Safety Reevaluation of Indonesian MTR-Type Research Reactor

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Abstract. Safety analyses of 30 MW MTR-type research reactor have been conducted for three important cases, i.e. loss of flow accident (LOFA), loss of off-site power and reactivity insertion accident (RIA) due to inadvertent control rod withdrawal. The transient calculations were done using PARET/ANL, and for reactivity insertion rate MCNP5 code was employed. For both LOFA and loss of off-site power, the reactor was assumed to have been operating at nominal power of 30 MW. On the other hand, during accident of inadvertent control withdrawal the reactor was assumed to have been operating at 1 MW. The analyses are intended to confirm that the most bounding design basis accidents, the reactor can be maintained safe. For all the aforementioned cases, The fuel and clad temperatures can be maintained well below safety criteria of 200 °C and 145 °C, respectively. In addition, the safety margin against flow instability (S) is kept well above the criterion of 1.48.

Keywords: MTR type, LOFA, loss of off-site power, RIA, rod withdrawal.

1. Introduction

Indonesian 30 MW MTR-type research reactor is pool type reactor cooled by downward forced convection during high power operation. The fuel material is plate type made of uranium silicide with 19.75 % U^{235} enrichment. The core is surrounded by beryllium reflectors consisting of block and element beryllium.

Safety evaluation of 30 MW MTR-type research reactor is performed for three cases. First, loss of flow accident which is caused by electrical or mechanical problem in the primary pump that leads to total primary pumps failure simultaneously. Second case is loss of off-site power that triggers the primary pumps to stop and at the same time the reactor scram. In the first case, the reactor scram is initiated by low-flow trip point, but for the second case the scram is due to loss of magnetic field in the control system. The third case is inadvertent control rods withdrawal inducing positive reactivity. All those calculations were done with PARET/ANL code. As for calculating the reactivity insertion rate from control rods, MCNP5 code was used.

The analyses are intended to demonstrate the level of safety for the most bounded design basis accidents of the reactor.

2. Governing Equations

The reactor power is calculated based on point kinetic equation:

$$\frac{d\Phi(t)}{dt} = \frac{[\rho(t) - \beta]}{\Lambda}\Phi(t) + \sum_{i=1}^{I}\lambda_i C_i(t) + S(t) \qquad (1)$$

$$\frac{dC_i(t)}{dt} = \frac{\beta f_i}{\Lambda}\Phi(t) - \lambda_i C_i(t) \quad ; \quad i = 1,2,..,I \qquad (2)$$

where:

t : time

Φ : reactor power
ρ : system reactivity
β : effective delayed neutron fraction
Λ : prompt neutron generation time
λ_i : group decay constant of group *i*
C_i : concentration of delay neutron precursor
f_i : delay neutron fraction of group *i*, β_i/β

The reactivity feedback is calculated as a summation of fuel expansion, moderator density change, fuel temperature (Doppler broadening effect):

$$\rho_c = \rho_{Rod} + \rho_{MD} + \rho_{Dop} \tag{3}$$

The reactivity term used in Eq.(1) is:

$$\rho(t) = \rho_{in} - \rho_c \tag{4}$$

Where ρ_{in} is external reactivity (eg. control rods).
The following partial differential equation is used for heat diffusion in fuel material:

$$\frac{\partial}{\partial t}\left[g(u,r)u(r,t)\right] = \nabla .k(u,r)\nabla u(r,t) + S(r,t) \tag{5}$$

u(r,t) is temperature as function of radial position *r* and time *t*. Volumetric heat capacity and thermal conductivity are expressed with *g(u,r)* and *k(u,r)*, respectively, as function of temperature and position. Volumetric heat source *S(r,t)* is assumed to be a separate function of space and time.

Hydrodynamic model in PARET is based on modified momentum integral that includes mass, momentum and energy conservation:

$$\frac{\partial \overline{\rho}}{\partial t} = -\frac{\partial G}{\partial z} \tag{6}$$

$$\frac{\partial G}{\partial t} + \frac{\partial}{\partial z}\left(\frac{G^2}{\rho'}\right) = -\left(\frac{\partial p}{\partial z}\right) - \frac{fv|G|G}{2De} - \overline{\rho}g \tag{7}$$

$$\rho''\left(\frac{\partial H}{\partial t}\right) + G\left(\frac{\partial H}{\partial z}\right) = \frac{q}{r_h} \tag{8}$$

where:
z = axial distance variable
$\overline{\rho}$ = volume-wighted two phase coolant density
G = coolant mass flow rate
ρ' = coolant effective density
p, f = pressure and friction factor
v = coolant specific volume
D_e = coolant channel equivalent diameter
g = gravity constant
ρ'' =effective slip flow density of coolant
H = coolant entalphy
r_h = coolant channel hydraulic radius
q = thermal energy obtained by coolant

3. Calculation Model and Safety Criteria

3.1. Calculation Model

For loss of flow accident, the reactor is assumed to have been operating at nominal power of 30 MW. Due to electrical problem in the primary pump, the total primary flow stops following coast down flow. When the flow reaches 80% of nominal flow rate, the reactor receives trip signal from low-flow trip point. The reactor than scram after delay time of 0.5 s. The coolant channels are divided into two channels, i.e. average and hot channels.

As for loss of off-site power, both the primary flow and reactor trip at the same time due to loss of electrical power. The reactor is assumed to have been operating at 30 MW. Due to the fly-wheel, the flow coast down stops totally at 90 s.

For reactivity insertion accident due to inadvertent control rods withdrawal, the reactor is assumed to have been operating at 1 MW. Calculation on the reactivity insertion rate is done with MCNP5 by multiplying the maximum gradient of control rod S-curve with motor speed. The first trip signal from positive floating limit value is assumed to fail to trip the reactor. And, the second trip signal from over-power signal of 114 % of nominal power eventually scram the reactor after delay time of 0.5 s.

3.2. Safety Criteria

For design basis accident, the maximum fuel and clad temperature may not exceed 200 °C and 145 °C, respectively. And for forced convection cooling mode, the minimum safety margin against flow instability (***S***) must be kept above 1.48. The value is calculated from:

$$S = \frac{\eta_C}{\eta_E} \tag{9}$$

Where η_C and η_E are calculated and experimental bubble detachment parameters, respectively. The calculated bubble detachment parameter is based on:

$$\eta(z) = \frac{[T_S(z) - T_C(z)]V(z)}{q''(z)} \tag{10}$$

Where

q'' : heat flux, w/cm^2
V : Coolant velocity, cm/s
z : distance from coolant inlet channel, cm
T_s, T_c : Saturated temperature and coolant bulk temp., K

4. Results and Discussions

4.1. Loss of Flow Accident

The transient starts at 5 s. During LOFA, the reactor power keeps operating until the primary flow reached 80% of nominal flow. Then after 0.5 s of delay, the reactor scram due to low-flow trip point. The natural convection flap will open when the flow reaches 15 % of nominal rate. The stagnation flow took place when the downward flow is equalized by upward flow from natural convection. From that point forward, flow reversal of natural convection will cool down the reactor core, as shown in Fig.1.

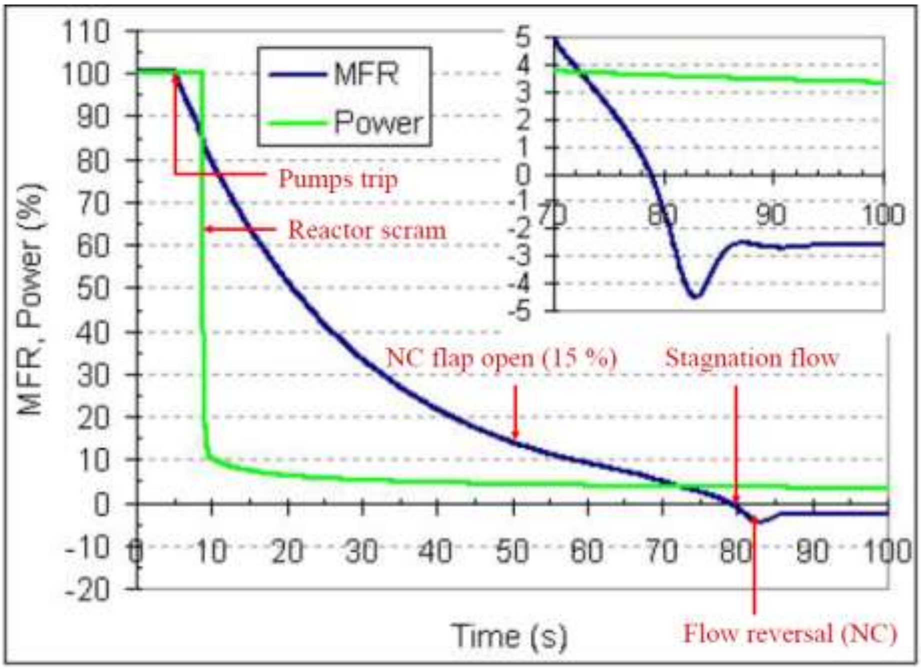

Fig. 1. Event sequence of LOFA

Fig. 2 shows the time history of fuel, clad and hot channel coolant temperatures. The temperatures reached the first peak just before the reactor scram. The first peak fuel temperature is 137.38 °C. As for the coolant temperature at the hot channel is 90.66 °C. The minimum safety margin against flow instability (S), as shown in Fig. 3, is 2.64. The figure also depicts time history of minimum burnout ratio (MNBR).

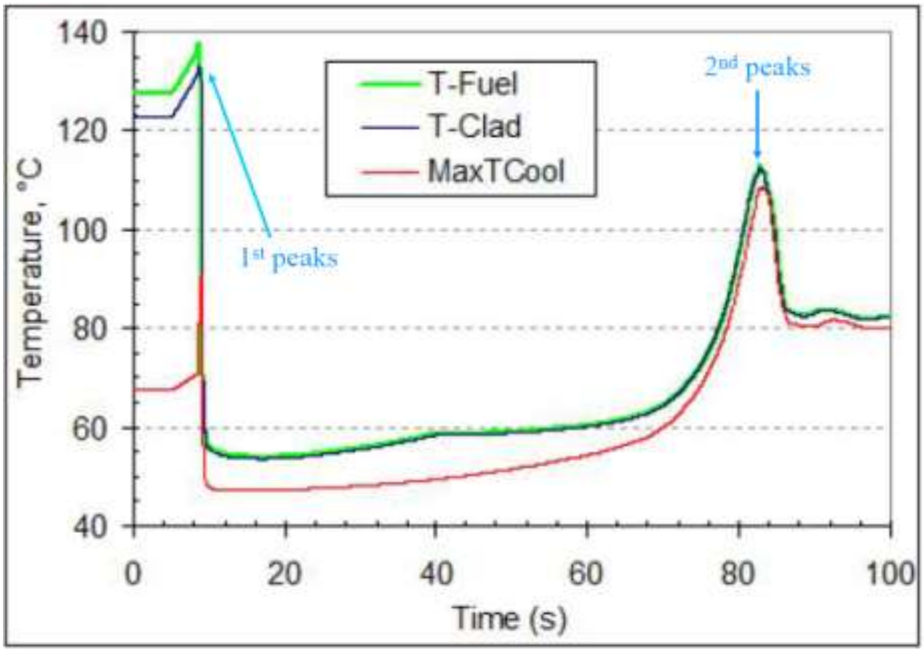

Fig. 2. Fuel, clad and hottest coolant temperature during LOFA

The second temperature peaks take place when the flow is stagnant. The second fuel peak temperature is 112.69 °C, and the maximum coolant temperature at hot channel reaches 108.64 °C.

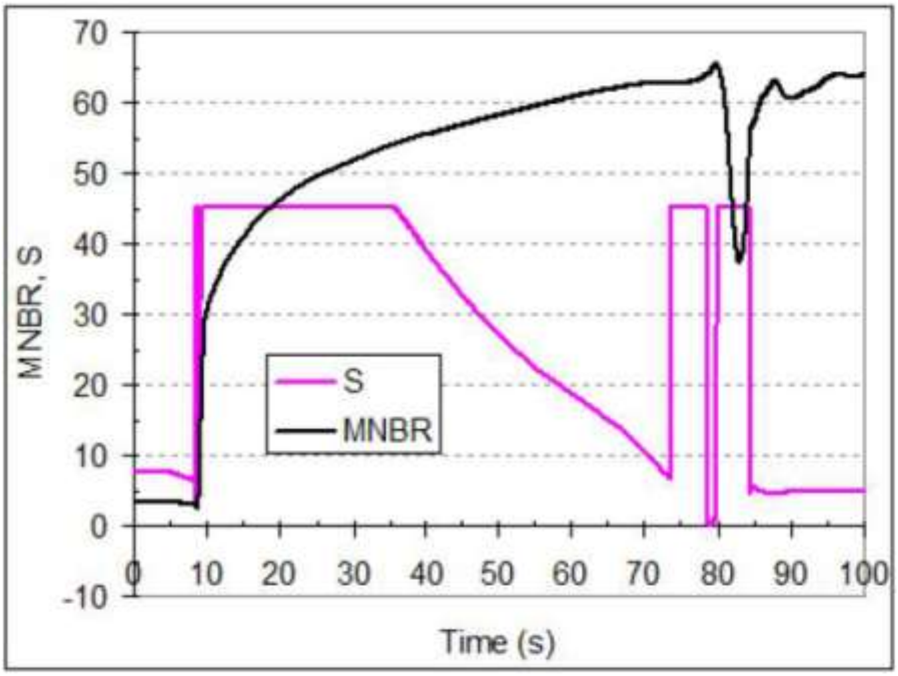

Fig. 3. Fuel, clad and hottest coolant temperature during LOFA

After undergoing reverse flow from upward to downward direction, the reactor decay heat is cooled down by stable natural convection that produces flow of around 2.66 % of nominal mass flow rate. Table 1. elaborates more detail safety parameters during the transient.

TABLE I: Sequence of event and safety parameters during LOFA

Time (s)	Parameter / phenomenon	value
Steady state		
0 – 5.0	Max . Fuel temp., °C	127.41
	Max . Coolant temp. At hot channel, °C	67.56
	Min. Flow stability parameter (S)	7.46
Transient conditions (first peak)		
8.45	Max. Fuel temp., °C	137.38
8.82	Min. Flow stability parameter (S)	2.64
8.85	Reactor scram	
8.89	Max . Coolant temp. At hot channel, °C	90.66
Second peak		
78.86	Stagnant flow	0.0
82.76	Max. Fuel temp., °C	112.69
83.26	Max . Coolant temp. At hot channel, °C	108.64
Stable condition with natural convection		
86.00	Core power, MW	1.04
	Max . Fuel temp., °C	84.26
	Max . Coolant temp. At hot channel, °C	82.92
	Mass flow rate of natural circulation, % of nominal MFR	2.66

4.2. Loss of Off-site Power

Unlike LOFA, during Loss of Off-site Power the reactor and primary pumps trip take place at the same time, as both are caused by loss of power as shown in Fig. 4.

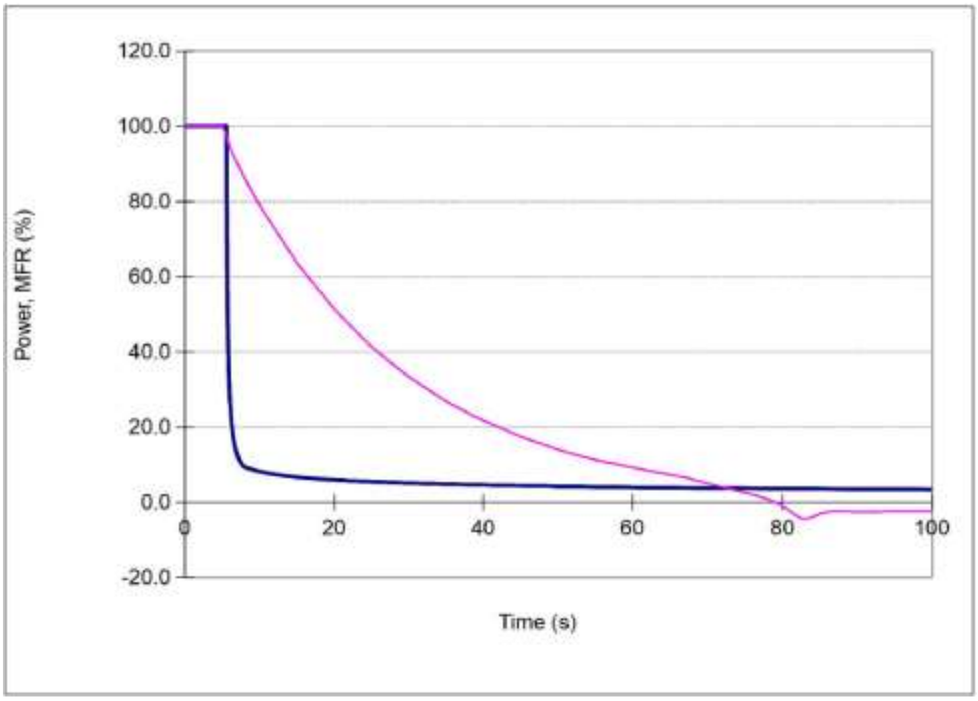

Fig. 4. The reactor power and mass flow rate

The first temperature peaks during loss of off-site power is lower than during LOFA, however the second peaks produce little bit higher. The fuel and clad temperature for both cases come very close, as the clad material is very thin and has good thermal conductivity.

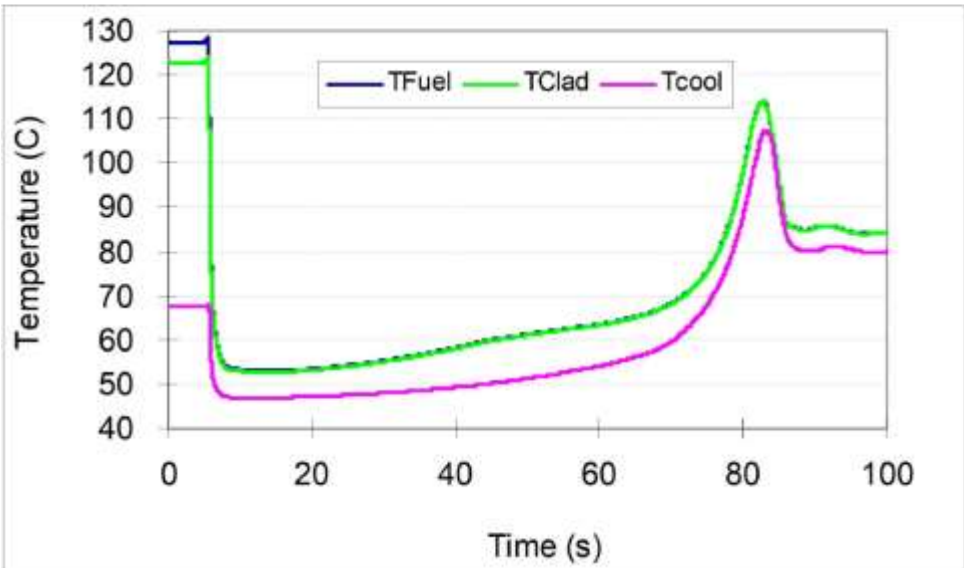

Fig.5. The reactor power and mass flow rate

4.3. Inadvertent Control Rods Withdrawal

The reactivity insertion rate of 2.2×10^{-4} /s, which was calculated using MCNP5, is simulated. The value came from multiplication of maximum S-curve gradient of control rods and motor speed. As shown in Fig. 6, the maximum gradient is located axially in the mid-plane where the neutron flux is maximum.

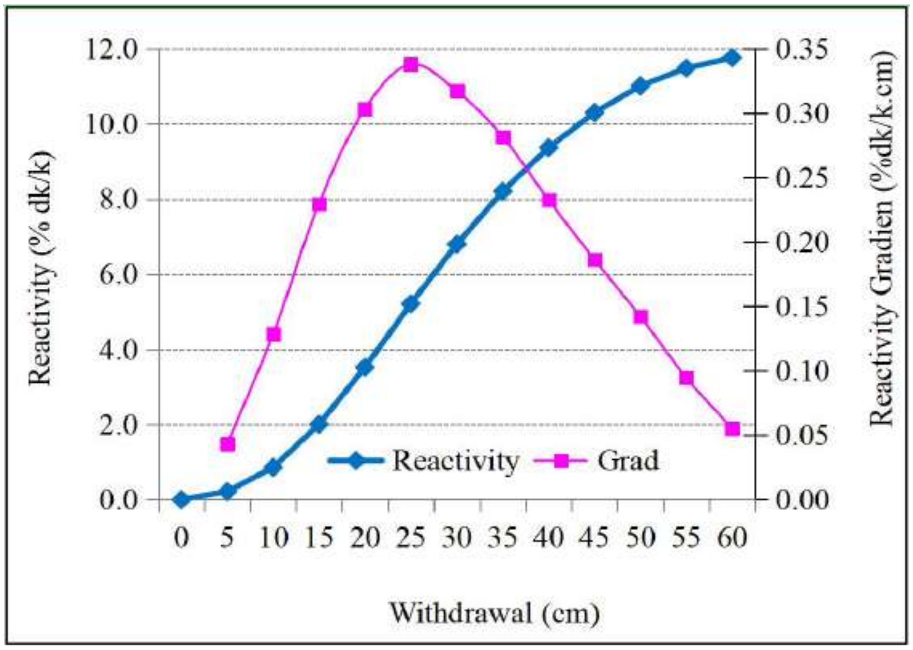

Fig.6. S-Curve control rod worth and its gradient

The transient starts at 5 s when the reactor is assumed to have been operating at 1 MW. As all control rods are inadvertently withdrawn with maximum speed, the reactor power starts to increase and reaches the first trip signal from floating limit value at t=18 s. The floating limit value is a trip signal aimed at limiting and controlling the power change both power increase (positive floating limit value) and power decrease (negative floating limit value) at high power operation mode.

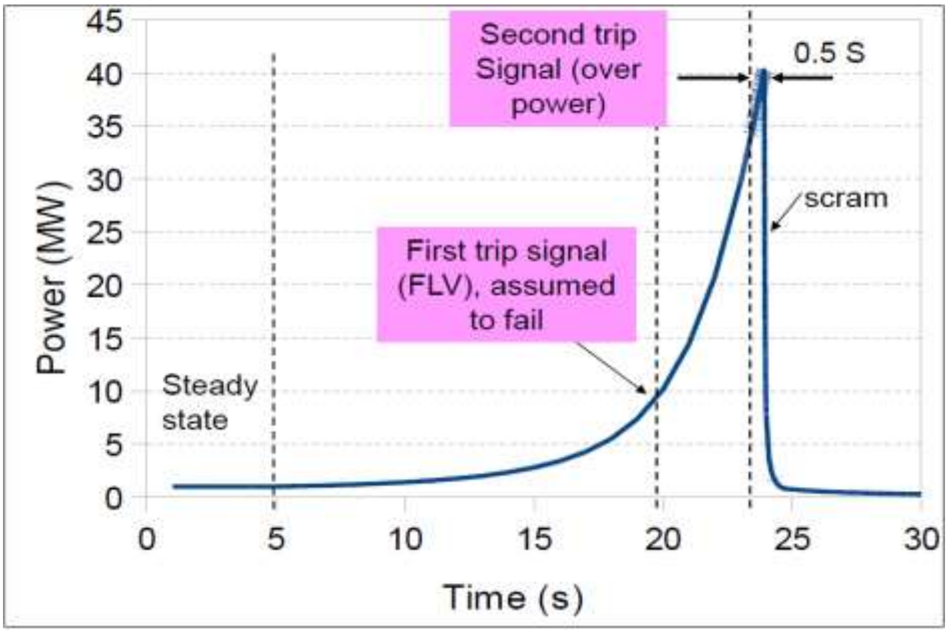

Fig.7. Event sequence of reactivity insertion accident

As the first trip signal is assumed to have failed to scram the reactor, eventually the second trip signal from over-power signal, which is 114 % of nominal power, scrams the reactor after 0.5 s of delay. The transient produces peak reactor power of 40.27 MW that leads to increase the fuel and clad temperatures of 185.32 °C and 138.04 °C, respectively. The maximum coolant temperature at the hot channel reaches 98.95 °C. As for minimum flow instability parameter (S), 2.25 is reached which is well above 1.48.

5. Conclusions

Two most bounding design basis accidents of LOFA and RIA, and Fukushima like loss of off-site power accident have been simulated using PARET/ANL code. All three simulated accidents have confirmed that all safety parameters can be maintained secured. The fuel and clad temperatures can be maintained well below safety criteria of 200 °C and 145 °C, respectively. In addition, the safety margin against flow instability (S) is kept well above the criterion of 1.48.

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