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Research Reactor Safety Analysis under Partial Loss of Flow without SCRAM

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ABSTRACT

The objective of the present study is to investigate the safety parameters and their safety margins for a typical MTR pool type research reactor during a partial loss of flow or core bypass without scram. The proposed reactor is a plate type fuel element with upward core flow. The code used in the analysis is RELAP5/MOD3.3, the well-known thermal hydraulic system code. In the proposed postulated accident, an exponential decrease of the core flow rate is assumed to occur with a time constant of 25 s to 20, 40, 60, and 80% of the nominal core flow. Study results for thermal hydraulic reactor behavior were validated by comparing RELAP5 results with published results of PARET code under the same accident conditions. Also, the safety parameters and margins were calculated. From comparisons, good agreements are found between RELAP5 code results and PARET code results for the maximum coolant, clad, and fuel temperatures, as well as minimum DNBR values. Deviations are found between the results of the two codes for flow instability parameter. But, both of the two codes concluded that fuel integrity criteria, in terms of thermal hydraulic instability, are not exceeded. In the second part of the study, the effect of decreasing the time constant on the transient results was studied.

1. INTRODUCTION

Due to the association of the reactors with dangerous radioactive isotopes and the huge amount of heat produced during the nuclear fission reaction, the safe utilization of nuclear reactors needs to be guaranteed. Although all safety features considered during modern research reactors design, various irregular events such as a loss-of-flow accident (LOFA), a loss-of-coolant accident (LOCA), or a reactivity initiated accident (RIA) can occur. A power excursion can be produced in the reactor core due to these events despite of the perfect operating order of the heat transport system, [1].

Core flow bypass and partial loss of flow are two issues with similar effects on reactor core cooling. The partial or total loss of flow is one of the postulated initiating events in the power and research reactors. The core bypass may be implemented intentionally in the reactor design or occurs as a result of a postulated event. In PWR reactors, some of the cold leg flow is intentionally directed upward to cool the pressure vessel internal upper structures including the control rods. Other flow is directed to cool the core surrounding structures such as, core barrel, baffles, and others. In research reactors, part of the core flow intentionally diverted to a decay tank for N-16 decay. All of these core bypasses are considered in the reactor design calculations. In other cases, the bypass occurs unintentionally due to a certain component failure such as circulation pump degradation, one of the flapper valves fail to tightly close, small break in the cold leg, etc. In general, many causes during reactor operation can lead to a LOFA occurrence, such as valve closure, pump failure, pipe blockage, heat exchanger blockage, loss of off-site power, etc. The hazard of a LOFA is that it can diminish fuel integrity due to the overheating of fuel that is produced from a low heat transfer coefficient in the reactor core. Research reactors are in many countries all over the world, [2]. The open pool reactors are the most popular and the most used among the

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different types of research reactors as they have great flexibility, great safety, and easy operation, [3]. Therefore, LOFA characteristics in pool-type research reactors, from thermal–hydraulic point of view, have been extensively discussed in many studies [4-13].

In the current study, the transient behavior of a typical MTR pool type research reactor is investigated during partial loss of flow or core bypass without scram, where unavailability of all safety systems in the reactor is assumed. RELAP5 code was used to perform the analyses in the current study.

2. REACTOR DESCRIPTION AND RELATED DATA

The reactor considered in the current study is a pool-type research reactor with an open water surface. The reactor is cooled by light water and reflected by beryllium reflector. The reactor core is composed of several assemblies of MTR fuel elements and graphite reflector blocs. Plate-type fuel elements are used with 19.7% enrichment of U-235. Each fuel element contains 19-plane fuel plates. The active length of the fuel plate is 0.8 m and its active width is 0.064 m. The core is configured as an array of (5 x 6), 29-fuel elements, cobalt box (the shaded box in Figure (1)), and fixed positions for control rods as shown in Figure (1). The rated power of the reactor is 22MW. The primary coolant system consists of upward forced convection of the light water coolant. To remove the heat generated in the reactor core, the primary cooling loop contains two branches with two pumps in each one, so one pump in operation and the other is in a standby mode, [2]. The main data of the reactor are given in Table (1).

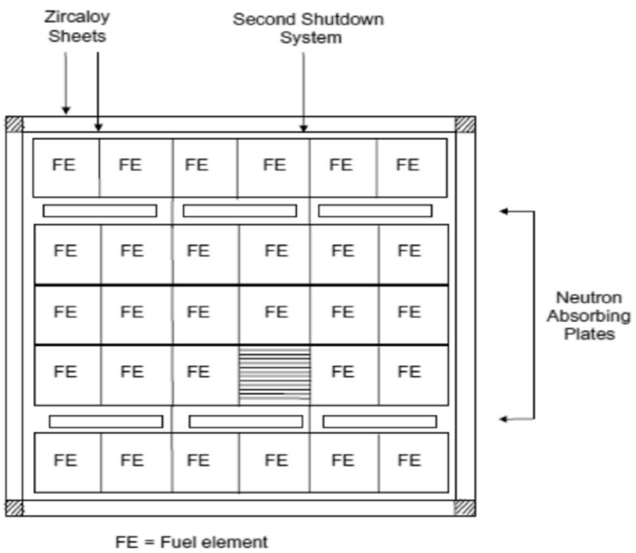


Fig. (1): Core Configuration

Table (1): Main Reactor Data, [2]

Parameter	Value
Reactor power (MW)	22
Type of coolant	Light water
Flow direction of coolant	Upward
Inlet temperature to the core (ºC)	40
Effective coolant flow in the core(m^3/hr)	1900
Reactor pool water level (m)	10.4
Thermal conductivity of fuel (W/m.K)	15
Thermal conductivity of cladding (W/cm K)	300
Reactor pressure, bar	2
Design peaking factor	3
Prompt neutron lifetime (ʌ), (µs)	75
Effective delayed neutron fraction (β_{eff}).	0.00705
Reactivity feedback coefficient of fuel temperature ($/ºC)	-3.12 x10^{-3}
Reactivity feedback coefficient of coolant temperature ($/ºC)	-1.3x10^{-2}
Reactivity feedback coefficient of void, $/%void	-0.2935
Shutdown reactivity worth	-10 $
Rate of flow decrease	exp(-t/25)
Starting point of reactor SCRAM	SCRAM disabled

3. SYSTEM NODALIZATION

The code used in the analysis is the thermal hydraulic system code RELAP5/MOD3.3. Figure (2) shows the RELAP5 nodalization of the research reactor adopted for the problem discussed in the present investigation. The nodalization focuses on the core zone, where coolant channels and heat structures are modeled. Also, the nodalization includes the main reactor components to allow acceptable simulation of the normal operation state, where, reactor core, lower and upper plenums with time dependent junction at the lower plenum in addition to two time dependent volumes was used. As shown in Figure (2), two channels represented the core, an average channel and a hot channel. One fuel plate represented the hot channel, while the rest of fuel plates (550 plates) represented the average channel. The control volume 100, in the figure, represents the flow source, where the control volume 300 represents the flow sink. Table (2) shows the main components of the reactor with their nodalization correspondences.

Table (2): Components of the nodalization

Component	Nodalization number
Flow source	TDV 100
Lower plenum	Branch 200
Average channel inlet nozzle	Pipe 205
Hot channel inlet nozzle	Pipe 206
Average channel	Pipe 215
Hot channel	Pipe 216
Bypass channel	Pipe 217
Upper plenum	Branch 220
Flow sink	TDV 300

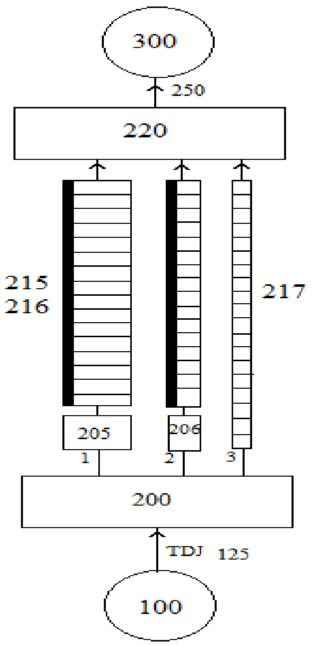


Fig. (2): System Nodalization

4. MODELING METHODOLOGY

RELAP5, the advanced thermal-hydraulic system code, is used to carry out the present analyses. A wide range of postulated accidents in power reactors is effectively simulated using RELAP5 code, as loss of flow (LOFA), loss of coolant (LOCA), loss of offsite power, loss of feedwater, and anticipated transients without scram (ATWS). The new versions of the code, such as Mode3.2 and Mode3.3, have extended capabilities covering the facilities of low pressure and temperature such as research reactors, [14]. The highly generic code RELAP5 is used to simulate a wide variety of thermal and hydraulic transients in both nuclear and non-nuclear systems, with the same efficiency of calculating the reactor coolant system behavior during transients, [15].

The transient analyses, in the current study, are performed using RELAP5/Mod 3.3 for the reactor full power operation mode. Ibrahim et. al., [2], previously performed the analysis for the current research reactor under study using PARET code. Comparisons between RELAP5 code results and the available results of PARET code were performed for validation purposes. In this stage, the same hot channel factors (a cosine shaped with a total hot channel power peaking factor of 3) and kinetics parameters are applied for both of the two codes.

At The beginning of the transient the reactor was running under steady-state conditions then core flow bypass starts. An exponential decrease of the core flow rate is assumed to occur with a time constant of 25 s to new steady bypass ratios of G/Go = 0.2, 0.4, 0.6, and 0.8, where G is the real steady state core flow rate and Go is the nominal core flow rate shown in Table (1). The SCRAM system is assumed to be unavailable. When the core flow bypass transient started, it will be detected by reactor protection systems through monitoring of different reactor operation parameters as core pressure drop, flow rate, and core temperature difference. In the current study, in order to obtain savior analytical results, unavailability of all reactor safety systems is assumed. The flow coast down time constant of the core pumps flywheels is used as the core flow bypass time constant to have more conservative results. Many events can cause core flow bypass as the primary cooling circuit small breaks inside the reactor main pool and flapper valves leakage. In the current work, a cosine shape with an extrapolated distance is considered for the axial heat flux distribution along the reactor core.

5. DESIGN GOALS

Design goals guarantee that appropriate margins are available for normal operation and transient conditions. For a safe reactor operation, the design goals must be fulfilled for any operable core configuration and must be verified whenever necessary. For the reactor under consideration, some design goals are required to be achieved. These goals, mainly, are Onset of Nucleate Boiling Ratio (ONBR), Departure from Nucleate Boiling Ratio (DNBR) and flow instability or Flow Redistribution Ratio (RDR). RELAP5 code does not support some critical phenomena calculation as flow instability or ONB. But the code is capable of control systems simulation normally used in hydrodynamic systems. It contains several types of control components. Each control component describes a control variable as a particular function of time-advanced quantities. The time advanced quantities include quantities from hydrodynamic volumes, pumps, junctions, heat structures, valves, trip quantities, reactor kinetics, and the control variables themselves. In this way, control variables are established from components that perform simple, basic operations, [16]. Using this technique, different correlations can be adapted to the code to calculate any desired parameter during problem simulation.

The ONB is not a restrictive criterion in the fuel element design. But, it is a heat transfer regime that should be identified for appropriate hydraulic and heat transfer considerations. Under ONB conditions, the clad surface temperature over which nucleate boiling will happen for a given local coolant pressure and surface heat flux can be stated by the correlation settled by Bergles and Rohsenow, [17]:

$$Ts = Tsat + \frac{5}{9}\left(\frac{9.23\,q}{P^{1.156}}\right)^{\frac{P}{2.16}^{0.0234}} \quad (1)$$

Where, Ts = surface temperature, ºC, at which ONB occurs, Tsat = saturation temperature, ºC, q = local heat flux, W/cm2, P = local pressure, bar absolute. This correlation is applicable down to the low pressure systems and is commonly used for research and test reactors, [17].

During reactor design, Departure from Nucleate Boiling (DNB) is potentially a limiting design constraint. So, acceptable data on burnout heat flux are needed. Optimization of core cooling against other neutronic, economic and materials constraints can best be achieved by careful use of standard, experimentally-deduced DNB correlations, [17]. For calculation of critical heat flux, Mirshak correlation [18] is used. The correlation is given as:

$$q_c = 151.(1 + 0.1198U)(1 + 0.00914\,\Delta T_{sub})(1 + 0.19\,P) \quad (2)$$

Where, qc = critical heat flux, W/cm^2, U = coolant velocity, m/s. ΔTsub = exit water subcooling, ºC, P = pressure, bars absolute.

Flow instabilities are unwanted in heated channels because flow oscillations affect the local heat transfer characteristics and may cause an early burnout [17]. For practical purposes in MTR reactors, the critical heat flux causes the onset of flow instability to occur is more limiting than the heat flux for stable burnout [2]. Whittle and Forgan correlation [19] is adapted to RELAP5 code using control variable method to calculate the flow instability parameter. Two forms of Whittle and Forgan correlation are used. The first form is that given in [17] where, the average heat flux ($\overline{q}_c$) at onset of flow instability can be expressed in terms of coolant velocity (U), channel geometry (W: width, tw: channel thickness, LH: heated length), temperatures (T_{sat}: saturation temperature, T_{in}: inlet temperature), and fluid properties (ρ: density, C_p: specific heat) as follows:

$$\overline{q}_c = R\,\rho\,C_p \frac{Wt_w}{W_H L_H}\,U\,(T_{sat} - T_{in}) \quad (3)$$

Where,

$$R = \frac{1}{1 + \eta\frac{D_H}{L_H}}$$

A value of η = 25 was determined as a best fit to Whittle and Forgan's data [17].

The second form of Whittle and Forgan's correlation is the form used by CNEA and INVAP [20] as follows:

$$P_{RD} = P_{sat}\,R \quad (4)$$

Where:

$$P_{sat} = \rho\,C_p Q_{ch}(T_{sat} - T_{in})$$

$$R = \frac{1}{(1 + T_r)}$$

$$T_r = 3.15\,DH_{ch}/L_m Gr^{0.23}$$

$$Gr = 1.08\,\rho\,V_{ch}$$

Where, P_{RD} is the power that causes flow instability or flow redistribution, DH_{ch} is the channel hydraulic diameter, L_m is the fuel length, and V_{ch} is the coolant velocity in the channel.

The above correlations, required to calculate the values of the design goals, were adapted to RELAP5 code using control variable methodology to obtain the related best estimate values of these design goals, as stated before.

6. RESULTS AND DISCUSSION

6.1. Model comparison with previous published studies

As stated above, RELAP5/Mod 3.3 code is used in the current study to perform the transient analysis for the reactor under consideration. The reactor core is simulated by two channels; a hot and an average channel. Where, 17 radial meshes and 28 axial volumes are used. The transient starts with steady-state conditions then core flow bypass occurs when the transient time reaches 100 s. At this time, trip of the main coolant pump takes place and a fast decrease of the core coolant flow rate occurs. An exponential decrease of the flow rate is assumed with a time constant of 25 s to a new stable bypass ratios of G/Go = 0.2, 0.4, 0.6, and 0.8, as mentioned before. The results of RELAP5 code of the present study are compared with the results of PARET code of Ref. [2] for validation purposes. The results are shown in this section.

Figure (3) gives reactor power transient response for core flow decrease with ratios of G/Go = 0.2, 0.4, 0.6, and 0.8. The reactor power falls from the steady state value of

22 MW to new steady-state values, as shown in the figure; since a negative induced reactivity due to feedbacks is applied. As no external reactivity is inserted to the reactor core, the reactor power is controlled by the feedback reactivity only. Total reactivity feedback for cases of G/Go = 0.2, 0.4, 0.6, and 0.8 is given in Figure (4).

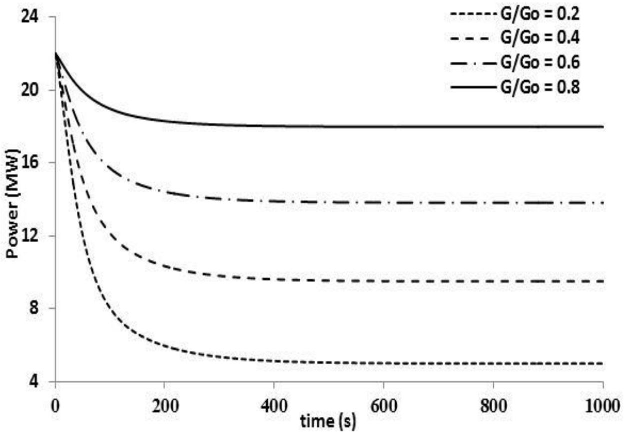


Fig. (3): Reactor power transient response for flow bypass cases of G/Go = 0.2, 0.4, 0.6, and 0.8

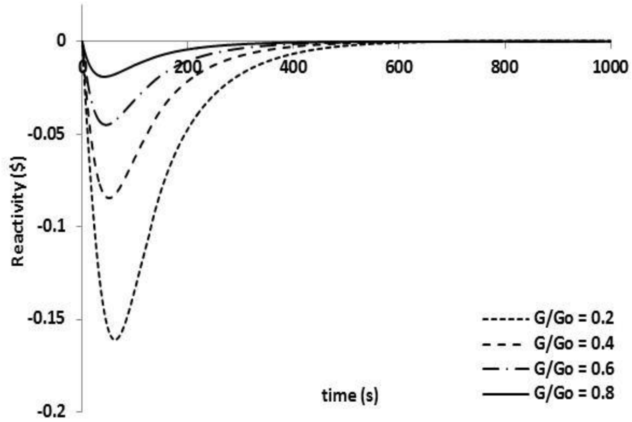


Fig. (4): Total reactivity feedback transient response for flow bypass cases of G/Go = 0.2, 0.4, 0.6, and 0.8

Figures (5-a, b, c and d) show the transient response of maximum temperatures for fuel, clad, and coolant for bypass ratios of 0.2, 0.4, 0.6, and 0.8, respectively. As shown in the figures, the temperatures of fuel, clad and coolant increase with flow reduction from the steady state values and reach a maximum then the temperatures decrease again and stabilize at new steady state values.

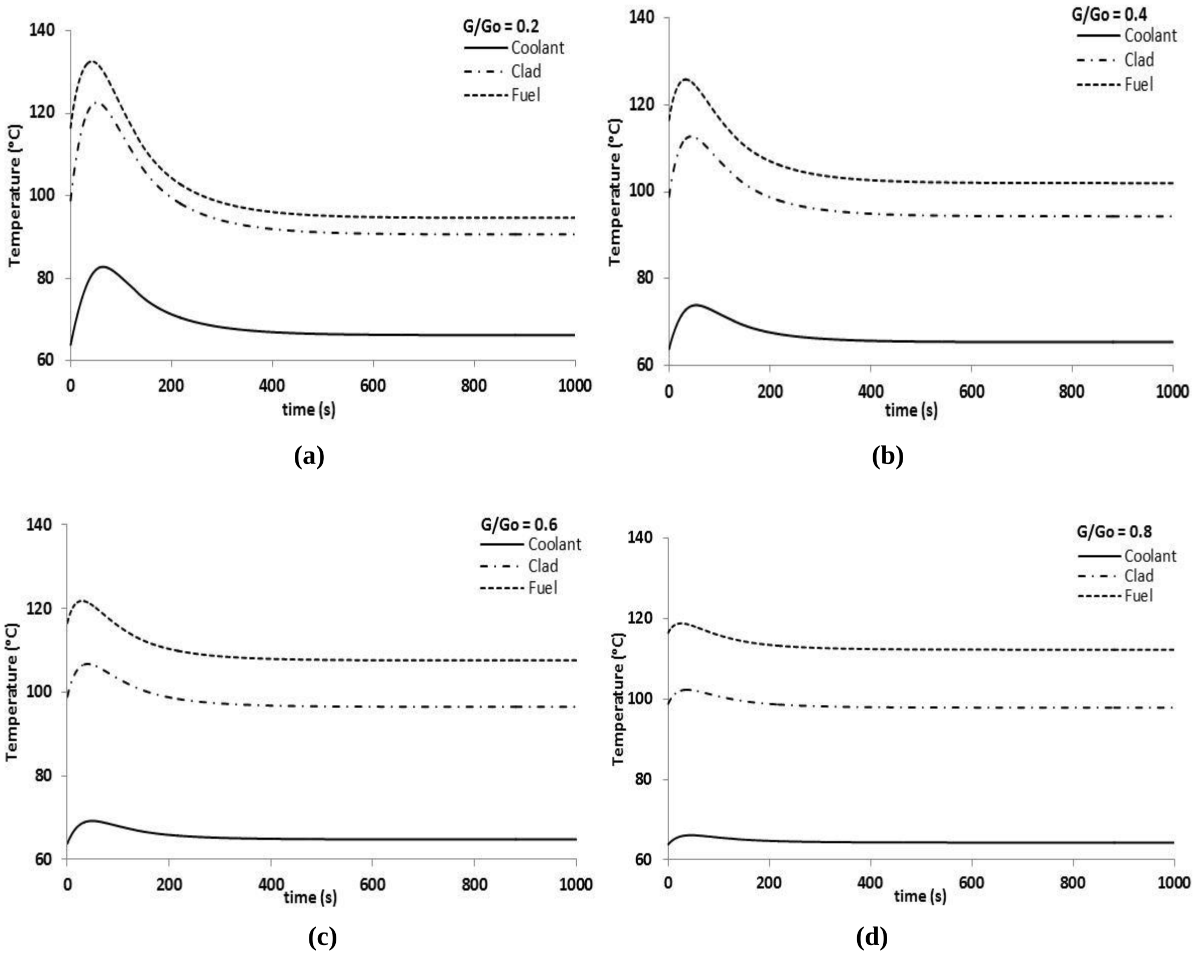


Fig. (5): Transient response of maximum temperatures of fuel, clad, and coolant for flow bypass cases of G/Go = 0.2, 0.4, 0.6, and 0.8

Bergles and Rohsenow correlation, equation (1), is adapted to RELAP5 code using control variable technique, as mentioned earlier, to calculate the ONB temperatures for flow bypass decrease with the prescribed ratios. The transient response of ONB temperatures for bypass ratios of 0.2, 0.4, 0.6, and 0.8 is given in Figure (6). Comparing the curves of Figure (6) by their related surface temperature curves of Figure (5) shows that no subcooled boiling occurs in the reactor hot channel for all cases of core bypass considered. Due to its inherent safety features, the reactor reduces its power without outside effects.

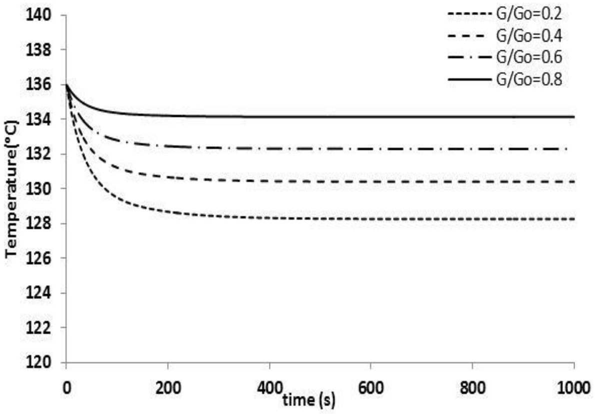


Fig. (6): ONB temperatures transient response for flow bypass cases of G/Go = 0.2, 0.4, 0.6, and 0.8

Mirshak correlation, equation (2) is introduced to RELAP5 code using control variable technique to calculate the departure from nucleate boiling heat flux for flow bypass decrease with ratios of G/Go = 0.2, 0.4, 0.6, and 0.8. With the reduction of the bypass ratio, the value of the new steady state reactor power falls and the reactor core becomes more stable, as noticed from the values of the departure of nucleate boiling ratio (DNBR) shown in Figure (7).

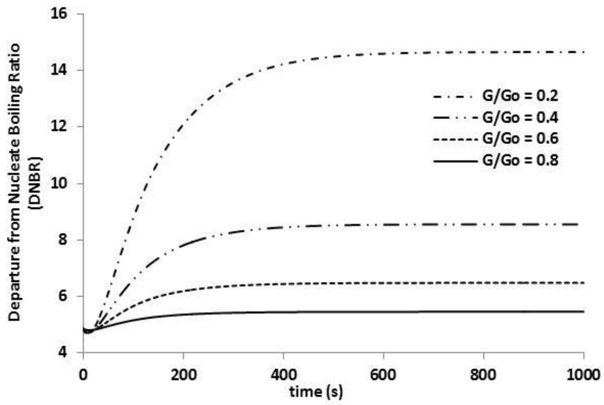


Fig. (7): DNBR transient response for flow bypass cases of G/Go = 0.2, 0.4, 0.6, and 0.8

Results of this study for the maximum fuel, clad, and coolant temperatures, as well as minimum DNBR values, during transient phase, are compared with the results of PARET code given in Ref. [2], as shown in Table (3). Where, the new steady state values for these parameters are compared with their related results of PARET code as given in Table (4). As shown from Tables (3) and (4), good agreements are found between the results of the two codes for core flow bypass decrease with ratios of G/Go = 0.2, 0.4, 0.6, and 0.8, during transient phase and for the new steady state values as well.

Table (3): Comparison of thermal hydraulic data during transient phase between PARET code and RELAP5 code (Time is between brackets)

Bypass ratio (G/Go)	**0.2**		**0.4**		**0.6**		**0.8**	
Code	RELAP5	PARET	RELAP5	PARET	RELAP5	PARET	RELAP5	PARET
T_{Fuel}, max (°C)	133.58 (42.05)	145.36 (40.02)	126.36 (33.8)	134.94 (23.01)	122.1 (28.2)	126.01 (13.01)	119 (24)	118.8 (6.01)
T_{Clad}, max (°C)	124.0 (52.05)	137.93 (40.54)	113.74 (43.6)	124.17 (23.02)	107.5 (29.6)	113.02 (13.01)	103 (35.2)	104.31 (6.01)
T_{Out}, max (°C)	82.75 (60.4)	98.19 (40.54)	74.0 (53.4)	82.41 (23.51)	69.22 (47.8)	74.03 (13.01)	66.11 (43.6)	68.27 (6.01)
ρ, max ($)	-0.161 (60.4)	- 0.2749 (40.54)	0.085 (50.6)	- 0.14936 (23.01)	-0.045 (42.2)	- 0.082344 (13.01)	-0.01905 (40.8)	- 0.03578 (6.01)
DNBR, min	4.69 (9.1)	5.38 (18.52)	4.72 (10.1)	5.38 (18.51)	4.766 (11.4)	5.445 (13.01)	4.81 (10)	5.76 (6.01)

Table (4): Comparison of thermal hydraulic data after the new steady state established between PARET code and RELAP5 code

Bypass ratio (G/Go)	0.2		0.4		0.6		0.8	
Code	RELAP5	PARET	RELAP5	PARET	RELAP5	PARET	RELAP5	PARET
Power (MW)	4.99	4.9	9.516	9.44	13.825	13.79	18.0	17.98
T_{Fuel}, max (°C)	95.875	93.56	102.86	99.78	108.07	104.51	112.54	108.69
T_{Clad}, max (°C)	91.92	90.26	95.38	93.3	97.34	95.05	98.564	96.22
T_{Out}, max (°C)	66.28	66.61	65.43	65.63	64.82	64.95	64.29	64.4
DNBR	14.65	14.77	8.56	9.39	6.49	7.61	5.46	6.74

Table (5): Comparison of minimum flow instability ratio (RDR) between PARET code and RELAP5 code for bypass ratios G/Go = 0.2, 0.4, 0.6, 0.8

Bypass ratio (G/Go)	0.2			0.4			0.6			0.8			Steady-state		
	RELAP5		PARET	RELAP5		PARET	RELAP5		PARET	RELAP5		PARET	RELAP5		PARET
	1st	2nd		1st	2nd		1st	2nd		1st	2nd		1st	2nd	
Minimum RDR	1.285	1.303	1.18	1.609	1.653	2.36	1.864	1.932	3.54	2.09	2.180	4.71	2.30	2.41	N/A

The previous two forms of Whittle and Forgan correlation, equations (3) and (4), are adapted to RELAP5 code using control variable method to calculate the values of flow instability or flow redistribution ratio.

Figure (8) gives the transient response of flow redistribution ratio, equation (4), for flow decrease with ratios of G/Go = 0.2, 0.4, 0.6, and 0.8 during the transient scenarios. From this figure it is noticed that the minimum flow instability ratio is more than one for all cases and consequently fuel integrity criteria in terms of thermal hydraulic instability are not exceeded. But, the flow instability design criterion, of a value 2 for this reactor, is violated for the three cases of 20%, 40% and 60% core flow bypass. On the other hand, the fuel integrity is preserved for all studied cases, as shown from Figure (8).

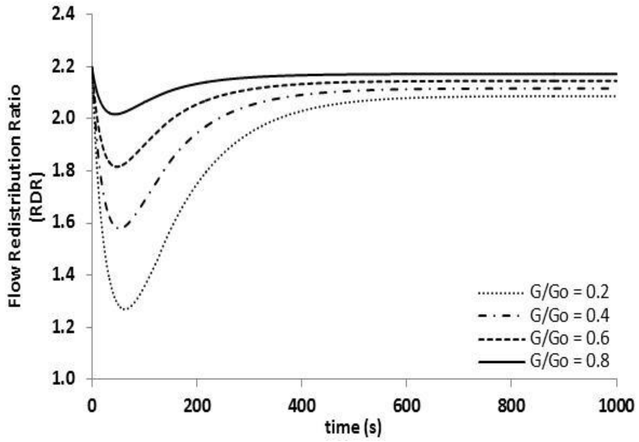


Fig. (8): Transient response of flow redistribution ratio (RDR) for flow bypass cases of G/Go = 0.2, 0.4, 0.6, and 0.8

The minimum flow instability ratios currently studied are compared with the results of PARET code given in Ref. [2], as shown in Table (5). Results of the two forms of Whittle-Forgan correlation that are adapted to RELAP5 by control variable method are shown in the table as 1st and 2nd, that refer to the first and second forms of the correlation, equation (3) and equation (4), respectively.

As shown in Table (5), results obtained from RELAP5 for the two forms of Whittle-Forgan correlation are too close to each other. The flow instability design criterion of 2, for this reactor, is exceeded for cases of core flow bypass of 20, 40 and 60 % of the core flow. Although deviations are found between RELAP5 results and PARET code results for flow instability parameter, both of the two codes concluded that the fuel integrity criteria in terms of thermal hydraulic instability are not exceeded.

6.2. Effect of changing the time constant on the results of the transient

In this part of the study, the effect of decreasing the time constant on the transient results was studied. An exponential decrease of the core flow rate is assumed to occur with a time constant of 10 seconds to a new steady state value of 20% of the nominal core flow rate, where the core flow of 20% of the core flow was the worst condition between the four studied flow values as deduced from the results of the first part of the study. Comparisons between the results for the two time constants of 25 and 10 seconds were given in this section.

Figure (9) gives reactor power transient response for core flow bypass decrease with a ratio of G/Go = 0.2, for the two time constants of 25 and 10 seconds. The reactor power falls from the steady state value of 22 MW to new steady-state values, as shown in the figure. Since the flow rate is decreasing more rapidly with time during the transient for time constant t= 10 s than for t=25 s, then the temperatures of fuel plate and coolant start to increase faster and the induced negative reactivity for t=10 s is larger, as the reactor has negative reactivity feedback coefficients. So, reactor power decreases faster for the time constant of 10 s by the effect of the feedback reactivity, as shown in Figure (9). Total reactivity feedback for both time constants is given in Figure (10).

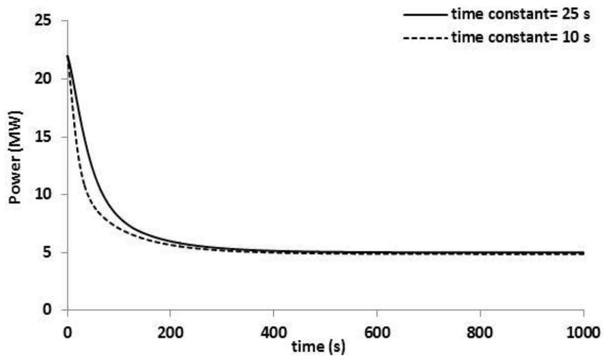


Fig. (9): Reactor power transient response for G/Go = 0.2 for time constants of 25 s and 10 s

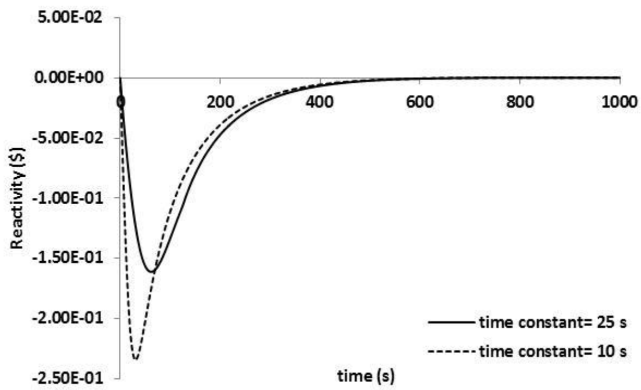
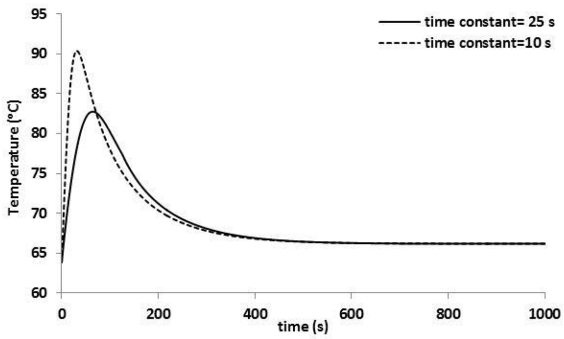
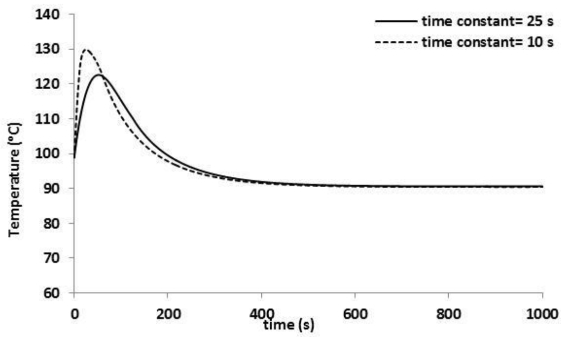


Fig. (10): Total reactivity feedback transient response for G/Go = 0.2 for time constants of 25 s and 10 s

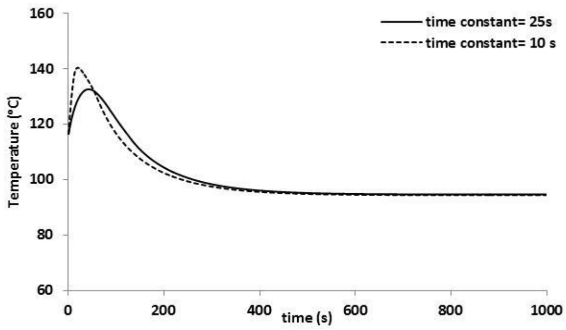
Figures (11-(a), (b), and (c)) indicate the maximum temperatures transient response of coolant, clad and fuel, respectively, for core flow bypass decrease with a ratio of G/Go = 0.2, for the two time constants of 25 and 10 seconds. As shown in the figures, coolant, clad and fuel temperatures increase with flow reduction from the steady state values and reach a maximum then the temperatures drop again and stabilize at new steady state values. The maximum attained values for all temperatures are larger for time constant t= 10 s, since a faster decrease in flow rate is encountered in this case than for t = 25s.



(a)



(b)



(c)

Fig. (11): Transient response of maximum (a) coolant, (b) clad, and (c) fuel temperatures for G/Go = 0.2 for time constants of 25 s and 10 s

The transient response of ONB temperatures for core flow bypass decrease with a ratio of G/Go = 0.2, for the two time constants of 25 and 10 seconds is given in Figure (12). Comparing the curves of Figure (12) by their related surface temperature curves of Figure (11-b) shows that, for both of the two time constants, no subcooled boiling occurs in the reactor hot channel, since onset of nucleate boiling temperature limits still higher than the obtained surface temperatures for both cases.

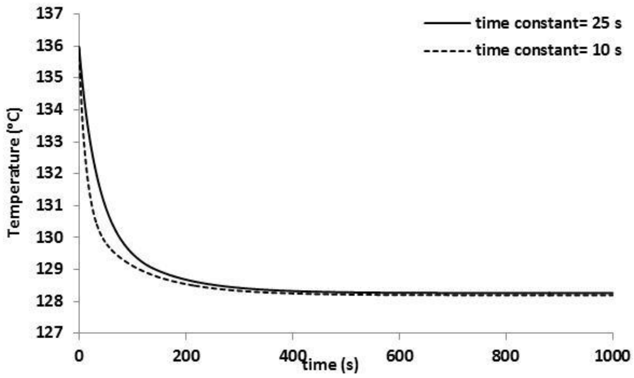


Fig. (12): ONB temperatures transient response for G/Go = 0.2 for time constants of 25 s and 10 s

Figure (13) gives transient response of DNBR for core flow bypass decrease with a ratio of G/Go = 0.2, for the two time constants of 25 and 10 seconds. As the core flow reduces, the reactor core is more stable at the new power levels and higher values of departure of DNBR were achieved. However, lower values of DNBR for time constant t=10s than for t=25 s were obtained as given in Figure (13).

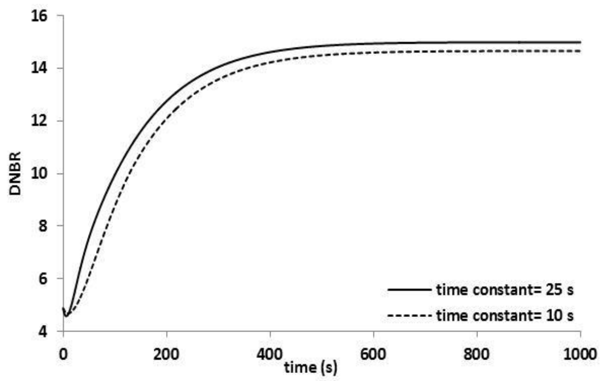


Fig. (13): DNBR transient response for G/Go = 0.2 for time constants of 25 s and 10 s

Figure (14) gives the transient response of flow redistribution ratio for core flow bypass decrease with a ratio of G/Go = 0.2, for the two time constants of 25 and 10 seconds. From this figure it is noticed that the flow redistribution design criterion, of a value 2 for this reactor, is violated for both cases of time constant. Although, the minimum flow instability ratio is more than one for both values of time constant; i.e., fuel integrity criteria in terms of thermal hydraulic instability are not exceeded, but much lower value for flow instability parameter is noticed with time constant t=10 s. So, the reactor stability is more threatened in this case, since the minimum flow instability ratio is very close to 1.

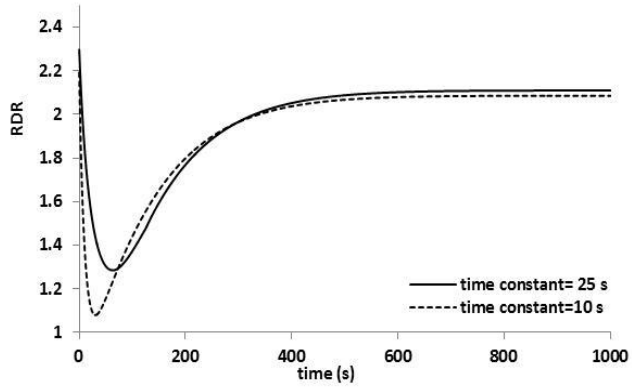


Fig. (14): Transient response of flow redistribution ratio (RDR) for G/Go = 0.2 for time constants of 25 s and 10 s

As seen in analysis of the previous part, decreasing the time constant from 25 to 10 seconds has a significant effect on the transient results of the problem. So, in the next part the effect of changing the time constant with values of 40, 30, 25, 20, 15, 10 and 5 seconds on the operating and safety parameters is given for a core flow of 20% of the nominal core flow rate.

Figure (15) shows the effect of changing the time constant on core flow rate for core flow bypass decrease with a ratio of G/Go = 0.2. As shown from Figure (15) that gradual decrease in flow rate occurs with increasing time constant from 5 s to 40 s. Consequently, the maximum values for coolant, clad and fuel temperatures, during the transient phase, decrease with increasing the time constant as given in Figure (16). As a result, negative reactivity feedback decreases with increasing the time constant. Figure (17) shows peak values for reactivity feedback during the transient phase for different values of time constant for G/Go = 0.2. As the negative reactivity feedback decreases, higher values of reactor power during transient phase are noticed as time constant increases. Figure (18) gives reactor power during steady state phase for different values of time constant for G/Go = 0.2.

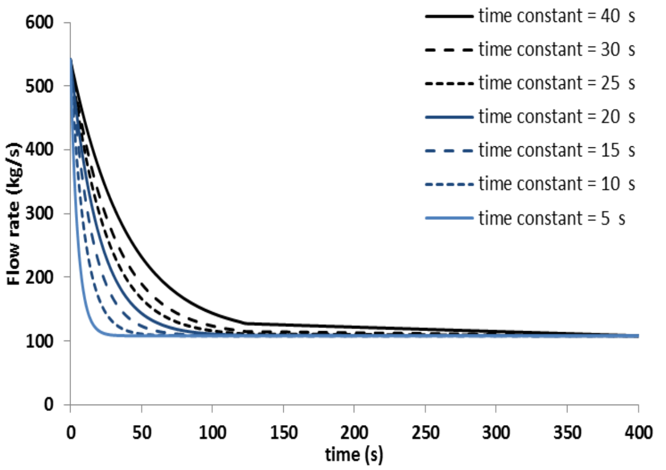


Fig. (15): Transient response of core flow rate for G/Go = 0.2 for different values of time constant

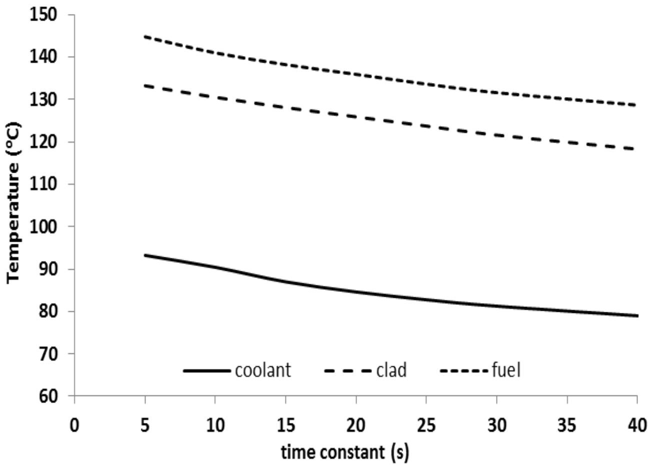


Fig. (16): Maximum coolant, clad, and fuel temperatures for G/Go = 0.2 for different values of time constant

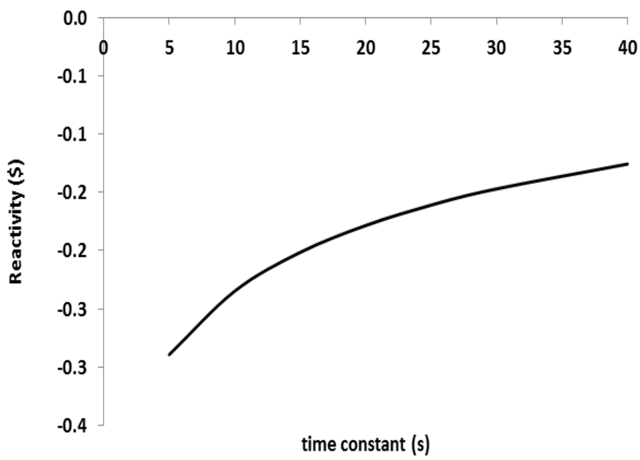


Fig. (17): Maximum negative reactivity feedback for G/Go = 0.2 for different values of time constant

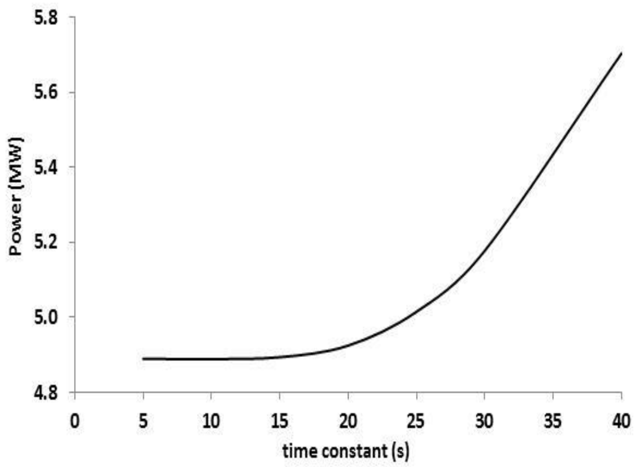


Fig (18): new steady state values for reactor power for G/Go = 0.2 for different values of time constant

Figure (19) gives minimum DNBR for core flow decrease with a ratio of G/Go = 0.2, for different values of time constant. As the time constant increases, gradual decrease in flow rate is encountered. The reactor core becomes more stable at the new power levels and higher values of DNBR were achieved.

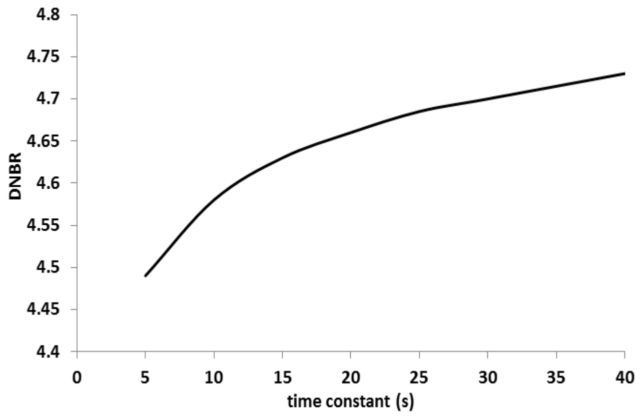


Fig. (19): Minimum DNBR for G/Go = 0.2 for different values of time constant

Figure (20) gives minimum flow redistribution ratio for core flow decrease with a ratio of G/Go = 0.2, for different values of time constant. From this figure it is noticed that the flow redistribution design criterion, of a value 2 for this reactor, is violated for all values of time constant in the prescribed range, for G/Go = 0.2. Although, the minimum flow instability ratio is more than one for time constant values greater than 6; i.e., fuel integrity criteria in terms of thermal hydraulic instability are preserved for time constant values greater than 6. But, much lower values for flow instability parameter are noticed with decreasing time constant. So, the reactor stability is more threatened with time constant decrease.

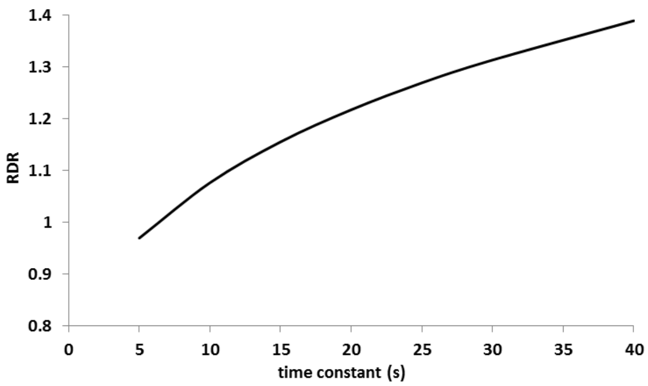


Fig. (20): Minimum flow redistribution ratio (RDR) for G/Go = 0.2 for different values of time constant

7. CONCLUSION

Core flow bypass and partial loss of flow are two issues with similar effects on reactor core cooling. This transient is one of the anticipated operational occurrences that can take place one or several times during the lifetime of the reactor. Many causes can lead to core flow bypass as small breaks of the primary cooling circuit inside reactor tank and flapper valves leakage which are typical flow bypass that can occur in the considered reactor. In the current study, thermal hydraulic behavior of an MTR reactor during core flow bypass, under the conditions of safety system unavailability, is considered. As core bypass takes place, an exponential decrease of core flow rate with a time constant of 25 s is assumed. RELAP5 thermal hydraulic code is used in the current study to perform the required analysis. The study results shows that for cases of core bypass ratios of G/Go = 0.2, 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. Study results were validated by comparing RELAP5 results with published results of PARET code under the same accident conditions. From comparisons, good agreements are found between RELAP5 code results and PARET code results for the maximum fuel, clad, and coolant temperatures, as well as minimum DNBR values, during transient phase and the new steady state phase. Deviations are found between the results of the two codes for flow instability parameter. But, both of the two codes concluded that no exceed of fuel integrity criteria in terms of thermal hydraulic instability is observed.

Also, comparison between reactor behavior under core flow bypass decrease with a ratio of G/Go = 0.2, for two different time constants of 25 and 10 seconds of flow reduction is performed. Since the flow rate is decreasing more rapidly with time during the transient for time constant t= 10 s than for t=25 s, the temperatures of fuel plate and coolant begin to increase faster and the induced negative reactivity for t=10 s is larger, as the reactor has negative reactivity feedback coefficients. So, reactor power decreases faster for the time constant of 10 s by the effect of the feedback reactivity. As a consequent, lower values for flow instability parameter is noticed for time constant t= 10 s and the reactor stability is more threatened in this case, since the minimum flow instability ratio is very close to 1.

Also, the effect of changing the time constant with values of 40, 30, 25, 20, 15, 10 and 5 seconds on the operating and safety parameters is given for a core flow of 20% of the nominal core flow rate. It is concluded from this part that the reactor core became more stable as the time constant increase, since gradual decrease in flow rate is encountered and higher values of safety parameters are noticed. Reactor stability is threatened with time constant decrease, since flow redistribution ratio is smaller than 1 for time constant below 6 seconds.

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