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Safety enhancement of the Egyptian Second Research Reactor to reduce the effect of loss-off site power supply

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Abstract

The studying and analysis of the nuclear abnormal conditions have become an important demand to achieve highly safety standards in the nuclear reactors (research or power), where the nuclear accidents have been a serious concern since the construction of the first nuclear reactor in 1954 and after the world faced some major nuclear accidents as Three Mile Island in 1979, Chernobyl in 1986 and Fukushima in 2011. This paper aims to analyse a failure in the natural circulation core cooling system that could lead to temperature rise during shutdown regime. According to this scenario, our study suggests a solution that involves the addition of a small pump to the core cooling circuit in case of flapper valve malfunction due to any reason. This potential scenario is applied theoretically in the ETRR-2 (Egyptian Testing and Research Reactor Number 2). The RELAP5-3D simulation code is used to simulate and model the thermal behaviour of the core condition during this abnormal scenario and to evaluate the proposed solutions. The proposed emergency pump may be designed and estimated to permit opening flapper valve as well. This approach resolves the issue of flapper valves malfunction, which makes the reactor shutdown safe.

Keywords: Nuclear reactors; Safety of research reactors; Loss of off-site power supply accident analysis; Modelling

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1. Introduction

This paper suggests solutions to modify the core cooling system in a natural regime and to enhance the power supply operation to avoid core damage. An accident is modelled on ETRR-2 (Egyptian Testing Second Research Reactor) and the RELAP5 (Reactor Excursion and Leak Analysis Program) code is used to simulate the behaviour of the core cooling during the loss of off-site power supply, as well as to evaluate the proposed solutions. Accidents in research reactors may not have the same harmful risk level to the public as compared to accidents in power reactors. They may pose a greater hazard to operators; therefore, design limits of research reactors have been set to prevent the occurrence of accidents and mitigate their hazards. Most research reactors have a compact core structure with multi-parallel cooling channels. Due to the high heat flux and the low cooling capacity of the channels, the critical temperature of the heating surface is exceeded within a few seconds, affecting the fuel plates, which may be damaged. To detect the extent of damage resulting from accidents, many accidents should have been simulated and analysed to take precautions and find solutions to prevent their occurrences and their damages [1]. One of the important accidents is the loss of off-site power supply (LOPS), which leads to the shutdown of all the systems and automatic insertion of the absorbing rods into the core. As no electricity is available to operate the core cooling system, the pumps will coast down for 90 s and stop. Once the primary flow has been diminished, the natural convection mode is triggered mechanically to remove the residual decay heat by opening the flapper

Nomenclature

A – heat transfer area, m^2
C_p – specific heat at constant pressure, J/(kg K)
D – hydraulic diameter, m
E_{node} – energy content of the element, J/s
F – volume flow rate, m^3/s
g – gravitational acceleration, m/s^2
G – heat generated in element, W/m^3
h – convective heat transfer coefficient, W/(m^2 K)
K – conductive heat transfer coefficient, W/(m K)
L – channel length, m
m – mass of coolant, kg
ppf – power peaking factor
q – heat flux, W/m^2
Q – rate of heat transfer, J/s
Re – Reynolds number
t – time, s
T – temperature, °C, K
w – coolant mass flow rate, kg/s
x – thickness of control volume, m

Greek symbols

ρ – density, kg/m^3

Subscripts and Superscripts

c – coolant
cl – clad
$cond$ – conduction
f – fuel
i – inlet
o – outlet
p – pressure

Abbreviations and Acronyms

IAEA – International Atomic Energy Agency
ETRR – Egyptian Testing and Research Reactor
LOPS – loss of power supply
MTR – material testing reactor
SSR – steady state reactor
TTR – Tehran Research Reactor

valves installed on the primary pipe.

If LOPS happen and the flapper valves fail to open (stuck), the temperature of the water around the fuel will increase and the steam may be formed inside the coolant channel. At this moment, the core is in a dangerous situation, and this may lead to a severe accident. A severe accident refers to an event with an extremely low probability of occurrence, but that can cause significant damage to the reactor core [2]. There are many studies about the reasons, consequences and results of LOPS on the nuclear reactors, and how to enhance the emergency systems are investigated continuously [3]. The study in this paper focuses on loss of power supply accident during the reactor operation, which touches the beyond-design basis accident (failure of both flapper valves, which are responsible for the natural convection mode beginning). A beyond-design basis accident is a sequence of events that involves failures on the reactor protection system, the safety systems or engineered safety features, which lead to core damage [4]. This type of accident is studied and analysed to find solutions to prevent the resulting damages [5]. To evaluate the reactor performance during accidents, a thermal-hydraulic code may be used for simulating the transient behaviour of the core. The RELAP5 code is chosen for simulating the accidents and the proposed solutions, which are mentioned in this paper. The RELAP5 code is a best-estimate code, and this means that the accident simulation and the results are more realistic [6]. The RELAP5 code was developed at the Idaho National Laboratory (INL) for light water reactors during postulated accidents. It is based on a nonhomogeneous and non-equilibrium model for a two-phase system that is solved by a fast and partially implicit numerical scheme. It models the loss-of-coolant accidents, loss of off-site power, loss of feed water and loss of flow accidents [7]. The potential of the alternative sequence to cool down the reactor in the event of loss of off-site power supply was studied in [8]. It is interesting to integrate the alternative procedure into the reactor operation, in order to improve the reliability of the decay heat removal (DHR) function by adding a medium-strength safety defence line. The approach presented in [9] adapted both the deterministic and the probabilistic methods to determine the likelihood of the LOPS accident. Two applicable codes were used to model and investigate the LOPS accident: the probabilistic safety assessment package (PSAPACK) of the International Atomic Energy Agency (IAEA) and the TR22M21 code, which was developed by the author to simulate the reactor performance during an accident. The coolant system for a research reactor is designed and constructed to provide adequate cooling to the reactor core, where the primary cooling system is not designed for cooling the core after shutdown, and a reliable separate system shall be provided for the removal of residual heat. For the reactor systems that use flappers or equivalent systems for the transition from forced to natural circulation cooling, or for operation with natural circulation cooling, and for which this mode is part of the safety system (or is considered an engineered safety feature), the single failure criterion will be applied. Instrumentation to verify their functioning and to provide signals to the reactor protection system shall be provided [10,11]. The Serpent and DYN3D codes were extensively compared with a variety of static and burnup calculations as defined in the IAEA benchmark for a 10 MW pool-type material testing reactor (MTR) [12]. The safety classification of structures, systems and components (SSCs) for pool-type research reactors in Korean design is proposed based on the IAEA methodology [13]. It recommends that the SSCs of pool-type research reactors be categorised, classified and selected based on a graded approach. The Tehran Research Reactor (TRR) safety is studied with the second shutdown system (SSS) performing level one of probabilistic safety assessment (PSA) using the SAPHIRE (Systems Analysis Programs for Hands-on Integrated Reliability Evaluations) code, based on the selected initiating events [14]. It is seen that with the existence of the SSS, the failure rate of the reactor shutdown system decreases at least to 3.20×10^{-4} of its previous value. The core damage frequency conservatively turns out to decrease at least by half of the value previously reported to be 8.37×10^{-6}/y. In order to enhance the TRR, and the reactor personnel and other public safety, equipping the TRR with an SSS is a rational plan. This engineered safety feature helps to avoid accidents. Study of MARIA Research Reactor, operated by the National Centre for Nuclear Research in Poland, provides the details of probabilistic analysis performed within the safety classification process [15]. It presents insight from the implementation of the procedure in the safety classification for the MARIA Research Reactor. The methods for common-cause failure (CCF) parameters estimation and mathematical models for estimation of the probability of occurrence of common cause events were studied for Romania-TRIGA Steady State (SSR) 14 MW Reactor [16]. It describes data collection and statistics referring to the potential common-cause failure for components of the reactor, which involve pumps, control rods, valves, fans, ejector, by taking into account the event attributes like root cause, coupling factor, detection method and corrective action taken. A fuzzy fault tree analysis (FFTA) to evaluate the performance of the primary cooling systems of the G.A. Siwabessy Multipurpose Research Reactor (RSG-GAS), which belongs to the National Nuclear Energy Agency of Indonesia (BATAN), is presented in [17]. Expert justifications were used to evaluate the failure occurrences of basic events in the primary cooling system of the RSG–GAS through questionnaires. The assessment by experts is in the form of qualitative data, which is then converted into quantitative data by applying FFTA. The development and application of the coupled neutron kinetics (NK) and thermal-hydraulic (TH) code THERMO-T to the analysis of protected reactivity insertion accidents and loss-of-flow accidents in a typical research reactor (RR) with standard materials testing reactor plate-type fuel elements is presented in [18]. The steady-state and transient neutron and thermo-hydraulic analysis of an MTR core defined in the IAEA 10 MW benchmark using Serpent2/SubChanFlow coupled code is carried out in [19]. The subchannel thermal hydraulic code SubChanFlow was extended and validated using relevant tests [20]. The SubChanFlow extension consists of the addition of a heat conduction module for a thin plate and the implementation of heat transfer correlations for rectangular channels. For the validation of the modified SubChanFlow code, temperature data from the RA-6 experiment obtained in a Reynolds range of $7\times10^3 < Re < 1.4\times10^4$ were used. The study in [21] deals with the extension of the thermal-hydraulic model of the CATHARE2 code to fast transient configurations. The extended CATHARE2 model is validated against experimental results from SPERT IV D-12/25: Benchmark.

2. Materials and research methods

2.1. Mathematical model

In this section, equations that govern the heat transfer by conduction in a symmetrical plane wall, using the energy balance approach from the core centreline to the coolant across the clad, are illustrated. The scheme used in RELAP5 follows the finite volume methodology. The energy balance method is based on subdividing the fuel plate into two control volumes and then applying an energy balance on each element, as shown in Fig. 1. This is done by first selecting nodes at which the temperatures are to be determined and then forming control volumes over the nodes. Fuel, cladding and coolant temperatures have been calculated using the equations below. Figure 1 shows a scheme of the different regions in the fuel plate adopted for calculation, including the fuel region and aluminium clad region.

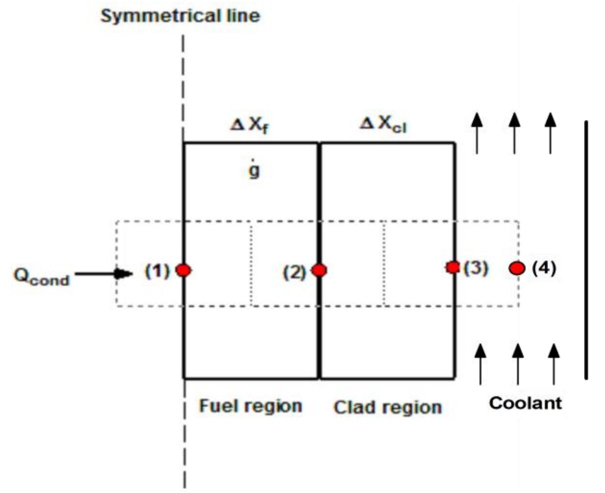

Fig. 1. Fuel plate model.

The heat conduction equation in the fuel is [22]:

$$Q_{cond} = \frac{\Delta E_{node}}{\Delta t} + G, \tag{1}$$

where Q_{cond} is the rate of heat transferred by conduction, G is the heat generated from fuel per unit volume, E_{node} is the energy content of the element and t is the time.

The general differential equation is

$$\rho_f\, C_{pf} \frac{dT_f}{dt} = K_f \frac{d^2 T_f}{dx^2} + G. \tag{2}$$

From the Taylor expansion theorem

$$T_{i+1} = T_i + \Delta x \frac{dT_i}{dx} + \frac{\Delta x^2}{2} \frac{dT_i^2}{dx^2} \tag{3}$$

The differential equation that describes the fuel centreline node, No. (1) and $i = N$, is expressed as

$$2K_f \frac{dT_i}{dx_f} + G\frac{\Delta x_f}{2} = \rho_f\, C_{pf}\, \Delta x_f\, \frac{dT_i}{dt}. \tag{4}$$

The differential equation of the fuel clad interface for the node No. (2), where $1 \le i \le N$, is:

$$K_f \frac{dT_i}{dx_f} + K_{cl} \frac{dT_i}{dx_{cl}} + G\frac{\Delta x_f}{2} = \left(\rho_f\, C_{pf} \frac{\Delta x_f}{2} + \rho_{cl}\, C_{pcl} \frac{\Delta x_{cl}}{2}\right) \frac{dT_i}{dt}, \tag{5}$$

$$C_{pc} \frac{dT_i}{dt} = h\, A\,(T_{i-1} - T_{i+1}) - wC_{pc}\,(T_{co} - T_{ci}), \tag{6}$$

where T is the temperature, K is the conductive heat transfer coefficient, h is the convective heat transfer coefficient, Δx is the thickness of the control volume, ρ is the density, m is the mass of coolant, w is the coolant mass flow rate and C_p is the specific heat at constant pressure. Subscripts f, cl, ci and co denote fuel, clad, coolant inlet and outlet, respectively.

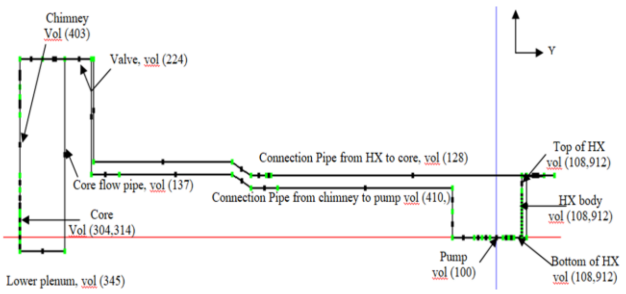

Fig. 2. RELAP5 nodalisation of cooling system;
HX – heat exchanger, vol – volumes of the node.

2.1.2. Nodalisation description

Nodalisation of the cooling system in RELAP5 is illustrated in Fig. 2. Component No. 100 represents the main pump. For simplification, it is modelled as a single volume, incorporating only the pump's geometric characteristics. Pipes 108 and 912 represent the internal structure of the heat exchanger. Each pipe is axially divided into 10 nodes, corresponding to the physical height of the heat exchanger. Pipes 128 and 137 model the internal connections from the heat exchanger outlet to the core inlet. These components are responsible for transporting the cooled water back to the core. Component 345 is a single-volume node that represents the lower plenum of the core. At this junction, the coolant flow splits into two streams: a larger portion directed to the average core channels and a smaller portion to the hot channel. Pipes No. 304 and 314 represent the hot and average core channels, respectively. Each channel is axially segmented into 21 nodes, representing the 0.8 m active length of the fuel elements. Radially, the channel is divided into six regions: two for cladding and four for fuel. Convective heat transfer boundary conditions are applied at the cladding surface. The coolant channel is modelled as a pair of half fuel plates, one from each side, forming a complete coolant passage.

3. Reactor description

Egyptian Second Research Reactor (ETRR-2) is an open pool type reactor, cooled and moderated by light water and reflected by beryllium. The nominal power is 22 MW. The reactor fuel is a plate-type fuel of 19.7% enrichment. The core is cooled by forced circulation upwards to avoid a cooling flow inversion in the case that the reactor is cooled by natural convection. The

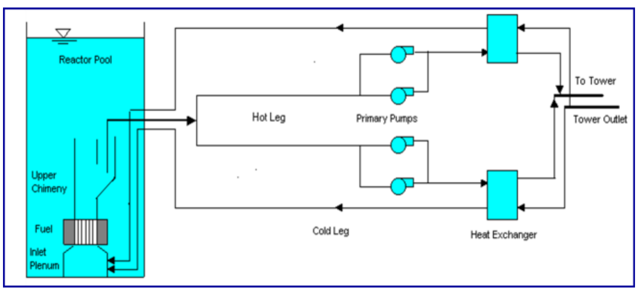

Fig. 3. Reactor core cooling system.

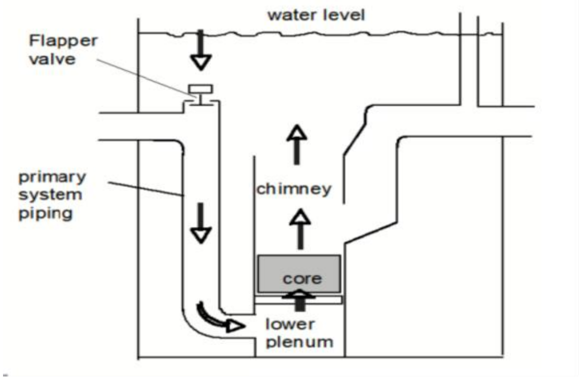

Fig. 4. Position of the flapper valve.

core coolant system with a flow rate of 1950 m^3/h is split into two 50% capacity branches; each branch has two 50% centrifugal pumps and one plate-type heat exchanger. Figure 3 shows the primary core cooling system [23]. The primary pipes have two flapper valves. These valves open by gravity forces in case there is no flow from primary pumps, as natural circulation cooling is established. Figure 4 shows the position and the working mechanism of a flapper valve. According to the quick reference of the emergency of ETRR-2, in case of a LOPS accident, the primary and secondary cooling systems will be stopped. The flapper valves must be opened after 90 s; otherwise, a malfunction may occur, but the operators can open them manually using operation tools. Flapper valves are mechanically designed to be closed under forced flow and opened by gravity; they may stack due to a failure in their seal part and cause a consequence of accident.

4. Results and discussion

4.1. Validation

The code is validated against core inlet temperature in normal operation, as shown in Fig. 5, and the reactor power status after shutdown – Fig. 6. It is shown from the two figures that the experimental values and the predicted values from RELAP5 are in good agreement [24].

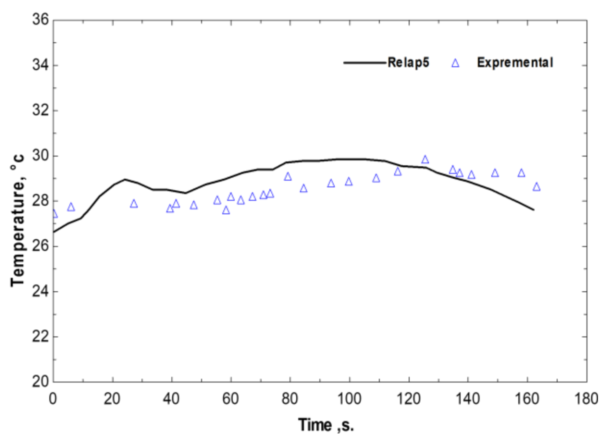

Fig. 5. Change in core inlet temperature over time.

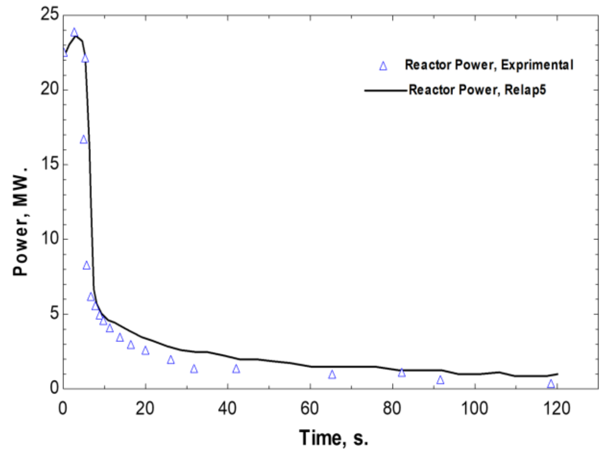

Fig. 6. Change of reactor power with time after shutdown.

4.2. Loss of power supply scenario

This event is analysed in an isolated manner; therefore, it would not be included as a cause of other events, where the absence of electric supply may be a contributor. Based on the safety analysis report, in a normal situation, if the external power supply is lost during the reactor operation, the primary and secondary pumps are stopped. At the same time, the reactor protection system (RPS) actuates the reactor shutdown system, which automatically inserts the control rods into the core, leading to a fast shutdown of the reactor in about 0.7 s. After the pumps tripped, the coast-down flow of the main pumps feeds the core with coolant for 90 s before it stops completely. Figure 7 shows the coast-down flow of the primary pump following the reactor shutdown.

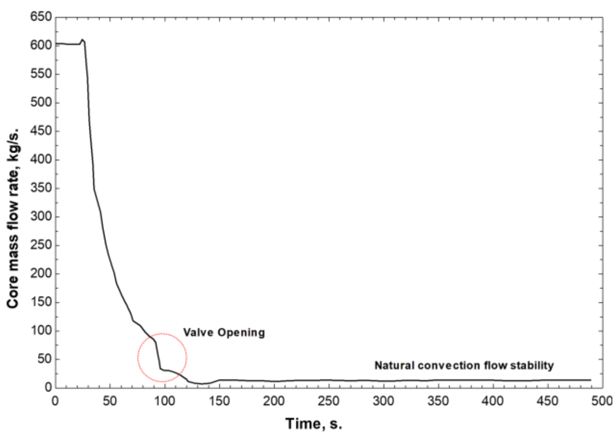

Fig. 7. Core flow after the pumps are tripped and the valves are opened.

Table 1. Sequence of events after LOPS accident.

Time (s)	Events
0.0 (initiative event)	• LOPS initiates • Core cooling system pump stops
0.7	• Fast shutdown (scram)
0.7–90	• Pump coast down
90	• Flapper valves opens and natural convection mode establishes

Normally, the flapper valves close during forced convection, but there is a special requirement that the valve should start to open when the pressure difference across it becomes less than 350 Pa. This condition is achieved 90 seconds after the pumps are tripped [18]. Table 1 shows the sequence of events after the LOPS event.

Figure 7 shows the core flow during normal conditions. After the flapper valves are opened, a marked drop in the core flow is observed, where the flow through the valve was initiated in an unexpected direction due to the pressure difference. After a while, natural circulation prevails in the expected direction (downwards inside the primary pipe) and the natural convection is established. The prediction obtained from the simulation carried out focuses on the hot channel, where it has the worst conditions in the core, and its safety means the core is safe. The hot channel is geometrically equal to a single fuel plate, the axial power is a cosine shape distribution, and the fraction of power allotted to hot channels was selected to produce a maximum allowable heat flux:

$$q_{max} = q_{ave}\,\mathrm{ppf}, \tag{6}$$

where q_{max} is the maximum heat flux, q_{ave} is the average heat flux, and ppf is the power peaking factor (ppf $\leq$ 2.3).

Figure 8 shows the variability of the clad's hottest point temperature. As a result of the flow drop at the beginning of the valves' opening, the temperature of the hottest point of the clad increases to about 75°C, and it still does not exceed the safety limits, 125°C.

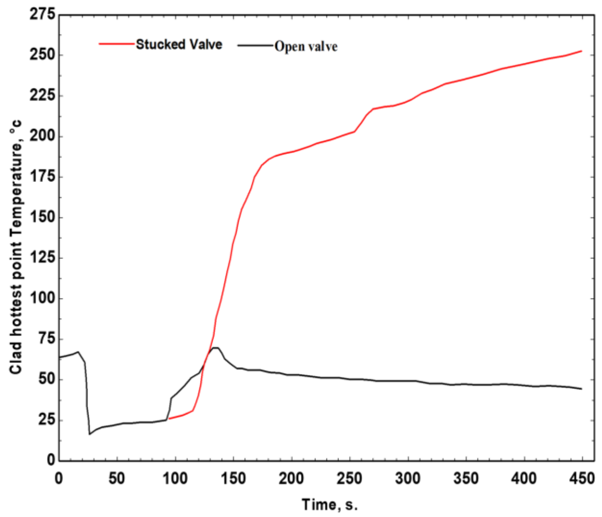

Fig. 8. Clad hottest point temperature.

4.3. LOPS scenario with flapper valves fail

If loss of power supply is followed by a failure of the flapper valves, the natural convection regime cannot be established, and the core flow will be zero. This situation is very serious because the coolant will heat up and the steam will be formed in a periodic way inside the coolant channels. In the first case, when the power drops below the threshold value after shutdown, the fuel will be cooled by a pulsed boiling mechanism, the steam is expelled, and the water replenishes the coolant channel again. The second situation is for the powers above the threshold power. Then, the pulsed boiling mechanism is not enough to cool the fuel plate and it will burn out. This paper is focused on the second situation, where there is a risk of danger in the reactor, and diverse emergency plans must be found to avoid any surprising situations. Figure 6 shows the temperature of the clad compared with its temperature in the case of valve opening. Figure 8 shows that in the case of the valve failure, the fuel is heated up after the main pumps stop completely (after 90 s), and its temperature increases rapidly to a dangerous value exceeding the safety limits (≤ 125°C). According to safety objectives and engineering design requirements, the reactor is in a critical situation, and this issue must be solved before it occurs.

4.4 Proposed solutions

As a result of valve failure, the flow to the core is zero, and the temperature of the clad increases to critical values, as shown in Fig. 8, and these values are sufficient to boil the coolant. When the new pump starts to work, the temperature of the core drops slowly. After a while, the flow of the new pump becomes steady-state and the temperature of the core decreases rapidly.

A small pump is installed with a capacity of about 3.0 m^3/h on the primary line and operates after ensuring the failure of the flapper valve (at 90 s after immediate shutdown). Figure 9 shows the position of that small pump, which is called the emergency pump. As shown, it is installed parallel to the main pump. This pump is simulated to operate after 1 min, 2 min and 3 min from the failure of the valve. This simulation is important to determine the optimum time to operate the pump. The emergency pump runs with the flapper valves after 90 s of a shutdown. If the valve fails to open, the pump already runs in a diverse way. But the mechanism of valve opening depends on the pressure difference inside and outside the primary pipe (less than 350 kPa). Thus, the operation of the small pump at the time of the valve opening will not affect the valve function, because the pump is selected and designed to provide a head of water column below the head at the flapper valve position. Figure 10 illustrates the configuration, which permits the flapper valve to open and keep the circulation of coolant inside the cooling loop. If we suppose that the valve opens and the small pump runs at the same time, most of the pump flow will go through the cooling loop. This solution is also helpful, as shown in Fig. 11. Peaks of the curves shown in Figs. 12 and 13 indicate the start time of the pump operation, and the drop at the end of the curves indicates

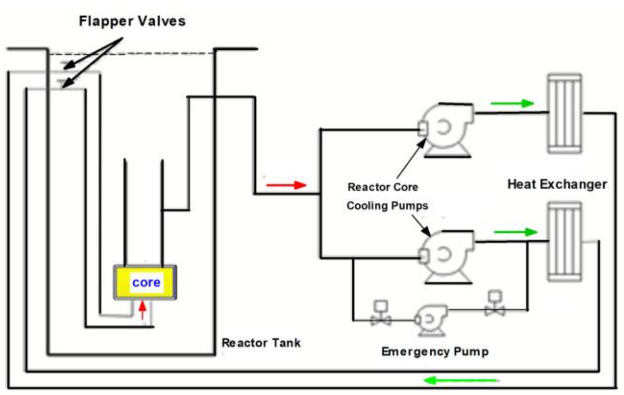

Fig. 9. Position of the emergency pump.

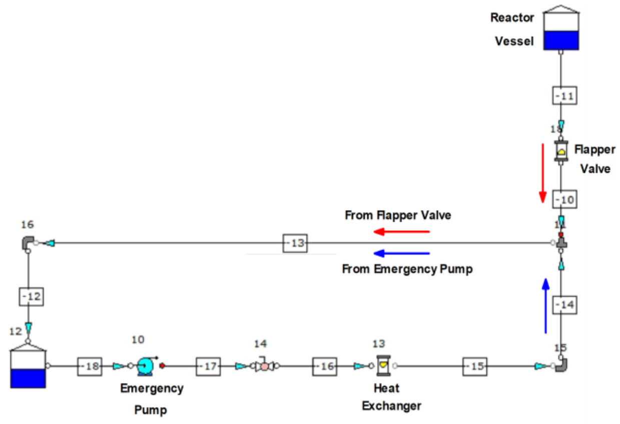

Fig. 10. Hydraulic layout of both the pump and the flapper valve.

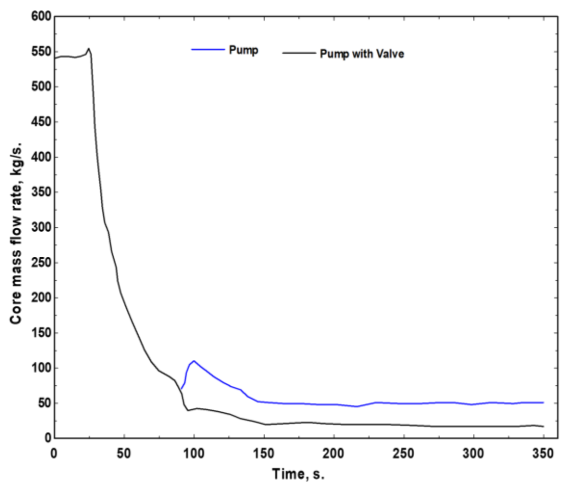

Fig. 11. Change in the outlet core coolant temperature over time for three cases.

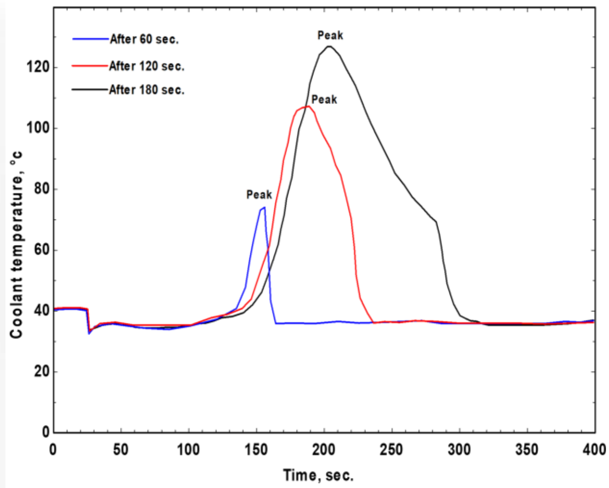

Fig. 12. Change in the coolant hottest temperature vs. time.

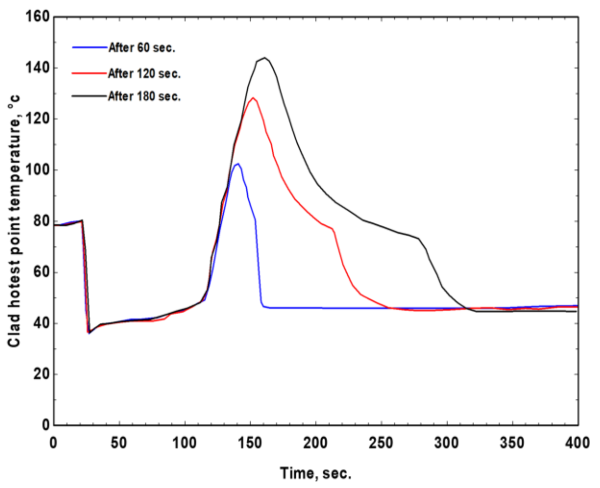

Fig. 13. Change in the clad hottest temperature vs. time.

the stability of the new pump flow. The peak values refer to the maximum core temperature reached in the different cases. It is found that, if the pump runs 1 min, 2 min and 3 min after the valve failure, the maximum core temperature does not exceed the safety values (125°C). Thus, it is recommended to operate the small pump after 1 min or 2 min of valve failure.

5. Conclusions

The reactor is capable of bringing and restoring a safe shutdown state by the passive natural convection safety system and inherent safety characteristics. If, for any reason, the passive and inherent safety systems have a malfunction, there is a need to use active safety systems. This situation is studied in this manuscript. It is supposed that the reactor is forced to shut down due to loss of power supply, and the flapper valve, which is responsible for natural convection, fails to open due to being stuck. It is recommended to install the proposed emergency small pump, with a capacity equal to that realised by a natural circulation laminar flow on the primary cooling system 50 m^3/h, to pressurise the core coolant after two minutes of the flapper valve failure. Modelling and simulation results indicate that the heat transfer and cooling are realised after the flapper valve is stuck with satisfactory temperatures prediction below safety limits, with a good margin. It is also recommended to connect an alternative power supply, such as a diesel generator, to run the emergency pump to guarantee its operation at any time.

References

[1] Dorsselaere, J.-P., Albiol, T., & Micaelli, J.-C. (2011). Research on severe accidents in nuclear power plants. In *Nuclear Power – Operation, Safety and Environment*. Intech, 155e182. doi: 10.5772/17552

[2] Volkanovski, A., Ballesteros Avila, A., Peinador Veira, M., Kančev, D., Maqua, M., & Stephan, J.-L. (2016). Analysis of loss of offsite power events reported in nuclear power plants. *Nuclear Engineering and Design*, 307, 234−248. doi:10.1016/j.nucengdes.2016.07.005

[3] Kim, S. H., Futami, T., Chang, S. H., & Jeong, Y. H. (2015). Safety Enhancements for PHWRs based on macroscopic losses of the Fukushima accident. *Science and Technology of Nuclear Installations*, 520756. doi: 10.1155/2015/520756

[4] Mohsendokht, M., Hadad, K. & Jabbari, M. (2018). Reducing the loss of off-site power contribution in the core damage frequency of a VVER-1000 reactor by extending the house load operation period. *Annals of Nuclear Energy*, 116, 303−313. doi: 10.1016/j.anucene.2018.01.030

[5] Gjorgiev, B., Volkanovski, A., Kancev, D., & Čepin, M. (2014). Alternative off-site power supply improves nuclear power plant safety. *Annals of Nuclear Energy*, 71, 304−312. doi: 10.1016/j.anucene.2014.04.011

[6] Petruzzi, A., & D'Auria, F. (2008). Thermal-hydraulic system codes in nuclear reactor safety and qualification procedures. *Science and Technology of Nuclear Installations*, 460795. doi: 10.1155/2008/460795

[7] RELAP5-3D Code Development Team. (2005). *RELAP5-3D code manual volume I: code structure, system models and solution methods*. INEEL-EXT-98-00834, Rev. 4.0.

[8] Grange, A., Marrel, A., Droin, J.B., Bertrand, F., Boutin, O., & Ferrasse, J. H. (2019). Loss of off-site power transient analysis for a sodium-cooled fast reactor equipped with a gas power conversion system and preliminary optimisation of its operation. *Nuclear Engineering and Design*, 355, 110315. doi: 10.1016/j.nucengdes.2019.110315

[9] International Atomic Energy Agency (2016). *Safety of research reactors: Specific safety requirements (IAEA Safety Standards Series No. SSR-3)*. IAEA, Vienna.

[10] International Atomic Energy Agency (2014). Safety of nuclear power plants. *Safety Reports Series*, 80. IAEA, Vienna.

[11] International Atomic Energy Agency (2016). Safety of nuclear power plants: Design, *Specific Safety Requirements (No. SSR-2/1)*. IAEA, Vienna.

[12] Margulis, M., & Gilad, E. (2016). Monte Carlo and nodal neutron physics calculations of the IAEA MTR benchmark using Serpent/DYN3D code system. *Progress in Nuclear Energy*, 88, 118−133. doi: 10.1016/j.pnucene.2015.12.008

[13] Kim, T.-R. (2016). Safety classification of systems, structures, and components for pool-type research reactors. *Nuclear Engineering and Technology*, 48(4), 1015−1021. doi: 10.1016/j.net.2016.02.009

[14] Boustani, E., & Khakshournia, S. (2017). Enhancing Tehran Research Reactor safety through a second shutdown system: A probabilistic safety assessment. *Progress in Nuclear Energy*, 100, 380−388. doi: 10.1016/j.pnucene.2017.07.006

[15] Kałowski, J., & Kowal, K. (2023). An integrated risk-informed safety classification for unique research reactors. *Nuclear Engineering and Technology*, 55, 1814−1820. doi: 10.1016/j.net.2023.01.017

[16] Radu, G., Mladin, D., & Prisecaru, I. (2013). Analysis of potential common cause failure events for Romania−TRIGA 14 MW reactor. *Nuclear Engineering and Design*, 265, 164−173. doi: 10.1016/j.pnucene.2013.06.027

[17] Hermansya, H., Kumaraningrum, A.R., Purba, J.H., & Yohda, M. (2020). Safety analysis technique for systems with limited data: Case study of the multipurpose research reactor in Indonesia. *Energies*, 13(8), 1975. doi: 10.3390/en13081975

[18] Margulis, M., & Gilad, E. (2016). Development and verification of the dynamic system code THERMO-T for research reactor accident analysis. *Nuclear Technology*, 196(2), 377−395. doi: 10.13182/NT1623

[19] Almachi, J. C., Sánchez-Espinoza, V. H., & Imke, U. (2022). High-fidelity steady-state and transient simulations of an MTR research reactor using Serpent2/Subchanflow. *Energies*, 15(4), 1554. doi: 10.3390/en15041554

[20] Almachi, J. C., Sánchez-Espinoza, V., & Imke, U. (2021). Extension and validation of the SubChanFlow code for the thermos-hydraulic analysis of MTR cores with plate-type fuel assemblies. *Nuclear Engineering and Design*, 379, 111221. doi: 10.1016/j.nucengdes.2021.111221

[21] Labit, J.-M., Seiler, N., Clamens, O., & Merle, E. (2021). Thermal-hydraulic two-phase modeling of reactivity-initiated transients with CATHARE2 − Application to SPERT-IV simulation. *Nuclear Engineering and Design*, 381, 111310. doi: 10.1016/j.nucengdes.2021.111310

[22] Bergman, T.L., Lavine, A.S., Incropera, F.P., DeWitt, D.P. (2011). *Fundamentals of heat and mass transfer* (7th ed.). Wiley & Sons.

[23] INVAP S.E. (1996). *ETRR-2 Detail Engineering Design*, NC-NSRC, Atomic Energy Authority, Cairo, Egypt.

[24] Abdelrazek, I., Abo Elnour, A.G., & Gaheen, M.A. (2015). Thermal hydraulic analysis of ETRR-2 using RELAP5 code. *Kerntechnik*, 80(1), 32−39. doi.org/10.3139/124.110475