ACTA TECHNICA CORVINIENSIS – Bulletin of Engineering | e–ISSN: 2067 – 3809
Tome XIX [2026] | Fascicule 1 [January – March]

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ANALYSIS OF LOSS OF FLOW ACCIDENT IN A TYPICAL RESEARCH REACTOR WITH DELAYED SCRAM SIGNAL

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Abstract: The reliable performance of safety systems and inherent design features is essential to ensure that the nuclear reactor core remains within acceptable thermal–hydraulic limits during both normal operation and off–normal conditions. This work examines the behaviour of a typical material test reactor (MTR) subjected to a loss of flow accident (LOFA) while introducing SCRAM signals after different delay intervals. Reactor responses were evaluated using the PARET/ANL code for SCRAM times of 5 s, 10 s, 20 s, 40 s, and 60 s following pump coast–down. The study assesses the evolution of fuel, cladding, and coolant temperatures, along with the transient characteristics of the flow instability ratio and critical heat–flux ratio. Results demonstrate the sensitivity of thermal margins to delayed shutdown actions and highlight the importance of timely reactivity insertion in mitigating LOFA consequences.
Keywords: LOFA, Flow instability, Thermal–hydraulics, PARET and MTR reactors

INTRODUCTION

Maintaining adequate safety margins is essential to the design and licensing of research reactors. Safety analyses typically address a spectrum of postulated events, including reactivity–related transients and accidents involving reduced coolant flow. Reactivity insertion events are commonly classified as either protected transients, where shutdown systems act promptly, or unprotected transients, which rely solely on inherent feedbacks. While unprotected reactivity events have been widely studied to establish safety thresholds, fewer investigations have focused on unprotected or delayed–SCRAM loss of flow accidents (LOFAs). A LOFA can arise from pump malfunction, loss of electrical supply, blockage within the primary loop, or mechanical failure. During pump coast–down, the reactor gradually loses its forced convection heat removal capability and becomes increasingly dependent on natural circulation. If natural convection cannot adequately remove heat, boiling may occur within the channels. Previous studies, including those by Housiadas [1], Muhammad [2], and Kazeminejad [3], have addressed LOFA behavior in different reactor designs using PARET or other thermal–hydraulic models. Additional investigations—such as those by Salama and El–Morshedy [4-6] have analyzed flow blockage, flow inversion, and transient thermal–hydraulic features in MTR–type reactors. Building upon these efforts, the present study evaluates LOFA progression in a typical MTR reactor, specifically examining the influence of delayed SCRAM initiation. The analysis is performed using the PARET/ANL code, which provides coupled neutronic and thermal–hydraulic modelling capabilities.

CORE CONFIGURATION

The reactor modelled in this work is a standard MTR–type unit cooled by upward–flowing light water and surrounded by a beryllium reflector. The rated power is 22 MW. Fuel elements consist of aluminum–clad U_3O_8-Al dispersion plates arranged in a 5×6 core lattice containing 29 elements and a cobalt irradiation box. Each fuel element contains 19 plates within an 8×8 cm cross–section and an active length of 80 cm. Coolant circulation is provided through two primary pumps and two heat exchangers. Shutdown mechanisms include bottom–mounted control rods and a pneumatic fast–insertion system. Key reactor specifications are presented in Table 1.

PARET MODELING

The transient is initiated from steady-state operation due to pump failure, causing a loss-of-flow accident (LOFA) with no safety or shutdown systems active. The coolant flow decays exponentially with a 25 s time constant (Figure 1), and the axial heat flux in the core follows a chopped cosine distribution, with the hot channel representing the peak flux.

$$q(z) = q_0 \cos\left(\frac{\pi(z - L/2)}{L_p}\right) \quad (1)$$

Where $L_p = L + 2e$, q_0 is the maximum axial heat flux and is given by:

$$q_0 = q_{ave} * PPF_T \quad (2)$$

q_{ave} is the average surface heat flux in the core, PPF_T is the total power peaking factor and e is the extrapolated distance.

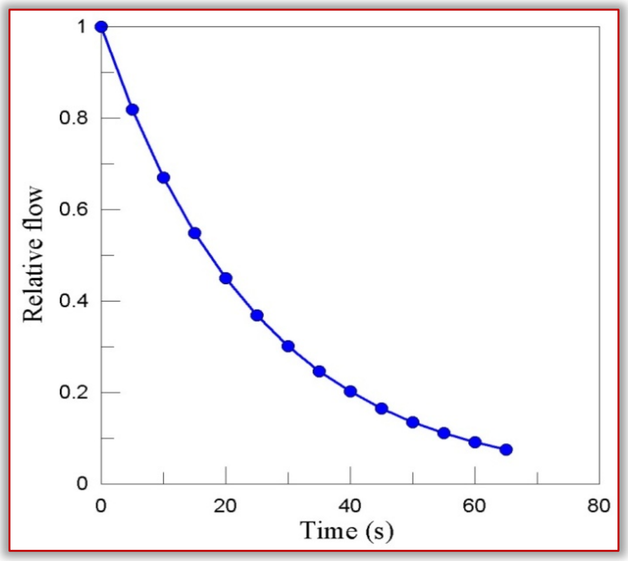

Figure 1. Relative flow coast–down in the reactor core

The specified conditions for the unprotected loss-of-flow transient are summarized in Table 2. The PARET code and simulation methodology have been validated in previous studies [7] through comparison with FLUENT CFD results. Thermal-hydraulic and transient analyses were performed using PARET [8], a code originally developed for power reactors to evaluate the impact of nondestructive accidents in research and test reactor cores. PARET integrates neutronics, hydrodynamics, and heat transfer using point kinetics, one-dimensional hydrodynamics, and one-dimensional heat conduction. Initially developed for SPERT-III experiments [9], the code was later enhanced by Woodruff [10] to include flow correlations and a properties library suitable for the lower pressure, temperature, and flow conditions characteristic of research reactors.

Table1. Reactor main data [11]

Parameter	value
Power, (MW)	22
Coolant	Light water
Direction of flow	upward
Inlet temperature, (°C)	40
Core flow rate, (m3/h)	1900
Level of water in the pool(m)	10.4
Thermal conductivity of fuel (W/m .K)	15
Thermal conductivity of clad (W/cm K)	300
Operating pressure (Bar)	2
Power peaking factor	3
Prompt neutron lifetime (Λ) (μs)	75
Effective delayed neutron fraction (βeff)	0.00705
Fuel temperature feedback coefficient ($/°C)	−3.12 ·10−3
Coolant temperature feedback coefficient ($/°C)	−1.3·10−2
Void reactivity feedback coefficient ($/%void)	−0.2935
Available shutdown reactivity worth ($)	−10
Nominal core outlet temperature (°C)	50
Core flow reduction rate	exp(−t/25)
Reactor SCRAM initiation point	SCRAM disabled

The core was modelled using a two-channel approach: one channel represented the hot plate and its flow path, while the other averaged the remaining fuel plates. The axial power distribution was divided into 21 regions and approximated by a chopped cosine profile, with a total power peaking factor of 3 for the hot channel. For computational simplicity, all reactivity feedback coefficients were treated using linear approximations. The PARET code employed the Dittus-Boelter correlation for single-phase flow, the McAdams correlation for two-phase flow, and its original models for transient two-phase flow and departure from nucleate boiling (DNB) calculations [8].

Table 2. Conditions specified for the loss of flow accident.

Condition	Value
Initial reactor power, Mw	22
Flow coast down rate	exp(−t/25)
Reactor scram initiation point	SCRAM after delayed time
Total power peaking factor of the hot channel	3

Reactor point kinetic model

PARET uses point kinetics to compute reactor power and includes feedback effects in the total reactivity. For this study, six delayed neutron groups are considered. In unprotected loss-of-flow transients, where no external reactivity is applied, the reactor power is governed entirely by feedback-driven reactivity changes.

$$\frac{dP(t)}{dt} = \frac{\rho(t) - \beta}{\Lambda} P(t) + \sum_{i=1}^{n} \lambda_i C_i(t) + S(t) \quad (3)$$

$$\frac{dC(t)}{dt} = \frac{\beta f_i}{\Lambda} P(t) - \lambda_i C_i(t) \quad (4)$$

where $\rho(t)$is the reactivity of the system as a function of time, β is the effective delayed neutron fraction, Λ is the prompt neutron generation time, $\lambda_i$decay constant for group i, $C_i$concentration of delayed neutron precursors of group i, $f_i$is the fraction of delayed neutrons of group $i, \beta_i/\beta$, and $P(t)$is the reactor power as a function of time.

Thermal hydraulics

The coolant flow in the channel gaps between fuel plates is analyzed using the modified momentum-integrated model, which is based on the conservation equations for mass, momentum, and energy.

$$\rho'' \frac{\partial E}{\partial t} + G \frac{\partial E}{\partial z} = q \quad (5)$$

$$\frac{\partial \bar{\rho}}{\partial t} = -\frac{\partial G}{\partial z} \quad (6)$$

$$\frac{\partial G}{\partial t} + \frac{\partial}{\partial z}\left(\frac{G^2}{\rho'}\right) = -\frac{\partial p}{\partial z} - \left(\frac{f}{\rho}\right)\left(\frac{|G|G}{2D_e}\right) - \bar{\rho} g \quad (7)$$

where

$$\bar{\rho} = \rho_l(1-\alpha) + \rho_v \alpha \quad (8)$$

$$\frac{1}{\rho'} = \frac{(1-x^2)}{\rho_l(1-\alpha)} + \frac{x^2}{\rho_v\alpha} \quad (9)$$

$$\rho'' = [\rho_l x + \rho_v(1-x)]\frac{\partial\alpha}{\partial x} \quad (10)$$

where G is the mass flow rate, p is the pressure, E is the enthalpy, f is the friction factor, D_e is the equivalent hydraulic diameter, ρ is the average density, ρ' is momentum density, ρ'' is the slip flow density, x is the vapor weight fraction (quality), α is the vapor volume fraction (void fraction).

The PARET heat transfer model uses a simplified one-dimensional approach, with heat conducted radially from the fuel through the cladding into the coolant. Heat transfer within the fuel plates is determined by solving the radial conduction equation.

$$\frac{\partial(\rho c_p T)}{\partial t} = \frac{\partial}{\partial x}\left(k\frac{\partial T}{\partial x}\right) + f_s S \quad (11)$$

where T is the temperature, x is the radial coordinate, ρc_p is the volumetric heat capacity, k is the thermal conductivity, S is the heat source per unit volume, and $f_s$heat source flag such that in $f_s = 1$ in the fuel meat and $f_s = 0$ in the clad. Axial heat conduction along the fuel plate length is neglected, as all heat in the PARET model is assumed to transfer radially through the cladding into the coolant.

A key aspect of the heat conduction solution is the evaluation of the convective heat transfer coefficient. PARET incorporates thermal-hydraulic correlations to determine heat transfer coefficients across different regimes, including natural convection, single-phase forced convection, nucleate boiling, and transient film boiling.

RESULTS

The PARET code employs a two-channel model, with one channel representing the highest-temperature region in the reactor, referred to as the hot channel. The remaining channels, including the average channel, operate at lower temperatures. Consequently, if the hot channel satisfies the operational limits and conditions (OLC), all other channels are also expected to comply. For this reason, the analysis and results focus exclusively on the hot channel.

Steady state

Before the transient, the reactor core is at steady-state, with maximum fuel and cladding temperatures, coolant outlet temperature, DNBR, and reactor power listed in Table 3.

Table 3. Steady state thermal hydraulic data of the hot channel.

Parameter	Value, PARET
Reactor Power (MW)	22
Tmax, fuel (°C)	112.29
Tmax, clad (°C)	96.947
T_{out}, max (°C)	64.11
DNBR	6.2427

Transient state

The transient begins from steady-state reactor operation, with the coolant flow rate decreasing exponentially with a time constant of 25 s. The unprotected loss-of-flow accident (LOFA) sequence involves a reduction in reactor flow rate due to pump coast-down, an increase in the power-to-flow ratio (P/G), and a rise in core temperatures, which induces negative reactivity.

— Onset of nucleate boiling (ONB) limit

The onset of nucleate boiling (ONB) is considered as the first indication of the potential for critical phenomena and the heat flux that initiates ONB is frequently used as a thermal design constrain. The ONB is taken as a warning in steady state conditions as doesn't actually correspond to any critical event. The time from steady state up to the onset of nucleate boiling for different delayed SCRAM signal are shown in Table 4.

Table 4 Time to reach onset of nucleate boiling (ONB) for different delayed SCRAM signals

Time for delayed SCRAM signals	5	10	20	40	60
Time of ONB	–	–	–	26.121 – 40.854 s	26.121– 56.91 s

— Delayed SCRAM Scenarios

A key safety function of the MTR reactor is to safely shut down the reactor and mitigate accident consequences. This study investigates the impact of delayed SCRAM signals on the progression of a loss-of-flow accident, with delays of 5, 10, 20, 40, and 60 s. Reactor shutdown is achieved by inserting a negative reactivity of -4$ into the core, which is conservative compared to the first shutdown system's available margin of -10$. Delayed SCRAM signals may arise from response latency in the reactor protection system (RPS) or be manually initiated by the reactor operator.

— Fuel, clad and coolant temperatures during different delayed SCRAM signals

As coolant flow decreases before shutdown, fuel, cladding, and coolant temperatures rise, inducing negative reactivity feedback that reduces reactor power. In this self-limiting transient, power is governed solely by feedback effects from fuel (Doppler), void, and coolant temperatures. The Doppler effect broadens absorption resonances, increasing neutron capture, while higher moderator temperatures harden the neutron spectrum, enhancing resonant absorption in ^{235}U, ^{239}Pu, and ^{232}Th. Thermal expansion of the coolant further reduces moderation, collectively contributing to the negative reactivity and limiting the transient.

Figures (2-4) depict the transient responses of maximum fuel, cladding, and coolant temperatures for delayed SCRAM times of 5, 10, 20, 40, and 60 s. Cladding and coolant temperatures rise from their steady-state values prior to the transient and continue to increase until the reactor trip, at which point fuel, cladding, and coolant temperatures drop sharply. Following the reactor trip, as the coolant flow continues to decline, temperatures rise again to a second peak before decreasing to a new steady-state level. Table 5 summarizes the peak cladding and coolant temperatures immediately before and after the reactor trip, as well as the second peak observed post-trip.

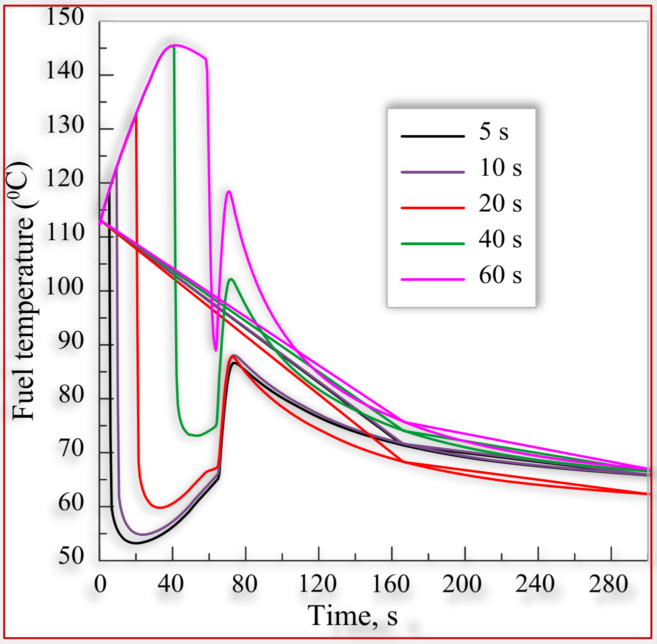

Figure 2. Transient response of fuel temperature for different delayed SCRAM signals

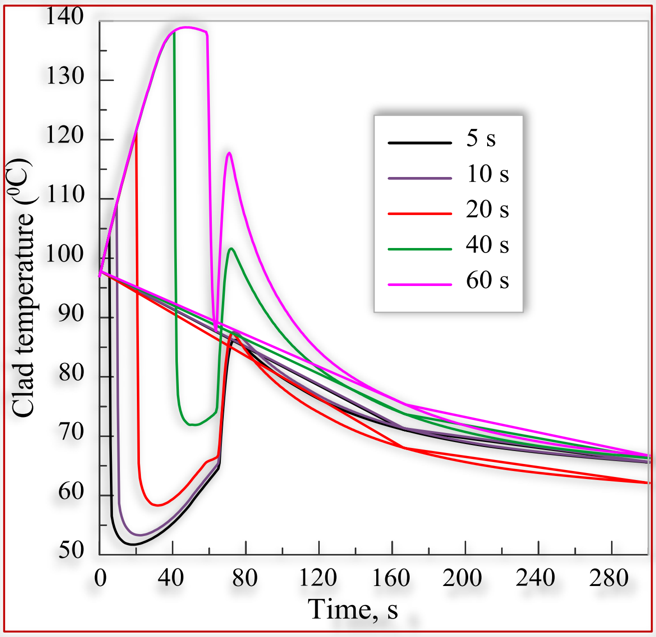

Figure 3. Transient response of fuel temperature for different delayed SCRAM signals

The thermal hydraulics data at the new steady states for all the cases of delayed scram scenarios are the same because the reactor power and core flow rate are approximately the same at the end of the transient. The new steady state clad and coolant temperature for all cases of flow trip scenarios are 57.354 °C and 52.566 °C respectively.

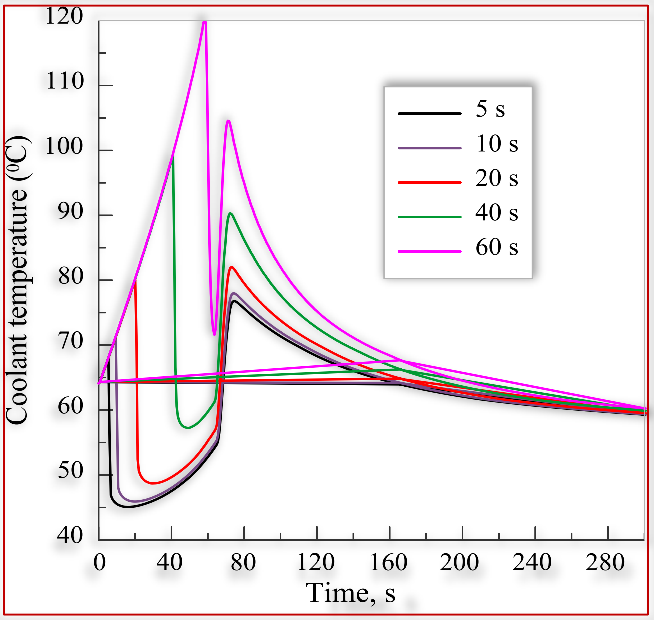

Figure 4. Transient response of fuel temperature for different delayed SCRAM signals

Table 5. Thermal hydraulics data during transient phase for different delayed SCRAM signals

Delayed SCRAM time	5 s	10 s	20 s	40 s	60 s
T_{Clad}, max (°C) before SCRAM	103.97	108.98	121.26	138.14	138.92
T_{Out}, max (°C) before SCRAM	67.98	71.203	80.043	99.208	119.72

— Reactivity and reactor power for different delayed SCRAM signals

Changes in reactor core temperature directly affect reactivity, either by modifying material densities through thermal expansion or phase changes, or by altering atomic motion. During transients, these effects are initially counteracted by inherent feedback mechanisms before SCRAM activation. In the event of a loss-of-flow accident, reactor power decreases as reactivity is reduced. At the precise moment of a flow trip, when negative reactivity is externally introduced, reactor power and reactivity experience a sharp decline.

Figures 5 and 6 show the transient responses of reactor power and reactivity for SCRAM delays of 5, 10, 20, 40, and 60 seconds. As fuel and coolant temperatures rise, negative reactivity is induced in the core, reducing neutron moderation and thereby limiting the uranium-235 chain reaction, which causes the observed drop in power. The SCRAM system further accelerates the power reduction by inserting additional negative reactivity. Ultimately, a new steady-state reactor power of 0.334 MW is reached in all scenarios.

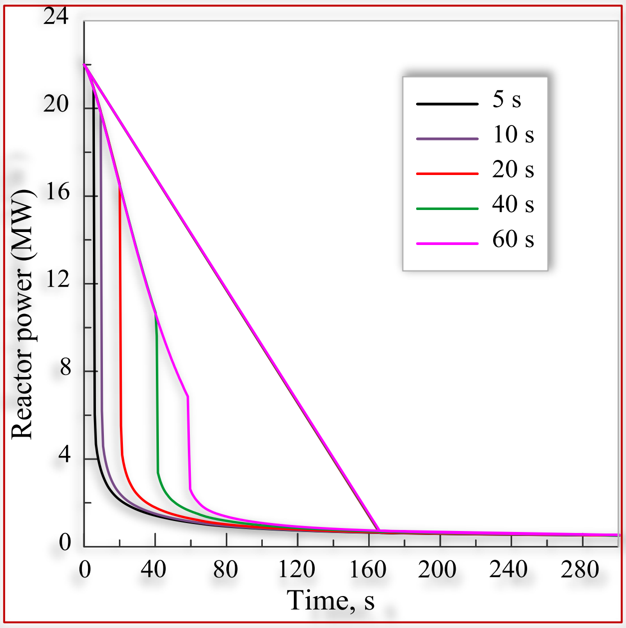

Figure 5. Transient response of reactor power for different delayed SCRAM signals

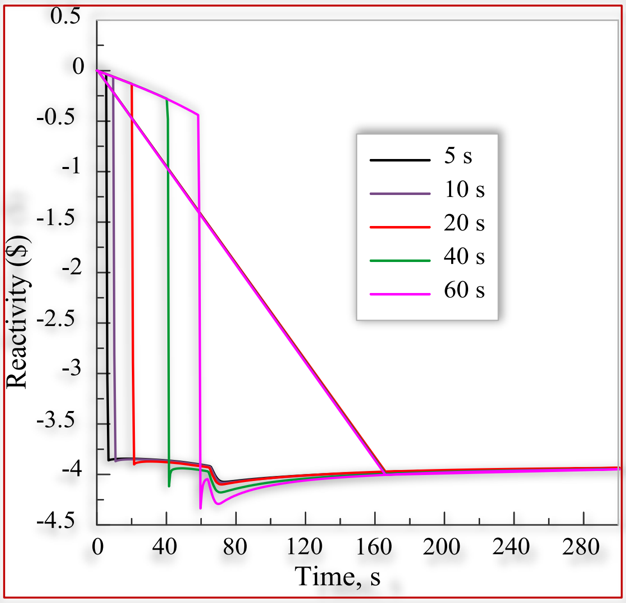

Figure 6. Transient response of reactivity for different delayed SCRAM signals

— **Flow instability**

Material testing reactors (MTRs) with 1-50 MW power, cooled and moderated by low-pressure water, are limited by thermal-hydraulic constraints due to the onset of flow instability (OFI) [5,7]. OFI restricts reactor power and can compromise core safety by reducing local heat transfer and lowering the burnout heat flux. Consequently, plate-type fuel design prioritizes the critical heat flux that triggers flow instability over stable burnout conditions. In transient analyses, the PARET code applies heat transfer, flow instability, and DNB correlations, with the Forgan correlation used to evaluate OFI.

Figure 7 depicts the transient response of the flow instability ratio (FIR) for various SCRAM delay times. For delays equal to or exceeding 54.1 s, the FIR drops below unity, indicating the onset of flow instability.

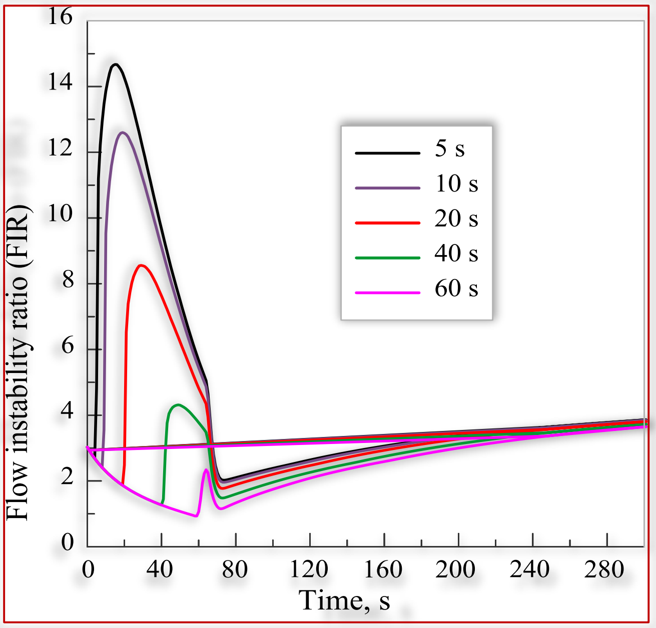

Figure 7. Transient response of flow instability ratio (FIR) for different delayed SCRAM signals

Table 6. Minimum flow instability ratio (FIR) different delayed SCRAM signals

Time for delayed SCRAM signals	5	10	20	40	60
FIR_{min}	2.023	1.96	1.77	1.26	0.933

Figure 8 shows the transient response of the critical heat flux ratio (CHFR) for various SCRAM delay times. From a thermal-hydraulic standpoint, the reactor remains stable, with CHFR as the key safety parameter, as it eventually settles at a new steady-state power level below the initial value. This emphasizes the crucial role of negative feedback in controlling reactor behaviour during the accident progression.

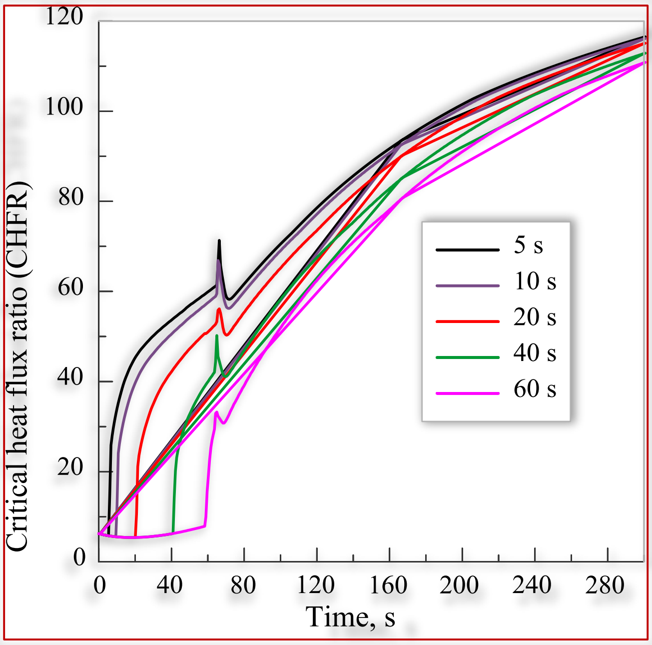

Figure 8. Transient response of critical heat flux ratio (CHFR) for different delayed SCRAM signals

CONCLUSIONS

The primary objective of reactor engineering and inherent safety systems is to ensure that nuclear fuel remains in a thermally stable state, maintaining sufficient safety margins under a range of operational conditions, including normal, abnormal, and emergency scenarios. This study presents an analysis of a loss-of-flow accident (LOFA) in a standard material test reactor. The impact of LOFA progression on key reactor safety parameters is evaluated by considering SCRAM initiation at various delays following the onset of the transient. Specifically, delay times of 5, 10, 20, 40, and 60 seconds from LOFA initiation are investigated. Calculations are performed using the nuclear reactor analysis code PARET. The effects of these accident scenarios on fuel, cladding, and coolant temperatures are quantified and discussed. Additionally, the transient responses of the flow instability ratio and the critical heat flux ratio are evaluated and presented.

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ACTA TECHNICA CORVINIENSIS
Bulletin of Engineering
ISSN: 2067–3809