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Reactor Core Behavior of NUR Research Reactor during a Protected Fast Loss of Flow Accident

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Abstract: The best estimate thermal hydraulic codes are the most used code for thermal hydraulic and safety analysis of nuclear reactors, regarding their ability to provide a more realistic nuclear reactor behavior during normal and transient operating conditions. Moreover, the use of such codes is essential in order to ensure safe and reliable operation of nuclear reactors. In this study, for safety assessment purposes, the code system Relap5/mod3.2 was used to predict the thermal-hydraulic behavior of the NUR research reactor during a protected fast loss of flow accident. This was accomplished through the reactor core modeling by using a one-hot channel and an average one as well to present the remaining channels in the reactor core. Then, the transient variation of fuel, clad and coolant temperatures in both channels, further to, the safety criteria: - The critical heat flux ratio (CHFR) and – The onset of Flow Instability Ratio (OFIR) were evaluated and analyzed from safety point of view, where it was concluded that the occurrence of the protected fast loss of flow accident in NUR research reactor does not lead to any safety issue or damage to the fuel. This justified by the fact that, the clad peak values are 73.2°C and 55.3°C respectively for the hot and average channels in their first peak while they become equal to 72.7°C and 62.0°C in their second peak which are far below the limit value (450°C) during transients, in the other hand, the lower values of CHFR are 4.6 and 9.9 respectively for the hot and average channels in their first minimum point while they become equal to 3.4 and 7.7 in their second minimum point which does not fall below the imposed safety limit 1.5 during transients.

1. Introduction

The research reactors are widely used in the world, regarding their crucial role in scientific research and development of the specific use of atomic energy for the benefit of humanity. This includes training and education, radioisotope production, silicon doping, and materials testing (IAEA, 2014). However, more attention should be paid to the safety analysis of research reactors (IAEA, 1992).

For this purpose, many authors used the best estimate system codes, such as, RELAP5 (RELAP5/MOD3 Code Manual, 1999) in order, to perform thermal hydraulic and safety analysis of research reactors during steady state and transient conditions (Hamidouche et al., 2004; Hedayat et al., 2007; Hamidouche et al., 2009; Omar et al., 2010; Chatzidakis et al., 2012; Karimpour & Esteki, 2014; Arshi et al., 2017; Hedayat, 2017; Azzoune et al., 2018; Guo et al., 2018; Rawashdeh et al., 2022; Hedayat & Davari, 2022; Cheridi et al., 2023; Azzoune et al., 2025). In addition to their capability to model the entire primary circuit, the codes system can provide more realistic nuclear reactor behaviour; therefore, more reliable safety analysis can be conducted with this type of thermal-hydraulic codes for nuclear reactors.

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Moreover, the Subchannel thermal hydraulic codes, such as, PARET (Obenchain, 1969), PLTEMP (Kalimullah et al., 2021) and SUBCHANFLOW (Almachi et al., 2022), have been used by many researchers, with the aim to conduct safety analysis for research reactors during normal or abnormal operating conditions (Boulaich et al., 2010; Rahman et al., 2012; Chatzidakis et al., 2014; Hainoun et al., 2014; Hammoud et al., 2014; Boulaich et al., 2015; Nasir et al., 2015a; Nasir et al., 2015b; Ibrahim et al., 2018; Puig & Dennis, 2019; Al-Zain et al., 2023; Almachi et al., 2024; Ferraro et al., 2024; Khakim et al., 2024; Hastuti et al., 2025). These kinds of thermal-hydraulic codes, which are designed for a research reactor, have limited modelling capability in the reactor core; some of them can provide a multidimensional reactor core behaviour, which gives a detailed insight into the temperature distribution within the reactor core; therefore, a deeper safety analysis can be performed.

Regarding the importance of the safety analysis of accident occurrences in nuclear reactors by using best estimate system codes. In the present study, the system code Relap5/mod3.2 is used to perform a detailed thermal-hydraulic and safety analysis of the NUR research reactor during a protected fast loss of flow accident (FLOFA), involving the prediction of the safety margins OFIR and CHFR, which has not yet been done for the MTR research reactor. This was achieved through the reactor core modeling by using two channels, the first one is considered as the hot channel, while the second one is the average channel (Azzoune et al. 2010; Park et al. 2013). Then, the temperature transient variations of fuel, clad and coolant, further to, the safety criteria CHFR and OFIR were evaluated in both channels and analyzed from safety point of view, where it is concluded that the occurrence of the protected fast loss of flow accident (FLOFA) in NUR research reactor does not lead to any significant consequences or damage to the reactor core. Regarding the fact that during the occurrence of the protected FLOFA, the cooling system of the NUR research reactor can ensure adequate reactor core cooling and maintain the clad temperature far below the limit temperature during transients 450°C.

2. Description of NUR research reactor

The NUR research reactor is a 1 MW open pool research reactor, its power is generated by low-enriched uranium MTR fuel type, and the light water is used in the reactor as coolant and moderator. The reactor core is composed of 12 standard fuel elements and 5 control fuel elements as well, which consist of 19 fuel plates and 14 fuel plates, respectively. The reactor is dispose with several irradiation boxes destined to radioisotope production and is surrounded by Graphite blocks used as a neutron reflector (Figure 1).

The NUR research reactor cooling system is composed of two circuits: the primary circuit between the reactor pool and a plate heat exchanger, while the second one is between the plate heat exchanger and the cooling towers (Figure 2). The cooling system ensures the reactor cooling through two different cooling modes: a downward forced convection cooling mode used during normal operating conditions, and also an upward natural convection cooling mode used for decay heat removal after reactor shutdown. During the downward forced convection cooling mode, the pump of the primary circuit circulates the light water through the reactor core at a flow rate of 220m3/h, in order to evacuated it's heat to the secondary circuit via the plate heat exchanger, then, the ordinary water circulates in the secondary circuit, dissipating this heat to the atmosphere through the cooling towers. While, during the natural convection cooling mode, both pumps of primary and secondary circuits are stopped, so the Flap valve is opened, then, the light water in the reactor pool is circulate passively through the reactor core by the buoyancy force resulted from the water density difference between both core and pool regions, so this circulation permit to the water to evacuated the heat from the reactor core to the water volume in the reactor pool.

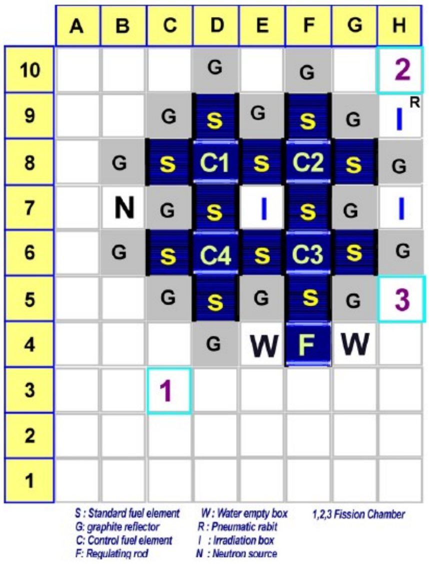

Fig. 1. Cross-sectional view of the reactor core (Azzoune et al., 2010).

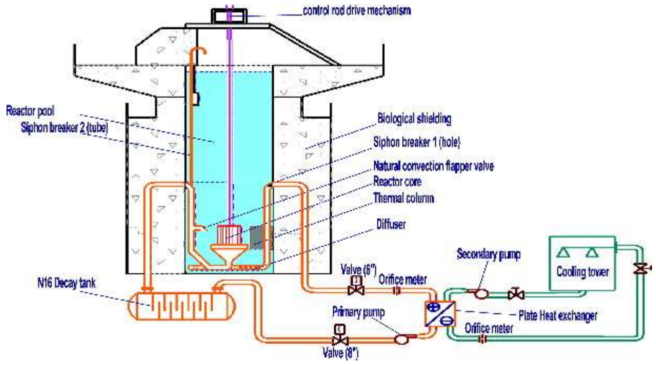

Fig. 2. Scheme of the NUR primary and secondary cooling circuits (Hammoud et al., 2014).

The main design data, fuel dimensions, and operating conditions of the NUR research reactor are summarized in Table 1.

Table 1. Summary of the NUR research reactor main operating and design data (Hammoud et al., 2014; Azzoune et al., 2010).

Core material	
Nuclear fuel type	MTR
Coolant/Moderator	Light water
Reflector	Graphite/light water
Material of nuclear fuel/Enrichment	U_3O_8-Al/19.75%
Number of standard fuel elements (SFE)	12
Number of control fuel elements (CFE)	5
Number of fuel plats per SFE	19
Number of fuel plats per CFE	14
Core thermal-hydraulics	
Fuel thermal conductivity (W/m K)	15
Clad thermal conductivity (W/m K)	180
Inlet coolant temperature (°C)	40
Inlet coolant pressure (bar)	1.7
Total flow rate	220 m^3/h
Fuel assembly dimensions	
Active length (mm)	615
Meat width (mm)	60
Meat thickness (mm)	0.7
Clad thickness (mm)	0.4
Plate thickness (mm)	1.5
Channel width (mm)	66
Channel thickness (mm)	2.7
Core kinetics	
Effective delay neutron fraction	0.008
Prompt neutron generation time (μs)	65.46
Void feedback coefficient ($/%void)	0.342
Doppler feedback coefficient ($/°C)	-2.54E^{-3}
Coolant temperature feedback($/°C)	-2.30E^{-2}

3. Relap5 model for NUR reactor core

This model focused only on the core and pool regions as presented in Figure 3. These two regions are modeled by using different hydrodynamics components such as pipes, Branches and single volumes. The core region is modeled by a one-hot channel P203 represents the hottest channel, and an average one as well P204 represents the whole remaining channels in the reactor core, while the pipe P051 is used for the Bypass. These pipes are connected to each other at the top and the bottom by using the single volumes (SV036 and SV037). The generated heat from the fuel plates in the hot channel and the average one are modeled by using two different heat structures, where both radial and axial peaking factors are calculated separately by using a neutronic code (Meftah et al. 2006; Azzoune 2007). The pool region is mainly modeled by the pipe P255 and the single volume SV257, while the branch B256 is used as a connection between them. The Flap valve of the natural circulation is modeled by a Trip valve, which ensures water circulation between the pool and the reactor core during the natural convection cooling mode. Pipe P211 presents the aspiration pipe of the primary circuit. Moreover, the time-dependent volume TDV400 was used to define the atmospheric conditions at the top of the reactor pool, while the time-dependent volumes TDV510 and TDV520 were used to define the reactor operating conditions (pressure and temperature).

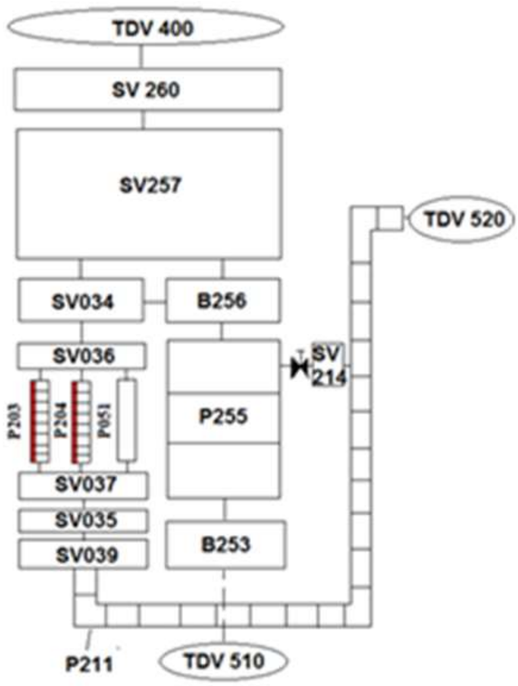

Fig. 3. Scheme of NUR reactor core Nodalisation by the code system Relap5/mod3.2.

4. The Safety Assessment of Nuclear Reactors

The safety assessment of nuclear reactors during steady state and transient conditions, requires the evaluation of safety criteria, such as (ONBR, OFIR, CHFR,..), which allow us, to verify that fuel overheating especially during transient conditions, will not lead to the occurrence of some critical phenomena, that could cause a rapid degradation of the heat removal from the reactor core and consequently fuel damage. The following things were considered in this study.

4.1. The Flow Instability Ratio

The flow instability ratio (OFIR) is defined by the ratio between the heat flux of onset of flow instability (q''_{OFI}) and the local heat flux (q''_z) (INVAP, S. E 2001).

$$\mathrm{OFIR} = \frac{q''_{OFI}}{q''_z} > \begin{cases} 2, for\ steady\ condtions \\ 1.3, for\ transien\ conditions \end{cases}$$

To determine the heat flux of the onset of flow instability, Whittle & Forgan's correlation (1967) is used (Olson & Kalimullah, 2018).

$$q''_{OFI} = \frac{R\,\rho_c\,c_c\,v_c\,D_h}{4L_h}\,(T_{sat} - T_{inlet})$$

$$R = \frac{1}{1 + \eta \frac{D_h}{L_h}} \tag{1}$$

where $q''_{OFI}\left(\frac{w}{m^2}\right)$ is the onset flow instability heat flux, $\rho_c\left(\frac{Kg}{m^3}\right)$ is the coolant density, $c_c\left(\frac{J}{Kg\,°K}\right)$ is the coolant heat capacity, $v_c\left(\frac{m}{s}\right)$ is the coolant velocity, $D_h\ (m)$ is the hydraulic diameter, $L_h(m)$ is the channel heated length, and η is a coefficient that can be taken equal to 25 or calculated by $\eta = 3.15\,(1.08\ \mathrm{G})^{0.29}$(INVAP, S. E 2001).

4.2. The Critical Heat Flux Ratio

The critical heat flux ratio (CHFR) is defined by the ratio between the critical heat flux (q''_{CHF}) and the local heat flux (q''_z)(INVAP, S. E 2001),

$$\mathrm{CHFR} = \frac{q''_{CHF}}{q''_z} > \begin{cases} 2, for\ steady\ condtions \\ 1.5, for\ transien\ conditions \end{cases}$$

To evaluate the critical heat flux, the Kaminaga Correlation (Jo et al., 2013) is used.

$$q''^{*}_{CHF,1} = 0.005|G^*|^{0.611}\left(1 + \frac{5000}{|G^*|}\Delta T^*_{sub,out}\right) \tag{2}$$

$$q''^{*}_{CHF,2} = \frac{A_f}{A_h}\Delta T^*_{sub,in}|G^*| \tag{3}$$

$$q''^{*}_{CHF,3} = 0.7\frac{A_f}{A_h}\frac{\sqrt{\frac{W}{\lambda}}}{\left[1 + \left(\frac{\rho_g}{\rho_f}\right)^{\frac{1}{4}}\right]^2}\left(1 + 3\Delta T^*_{sub,in}\right) \tag{4}$$

Where: $q''^{*}_{CHF} = \frac{q''_{CHF}}{\left(h_{fg}\sqrt{\lambda(\rho_f - \rho_g)\rho_g g}\right)}$, $G^* = \frac{G}{\sqrt{\lambda(\rho_f - \rho_g)\rho_g g}}$, $\Delta T^*_{sub} = \left(\frac{C_p\,\Delta T_{sub}}{h_{fg}}\right)$, $\lambda = \left(\frac{\sigma}{(\rho_f - \rho_g)\rho_g}\right)^{1/2}$, and, A_f is the flow area and A_h is the heated area.

For downward flows

If $G^* \geq G^*_1$, then $q''_{CHF} = \min\,(q''_{CHF,1}, q''_{CHF,2})$

If $G^* < G^*_1$, then $q''_{CHF} = \max\,(q''_{CHF,2}, q''_{CHF,3})$

For upward flows

If $G^* \geq G^*_1$, then $q''_{CHF} = \min\,(q''_{CHF,1}, q''_{CHF,2})$

If $G^* < G^*_1$, then $q''_{CHF} = \max\,(q''_{CHF,1}, q''_{CHF,3})$

With $G^*_1 = \left(\frac{0.005}{\left(\frac{A_f}{A_h}\right)\Delta T^*_{sub,in}}\right)^{\frac{1}{0.389}}$, and when, $G^* < G^*_1$, $\Delta T^*_{sub,out}$ in the equation (2) should be set to zero.

5. The Fast Loss of Flow Accident (FLOFA)

During this event, it was expected that the mass flow rate would decrease exponentially according to the equation (Hammoud et al., 2014), $\dot{m} = \dot{m}_0 \exp\left(-\frac{t}{\tau}\right)$, and when the mass flow scram condition is reached, after we lose an amount of 20% from the initial mass flow rate. Then a reactivity rate of -10$/0.5s is inserted in the reactor core with a delay time of 0.25s, so the reactor power will drop from the initial operating power, which is taken to be 1.2 MW (120% of the nominal power), until the reactor power is generated from decay heat only and the reactor has sustained in the shutdown state. However, the flow rate continues to decrease, and when 80% has been lost from the initial mass flow rate, the natural convection valve is opened to allow the inversion of flow and passive cooling of the reactor core. The parameter (τ) is the time constant for the pump coast down, which is taken equal to 2.2s (Hammoud et al., 2014).

6. Results and discussions

After the FLOFA is initiated at t=5s, the mass flow rate decreased rapidly through the reactor core, which led to temperature increases in both hot and average channels. In Figure 4, the temperature transient variation of fuel, clad, and coolant is presented for the hot channel. It’s obvious that the temperature profiles characterized by two peaks. At the first peak, which occurs at the scram point (t=5.9s), the temperature of fuel, clad, and coolant reach their highest values 75.0°C, 73.2°C, and 53.6°C, respectively. After that, they dropped sharply proportionally to the reactor power, before increasing again due to the effect of decay heat until reaching the maximum values 72.8°C, 72.7°C, and 64.9°C, respectively, in the second peak. Thereafter, they are decreased slowly until stabilizing after the establishment of the natural convection cooling mode.

Figure 5 shows the temperature transient variation of fuel, clad, and coolant in the average channel. It is clear that their temperature profiles are similar to those of the hot channel, where the temperature at the scram point is equal to 56.1°C, 55.3°C, and 46.1°C, respectively, for fuel, clad, and coolant, while they become equal to 62.0°C for fuel and clad and 58.4°C for the coolant in the second peak.

Figure 6 shows the rector power transient variation, which drops sharply from its nominal value (1.2MW) after scram initiation, to reach its minimal value after reactor shutdown. Even after reactor shutdown, there is heat generated by the fuel (decay heat), which is about 50 kW; this amount of heat is evacuated from the NUR reactor core by the natural convection cooling mode.

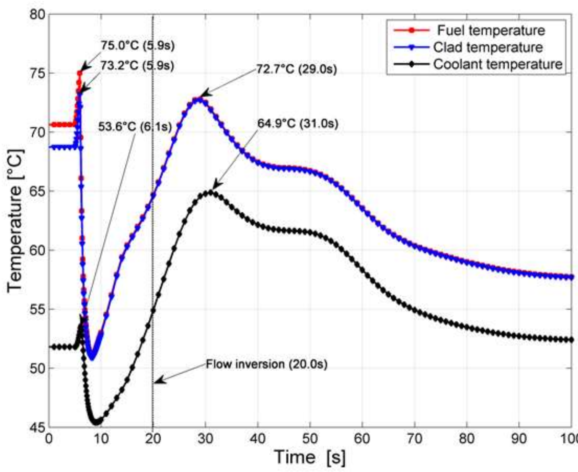

Fig. 4. The temperature transient variation in the hot channel.

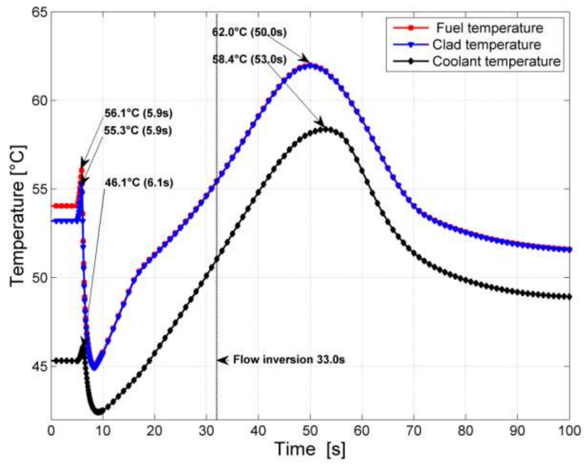

Fig. 5. The temperature transient variation in the average channel.

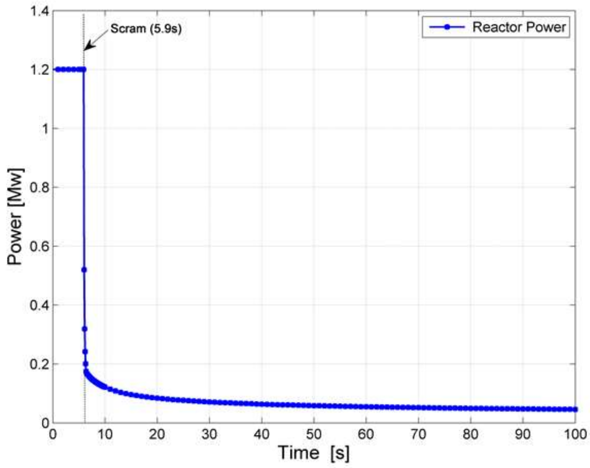

Fig. 6. The reactor power transient variation during FLOFA.

Figures 7 and 8 illustrate the transient variation of mass flow rate, respectively, in the hot and the average channels. After the accident initiation at t=5.0s, the mass flow rates in both channels decreased rapidly from their

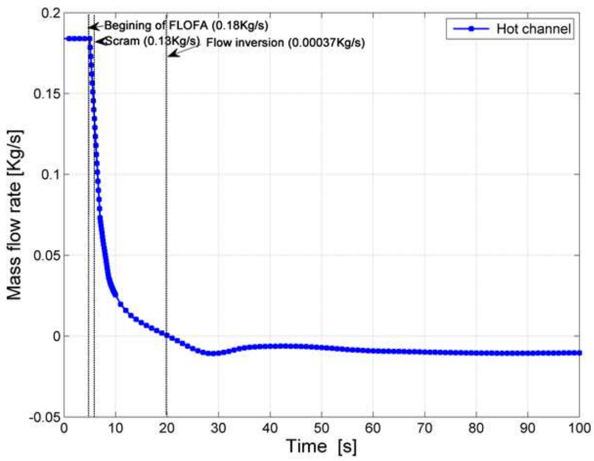

Fig. 7. The mass flow rate transient variation in the hot channel.

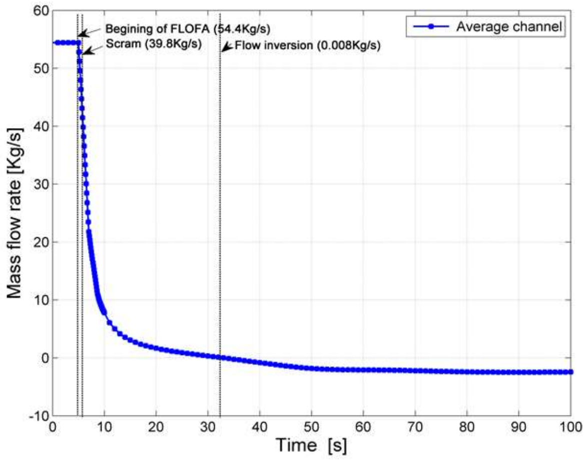

Fig. 8. The mass flow rate transient variation in the average channel.

nominal values which are about 0.15350 kg.s^{-1} and 55.0 kg.s^{-1}, respectively in the hot and average channels, but after the flow inversion they stabilized at their minimum values which are about 0.01 kg.s^{-1} for the hot channel and 2.5 kg.s^{-1} for the average one.

Figures 9 and 10 show the transient variation of the safety criteria CHFR and OFIR in the hot and the average channels. It's obvious that both safety criteria decrease slightly after the accident initiation until they become equal to 4.9 and 9.9 for CHFR and 2.6 and 5.8 for OFIR, respectively, in the hot and average channels at the scram point (5.9s). After this point, they sharply increased inversely proportional to the reactor power, before decreasing again proportionally to the mass flow rate, until reaching their lowest values at the flow inversion moments, to become equal to 3.4 and 7.7 for CHFR respectively in the hot and average channels, while the OFIR drops below the safety limit 1.3, which indicate that a

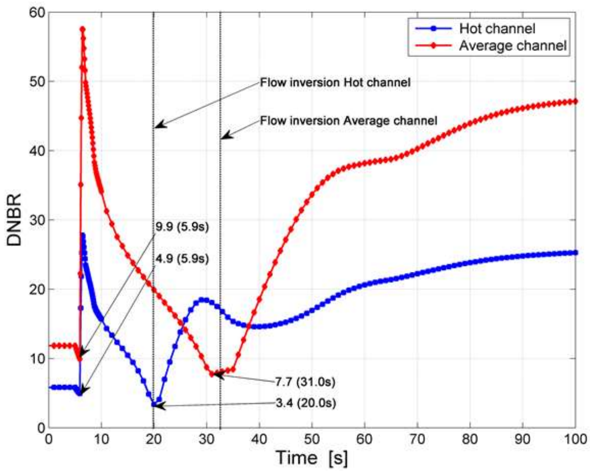

Fig. 9. The transient variation of the minimum of CHFR in the hot and average channels.

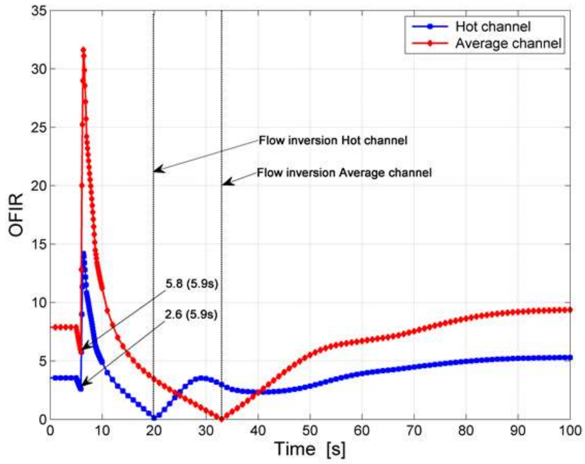

Fig. 10. The transient variation of the minimum of OFIR in the hot and average channels.

flow instability has occurred, that due to the flow inversion from downward forced convection to upward natural convection cooling mode.

Figures 11 and 12 present the transient variation of the heat fluxes q_{CHF}, q_{OFI} and the maximum q_z in both the hot and the average channels. It's evident that the q_z variation is proportional to the reactor power, while q_{CHF} *and* q_{OFI} are proportional to the mass flow rate. After the scram point (5.9s), q_z drop sharply and become almost constant, while q_{CHF} *and* q_{OFI} are decreased rapidly until they reach their lowest values at flow inversion moments, which are 20.s and 33.0s, respectively, in the hot and average channels.

Table 2 shows a summary of the interest parameters variation in both channels during the FLOFA, including fuel, clad, and coolant temperatures, further to the safety criteria OFIR and CHFR. All these parameters are character-

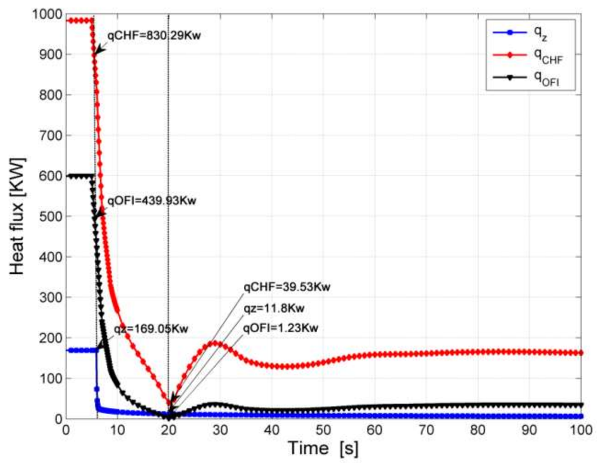

Fig. 11. The transient variation of the heat fluxes q_z, q_{CHF} and q_{OFI} in the hot channel.

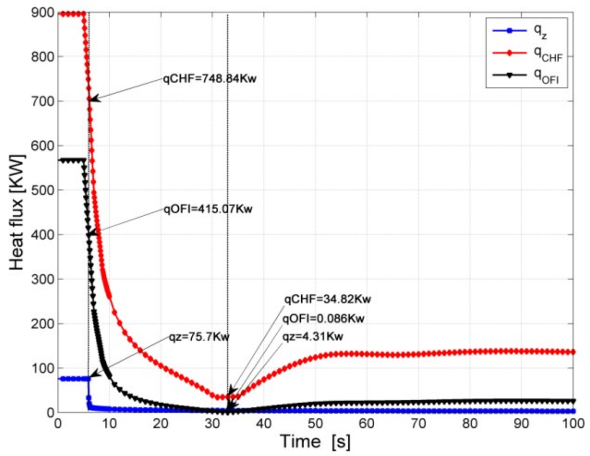

Fig. 12. The transient variation of the heat fluxes q_z, q_{CHF} and q_{OFI} in the average channel.

Table 2. Summary of results analysis during FLOFA.

	Hot channel	Average channel
Trip time (s)	5.9	
Power at scram (MW)	1.2	
1st peak Fuel temperature (°C)	75.0 (5.9s)	56.1 (5.9s)
1st peak Clad temperature (°C)	73.2 (5.9s)	55.3 (5.9s)
1st peak Coolant temperature (°C)	53.6 (6.1s)	46.1 (6.1s)
1st minimum of OFIR	2.6 (5.9s)	5.8 (5.9s)
1st minimum of CHFR	4.9 (5.9s)	9.9 (5.9s)
Flow inversion time (s)	20.0	33.0
2nd peak Fuel temperature (°C)	72.8 (29.0s)	62.0 (50.0s)
2nd peak Clad temperature (°C)	72.7 (29.0s)	62.0 (50.0s)
2nd peak Coolant temperature (°C)	64.9 (31.0s)	58.4 (53.0s)
2nd minimum of OFIR	0.1(20.0s)	0.02 (33.0s)
2nd minimum of CHFR	3.4 (20.0s)	7.7 (31.0s)

-rized during the FLOFA either by two peaks, like the temperatures, or by two minimum points, like the safety criteria. After the initiation of the accident at t=5.0s, the mass flow rate decreased rapidly, this yield to a fast increase in the fuel, clad, and coolant temperatures, where they became equal to 75.0°C, 73.2°C, and 53.6°C, respectively in the hot channel, and 56.1°C, 55.3°C and 46.1°C in the average channel at the first peak. However, when the mass flow rate decreases below 20% of their initial value 54.4Kg/s, the reactor scrammed, so the reactor power is sharply dropped, witch lead to a sharp drop of fuel, clad and coolant temperatures as well in both channels, but they increased again due to the decay heat effect, until they reaching their second peak after flow inversion, where they respectively becomes equal to 72.8°C, 72.7°C and 64.9°C in the hot channel and 62.0°C, 62.0°C and 58.4°C in the average channel.

For the safety criteria OFIR and CHFR, their variation during the FLOFA is inversely proportional to the temperatures. Therefore, they characterized by two minimum points rather than two peaks. However, this study is more concerned about the CHFR, which must not fall below the imposed safety limit of 1.5, to avoid any safety issue in the reactor. At the first minimum point, which occurs at the scram point (5.9s), the CHFR is equal to 4.9 and 9.9, respectively, in the hot and average channels. While the lowest values are reached at the second minimum point after reactor scram, to be equal to 3.4 and 7.7, respectively, in the hot and average channels.

7. Conclusion

This work aims to study the occurrence of a protected fast loss of flow accident (FLOFA) in the NUR research reactor by using the code system Relap5/mod3.2. The use of such code allows us to predict the reactor core thermal-hydraulic behavior, such as the transient variation of the coolant, clad, and fuel temperatures, further to the mass flow rate and reactor power. This step is essential for the safety assessment of the NUR research reactor during the occurrence of the considered accident. For this purpose, the NUR research reactor was modeled by the code system Relap5/mod3.2, according to the nodalization scheme presented in Figure 3, and more specifically, the reactor core was modeled by using a one-hot channel and an average one as well to present the remaining channels in the reactor core. Then, the transient variation of the coolant, clad, and fuel temperatures, the mass flow rate, and the reactor power, further to the safety criteria CHFR and OFIR in the hot and average channels, were graphically presented and discussed.

The temperature profiles of coolant, clad, and fuel were mainly characterized by two peaks, the first one occurring at the scram point, while the second occurs after flow inversion. However, the clad temperature at the two peaks for both hot and average channels stayed well below 450°C, which is considered the safety limit of research reactors during transients.

The safety criteria CHFR and OFIR were mainly characterized by two minimum points, the first one occurred at the scram point, while the second occurred at the flow inversion moment. However, the CHFR did not fall below the imposed safety limit (1.5) in both channels, contrary to the OFIR, which fell below the safety limit (1.3) at t=20.0s and t=33.0s, which correspond to the points of flow inversion, respectively, for the hot and average channels.

However, this safety analysis focused only on the CHFR safety criteria, which unexpectedly reached their lowest values, although the reactor was in a shutdown state; nevertheless, is not fall below the imposed safety limit (1.5). Therefore, the occurrence of the protected fast loss of flow (FLOFA) in the NUR research reactor would not cause any safety issue or fuel damage.

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Nomenclature

A	Areas, m^2
C	Coolant heat capacity, J kg^{-1} °K^{-1}
D	Hydraulic Diameter, m
L	Length, m
m	Mass flow rate, kg s^{-1}
G	Mass flux, kg m^{-2} s^{-1}
g	Gravitational Acceleration, m s^{-2}
v	Coolant velocity, m s^{-1}

Greek symbols

ρ	Coolant density, kg m^{-3}
σ	Water Surface Tension, Nm^{-1}
τ	Time constant, s
η	Coefficent

Subscripts

c	Coolant
f	Fuel
g	water vapor
h	Heated
*	Dimensionless
Sub	Sub cooled

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