MTR research reactor by focusing on identifying safety limit and margin. In this research is discussed the dynamic analysis for conceptual core design of MTR research reactor used MTR-DYN code[9]. Before the analysis is done, it was started from design calculations and carried out by means of WIMSD-5B[10] for cross-section generation as an input to Batan-3DIFF diffusion code for core calculation[11] to determine the integral and differential control rod worth. The macroscopic cross-section was also needed to calculate the fuel and moderator reactivity coefficient and delayed neutron fraction. To determining the maximum speed of control, it is used a period-reactivity relation. The MTR-DYN code is used to determine the thermal-hydraulic parameter to analyze core dynamic as a rapid transient initiated by positive reactivity in the core.

The WIMSD-5B code is used for calculating group constants for different materials in the MTR type research reactor core. Batan-3DIFF and MTR-DYN codes are used for core calculation. These codes which used for the neutronics and steady state thermal hydraulic and dynamic parameters which had been verified using RSG-GAS core[12].

METHODOLOGY

The methodology to achieve the research objective can be divided into two types of calculations. This section elaborates these two calculation types.

A. Cell calculation

To solve the neutron kinetics equations, the macroscopic cross section library for various materials in the core is set-up. For this purpose, the WIMSD-5B lattice code is used. In practice, the cells which may correspond to any region of the core (fueled and non-fueled) are identified. When defining the unit cell dimensions, the principle of conservation of volume ratio of the different material in the fuel assembly is considered. The fuel assembly in the core configuration is showed in Figure 1. The fuel cell dimensions are calculated taking into account the fuel meat conservation criteria. The unit cell for fuel element (FE) is showed in Figure 2. An extra region is added accounting for the remaining water and aluminum in the same proportions as in the physical fuel element , this region includes the aluminum in the plates beyond the width of the meat and the aluminum side plates, the water beyond the width of the meat, and the water channels surrounding the fuel element. In the particular case of control element, the super-cell option of WIMSD-5B is used. The representative cell is modeled with 15 regions. The control rod and box absorber are showed in Figure 3 and 4. The macroscopic cross-section data were generated by WIMSD-5B code as a function of burn-up and fuel and moderator temperatures. Different burn-up values, ranging from 0% to 90 % were considered in order to generate all conditions, beginning of cycle and end of cycle cores.

Fuel and moderator temperatures were chosen in order to cover a large set of core condition for normal and transient conditions. The macroscopic cross-section also was performed to determine the average speed of neutrons. The cross-section generation is done actually in 4 neutron energy group, but in the determination of the average speed of neutrons is carried out in 69 group neutron energy. The average speed of neutrons in 4 groups of power calculated by weighting the speed of neutrons in 69 energy groups with a average cell flux. The speed of neutron was used to calculate total delayed neutron fraction.

Steps of cell calculation are as follows:[13]

a.The macroscopic x-section as a function of burn-up is generated under ambient conditions (20^{o}C).

b.The macroscopic x-section for fuel elemen is generated as a function of temperatures (50^{o}C, 100^{o}C, 150^{o}C, 200^{o}C).

c.The macroscopic x-section for moderator is generated as a function of temperatures (35°C, 45°C, 60°C, 100°C). In this step the moderator density unchanged

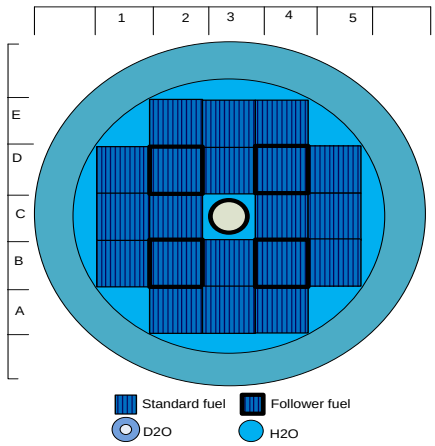

Fig. 1. Core configurations of the MTR type research reactor [2]

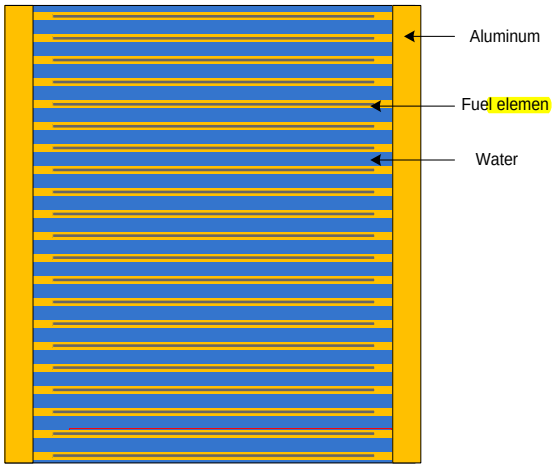

Fig. 2. Standard fuel element [2]

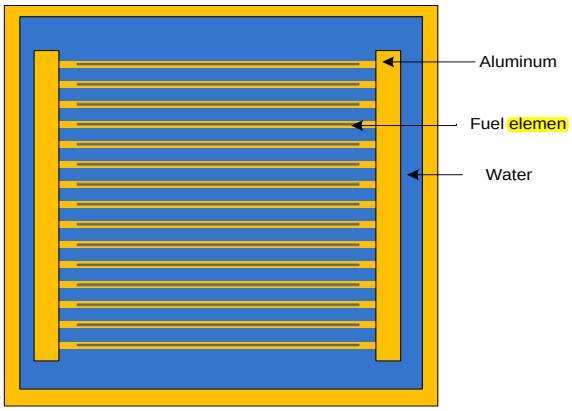

Fig. 3. Fuel follower element[2]

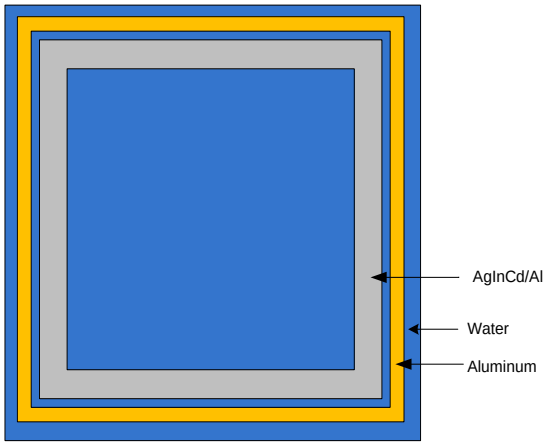

Fig.4. Control rod element [2]

B.Core calculation

Core calculation was done by using Batan-2DIFF and Batan-3DIFF codes for coefficient reactivity, integral and differential control rod worth. Batan-2DIFF was also used to calculate total delayed neutron fraction which is needed as an input for MTR-DYN code for dynamic analysis. 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 using 3-dimensional multigroup neutron diffusion with finite difference method. Flowchart of N and steady state T/H calculations are shown in Figure 5. All calculations were carried out by the adiabatic method (AM) [14].

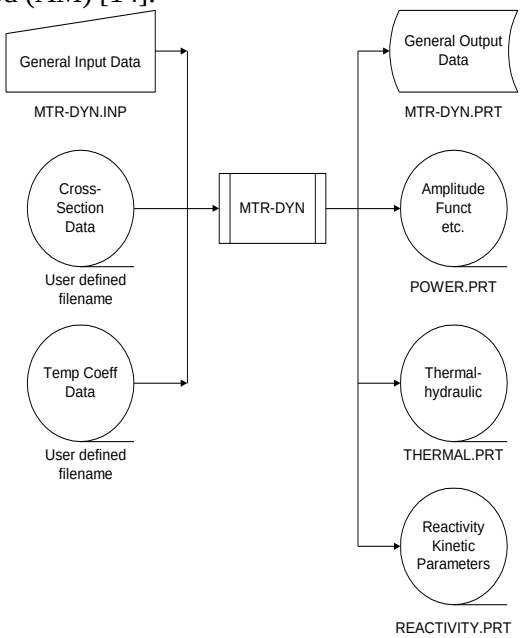

Fig. 5. MTR-DYN code input/output file structure [14]

RESULTS AND DISCUSSION

Neutronic parameter of MTR type research reactor are showed in Table 1. The result showed that the radial power peaking factor (PPF) value <1.4, as well as the maximum neutron flux in the center of the core already met the acceptance criteria > 1.0 x 10^{15} n / cm^2s. The value of the PPF on the equilibrium core is far below the safety limit. The PPF value will change due to the pattern and materials exchange in the core. It can be said that the PPF is affected by fuel management. Therefore the primary factor in determining the maximum value of PPF is a burn-up distribution in the fuels and around the control rod positions. Radial PPF greatest value is 1.24 with the largest fraction and is close to the control rod (D-2) position at equilibrium core. The maximum radial PPF value is less than the value of safety limit 1.4.

Table 1. Neutronic parameters of MTR research reactor

No.	Parameters	Uranium density 3.96 [gU/cm^3]
1.	Excess reactivity[%$\Delta k/k$]	11.32
2.	Contol rod worth [%$\Delta k/k$]	-25.60
3.	Maximum radial power peaking factor	1.23
4.	Cycle length reactor operation (days)	15
5.	Maximum axial power peaking factor	2.1
6.	Maximum thermal neutron flux in the irradiation position (E15 n/cm^2s)	0.579
7.	Maximum thermal neutron flux in the center (E15 n/cm^2s)	1.08
8.	Power density (W/cm^3)	635

a. Reactor kinetic parameter

Reactor kinetic parameter such as delayed neutron fraction and prompt neutron lifetime are needed to determine for dynamic analysis. Table 2 showed that calculation result of delayed neutron fraction and prompt neutron lifetime for research reactor MTR type using U9Mo-Al fuel with 3.96 g/cc of density. The bigger delayed neutron fraction is the easier for the core to be controlled when the transient occurs in the reactor. Compare the value with the RSG-GAS core is 7.16 E-03. Reactor kinetics parameters required in this study was effective delayed neutron fraction (β_k) and delayed neutron decay constant. Acceptance criteria for a kinetic parameter value such as effective delayed neutron fraction (β_{eff}) is $\beta_{eff} = \gamma\ \beta$, ($\gamma$ = 1.05 to 1.25) and the value β is 0.0064 for uranium fuel[15]. So the value of the effective delayed neutron fraction total should be in the range of 0.00672 to 0.00840. Beyond this value, the kinetic parameters β_{eff} will be rejected.

Table 2. delayed neutron fraction for density of 3.96 gram/cc

Group	Delayed neutron fraction (β_k)	Decay constant of delayed neutron (λ_k) s^{-1}
1	$2.61123.10^{-04}$	$1.29065.10^{-02}$
2	$1.53233.10^{-03}$	$3.11613.10^{-02}$
3	$1.69042.10^{-03}$	$1.34027.10^{-01}$
4	$2.65407.10^{-03}$	$3.31390.10^{-01}$
5	$7.56494.10^{-04}$	$1.46117.10^{-00}$
6	$2.95836.10^{-04}$	$3.81104.10^{+00}$
Total delayed neutron fraction : $7.19027.10^{-03}$		
Average decay constant : $7.84863.10^{-02}$ s^{-1}		
Prompt neutron lifetime : $55.49.10^{-6}$ s		

It is important to know the sign and magnitude of the various reactivity coefficients because these coefficients suggest the consequences of sudden changes in the operating parameters. A positive value for a reactivity coefficient means that a positive change in that parameter will increase reactivity and tend to increase power. A negative value for a reactivity coefficient means that a positive change in that parameter will decrease reactivity and tend to decrease power. In both cases, a larger absolute value of the reactivity coefficient becomes greater sensitivity to changes in that parameter. Batan-2DIFF code calculates accurately the fuel and moderator temperature coefficient of reactivity. The calculation result in Table 3 showed that the moderator temperature reactivity coefficient is negative. It meets the acceptance criteria as the moderator temperature is changed around the (35^oC-80^oC). When moderator temperature is changed, the fuel temperature is reminded constantly. Table 4 showed that fuel temperature is also negative when the fuel temperature was changed around (50^oC-200^oC). Both parameters (moderator and fuel coefficient of reactivity) are significant for feedback reactivity for dynamic analysis. The fuel temperature coefficient is a very prompt effect because fuel temperature changes quickly in a change in power. In an accident where the power increases, a negative fuel temperature reactivity coefficient provides a prompt negative feedback, which tends to bring power down.

Table 3. Moderator temperature reactivity coefficient

Moderator temperature [°C]	k-eff [%Δk/k]	Core reactivity [ρ] [%Δk/k]	Δρ [%Δk/k]	Moderator reactivity coefficient [(%Δk/k)/°C]
35	1.035108	3.39188	0	-
45	1.034265	3.31313	- 0.07875	- 7.876E-03
60	1.032981	3.19296	- 0.12019	- 8.013E-03
80	1.031249	3.01865	- 0.17431	- 8.751E-03

Table 4. Fuel temperature reactivity coefficient

Fuel Temperature [°C]	k-eff [%Δk/k]	Core reactivity [ρ] [%Δk/k]	Δρ [%Δk/k]	Fuel reactivity Coefficient [(%Δk/k)/°C]
50	1.044953	4.30201	0	-
100	1.043889	4.20448	-0.09753	-1.9506E-03
150	1.042898	4.11343	-0.09105	-1.8210E-03
200	1.042329	4.06118	-0.05225	-1.0450E-03

Figure 6 shows axial power peaking factor for hottest channels resulting from the calculation as the reactor operates at power of 50 MW. The maximum value of axial PPF is 1.9 as the position of control rod 25 cm of height from the core.

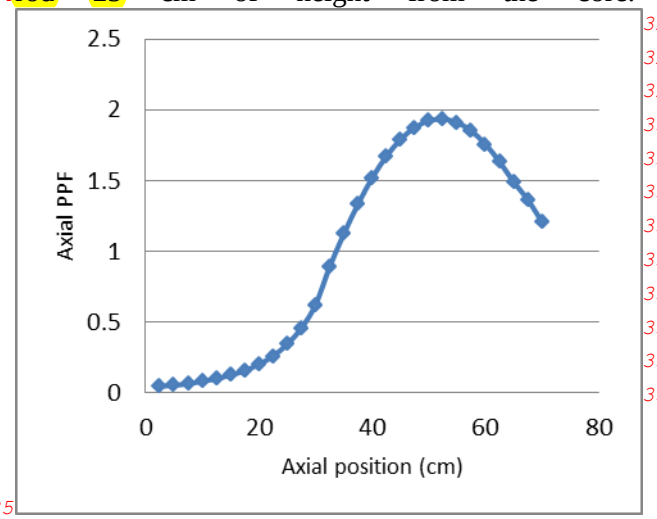

Fig. 6. Maximum axial power densities in D-2 core grid position

C. Integral and differential control rod worth

Reduction or increment of thermal utilization factor (f) is an important property of a control rod and depends on keff, whether the rod is inserted or withdrawn from the core. The change in Keff results in a change in the reactivity of the core. The worth of a control rod directly relates to its effect on reactivity and it has usually the same unit of that. The effectiveness, or worth, of a control rod, depends on mostly of the value of the neutron flux at the location of the rod. The change in reactivity caused by control rod motion is referred to as control rod worth. For a reactor with a single control rod like MTR type reactor, the control rod's worth has a maximum effect if placed in the center of reactor core having the maximum flux. The difference of the worth of the rod between the inserted or withdrawn rod from the reactor is dependent on the axial flux shape. Generally, the flux at the top and bottom of the reactor is typically less than that in the middle. Therefore, the rod worth per unit length of the control rod at the top and bottom of the core is less than that in the middle during insertion or withdrawal. This behavior is typically illustrated in the integral and differential rod worth curves as shown in Figures 7 and 8 respectively.

The integral control rod worth curve is particularly significant in research reactor operation. For a reactor which has a large amount of excess reactivity, several control rods are required [16]. To gain the full effectiveness of the rods and a relatively even flux distraction, the rods need to be distributed appropriately. The exact worth of each control rod is dependent upon the design of the reactor. The exact effect of control rods on reactivity can be usually estimated. For example, a control rod can be withdrawn in small increments, such as 5 cm, and the change in reactivity can then be determined following each increment of withdrawal. By plotting the resulting reactivity versus the rod position, a graph similar to that in Figure 7 is obtained. The graph depicts integral control rod worth over the full range of withdrawal. The integral control rod worth is the total reactivity worth at that particular degree of withdrawal and that is mostly defined to be the greatest if the rod fully withdrawn.

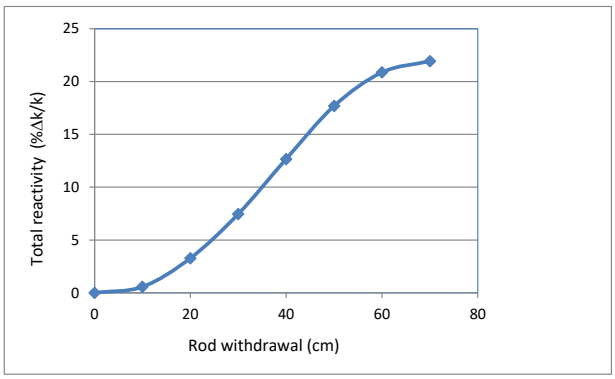

Fig 7. Integral control rod worth

The above Figure 7 shows that the gradient of the curve (Δρ/Δx) indicating the amount of reactivity inserted per unit of withdrawal is the highest when the control rod is mid out of the core. This occurs due to the area of the highest neutron flux near the center of the core. Therefore, the change of neutron absorption is the most one in this area. If the slope

of the curve for integral rod worth in Fig. 7 is taken into account, the rate of change of control rod worth will then be the function of control rod position.

A plot of the slope of the integral rod worth curve, so called the differential control rod worth, is then shown in Fig. 8. It is seen that at the bottom of the core, where there are few neutrons, the rod movement has little effect due to the little various change in reactivity worth per cm. As the rod approaches the center of the core, its effect becomes greater, and hence making the change in reactivity worth per unit length greater. However, at the center of the core, the differential rod worth is the highest depending on small rod motion. From the center of the core to the top, the rod worth per unit length is basically inversed to the rod worth per unit length from the center to the bottom.

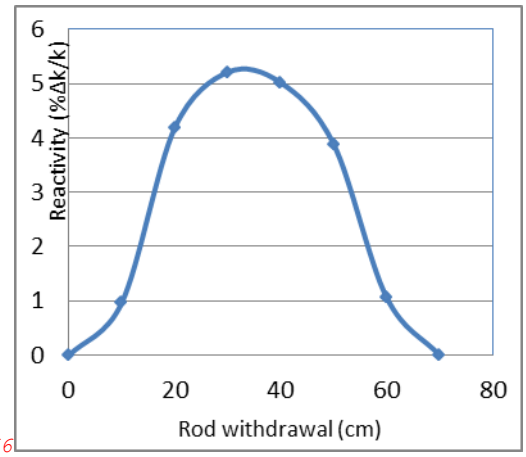

Fig 8. Differential of control rod worth

The control fuel element (CFE) are modeled in Batan-3DIFF code and based on the code, XS requirements, four groups of neutron energy for libraries of XS are provided. This was only done by changing the thermal capture cross-section of a region of concern by an amount such that the reactivity of inserted control rod would correspond to the benchmark. The control rod worth is estimated by calculating keff with all four absorber boxes inserted in the core. The results are shown in Fig. 7 indicating the axial power shape as a function of the control position in the reactor ranging from fully inserted to fully withdrawn positions. An axial peaking factor of 1.5 is obtained bounded by the limit. The effect of the insertion rate on the flux density is illustrated in Fig 7. In addition, Fig 8 shows the worth of fully inserted four control fuel elements. The result shows that the gradient is 0.594 %Δk/k/cm and by adding 15 % of the safety factor, that becomes 0.68 %Δk/k/cm. From Fig 8 showing the differential worth of control fuel elements, it can be said that the most effective control rod is at the position of 25-40 cm. In addition, the following factors surely affect the control rod worth, such as, burn-ups of neutron absorber, follower fuel and distribution of fuels in the reactor core.

d. Period- reactivity relation

To calculate the control rod velocity is used POKDYN program. POKDYN program written in Fortran consisting of main program, POKS subroutine, POKIN subroutines, and Function REACH Function that uses kinetic equations point to complete the calculation in terms of neutronic transient conditions and accidents because of reactivity insertion. The results of the calculations of period-reactivity relationship using the data of Table 2 , Table 3 and Table 4.

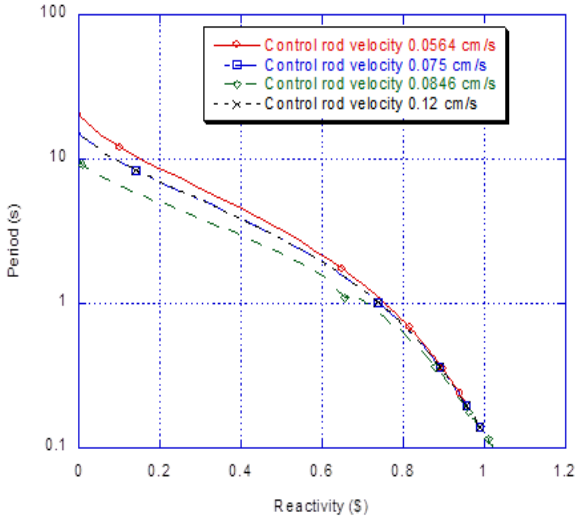

Fig 9. Maximum control rod speed

Reactivity can not be directly measurable and in most research reactors, procedures do not refer to it and most technical specifications do not limit it. Instead, they specify a limiting rate of neutron power rising (measured by detectors), commonly called a reactor period (especially in the case of MTR type reactor). The smaller the value of τe, the more rapid the change in reactor power. The reactor period may be positive or negative. As the reactor period is positive, reactor power increases or vice versa. If the reactor period is constant with time, as associated with exponential power change, the rate is referred to as a stable reactor period. However, if the reactor period is not constant but is changing with time, as for non-exponential power change, the period is referred to as a transient reactor period. According to German system (SUS), once reactor period (reactor doubling time) < 10 s, power increases fast and power scram is set [16].

A much more exact formula reactor period is based on solutions of six-group point kinetics equations. From these equations, an equation, called the in-hour equation (which comes from the inverse hour, when it was used as a unit of reactivity that

corresponded to e-fold neutron density change during one hour), may be derived. The reactor period, τe, or e-folding time, is defined as the time required for the neutron density to change by a factor e = 2.718. The reactor period is usually expressed in units of seconds or minutes. On the MTR type research reactor the acceptance criteria for a period > 10 s, so the Fig 9 showed the relation of period and reactivity for MTR type research reactor. The result showed that for control rod speed 0.0564 cm/s, the period more than 10 s. If the speed of the control rod changes to 0.075 cm/s, the period of the core is still more than 10 s. This also the same as that changes to again to 0.0846. However, when the speed of the control rod increases to 0.12 m/s, the period becomes less than 10 s. This means that the maximum control rod for the MTR type research reactor is 0.0846 cm/s.

e. Thermal-hydraulics and dynamics analysis

The coolant mass flow rate in the MTR type research reactor is limited by the flow instability phenomenon. Reactor with high thermal power can occur flow instability characterized by a flow excursion. When the flow rate and the heat flux are relatively 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 by the higher of fuel height so the maximum temperature of fuel and cladding will be reduced. Based on the calculation, the maximum coolant flow rate obtained at 900 kg/m^2s. By using the mass flow in the core is 900 kg/s, the maximum coolant velocity at fuel channel is 12.29 m/s.

Starting from the initial conditions of 45^oC for core temperature and 0.02% of nominal for the power(1 MW), a ramp reactivity of 51 pcm/s was inserted during 16 sec. The fuel and moderator temperature feedback coefficient of reactivity are noted in Table 3 and Table 4. Figure 10 and Figure 11 show the development of calculated relative reactor power and average core temperature. The comparison to the safety limit 210^oC of maximum fuel temperature depicts good calculation result compared with safety limit. Starting from the initial value the power increases to about 66 MW of power, the average core temperature escalates to about 76^oC. This behaviour stimulates the negative reactivity feedback effect that is able at this point to consume the entire available excess reactivity forcing the power to decrease and demonstrating the inherently safety features of conceptual reactor core. Sometime, it is difficult to measure cladding or fuel temperature in a research reactor. Thus, the calculated temperatures are used to evaluate the consequences of an accident. During the accident, the calculated maximum fuel and cladding temperature of the central channel amount to 196^oC and 192^oC respectively. Thus, the maximum clad temperature is far from the fuel temperature safety limit of 210^oC, ensuring that no sub-cooled boiling occurs. Hence, the transient indicates that the accomplished power peak of about 66 MW does not affect the fuel elements, since no DNB is to be expected under these conditions

During RIA events, the core becomes supercritical and generally core power rises to level beyond the heat removal system capability. For this reason, such events are considered one of the most severe transients that could lead to core damage. To understand the dynamics of such phenomena, it is necessary to identify the various key parameters that govern the power excursion shape, the inherent self-limiting behavior, the power, and the released energy. These later are mainly governed by the prompt neutron lifetime that used in the core. There are also the delayed neutron fraction, reactivity coefficient related to complex interactions of the physical process between kinetics and thermal-hydraulics phenomena as well as the response of the reactor control system. The MTR type research reactor core with 390 gram of fuel loading corresponds to ramps reactivity of 51 pcm/s. In this case, the role of prompt (Doppler reactivity) and delayed coolant temperature rise feedback effects and they are emphasized in all the consideration transients. The scram system is activated when the reactor power reaches at 59 MW. Since there is 0.5 seconds to actualizing the control rod scram system, the power increases to 66 MW. When the reactor power increases to 66 MW, the fuel, cladding, and coolant temperatures are 196^oC, 192^oC, 76^oC respectively, but they are still below the safety limit of 210^oC. The temperature of the fuel, coolant and cladding should be limited to maintain the integrity of the fuel. Table 5 showed that the maximum temperature in the fuel, cladding, and moderator. It was stated that reactions of γUMo alloys with aluminum started typically at the temperature of 645^oC[17]. It is also indicated a temperature of 641^oC for the melting point of the aluminum, and it is suggested that chemical reaction started soon after the melting phenomena. Thus, it is confirmed the validity of the fundamental assumptions of their approach to the detection of possible reactions in the U-Mo-Al system. The formation of a layer of liquid aluminum in contact with the γUMo particles promotes an effective contact between both

surfaces, enhancing the probability of reaction. So the fuel and cladding temperature are still far below the melting point of the aluminum as a cladding.

Figure 10 showed the power and reactivity profile after insertion of reactivity 51 pcm/s and after 16 seconds the maximum power reached 66 MW and then reactor scram by setting point at 118 % of maximum power. Figure 11 showed the temperature profile of fuel, cladding, and moderator. The maximum temperature is still below the safety limit for U-9Mo/Al fuel.

Table 5. Maximum temperature with various uranium density at fuel elements

Parameters	Uranium density [3.96 gram/cm^3]
Coolant temperature at core inlet [^{o}C]	44.5
Coolant temperature at core outlet [^{o}C]	60.50
Maximum temperature at coolant [^{o}C]	76.0
Maximum temperature at cladding [^{o}C]	192.0
Maximum temperature at fuel meat [^{o}C]	196.0

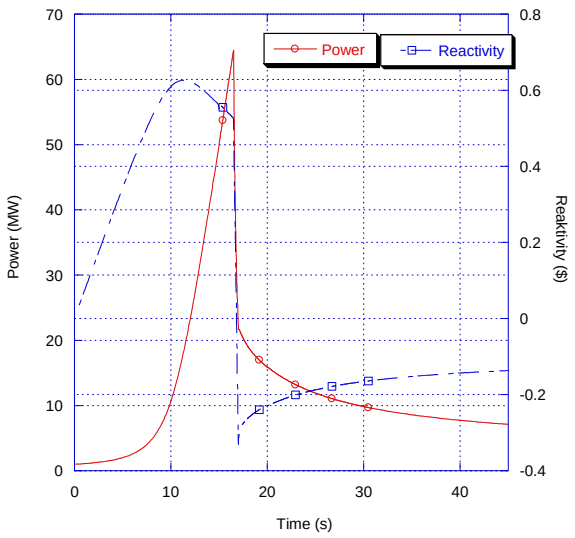

Fig 10. Power and reactivity of the core as transient

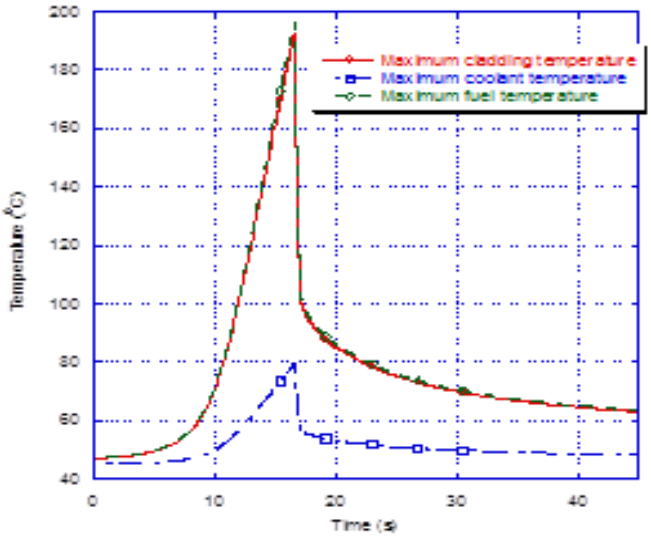

Fig 11. Maximum fuel, cladding and moderator temperatures

CONCLUSIONS

Core configuration including a number of in-core irradiation positions has a very much influence on the value and distribution of neutron fluxes either in-core or reflector region. To have an increase of neutron flux in the core region, it can be obtained by adding an in-core irradiation position. The outer irradiation positions have the biggest influence on increasing the neutron flux in the reflector region. For the MTR type research reactor, based on the equilibrium core, the best core is using fuel density of 3.96 gram/cm^3 that provides so many irradiation positions with highest neutron fluxes. Based on the calculation of MTR-DYN code, it is clear that uranium densities of 3.96 for U9Mo-Al fuel that are proposed in this research work can be utilized as a candidate fuel for the RRI reactor core. The neutronics and thermal-hydraulics criteria are fulfilled with no exceeding safety limit. Fuel cladding strain limit is not exceeding during anticipated transient with a thermal power of 50 MW. The dynamic characteristic of the core has to carry out with a core configuration of 3.96 gram/cm^3 at optimum uranium density of U9Mo-Al fuel for MTR research reactor without safety rod which proposed in this research can be applied with good safety features.

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