NUKLEONIKA 2000;45(4):229–233

ORIGINAL PAPER

Accidents of loss of flow for the ETTR-2 reactor: deterministic analysis

Ahmed Mohammed El-Messiry

Abstract The main objective for reactor safety is to keep the fuel in a thermally safe condition with adequate safety margins during all operational modes (normal-abnormal and accidental states). To achieve this purpose an accident analysis of different design base accident (DBA) as loss of flow accident (LOFA), is required for assessing reactor safety. The present work concerns this transients applied to Egypt Test and Research Reactor ETRR-2 (new reactor). An accident analysis code FLOWTR is developed to investigate the thermal behaviour of the core during such flow transients. The active core is simulated by two channels: 1 – hot channel (HC), and 2 – average channel (AC) representing the remainder of the core. Each channel is divided into four axial sections. The external loop, core plenums, and core chimney are simulated by different dynamic lumps. The code includes modules for pump coast down, flow regimes, decay heat, temperature distributions, and feedback coefficients. FLOWTR is verified against results from RETRAN code, THERMIC code and commissioning tests for null transient case. The comparison shows a good agreement. The study indicates that for LOFA transients, provided the scram system is available, the core is shutdown safely by low flow signal (496.6 kg/s) at 1.4 s where the HC temperature reaches the maximum value of 45.64°C after shutdown. On the other hand, if the scram system is unavailable, and at t = 47.33 s, the core flow decreases to 67.41 kg/s, the HC temperature increases to 164.02°C, and the HC clad surface heat flux exceeds its critical value of 400.00 W/cm^2 resulting of fuel burnout.

Key words design base accidents • LOFA • loss of flow • pump coastdown • reactor transients

Introduction

LOFA is one of the frequently occurring DBA [3]. It affects the fuel integrity due to its overheating resulting from a low coolant heat transfer coefficient and consequently low core coolability. However, the primary core cooling system (PCCS), should assure core cooling and provide adequate safety margin to critical phenomena as redistribution and departure from nucleate boiling for every operational mode [1, 7]. PCCS loses its function as a result of flow reduction, which may be due to loss of off-site power, pump failure, heat exchanger blockage, pipe blockage or valve closing. ETRR-2 PCCS is connected to a natural core circulation loop where the flow circulates in case of loss of the main core forced flow (Fig. 1). The present study serves as a material test and research of (MTR) swimming pool reactor type.

Mathematical modelling

Fig. 2 illustrates the nodelization scheme of ETRR-2 core and its PCCS. The FLOWTR describes the neutronic part of the core and the coupling feedback reactivity coefficients by the well-known point kinetic formula. The thermohydraulic part is simulated by applying mass, momentum and energy conservation laws to every node [9]. The total number of

A. M. El-Messiry
National Center for Nuclear Safety and Radiation Control,
Atomic Energy Authority, Nasr City 11762,
P.O. Box 7551, Cairo, Egypt,
Tel.: 00202/ 2740236, Fax: 00202/ 2740238,
e-mail: elmessiry@pcn.aea.sci.eg

Received: 13 September 1999, Accepted: 6 March 2000

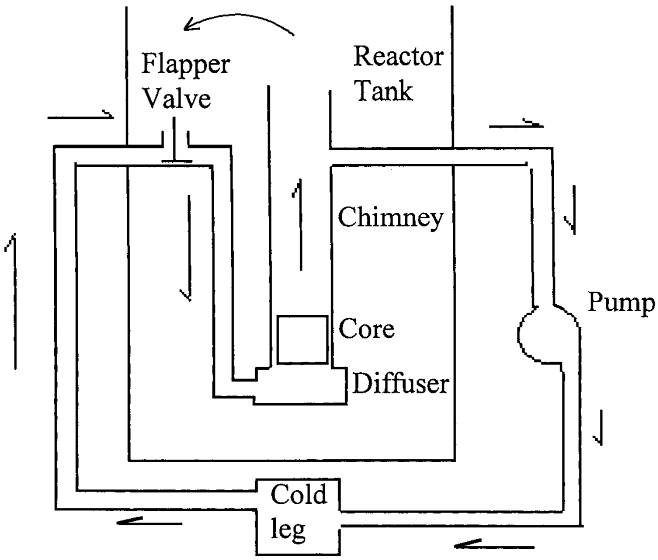

Fig. 1. The ETRR-2 flow circuit during LOFA.

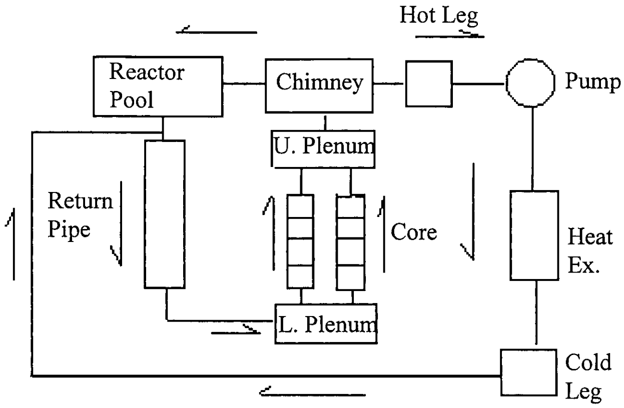

Fig. 2. The ETRR-2 nodalization model.

nodes is 28. Each channel (HC and AC) is divided into four axial divisions to take into account the cosine power shape. A heat slab structure has been assigned to each control volume of the core. The temperature transport delay in pipelines, the decay heat after shutdown and the reactor scram system modules are incorporated. The pump coast down module is derived by fitting the manufacture pump curve [6] with an exponential decay function:

$$(1) \qquad W_p = W_{po} \cdot \exp\left(-\frac{t}{T_{fp}}\right)$$

where: W_p – flow rate in the primary circuit, W_{po} – denotes the flow rate in the primary circuit at time t = 0, T_{fp} – time constant of flow decay of the primary circuit.

The flapper valve module is developed using data [5, 6] given in Table 1.

The equations employed are as follows:

$$(2) \qquad W = W_n + W_p$$

$$(3) \qquad W_n = 0.0 \qquad t <= 70.0$$

$$= W_{no} \frac{(t-t_n)}{t_{op}} \qquad 70.0 <= t <= 71.0$$

$$= 9.722 \qquad t >= 71.0$$

where: t – time in s, W_n – natural flow rate, t_n – time at which flapper valve starts to open, t_{op} – opening time.

The main thermohydraulic equations are given below, where the heat transfer correlations for forced convection (Sider Tate) [7], natural convection [8], boiling state (Mc-Adames) [4, 8] are used:

$$(4) \qquad Uallz(i,j) = \frac{1}{\frac{s}{2k_f} + \frac{c}{k_c} + \frac{1}{u(i,j)}}$$

$$(5) \qquad Qallz(i,j) = Uallz(i,j)\,[(T_m(i,5j) - T_f(i,5j-3)]\,A_s(i,j)$$

$$(6) \qquad \frac{dT_f(i,5j-3)}{dt} = \frac{Qallz(i,j)}{C_p M_{cz}(i,j)} + \frac{[T_f(i,5j-4) - T_f(i,5j+1)]W(i)}{M_{cz}(i,j)}$$

$$(7) \qquad \frac{dT_m(i,5j)}{dt} = -\frac{P_{oz}(i,j) - Qallz(i,j)}{C_p \cdot M_f(i,j)}$$

$$(8) \qquad h(i,j) = 0.023\,\frac{K}{D_e}\,R_e^{.8}(i,j)\;P_r^{.4}(i,j), \qquad R_e > 2000$$

$$(9) \qquad h_{bc}(i,j) = 0.074\,[T_c(i,j) - T_{sat}(i,j)]^{2..68}$$

$$(10) \qquad h_{bs}(i,j) = U(i,j) + 2.253\,[T_c(i,j) - T_{sat}(i,j)]^{2..96}$$

where: (i,j) – refers to channel i, and axial section j; D_e – equivalent diameter, R_e – Reynolds number, P_r – Prandtl number; T_f, T_c, T_m – axial coolant, clad, and fuel temperatures, P_{oz} – reactor power, C_p – coolant specific heat, W – coolant flow rate, c, s – clad thickness and fuel half thickness, k_c, k_f – clad and fuel thermal conductivity; h – forced heat transfer coefficient, h_{bc} – subcooled boiling heat transfer coefficient, h_{bs} – saturated boiling heat transfer coefficient, T_{sat} – saturation temperature, u – coolant heat transfer coefficient, Uallz – overall heat transfer coefficient, Qallz – overall heat transfer rate, A_s – surface area, M_{cz} – coolant mass, M_f – fuel mass.

Table 1. Flapper valve data.

Flow rate at which the valve starting opens	26.389 kg/s
Instant of opening	~70.0 s
Opening time	1 s
Natural flow during flapper valve complete opening	9.722 kg/s

The code is applicable over the expected range of operational parameters. It yields conservative predictions, mainly due to assumptions done in its development and upgrading as well as the built in solution routine by Runge Kutta, the built in safety factors, modelling options and used plant specific data.

Accidents scenarios

Once core flow is low enough (26.389 kg/s), the flapper valve placed on the return pipe of PCCS (Fig. 1), opens in one second and natural circulation is established through the core. The core flow continues to decrease while the natural circulation increases to its maximum 9.722 kg/s. The core flow is dominated by the natural flow, where the flapper valve is completely opened (Fig. 3).

The reactor will be scrammed if one of the following safety system setting is reached [6]:

1 – high temperature difference across the core (>11.0°C), 2 – low core cooling flow (<500 kg/s, nominal = 527.78 kg/s), 3 – high core outlet temperature (53.0°C), 4 – low core pressure drop, 5 – over power (>24 MW), 6 – low reactor period (<20 s).

According to the instrumentation response used in sensing the thermohydraulic and neutronic parameters, the triggers signals are delayed by 200 ms for thermohydraulic and by 25 ms for neutronic parameters [6].

Table 2. Verification results.

Parameter	THERMIC	RETRAN	FLOWTR
Maximum HC coolant temperature, °C	63.0	57.0	61.81372
Maximum HC meat temperature, °C	116.4	117.8	115.1691
Maximum HC clad temperature, °C	93.4	95.0	93.4738
HC heat flux, W/cm^2	117.0	–	116.9748
Core average heat flux, W/cm^2	39.0	–	38.9916

Results and discussion

The code is verified against results obtained using RETRAN code [5] and TERMIC code [6] for a null transient case. Table 2 summarizes the results.

For loss of flow transients with scram, the rector scrammed at t = 1.4 s is due to low flow signal (496.6 kg/s). Hence, power and temperatures drop suddenly to lower values. Temperature starts to increase again slowly to a maximum value and then it continues to decrease to 40.0°C (Figs. 3, 4 and 5). This variation refers to a sudden decrease in power, a continuous decrease in flow and consequently a decrease in heat transfer coefficients and rates (Fig. 6). The hot channel surface clad temperature reaches its maximum at 45.64°C (axial section number 3) at time 84.0 s, which is below the surface clad corrosion point at 105°C and the onset nucleate boiling temperature at 126°C (Fig. 5). This demonstrates a reactor safe shutdown.

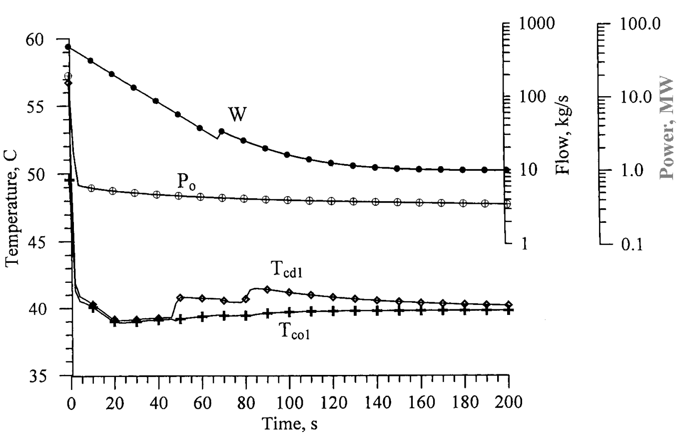

Fig. 3. Total flow W, power P_o, AC outlet temperature T_{co1}, and AC average clad temperatures T_{cd1} for LOFA with scram.

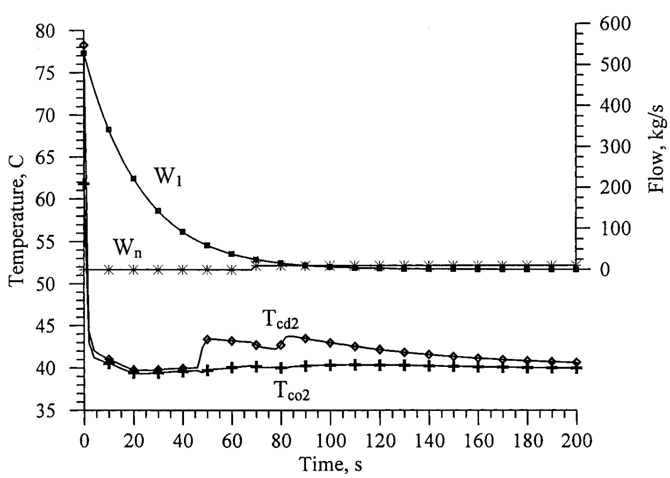

Fig. 4. Loop flow W_1 and natural flow W_n, HC outlet temperature T_{co2}, and HC average clad temperatures T_{cd2} during LOFA transient with scram.

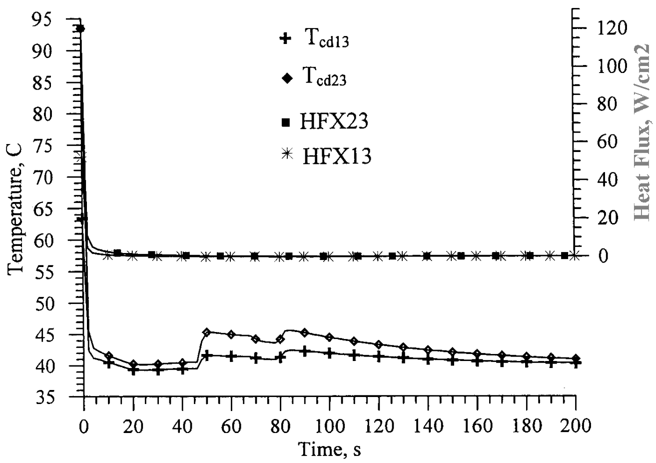

Fig. 5. Maximum AV and HC heat fluxes HFX13, HF23 and clad temperatures T_{cd13}, T_{cd23} for LOFA with scram.

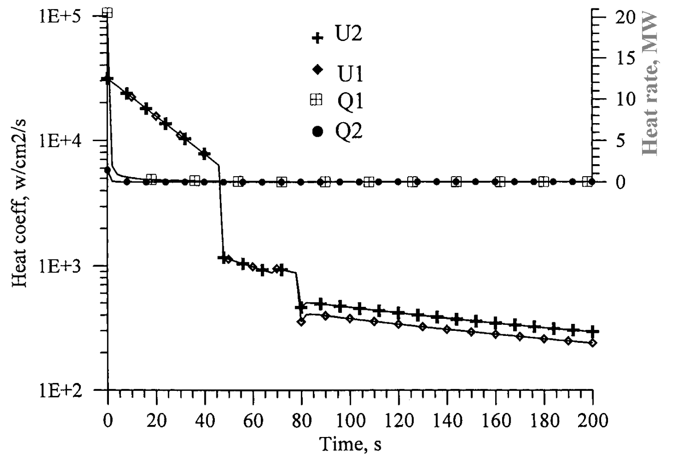

Fig. 6. AC, HC heat transfer coefficients U1, U2, and heat transfer rates Q1, Q2 for LOFA with scram.

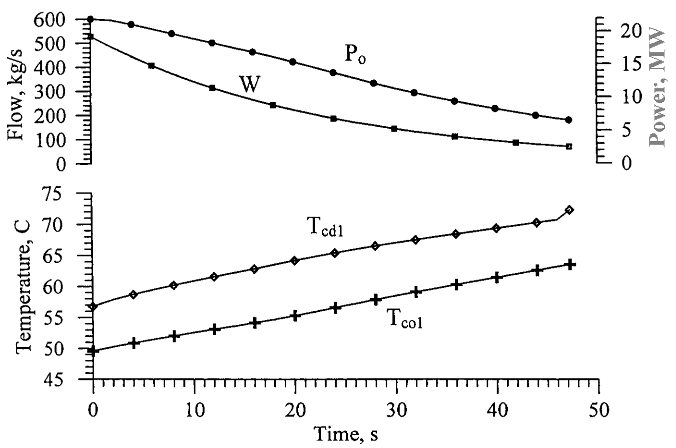

Fig. 7. Total flow W, power P_o, AC outlet temperature T_{co1}, and AC average clad temperatures T_{cd1} for LOFA without scram.

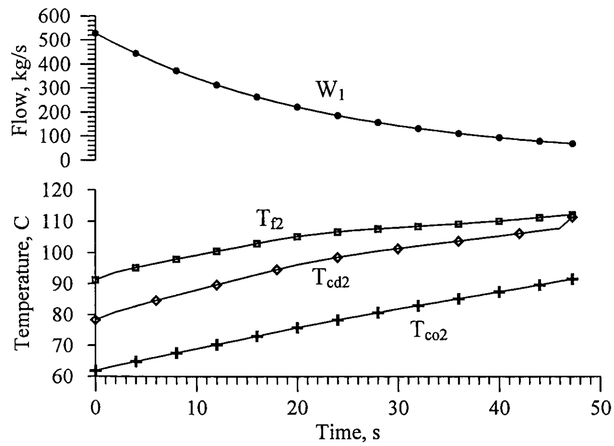

Fig. 8. Loop flow W_1, HC coolant outlet T_{co2}, average clad T_{cd2} and fuel temperatures for LOFA without scram.

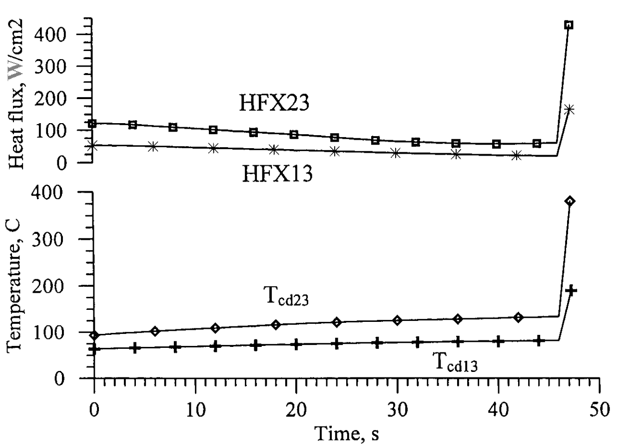

Fig. 9. Maximum AV and HC heat fluxes HFX13, HF23 and maximum clad temperatures T_{cd13}, T_{cd23} for LOFA without scram.

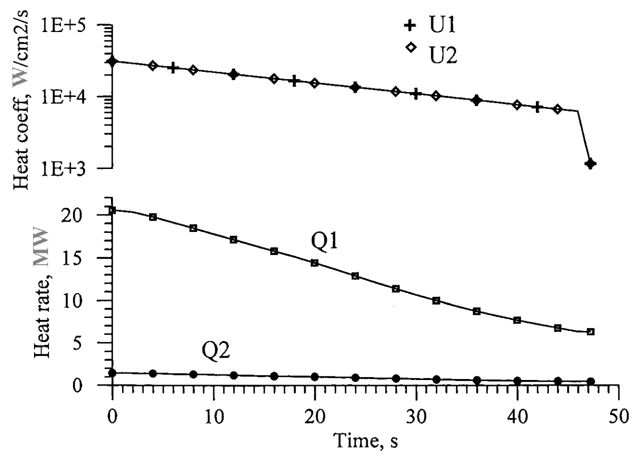

Fig. 10. AC, HC heat transfer coefficients U1, U2, and heat transfer rates Q1, Q2 for LOFA without scram.

On the other hand, for LOFA without scram, the core flow decreases, and consequently the heat transfer coefficients decrease. This results in the temperature increase and, in consequence, a power decrease due to negative temperature feedback coefficients (Figs. 7–10). While the core flow continues to decrease, the coolant heat transfer coefficients decrease sharply and coolant subcooled boiling starts at t = 32 s. As a result, the clad surface heat fluxes increase rapidly and exceeds the burnout value (400 W/cm^2) at 47.33 s causing a fuel damage. The HC clad surface temperature and heat flux approach values are 164.02°C and 400.31 W/cm^2, respectively, at burnout instant.

Conclusions

The concluding remarks are as follows:

1. The reactor scrammed safely due to LOFA without any clad corrosion or coolant boiling, in condition that the scram system is available (maximum surface clad temperature is 45.64°C after scram).
2. If the scram system fails, the hot channel clad surface temperature exceeds the onset nucleate boiling point and burnout values of 126°C and 164°C, respectively. Hence, coolant boiling and fuel failure occur.
3. During LOFA without scram, the heat flux decreases while the clad temperature increases continuously up to t = 47.7 s. After that the heat transfer coefficient undergoes a sudden decrease due to the very low flow, which causes a sharp increase in both clad temperature and heat flux to a limit of burnout values.

References

1. Code on the safety of nuclear research reactors: design (1992) Safety series no. 35–S1. IAEA, Vienna
2. El-Wakil MM (1971) Nuclear heat transport. McGraw Hill Intern. Text Book Co., Scranton
3. Gadalla AA, El-Fawal MM, (1996) PWRs loss of coolant accident analysis; DURFAN safety system code assessment. Arab Journal of Nuclear Science and Applications 31;1:147–160
4. Holman JP (1984) Heat transfer. 5th end. McGraw Hill Intern. Book Co., New York
5. INVAP SE (1996) ETRR-2 detail engineering design. Atomic Energy Authority, Cairo
6. INVAP SE (1997) Safety analysis report of ETRR-2. #0767–5325–3IBLI–001–10, Atomic Energy Authority, Cairo
7. Research reactor core conversion – guidebook (1992) TECDOC–643. IAEA, Vienna
8. Tong LS, Weisman J (1979) Thermal analysis of pressurized water reactors. 2nd. American Nuclear Society, Illinois
9. Woodruff WL (1984) A user guide for the current ANL version of the PARET code. Argonne National Laboratory, Argonne, IL, USA