Analysis of a Hypothetical LOCA in an Open Pool Type Research Reactor

R. Rachamin[1], A. Galperin[2], E. Elias[3]

1) Institute of Fluid Dynamics, Helmholtz-Zentrum Dresden-Rossendorf, POB 51 01 19, 01314 Dresden, Germany
2) Department of Nuclear Engineering, Ben-Gurion University of the Negev, POB 653, 84105 Beer-Sheva, Israel
3) Faculty of Mechanical Engineering, Technion - Israel Institute of Technology, 32000 Haifa, Israel

Corresponding author: r.rachamin@hzdr.de

Abstract. An analysis of a hypothetical loss of coolant accident (LOCA) in a pool-type research reactor is presented. The study was implemented for the Israel Research Reactor 1 (IRR-1), which is a 5MW reactor using highly enriched MTR-type fuel plates reflected by Graphite elements. The reactor core is cooled by downward forced flow of light water during normal operation and by upward natural convection flow through a safety flapper valve during shutdown. LOCA in pool-type research reactors may be initiated by various incidents such as ruptures and leakages from pipes and valves in the primary cooling system, ruptures of beam tubes or cracking of the pool wall caused by, e.g., strong earthquakes. Each one of these scenarios results in a rapid drop of the pool water level after reactor SCRAM. If water flow through the break persists, the core could eventually uncover completely and be exposed to the ambient air. The present study analyzes the possibility of passively cooling an exposed reactor core by thermal radiation and natural convection to air. The core uncover time is estimated by conservatively assuming that the LOCA was initiated by a guillotine break of a 10 inch outlet cooling pipe at the bottom of the pool, causing the core to uncover about 20 min after reactor SCRAM. Longer uncover times were used for parametric comparison. Since the Graphite reflector elements surrounded the core are typically solid that do not generate heat, they have the potential to act as a heat sink. The effect of the reflector on the core cooling was studied by comparing the total heat transfer from the core with and without considering the thermal contact between the core and the Graphite reflector elements. It is shown that for an uncover time of 20 min the core could reach its melting point if thermal contact with the Graphite is neglected. On the other hand, considering perfect thermal contact between the core and the Graphite reflector, the core temperature is predicted to remain indefinitely below the clad melting point (580 ^oC). The decay heat generation rate after reactor shutdown plays an important role in the analysis of LOCA. Several empirical correlations and theoretical models are available for predicting the decay heat after shutdown of a continuously operating power reactor. These correlations could not be simply applied for research reactors that work intermittently. A conservative decay heat generation curve was, therefore, estimated by comparing numerical results obtained by the BGCore computer code with available semi-empirical fitting functions and the ANS 5.1 standard curves. It has been shown that the BGCore computer code predict the decay heat generation rate with a small deviation from the corresponding semi-empirical functions results and the ANS 5.1 standard curves.

1. Introduction

Research reactors play an important role in the development of nuclear science and technology. They comprise a wide range of different reactor types that are not used for power generation. The primary use of research reactors is to provide a neutron source for research and various applications, including scientific education and training [1] [2]. For nuclear research and technology development to continue to prosper, research reactors must be safely and reliably operated. To ensure this, a set of postulated, severe accidents must be considered and analyzed.

One of the most common research reactor designs is a pool-type reactor. In a pool-type reactor, the core is a cluster of fuel elements sitting in a large open pool of water. Loss of coolant accident (LOCA) is one of the most important severe accidents that could challenge the safety limits of such a reactor type. There could be several primary causes to initiate a LOCA in a pool-type reactor, such as breaks in the piping system, ruptures of the beam tubes, and concrete wall failures of the reactor water pool. In case of a LOCA event, the pool water level will drop and the reactor will SCRAM (emergency shutdown of a nuclear reactor) automatically because of increased leakage from the upper surface of the core. If the water flow persists, it may lead to the most severe LOCA, where the core is uncovered completely and exposed to ambient air. Once this accident occurs, it may cause severe core damages, so it must be considered.

This paper presents the development of a mathematical model for the investigation of the behavior of a pool-type research reactor following a postulated complete LOCA event. The model was implemented to analyze the thermal behavior of the Israeli Research Reactor 1 (IRR-1) core following a LOCA. The paper is organized as follows. Section 2 gives a brief description of the IRR-1 core. Section 3 defines the accident scenario considered for the IRR-1 LOCA analysis. Section 4 describes different methods used for the assessment of decay heat generation rates, which plays an important factor in the analysis of LOCA. Section 5 is the major and the most important part of the paper. It describes the mathematical model that was developed for the LOCA analysis. Section 6 presents the results obtained from the LOCA analysis performed for the IRR-1. It assesses the possibility of passively cooling the IRR-1 core and determines the clad wall temperature in case of complete LOCA event. Finally, a summary and the main conclusions are given in Section 7.

2. The Israel Research Reactor 1 (IRR-1)

The IRR-1 [3], first operated in 1960, was donated by the USA government in the framework of "Atoms for Peace" program of President Eisenhower, to foster the use of nuclear technologies for the advancement of the Israeli economy. The IRR-1 is safeguarded by the IAEA. The main areas of application of the IRR-1 include research and training in nuclear engineering, neutron radiography and diffraction, activation analysis and changing colors of semi-precious and precious stones.

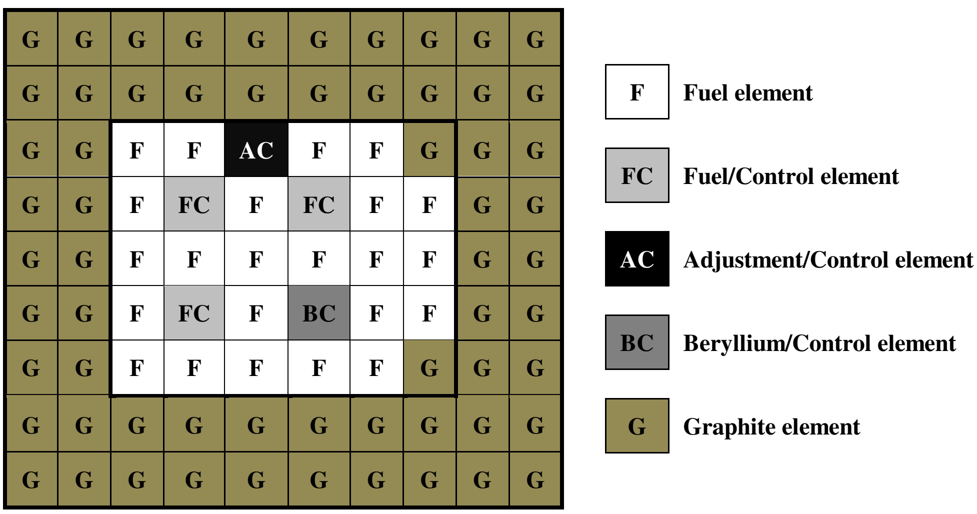

FIG.1. Typical IRR-1 core layout with Graphite elements reflector

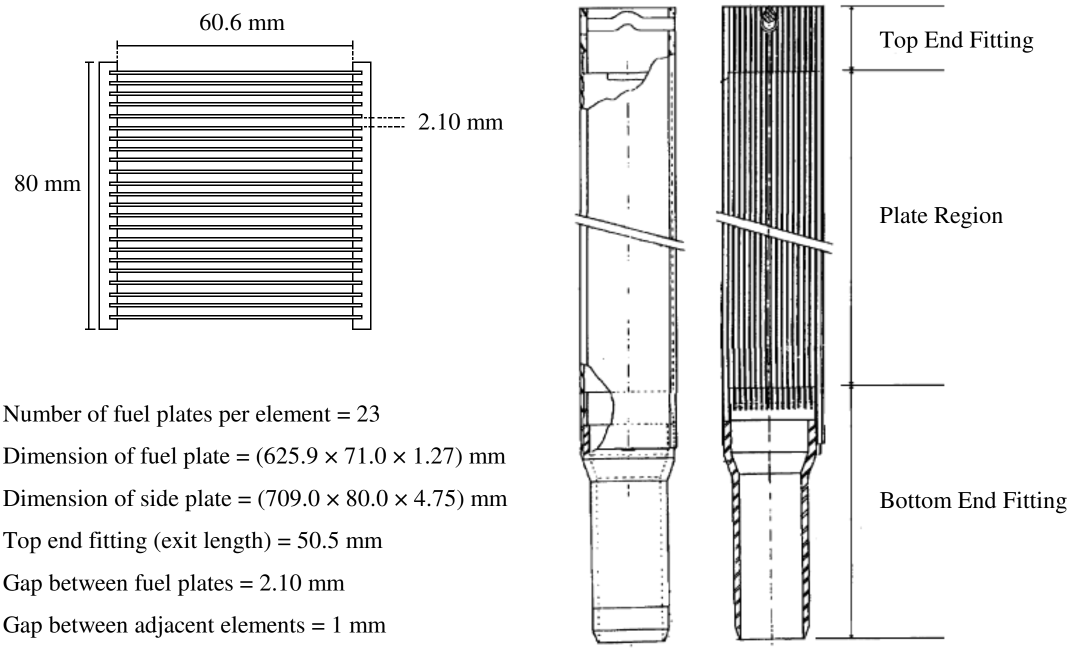

FIG.2. Standard IRR-1 fuel element

The IRR-1 is a typical swimming pool type research reactor utilizing plate type fuel cooled and moderated by light water. The reactor was designed to operate with highly enriched uranium (HEU) fuel at a power level of 5MWt. The reactor core is an assembly of standard fuel and control elements mounted on a grid plate. The allocation of the elements on the grid plate is shown in FIG. 1. The grid has 30 holes in a 5×6 pattern. The HEU core utilized 93% enriched fuel in the form of UAlx-Al. The fuel elements are of Materials Test Reactor (MTR) type. Each standard fuel element contains 232 gr of ^{235}U uniformly distributed in 23 flat plates. A schematic view of a standard fuel element is shown in FIG. 2. The fuel/control element contains 17 fuel plates and a rectangular passage for the movement of control rods. The control rods are inserted and removed vertically, from the top, by an electrical drive mechanism installed on a bridge above the reactor pool surface.

The reactor core is reflected by two rows of Graphite elements, to reduce neutron loss from the core, and it surrounded by light water. The water acts as a neutron moderator, cooling agent and radiation shield. The layer of water directly above the reactor core shields the radiation so completely that operators may work above the reactor safely. The reactor is easily accessible and the whole primary cooling system, i.e. the pool water, is under atmospheric pressure.

During normal operation, heat generated by nuclear reactions is removed by the downward forced flow of light water circulated by a primary cooling circuit. But during shutdown stage, the reactor core is cooled by upward natural convection flow through a safety flapper valve.

3. Accident Scenario

The present study assesses the possibility of passively cooling the IRR-1 core and determines the clad wall temperature in case of complete LOCA event. The following accident scenario was considered following an initiation of LOCA event:

- At time = 0 a reactor SCRAM occurs as a result of LOCA.

- About 20 min later the reactor core is completely uncovered and exposed to the ambient air. The core uncover time is minimized by assuming that the failure occurs at the largest pipe at the bottom of the pool, i.e. at the 10 inch diameter outlet water pipe.
- The uncovered core is cooled passively by upward natural convection flow of air and thermal radiation to containment chamber walls.

4. Decay Heat Source Estimation

Decay heat power after shutdown plays an important role in nuclear safety and, in particular, in the analysis of LOCA. It is of high importance to precisely estimate the decay heat generation rate at any time following irradiation of the fuel under any reactor conditions. The decay heat generation rate can be estimated by a variety of semi-empirical fitting functions, such as the Way-Wigner and the Patterson-Schlitz formulas. The Way-Wigner (WW) formula [4] [5] is the simplest semi-empirical fitting function to assess the decay heat production rate coming from fission-product decay process. The general form of this correlation is

$$\frac{P(T,t)}{P_0} = 0.0657 \cdot \left[t^{-0.2} - (t+T)^{-0.2}\right], \quad t > 10\,sec \tag{1}$$

where

$P(T,t)$	is the decay heat power
P_0	is the reactor operating power at steady-state
T	is the time, in seconds, of reactor shutdown measured from the time of start-up
t	is the time, in seconds, measured from shutdown

This formula is generally valid from 10 sec after shutdown up to several months after shutdown. The Patterson and Schlitz (PS) analytically fitted formula [4] [5], on the other hand, is valid for t after shutdown ranging from 0.1 sec up to 10^9 sec. The PS formula is described as follows

$$\begin{aligned}\frac{P(T,t)}{P_0} = &\left[0.1 \cdot (t+10)^{-0.2} - 0.087 \cdot \left(t + 2 \cdot 10^7\right)^{-0.2}\right] \\ &- \left[0.1 \cdot (t+T+10)^{-0.2} - 0.087 \cdot \left(t + T + 2 \cdot 10^7\right)^{-0.2}\right], \quad 0.1\,sec < t < 10^9\ sec\end{aligned} \tag{2}$$

Another method used for decay heat rate estimation is various computer codes, which perform detailed depletion calculation coupled with decay calculation. These computer codes are sometimes also using semi-empirical fitting functions, such as described above, for the transient analysis of nuclear reactor since they give estimates of decay heat generation rates with least computational effort. In this section, the BGCore computer code [6] was chosen to demonstrate the decay heat generation estimation. The BGCore is a software package for comprehensive computer simulation of nuclear reactor systems and their fuel cycles. It combines MCNP with a fuel depletion and decay module, which calculates the fuel isotopic composition. The BGCore system tracks a large amount of nuclide (~1700), which upgrades its capability in decay heat calculation.

TABLE I presents the relative decay heat generation (P/P_0) obtained using the BGCore code and the WW and PS formulas. For comparison, the values of P/P_0 obtained from the ANS 5.1

standard data [7] are also given in TABLE I. The P/P0 estimation was performed assuming that the reactor was operated for 219 days before shutdown.

TABLE I: Relative decay heat obtained by different methods for various post shutdown periods

Time after shutdown, sec	P/P_0, %			
	Semi-empirical fitting functions		BGCore	ANS 5.1
	WW	PS		
0.1	----	5.91	6.16	6.61
1	----	5.80	5.79	6.16
10	3.92	5.10	4.41	4.91
10^2	2.39	3.52	2.88	3.22
10^3	1.42	2.12	1.76	1.76
10^4	0.81	1.20	0.85	0.87
10^5	0.43	0.61	0.39	0.38
10^6	0.19	0.25	0.18	0.18
10^7	5.01×10^{-2}	5.03×10^{-2}	4.11×10^{-2}	3.70×10^{-2}
10^8	----	2.48×10^{-3}	1.88×10^{-3}	1.80×10^{-3}
10^9	----	8.91×10^{-5}	3.75×10^{-4}	6.00×10^{-3}

It is noted in TABLE I, that for post-shutdown times up to ~10^3, the estimated P/P0 values obtained from the BGCore code are smaller than that obtained from the ANS 5.1 standard curve. The difference between the two sets of values is mainly due to the fact that the ANS 5.1 standard data is based on typical uranium-oxide fuelled (3.5-4.5% enrichment) cores, whereas the IRR-1 has a higher value of enrichment (93%) and smaller core size. This leads to significant differences in the neutron spectrum and hence, in the production and decay rates of various radioisotopes. For very short values of post-shutdown times, the PS formula gives results quite close to the corresponding values obtained from BGCore. However, for longer periods, the PS formula significantly deviates from BGCore. The PS formula overestimates the decay heat compared to the other methods. For post-shutdown times, the WW formula gives P/P0 results quite close to those obtained from BGCore and the ANS 5.1 standard data. To summarize, the predictions of decay heat by the above two methods (semi-empirical fitting functions and the BGCore code) are found in reasonably good agreement with the results obtained by using the ANS 5.1 standard curve data.

5. Methodology

This section describes the mathematical model developed to estimate the heat removal from the IRR-1 core in case of a LOCA event. The model was developed under the assumption that the reactor core loses heat only by natural convection and thermal radiation. Conduction in the fuel plates only serves to equalize the element temperatures across the reactor core. Conduction losses to the grid plate are ignored. It is assumed that both the fuel plate temperature and residual heat generation are uniform.

Under the above assumptions, the average fuel temperature transient can be evaluated by the following energy balance equation

$$\left[q_{residual} - q_{radiation} - q_{convection} - q_{conduction}\right] = m \cdot c_p \cdot \frac{dT_w}{dt} \quad (3)$$

with initial condition $T_w = T_{w0}$ at $t = 0$.
Evaluations of individual terms of this equation are presented in the following sub-section.

5.1. Natural Convection Heat Losses

The convection heat losses from the reactor core, in case of LOCA, are by natural convection flow of air. There are three mechanisms of natural convection heat losses from the reactor core

1. from the channels between the fuel plates,
2. from the channels between adjacent elements, and
3. from the element's free faces on the sides of the reactor core.

The heat losses in the channels between the fuel plates of an element, are obtained by

$$q_{convection,\ channel} = \dot{m} \cdot c_p \cdot (T_w - T_\infty) \quad (4)$$

where

$\dot{m}$ is the mass flow rate in the channel
c_p is the specific heat of air
T_w is the plates temperature
T_∞ is the ambient air temperature

The mass flow rate is found from a force balance on a laminar, fully developed two-dimensional flow between parallel plates. To formulate an expression for the mass flow rate in the coolant channels between the fuel plates, several assumptions were made:

1. A pair of adjacent fuel plates forms a vertical coolant channel with symmetrically heated, isothermal walls.
2. The exit, entrance and inertia pressure drop are negligible.
3. The exit and mean bulk temperature along the whole length of the plate is equal to the plat wall temperature.
4. The chimney effect of the element walls above the top of the fuel plates (top end fitting in FIG. 2) is considered.

From those assumptions, the mass flow rate in the channels between the fuel plates of an element, takes the form [8]

$$\dot{m} = \frac{g}{\nu} \cdot (\rho_\infty - \rho) \cdot \left[\frac{L_1 + E}{\dfrac{12 \cdot L_1}{(N-1) \cdot S_1^3 \cdot W_1} + \dfrac{2 \cdot E \cdot f}{D_h^2 \cdot A_{et}}} \right] \quad (5)$$

where

N is the number of fuel plate in an element
L_1 is the fuel plate length
W_1 is the fuel plate width
S_1 is the spacing between the fuel plate

A_{et} is the cross-section area of the top opening of an element
E is the exit length of a fuel element (top end fitting in FIG. 2)
D_h is the fuel element exit hydraulic diameter
f is the fuel element exit friction coefficient
ν, ρ are the kinematic viscosity and density of air, respectively

The side plates of adjacent elements in the reactor core form a vertical channel with symmetrically heated, isothermal plates. To estimate the heat losses in the channels between adjacent elements, several assumptions were made:

1. The exit, entrance and inertia pressure drop values are negligible.
2. The exit and mean bulk temperature along the whole length of the plate is equal to the plate wall temperature.
3. Significant amount of the residual heat generated in the fuel plates is transferred by conduction to the side plates. This assumption can be justified by computing the temperature difference between the fuel mid-plate and the side plate of an element. A preliminary study showed that the temperature difference between the fuel mid-plate and the side plates is no more than 15 ^{o}C, and it decreases even more at longer time. Therefore, the temperature of the side plates will be close to that of the fuel plates, and it can be considered that all the heat generated in the fuel plates is practically transferred to the side plates.

In generally, this problem is identical to the once of a flow between fuel plates, described above. Therefore, Equation (4) may also be applied to evaluate the heat losses in the channels between the adjacent elements, were in this case, the mass flow rate takes the simple from

$$\dot{m} = \frac{g \cdot S_2^3 \cdot W_2}{12 \cdot \nu} \cdot (\rho_\infty - \rho) \tag{6}$$

where

W_2 is the side plate width
S_2 is the spacing between adjacent elements

Note that this expression can be derived from Equation (5), by setting the exit length, E, to zero.

The heat losses from the element's free face are obtained by

$$q_{convection,\ free\ faces} = \frac{\overline{Nu}_L \cdot k}{L_2} \cdot A_{fs} \cdot (T_w - T_\infty) \tag{7}$$

where

A_{fs} is the surface area of the element's free faces
k is the thermal conductivity of air
$\overline{Nu}_L$ is the Nusselt number

Various correlations available for natural convection flow on an isothermal vertical plate can be found in [8].

5.2. Thermal Radiation Heat Losses

Radiation heat transfer effects are often significant relative to natural convection. There are two mechanisms of thermal radiation heat losses from the reactor core

1. from the element's free faces to the containment chamber walls, and
2. from the top opening of the elements.

The radiation heat losses from the element's free faces to the containment chamber walls are obtained by

$$q_{radiation,\ free\ faces} = A_{fs} \cdot \sigma \cdot \varepsilon_{al} \cdot \left(T_w^4 - T_\infty^4\right) \tag{8}$$

where

ε_{al} is the emissivity of Aluminum ($\varepsilon_{al} = 0.1$), and

σ is the Stefan-Boltzmann constant ($\sigma = 5.67\times10^{-8}\ W/m^2 \cdot K^4$).

The heat losses from the top opening of the elements are obtained by

$$q_{radiation,\ top} = A_{et} \cdot \sigma \cdot \varepsilon_{eff} \cdot \left(T_w^4 - T_\infty^4\right) \tag{9}$$

where

ε_{eff} is the effective emissivity of the top opening of an element

The upper edges of the plates have the standard emissivity of Aluminum ($\varepsilon_{al} = 0.1$). When viewed longitudinally, the channels between the plates are considered as blackbody ($\varepsilon_b = 1$). Since, the cross-section area of the top opening of a fuel element is composed of, approximately, half plates and half open channels, the effective emissivity can be taken as 0.6. For the same consideration, the effective emissivity of a fuel/control element can be taken as 0.443.

Another mechanism of radiation heat transfer is the radiation exchange between opposing element faces, which is given by

$$q_{radiation,\ exchange} = \frac{A_s \cdot \sigma \cdot \varepsilon_{al}}{2-\varepsilon_{al}} \cdot \sum_{i=1}^{m}\left(T_0^4 - T_i^4\right) \tag{10}$$

where

A_s is the surface area of the element face

T_0 is the temperature of the element under study

T_i is the temperature of the i element facing the one under study

m is the number of opposing element faces

This mechanism does not remove any heat from the reactor core, but only tends to equalize the elements temperature.

6. Results and Discussion

The mathematical model described above was implemented to assess the decay heat removal from the IRR-1 core following a hypothetical LOCA. The IRR-1 core layout depicted in FIG.1 was considered. The 5×6 element core contains 23 MTR-type standard fuel elements (each consists of 23 fuel plates) and three control fuel elements (each consists of 17 fuel plates). The decay heat generated in these elements is presented in FIG. 3 as a function of time. The data of the decay heat source presented in this figure are attributed to the assumed operational schedule of the reactor, which is

- 580 fuel plates and steady-state power of 5 MWt,
- 8 hours of operation in a working shift per week,
- average fuel depletion = 30% of initial fissile content.

The reactor core is reflected by Graphite elements. Since the Graphite reflector elements surrounded the core are typically solid that does not generate heat, they have the potential to act as a heat sink. Therefore, the LOCA analysis was performed in two modes

1. Neglecting thermal coupling between the core and Graphite reflector elements.
2. Considering perfect thermal coupling of the core with the Graphite reflector elements.

In both of the modes analysis, the natural convection and radiation heat losses from the AC and BC elements were ignored.

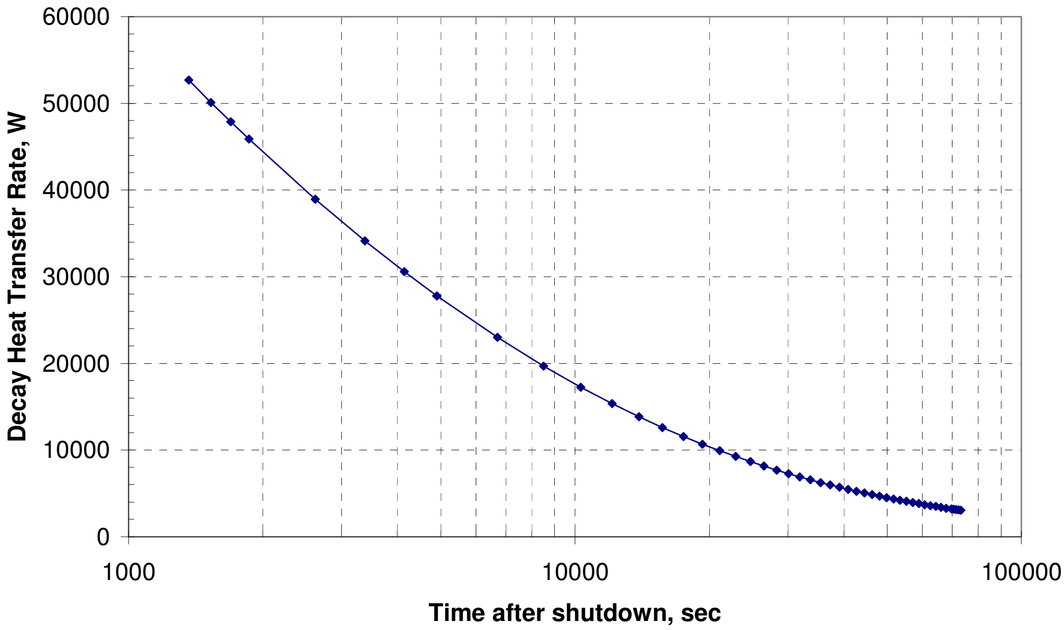

FIG.3. Decay heat generated in the IRR-1 core as a function of time

The temperature of the core resulted from the first mode analysis is shown as a function of time in FIG. 4. In a hypothetical LOCA, the IRR-1 core is assumed to uncover after 1200 sec. The results in FIG. 4 show that neglecting the Graphite heat sink, the IRR-1 core may reach melting point after about 3200 sec from the moment of reactor shutdown. The temperature of the core is also dependent on uncover time (or lag time), i.e. the time interval between reactor shutdown and total loss of water coolant. The temperature as a function of time for various lag times is shown in FIG. 5. It is shown in that for a lag time of 4 hours or more, the core temperature would not reach the clad melting point (580 ^{o}C) indefinitely. The temperature of the core resulted from the second mode analysis (perfect thermal coupling core/Graphite) is shown as a function of time in FIG. 6. Unlike the results of the first mode analysis, FIG. 6 shows that even for a lag time of 1200 sec the core temperature remains below the clad melting point.

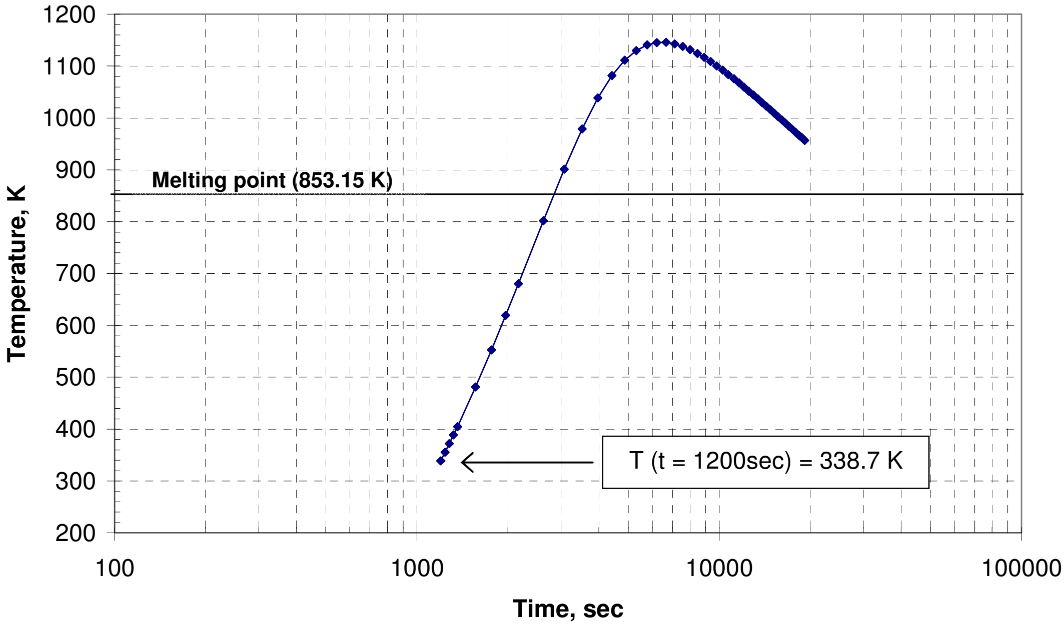

FIG.4. Temperature as a function of time, obtained for the case without the Graphite elements

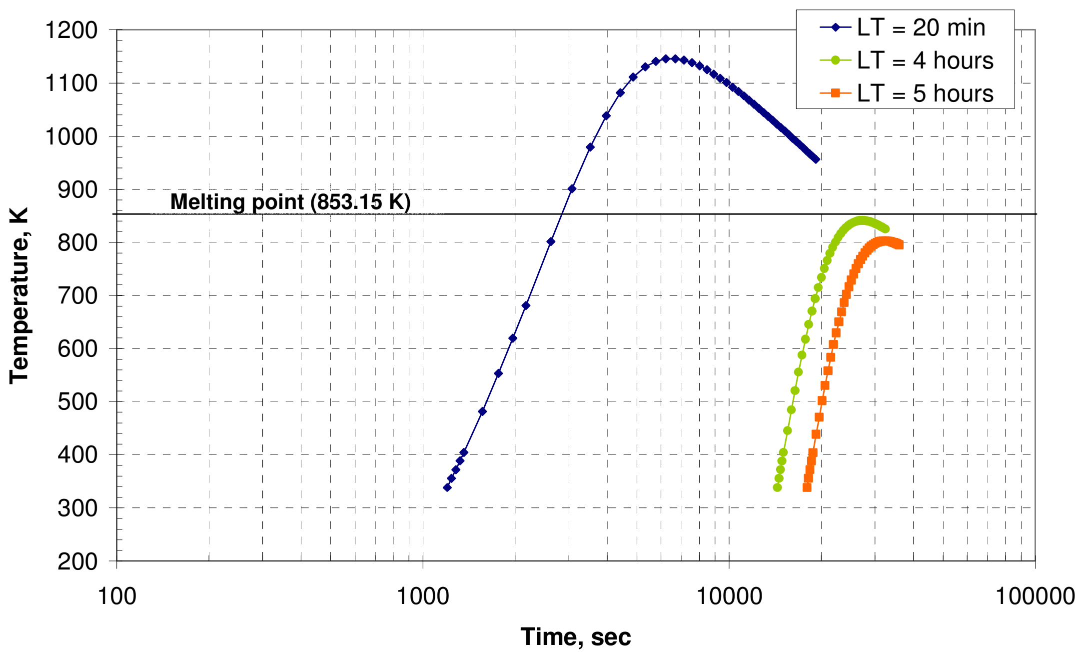

FIG.5. Temperature as a function of time for various lag times (LT), obtained for the case without the Graphite elements.

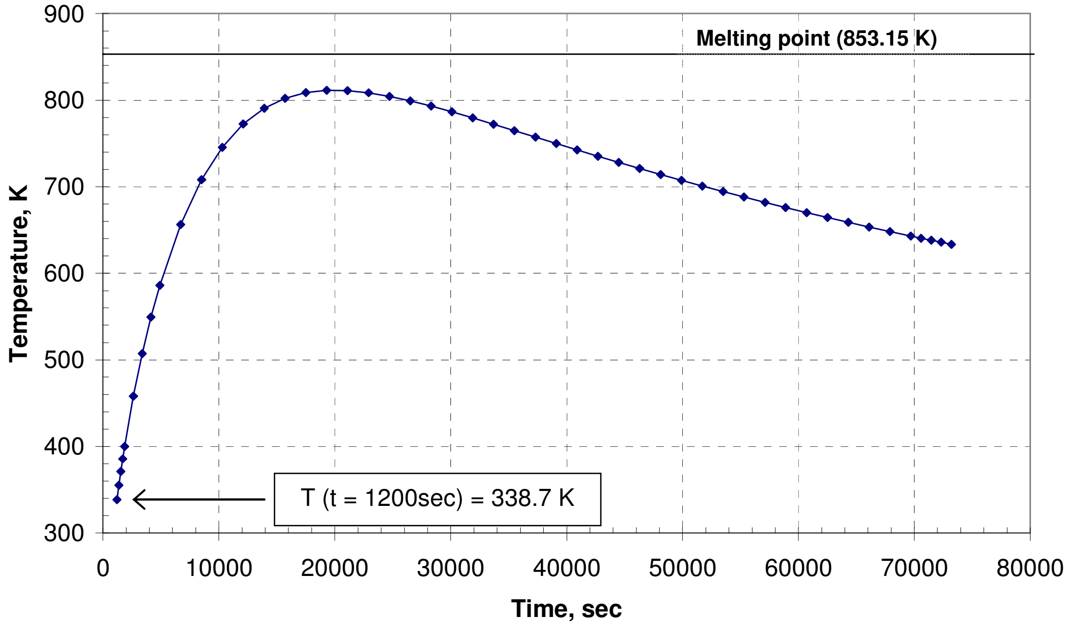

FIG.6. Temperature as a function of time, obtained for the case with the Graphite elements

7. Summary and Conclusions

This paper presents a mathematical model for the analysis of a hypothetical LOCA in a pool-type research reactor. The model was implemented for the IRR-1, which is a 5MW reactor using highly enriched MTR-type fuel plates reflected by Graphite elements. The LOCA analysis for the IRR-1 was performed for two modes of analysis: the first neglected thermal contact with the Graphite reflector elements and the other assumed perfect thermal transfer to the Graphite reflector elements. In both modes, the natural convection and radiation heat losses from the AC and BC elements were ignored. Therefore, the predicted results are thought to be conservative. The core uncover time was estimated by conservatively assuming that the LOCA was initiated by a guillotine break of a 10 inch outlet cooling pipe at the bottom of the pool, causing the core to uncover about 20 min after reactor SCRAM. It was found that for 20 min uncover time and no thermal contact with the Graphite reflector elements, the IRR-1 core is predicted to reach the melting point. On the other hand, for a core with perfect Graphite heat sink, the core temperature is predicted to remain indefinitely below the clad melting point (580 ^{o}C). The results found in this study can serve to guide operation and safety procedures of the IRR-1.

The paper also evaluates several methods used for the assessment of decay heat generation rates, which plays an important factor in the analysis of LOCA. The decay heat generation rates were estimated by two methods: semi-empirical fitting functions (WW and PS formulas) and BGCore computer code. The value of the decay heat at shutdown for continuous full power operation obtained from BGCore was 6.16% as compared to 6.61% obtained from the ANS 5.1 standard. These results were then compared with those obtained from the semi-empirical fitting functions to see how their predictions vary from each other. The results showed that the semi-empirical functions predict the decay heat generation rate with a small deviation from the corresponding BGCore results, which in turn agree reasonably well with the results obtained from the ANS 5.1 standard curve data.

8. Acknowledgment

This work was supported by the Israel Atomic Energy Commission and the Israel Ministry of Energy.

9. References

[1] "Research Reactors: Purpose and Future", International Atomic Energy Agency, Vienna, 2016. Available from: https://www.iaea.org/OurWork/ST/NE/NEFW/Technical-Areas/RRS/documents/RR_Purpose_and_Future_BODY.pdf

[2] "Applications of Research Reactors", International Atomic Energy Agency, Vienna, 2014. Available from: http://www-pub.iaea.org/MTCD/Publications/PDF/Pub1627_web.pdf

[3] The SNRC Reactor (Israeli Research Reactor 1 - IRR1), Soreq Nuclear Research Center [Online]. Available from: http://soreq.gov.il/mmg/eng/Pages/The-SNRC-Reactor.aspx

[4] Arkani, M., Gharib, M., 2008. "Residual Heat Estimation by using Cherenkov radiation in Tehran Rsearch Reactor", Nuclear Instruments and Methods in Physics Research A, August 2008, pp. 417-421.

[5] McMaster Nuclear Reactor, McMaster University; "Decay Heat Estimates for MNR"; Technical Report 1998-03; February 1999.

[6] Fridman, E., Shwageraus, E., Galperin, A., 2008, "Implementation of multi-group cross-section methodology in BGCore MC-depletion code", PHYSOR'08, Kursaal Conference Center, Interlaken, Switzerland, September 14-19.

[7] "Decay Heat Power in Uranium Fuelled Light Water Reactors"; ANSI, ANS 5.1-1979; American Nuclear Society, 1979.

[8] Rohsenow, W.M, Choi, H, 1961. "Heat, Mass and Momentum Transfer", Prentice Hall, New Jersey, 1961.