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Original Article

Thermal hydraulic analysis of core flow bypass in a typical research reactor

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ABSTRACT

The main objective of nuclear reactor safety is to maintain the nuclear fuel in a thermally safe condition with enough safety margins during normal operation and anticipated operational occurrences. In this research, core flow bypass is studied under the conditions of the unavailability of safety systems. As core bypass occurs, the core flow rate is assumed to decrease exponentially with a time constant of 25 s to new steady state values of 20, 40, 60, and 80% of the nominal core flow rate. The thermal hydraulic code PARET is used through these calculations. Reactor thermal hydraulic stability is reported for all cases of core flow bypass.

1. Introduction

Research reactors exist in many countries around the world. Many countries consider research reactors as an initial step towards constructing their nuclear power plants programs (NPPs). The neutrons generated in a research reactor have a lot of useful applications like, neutron scattering, non-destructive testing, materials testing, production of radioisotopes, research, and education. Safety calculations of research reactors always include reactivity insertion accidents (RIAs), and loss of flow accidents (LOFAs). One of the anticipating operational occurrence that can happen in research reactors is core flow bypass. The core primary cooling circuit flow bypass analysis does not receive a lot of attention in previous researches. To be more conservative, it is assumed that reactor safety systems are unavailable throughout the present study. Because of the complexity of reactor systems and the coupling between reactor kinetics and thermal-hydraulics, a lot of one-dimensional and zero dimension codes were developed to study the behavior of such systems during and after thermal-hydraulics transients. All one-dimensional codes couples conservation equations of mass and momentum in the coolant region with the energy equation in the fuel and clad regions by using steady-state heat transfer correlations formulas because no reliable transient correlations were developed. Housiadas [1] investigated the course of loss of flow transients in pool-type research reactors, with SCRAM disabled. The analysis is performed with a customized version of the code PARET. Flow instability analysis during the unprotected LOFA is also studied. The effects of using low and high enrichment uranium fuel on the uncontrolled loss of flow transients in a material test research reactor are studied by Muhammad [2]. The thermal hydraulic code PARET was used to carry out the calculations. Kazeminejad [3] investigated the loss of flow accident and flow inversion in a pool type research reactor, with SCRAM enabled. The analyses were performed by a lumped parameters approach for the coupled kinetic–thermal-hydraulics, with continuous feedback due to coolant and fuel temperature effects. El-Morshedy [4] studied the flow inversion phenomenon during LOFA with different core inlet temperatures in a typical MTR reactor with upward core cooling. El-Morshedy [5] developed a transient thermal-hydraulic model entitled Tank in Pool Reactor Thermal-Hydraulic Analysis (TPRTHA) to simulate the steady-state operation and loss of flow transient for a tank in a pool type research reactor.

The work presented in this paper focuses on the transient behavior of a typical MTR reactor as a result of main core cooling system flow bypass. All reactor safety systems are assumed to be unavailable. The thermal hydraulic analysis code PARET is used for the present calculations.

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2. Core configuration and reactor data

A typical research reactor is considered in this study. The core configuration of the reactor is shown in Fig. 1. The reactor core cooling system is presented in Fig. 2. The reactor is a light water, beryllium reflected, and open chimney in open tank type. The reactor full power is 22 MW. Plate-type fuel elements are used with 19.7% enrichment ratio in U-235. The fuel elements are (8 cm × 8 cm) boxes, each with 19-plane fuel plates. The fuel active length is 80 cm and the active width is 6.4 cm. The core configuration is (5 × 6) array, 29-fuel elements, cobalt box (the hatched box in Fig. 1, and fixed position control rods that are controlled from the bottom of the core as indicated in Fig. 1. The primary coolant system consists of a forced convective upward flow of the light water coolant. Two pumps and two heat exchangers are employed to circulate the coolant and to remove the heat in the primary loop as depicted in Fig. 2. The primary cooling loop in Fig. 2 contains two branches with two pumps in each one, so one pump in operation and the other is in a standby mode.

Coupled mechanisms and absorbing plates are used for controlling and shutting down the reactor. For fast insertion, a pneumatic system is used. The fast shutdown is carried out by means of a compressed air injection from a tank to the cylinder piston set and by disconnecting the electromagnet that holds the piston. A diverse second shutdown system is used when the control rods do not function to shutdown the reactor. It consists of four chambers for injection of gadolinium nitrate solution as illustrated in Fig. 2. The reactor main data are given in Table 1.

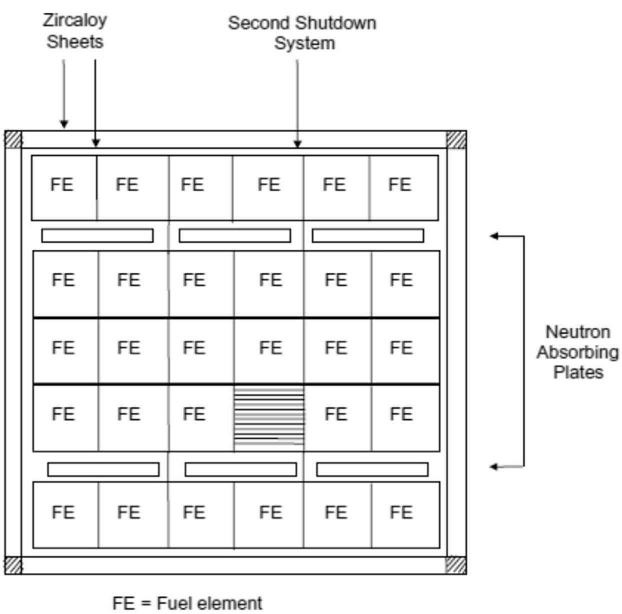

Fig. 1. Core configuration.

Table 1
Reactor main data.

Parameter	value
Rated power, (MW).	22
Coolant.	Light water
Coolant flow direction.	Upward
Nominal core inlet temperature, (°C).	40
Effective core coolant flow, (m^3/h).	1900
Water level in the reactor pool, (m).	10.4
Fuel thermal conductivity, W/m.K.	15
Cladding thermal conductivity (W/cm K).	300
Reactor system pressure, bar.	2
Design peaking factor.	3
Prompt neutron lifetime (Λ), (μs).	75
Effective delayed neutron fraction (βeff).	0.00705
Fuel temperature reactivity feedback coefficient, $/°C.	$-3.12 \cdot 10^{-3}$
Coolant temperature feedback coefficient, $/°C.	$-1.3 \cdot 10^{-2}$
Void reactivity feedback coefficient, $/%void.	−0.2935
Available shutdown reactivity worth.	−10 $
Flow reduction rate.	exp(-t/25)
Reactor SCRAM initiation point.	SCRAM disabled

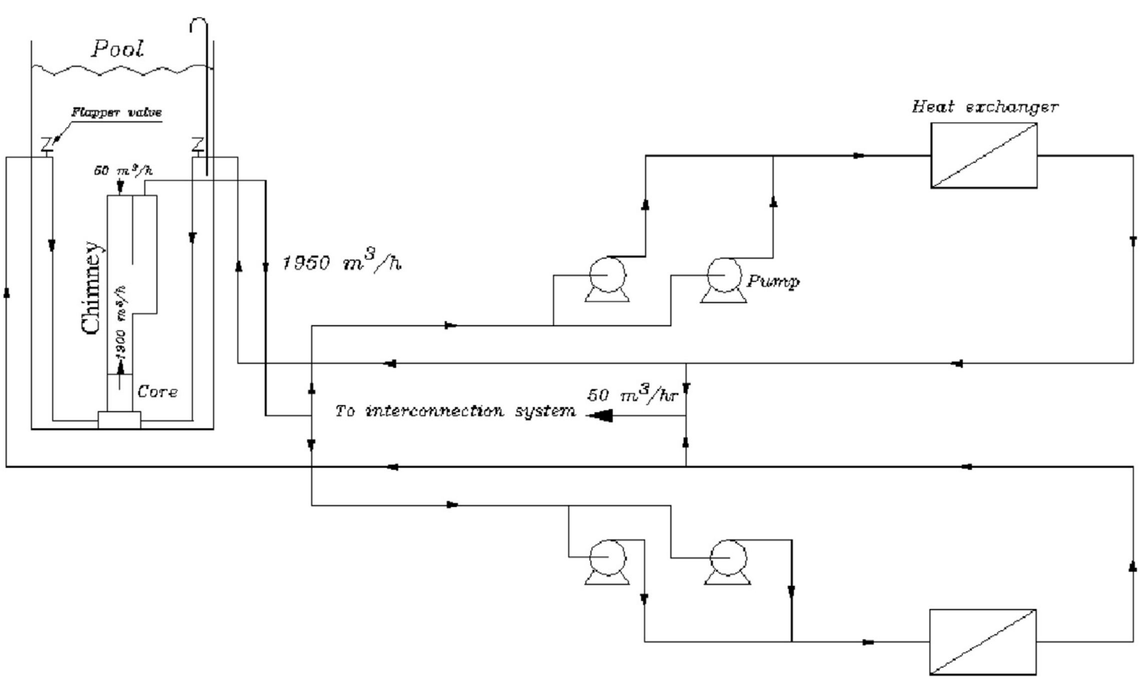

Fig. 2. Reactor core cooling system.

3. Modeling methodology

The transient events start when the reactor is running under steady-state conditions and core flow bypass occurs. The flow rate is assumed to decrease exponentially with time constant of 25 s to a new stable bypass ratios of $G/G_o = 0.2$, 0.4, 0.6, and 0.8, where G is the actual steady state core flow rate and G_o is the nominal core flow rate given in Table 1. It is assumed that the SCRAM system is unavailable. In case of the core flow bypass occurring, reactor protection systems will detect the core flow bypass by monitoring core flow pressure drop, core flow rate, and core temperature difference. In this study, all reactor safety systems are assumed to be unavailable in order to obtain savior analytical results. The core flow bypass time constant is used equal to the flow coast down of core pumps flywheel to be more conservative. Core flow bypass can result from a lot of events like small breaks of the primary cooling circuit inside the reactor main pool otherwise, the pool water level will decrease until the siphon breaker level changing the value of the core outlet pressure and consequently margins to critical phenomena will change. Flapper valves leakage were a typical flow bypass that occurred in the reactor under consideration. The axial heat flux distribution along the reactor core under this study is considered as cosine shape with an extrapolated distance. The input deck used in the present work is verified in Ref. [6].

The PARET code [7] is used to carry out the thermal hydraulics and transient analyses. It is a coupled neutronics, hydrodynamics, and heat transfer code employing point kinetics, one-dimensional hydrodynamics, and one-dimensional heat transfer technique. The code was developed for power reactors for the analysis of SPERT-III experiments [8] and was later customized [9] to include flow correlations, and a properties library that was considered more applicable to the low pressure, temperatures and flow rates encountered in research reactors.

A two-channel model was used to analyze the core, one channel representing the hot channel while the other average channel representing the remaining fuel plates in volume weighted sense. The axial source distribution was represented by 21 axial regions and a chopped cosine shape which has a total power peaking factor of 3 for the hot channel. The hot channel and associated hot plate are assigned with a feedback weight equal to 1/N where N is the total number of channels in the core, while the average channel and associated average plate are assigned with a weight equal to (1−1/N). Fuel and coolant temperatures of each axial node are weighted with the square of corresponding segments node power P_k, and weighting factors W_k. These factors are mathematically formulated as follows

$$W_k = \frac{P_K^2}{\sum_{k=1}^{Naxial} P_K^2} = \frac{Pf_K^2}{\sum_{k=1}^{Naxial} Pf_K^2} = \frac{f_K^2}{\sum_{k=1}^{Naxial} f_K^2} \quad (1)$$

The axial power shape is assumed cosine shape so f_k can be formulated as

$$f_k = \frac{\int_{z-\frac{\Delta Z}{2}}^{z+\frac{\Delta Z}{2}} \mathrm{P(z)dz}}{\int_{h_{ex}}^{h_{ex}+H} \mathrm{P(z)dz}} = \frac{\int_{z-\frac{\Delta Z}{2}}^{z+\frac{\Delta Z}{2}} Sin\left(\frac{\pi Z}{H+2h_{ex}}\right) dz}{\int_{h_{ex}}^{h_{ex}+H} Sin\left(\frac{\pi Z}{H+2h_{ex}}\right) dz} \quad (2)$$

Where: z refers to the axial direction of the fuel plate, H is the fuel active length, h_{ex} is the fuel element extrapolated distance, and k refers to the axial node number with a maximum of 21 axial node. Naxial is the total number of axial segments, P is the fuel plate power, and f_k is the normalized axial power fraction of k_{th} axial segment. The tabulated data of f_k as obtained from Eq. (2) are inserted into the input deck.

4. Results and discussion

A two-channel model is used in the PARET code. The hot channel is the place of the highest temperature in the reactor. All the other channels including the average channel have temperatures lower than that of the hot channel. Therefore, when the hot channel satisfies the limiting conditions, all the other channels will also satisfy them. Therefore, the results compared here are of the hottest channel only.

4.1. Reactor power and reactivity

Fig. 3 depicts the transient response of reactor power for core flow bypass ratios of $G/G_o = 0.2$, 0.4, 0.6, and 0.8. Changes in core temperature affect the reactivity due to changes in the coolant density (due to expansions or phase changes), and/or due to changes in the thermal movement of atoms. Density variations will change the material macroscopic cross sections, while thermal movement of nuclei will affect their microscopic cross sections (Doppler effect). Since the flow rate is decreasing with time during the transient, then the temperatures of fuel plate and coolant begin to increase. As the reactor has a negative reactivity feedback coefficients, a negative reactivity is produced. Thus, core flow bypass transient induces a negative reactivity into the reactor. Since the induced inherent reactivity due to feedbacks is negative, the reactor power decreases from the steady state value of 22 MW that was prevailed just before transient as shown in Fig. 3. The reactor power is controlled by the feedback reactivity only as no external reactivity is inserted to the reactor core.

Fig. 4 illustrates the transient response of total reactivity feedback for cases of $G/G_o = 0.2$, 0.4, 0.6, and 0.8. As the fuel and coolant temperatures increase, the feedback reactivity decreases and becomes negative. The reactor induces a negative reactivity and no

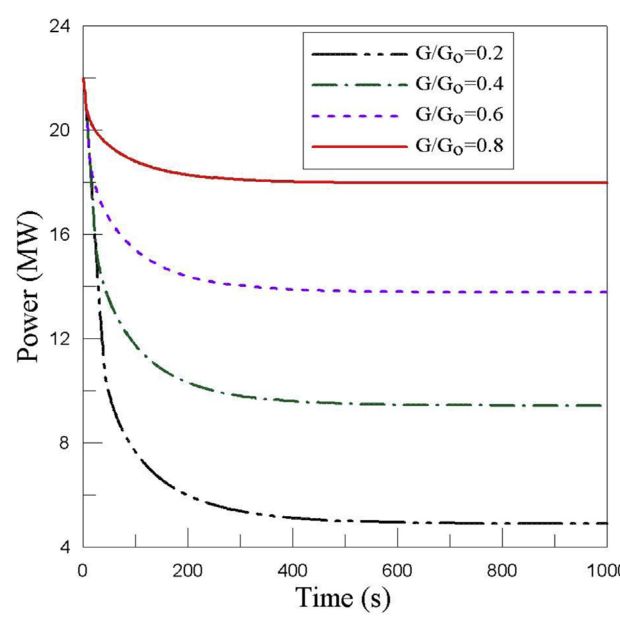

Fig. 3. Transient response of reactor power for cases of $G/G_o = 0.2$, 0.4, 0.6, 0.8.

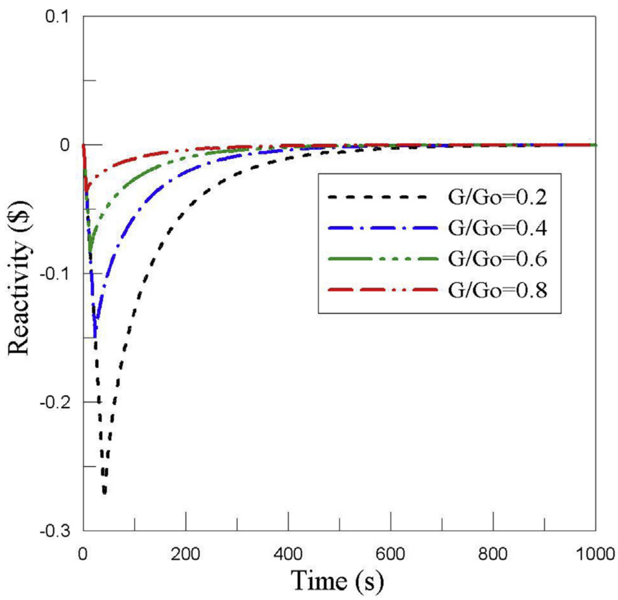

Fig. 4. Transient response of total reactivity feedback for cases of $G/G_o = 0.2, 0.4, 0.6, 0.8$.

longer being critical. Finally, a new balance is reached between the coolant flow rate and reactor power. The new steady state power levels are 4.9032, 9.4439, 13.788, and 17.98 MW for core flow bypass ratios $G/G_o = 0.2, 0.4, 0.6$, and 0.8, respectively. Since there is no external reactivity inserted into the core, the reactor is controlled by its inherent safety features.

4.2. Fuel, clad, and coolant temperatures

Figures (5-8) indicate the transient response of maximum fuel, cladding, and coolant region temperatures for bypass ratios of 0.2, 0.4, 0.6, and 0.8, respectively. The fuel plate and coolant temperatures increase from the steady state values and attain a maximum after that decrease and reach new steady state levels. The maximum fuel, clad, and coolant temperatures are given in Table 2. The reactor power decreases due to the inherent negative reactivity of the reactor and the fuel plate and coolant temperatures decrease and stabilize at a new steady state values. The new steady state fuel plate and coolant temperatures are shown in Table 3.

For cases of core bypass ratios of $G/G_o = 0.4, 0.6$, and 0.8, no subcooled boiling takes place in the reactor hot channel. The reactor reduces its power due to its inherent safety features without outside effects. For bypass ratio of $G/G_o = 0.2$, the coolant subcooled boiling ranges from simulation time of 27.02 up to 65.04 s and then single phase is established.

As the bypass ratio decreases, the new steady state reactor power value decreases. The reactor core is more stable at the new power levels as the bypass decrease as seen from the departure of nucleate boiling ratio DNBR values given in Table 3.

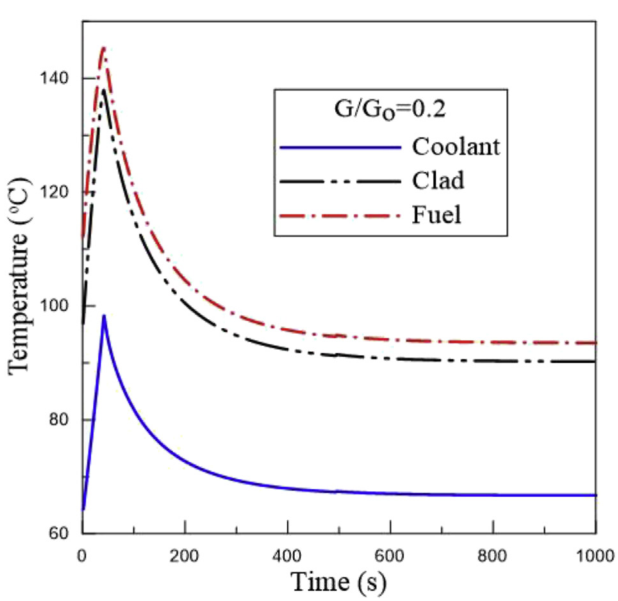

Fig. 5. Transient response of maximum fuel, clad, and coolant temperature for case of $G/G_o = 0.2$.

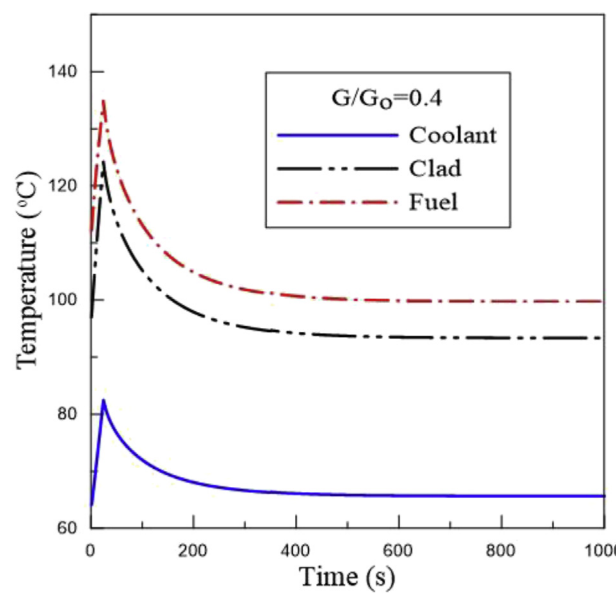

Fig. 6. Transient response of maximum fuel, clad, and coolant temperature for case of $G/G_o = 0.4$.

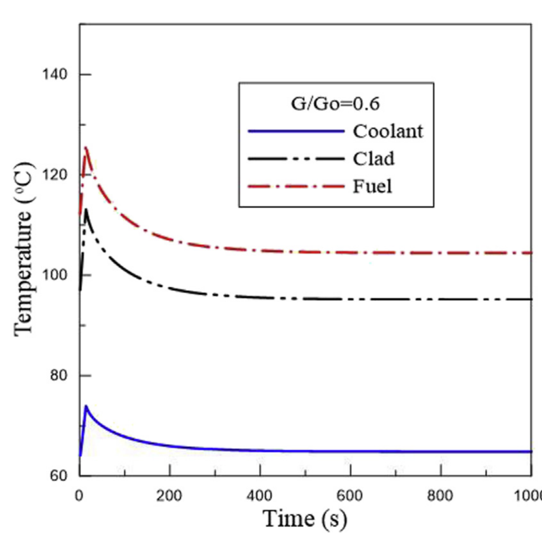

Fig. 7. Transient response of maximum fuel, clad, and coolant temperature for case of $G/G_o = 0.6$.

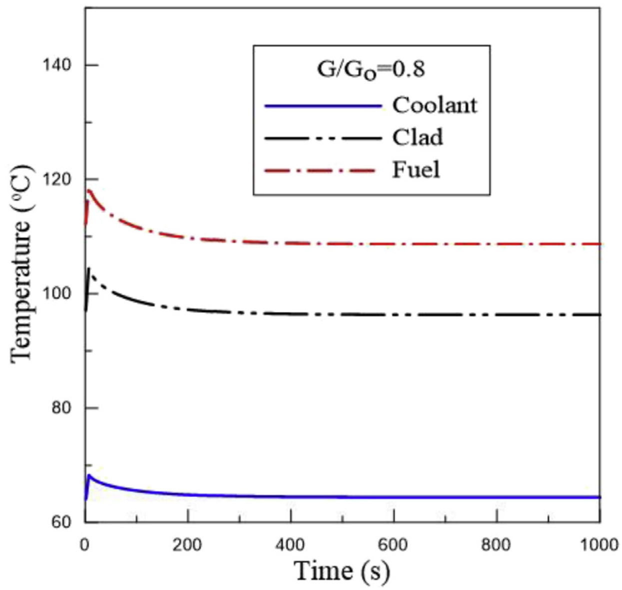

Fig. 8. Transient response of maximum fuel, clad, and coolant temperature for case of $G/G_o = 0.8$.

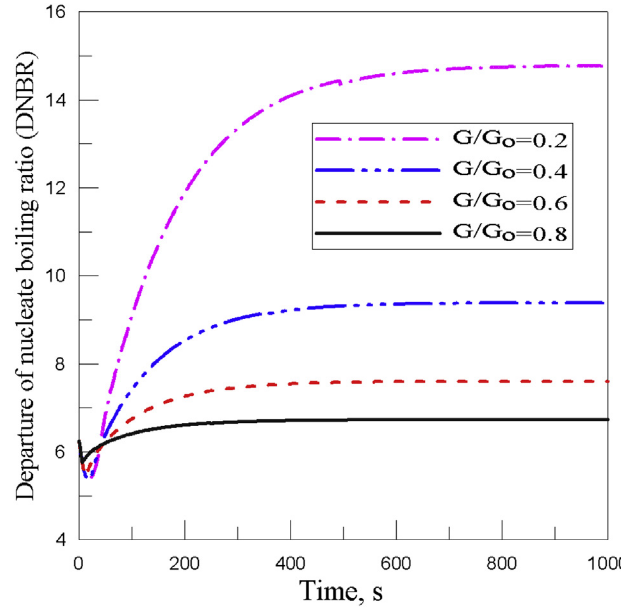

Fig. 9. Transient response of departure of nucleate boiling ratio (DNBR) for cases of $G/G_o = 0.2$, 0.4, 0.6, and 0.8.

Table 2
Thermal hydraulics data during transient phase. (Time is between brackets).

Bypass ratio G/G_0	0.2	0.4	0.6	0.8
T_{Fuel}, max (°C)	145.36 (40.02)	134.94 (23.01)	126.01 (13.01)	118.8 (6.01)
T_{Clad}, max (°C)	137.93 (40.54)	124.17 (23.02)	113.02 (13.01)	104.31 (6.01)
T_{Out}, max (°C)	98.19 (40.54)	82.41 (23.51)	74.03 (13.01)	68.27 (6.01)
ρ, max ($)	−0.2749 (40.54)	−0.14936 (23.01)	−0.082344 (13.01)	−0.03578 (6.01)
DNBR, min	5.38 (18.52)	5.38 (18.51)	5.445 (13.01)	5.76 (6.01)

Table 3
Thermal hydraulics data after the new steady state established.

G/G_o	0.2	0.4	0.6	0.8
Power (MW)	4.9	9.44	13.79	17.98
T_{Fuel}, max (°C)	93.56	99.78	104.51	108.69
T_{Clad}, max (°C)	90.26	93.3	95.05	96.22
T_{Out}, max (°C)	66.61	65.63	64.95	64.4
DNBR	14.77	9.39	7.61	6.74

4.3. Flow instability

Flow instabilities must be avoided in reactor heated channels because flow oscillations affect the local heat transfer characteristics and may induce a premature burnout. For practical purposes in MTR reactors, the critical heat flux that leads to the onset of flow instability is more limiting than the heat flux for stable burnout. The PARET code supports different varieties of heat transfer, flow instability, and Departure of Nucleate Boiling (DNB) correlations. Forgan heat transfer correlation is used to study the onset of flow instability during the transient scenarios. The flow instability ratios for core bypass ratios of $G/G_O = 0.2$, 0.4, 0.6, and 0.8 are presented in Table 4. The flow instability ratio is more than one and consequently no exceed of fuel integrity criteria in terms of thermal hydraulic instability and DNB is observed. The flow instability design criteria is 2 for this reactor so it will be exceeded only for case of 20% core flow bypass. However the fuel integrity is still maintained for this case, the design criteria is exceeded as shown in Table 4.

Table 4
Flow instability ratio for bypass ratios $G/G_o = 0.2$, 0.4, 0.6, 0.8.

$\frac{G}{G_0}$	0.2	0.4	0.6	0.8
FIR, min	1.18	2.36	3.54	4.71

Fig. 9 illustrates the transient response of departure of nucleate boiling ratio (DNBR) for cases of $G/G_o = 0.2$, 0.4, 0.6, and 0.8. The reactor still stable thermal hydraulically after the new steady state power levels have been established for core bypass ratios of $G/Go = 0.2$, 0.4, 0.6, and 0.8.

In this research, It is assumed that the core inlet temperature remains constant throught the problem simulation. If this does not occur, then the temperatures of all the core materials (fuel, clad and coolant) will be more than predicted in this work. This assumption is reasonable for fast transient and the early phase of the transient.

5. Conclusions

Core flow bypass is one of the anticipated occurrences that can occur one or several times during the reactor lifetime. Core flow bypass can result from a lot of events like small breaks of the primary cooling circuit inside reactor tank, flapper valves leakage which was a typical flow bypass that occurred in the reactor under consideration. In this work, core flow bypass is studied under the conditions of safety system unavailability. As core bypass occurs, the core flow rate is assumed to decrease exponentially with a time constant of 25 s to new steady state values of 20, 40, 60, and 80% of the nominal core flow rate. The thermal hydraulic code PARET is used through this study. Reactor stability is reported for all cases of core bypass. The present study concludes that the reactor is still safe for core bypass ratios of $G/G_o = 0.2$, 0.4, 0.6, and 0.8. These cases do not cause problems to the fuel integrity. The reactor power decreases due to the induced negative reactivity in the reactor and the fuel, clad, and coolant temperatures increases then decreases and stabilize at new levels.

Appendix A. Supplementary data

Supplementary data to this article can be found online at https://doi.org/10.1016/j.net.2018.08.021.

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