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Flow Blockage Accident Analysis of Tehran Research Reactor Fuel Assembly

J. Jafari* and S. Khakshournia

Reactors and Accelerators R & D School, Nuclear Science and Technology Research Institute, P.O. Box: 1439951113, Teheran- Iran

Abstract: Tehran Research Reactor (T.R.R.) is a pool-type, 5 MW thermal research reactor. One probable event is that if some external objects or debris fall down into the reactor core and cause obstruction of the coolant flow through one of the fuel assemblies, decreasing the surface flow area, ceases the coolant flow, and also raises the fuel and sheaths temperature. Thermal hydraulic analysis of this event has been studied using RELAP5 system code. This report is related to the partial and total obstruction of a single Fuel Element (F.E.) and cooling channel of 27 F.E. equilibrium core of the T.R.R. Such event may lead to severe accident for such type of research reactors, since it may cause a local dry out and eventually loss of the F.E. integrity. Two scenarios are analysed in order to emphasize the severity of the mentioned accident. The first is a partial blockage of hot F.E. which is considered for four different obstruction levels of the nominal flow area: 25%, 50%, 75% and 93%. The second is related to an extreme case which consists of the total blockage of the same F.E. The reactor power is derived through the kinetic point calculation in the RELAP5 code. The point kinetic feedbacks including the fuel temperature (Doppler coefficient) and the coolant density coefficient have been considered through the applied model. The main results obtained from the RELAP5 calculations are as follows: 1. In case when the flow blockage is under 93% of the nominal flow area of an average F.E., only the increase of the coolant and clad temperatures are observed with no integrity of the F.E. consequences. The mass flow rate remains sufficient enough and cools the clad safely 2. In the case of a total obstruction in the nominal flow area, it is seen that the severe accident is due to dryout conditions and reaches promptly, while melting of the cladding occurs.

Keywords: ***Tehran Research Reactor, Blockage of Flow, Fuel Element, RELAP5 Code***

1- Introduction

Among non-excursion accidents that may be considered, some of them are related to the flow loss which is considered to be the most significant event, and it should be realized in the safety evaluation of nuclear reactors. An other postulated accident for pool-type research and test reactors is blockage of the coolant channels that may cause flow loss in the blocked channels. Such an accident leads to the retention of coolant water in the coolant channels instead of natural or forced circulation of water through the channels. Once the blockage occurs, the coolant flow rate decreases and nearly the total heat of transferring mechanism in the blocked channels change to the conduction mechanism which could result in less heat transmission rate as compared with the convection mechanism. It is necessary to demonstrate that the heat of a residual, can be removed without the core melting hazard in such channel blockage accident.

So far, almost all of the research reactor safety analyses have been performed by using conservative computational codes (see [1] and references therein). Nowadays, on the one hand, there are some works showing capability of the thermal-hydraulic RELAP5 code system for analyzing transients under operating conditions of research reactors [1, 2, 3]. On the other hand, one can rarely find the transient studies in the available scientific literatures which are related to the flow assembly blockage accident [4, 5].

Due to the lack of study related to the flow blockage event in the fuel assembly of Tehran Research Reactor, in this paper we intend, as the first attempt, to use the thermal-hydraulic RELAP5/Mod3.3 system code for simulating partial and entire coolant channel blockage accident of the T.R.R. Hence, it is assumed that some granular materials such as sand or soil fall

*email: jljafari@aeoi.org.ir
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down into the open pool surface for any reason and cause blockage of some coolant channels of the T.R.R. core.

2- Summary Description of the T.R.R.

The T.R.R. is a 5 MW pool-type research reactor with plate type fuels and 20% enrichment [6]. The schema of the T.R.R. primary circuit is shown in Fig. 1. The main components consist of: two pools (stall and open), core, grid plate, flapper valve, piping, hold-up tank, main circulating pump and heat exchanger. The reactor core is cooled and moderated with light water and equipped with 22 standard fuel elements (S.F.E.), consisting of 19 fuel plates and 5 control fuel elements (C.F.E.) with 14 fuel plates. The core is installed in the stall pool; the coolant can flow back to the stall or open pool.

3- Description of RELAP5 System Code

The RELAP5 system code [7] is a qualified code which is verified and validated by international community for the safety analysis of the water cooled nuclear reactors [8]. The code is based on applying non-homogeneous, non-equilibrium set of six partial derivative balance equations related to the steam and liquid phases. An implicit finite difference scheme is used to solve the equations inside the controlled volume, where they are connected by junctions. The control volume has a direction associated with that it is positive from inlet to the outlet. The fluid scalar properties such as pressure, energy, density and void fraction are represented by the average values and viewed as being located at the control volume center. The fluid vector properties, i.e. velocities, are located in the junctions and are associated with mass and energy flow between control volumes. Control volumes are connected in series by using junctions to demonstrate flow paths. The loop components are modeled in one-dimensional sense, using staggered mesh for calculating temperatures and heat flux vectors. The heat structure is thermally connected with hydrodynamic control volumes through heat flux.

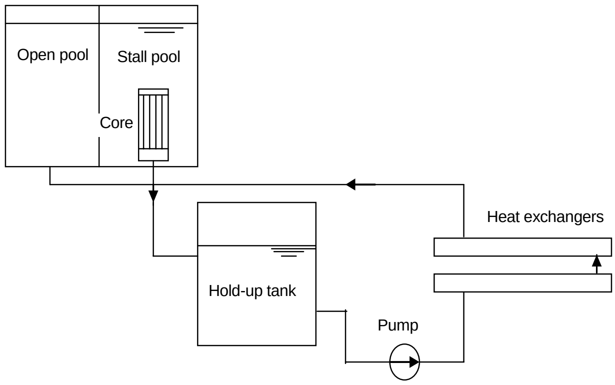

Fig. 1. Schema of T.R.R. primary circuit.

4- T.R.R. Nodalization Description

T.R.R. RELAP5 nodalizations are shown in Fig. 2. Time dependent volume (T.V.), time dependent junction (T.J.), tripe valves, motor valves, branch, pipe, and single junction, are the main RELAP5 components in this nodal scheme. Pool of the reactor is divided into three parts: upper part of the core (pipe 80), middle part of the pool including core of the reactor (pipes 100, 105, 110, 111) and volumes around the core (pipe 120) and lower part of the core (branches 130, 140, 150). The pipe 100 represents average core and the black column is the heat structure of the core. The pipe 105 represents one S.F.E. with average power and pipe 111 indicates hot fuel element which is considered as simulation of blockage event by using valve 98. The pipe 110 represents the bypass flow from the core. The valve 145 is a motor valve that simulates the flapper valve in the bottom of the core and also, numbers 190 and 500 are motor valves in the outlet and inlet of the pool respectively. The H.u.T. is modeled by a pipe and is divided into full (210) and empty (220) parts. The trip valve (250) connects these two parts. In the normal operation of the reactor, this valve remains closed; while an accident occurs, the valve opens and the coolant fills the empty part. The T.V. (240) and trip valve (230) simulate the atmosphere condition above the H.u.T. Technical characters of the main primary circulation pump (280) are introduced to the code exactly. The shell and tube heat exchangers are modeled by pipes 300-440 (shell side) and 700-870 (tube sides). The T.V.s (680 and 890) and T.J.s (690 and 880) provide fluid flow to the secondary side. The T.V. (50) and trip valve (60) simulates the atmospheric conditions above the pool.

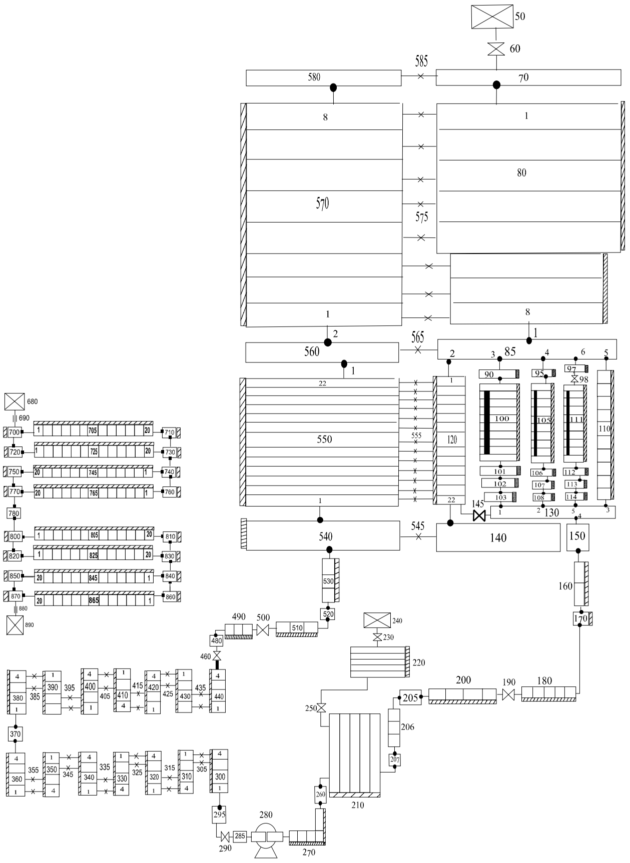

Fig. 2. T.R.R. RELAP5 nodal points.

5- Performed Calculations

The RELAP5 system code is used for thermo-hydraulic analysis of the blockage event. Partial and total obstructions in a single S.F.E. cooling channel from 27 F.E. equilibrium core of the T.R.R. are considered for analysis. This event can cause severe accident for this type of reactor, since it may lead to local dry out and eventually to the loss of the F.E. integrity. Two scenarios are analysed for emphasizing severity of the accident. The first one is a partial blockage of hot F.E. considering as four different obstruction levels: 25%, 50%, 75% and 93% of the nominal flow area. The second one is an extreme scenario consisting total blockage of hot F.E. This study constitutes the first step for outstanding work; it consists of performing 3-D simulation while using the Best Estimated coupled code technique. However, as an initial approach, instantaneous reactor power is derived through the point kinetic calculation included in the RELAP5 code.

6- Results of Calculations

Results obtained from the RELAP5 calculations for different levels of obstructions are shown in Figs. 3 to 13. Three initial obstructions of 25%, 50%, and 75% are shown: when transients start, reduction of the mass flow (Fig. 3) leads to an increase of clad temperature (Fig. 4), while reactor power remains nearly constant and no void formation is observed. In the next case with 93% of the nominal flow area obstruction when the transient is initiated, the decrease of mass flow in the obstructed channel (Fig. 5) causes an increase of clad temperature (Fig. 6) which leads to void formation (Fig. 7) with corresponding negative reactivity (Fig. 8). Consequently, the reactor power as shown in Fig. 9 exhibits self-shut- down behavior. It is seen that after about 200 seconds, the reactor power is low enough in such a way that the mass flow rate in the obstructed channel is sufficient for cooling down the fuel element and stopping the subsequent void production. In the extreme case of total blockage in the F.E. channel (Fig. 10) large vapor product (Fig. 11) occurs and leads to local dry out of the fuel plate (Fig. 12). In this case, although the large feedback is involved, the remaining reactor power which was calculated by the point kinetic model causes the clad temperature to reach its melting point which causes the loss of integrity in the F.E. and the release of radioactive materials to the environment (Fig. 13).

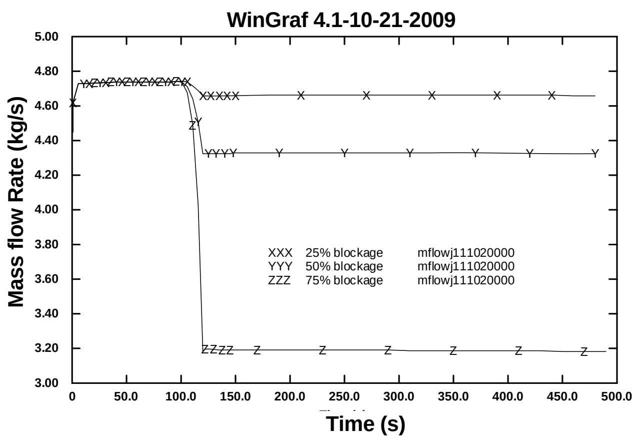

Fig. 3. Mass flow in the partially obstructed channel of T.R.R.

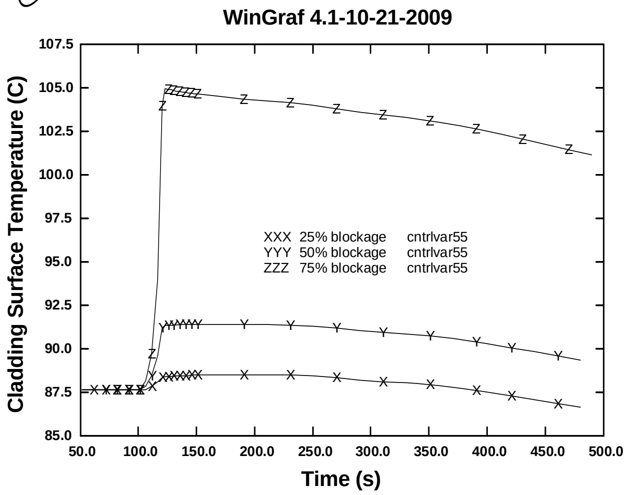

Fig. 4. Clad Temp. in the partially obstructed channel of T.R.R.

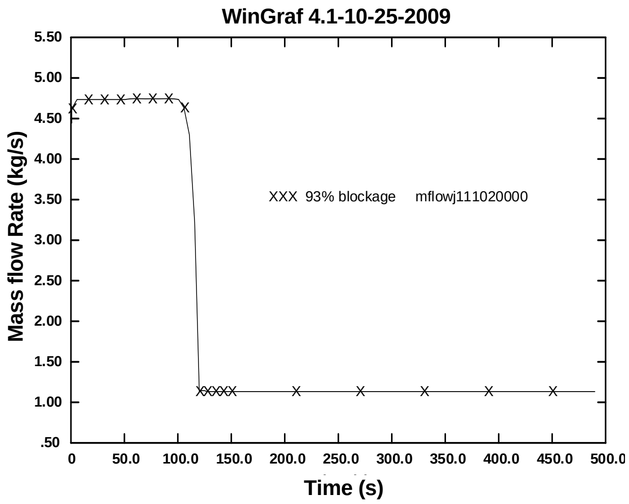

Fig. 5. Mass flow in the partially obstructed channel of T.R.R.

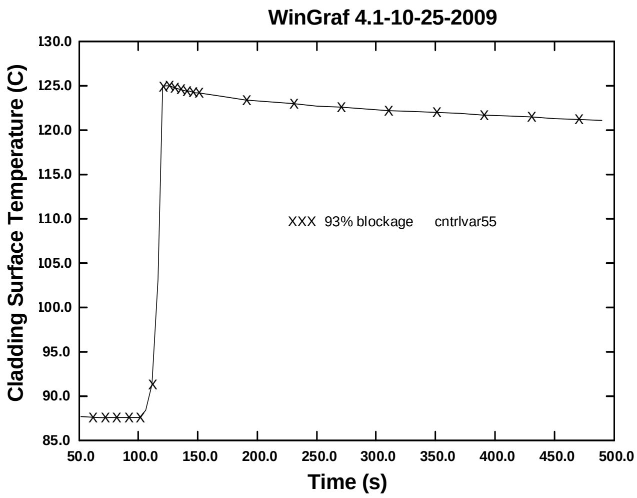

Fig. 6. Clad temperature of partially obstructed channel of T.R.R.

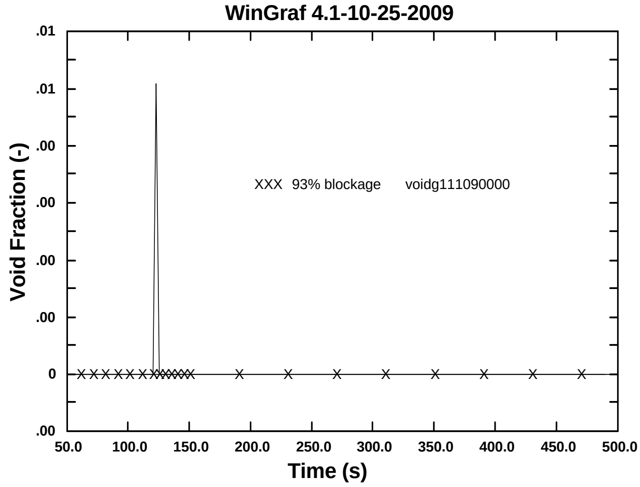

Fig. 7. Void fraction in the partially obstructed channel of T.R.R.

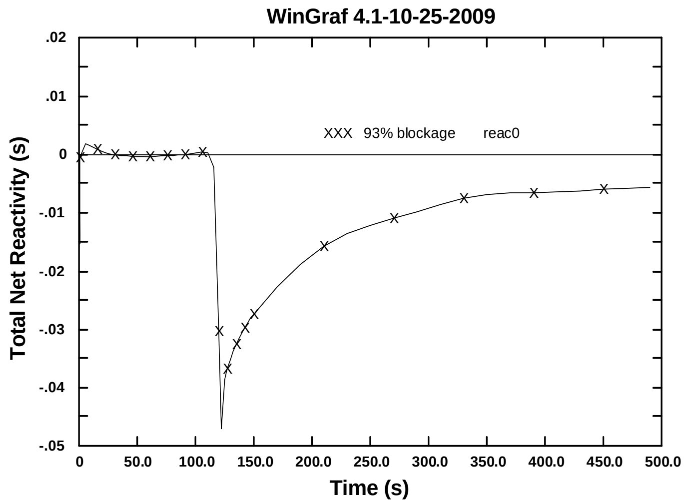

Fig. 8. Total reactivity in T.R.R.

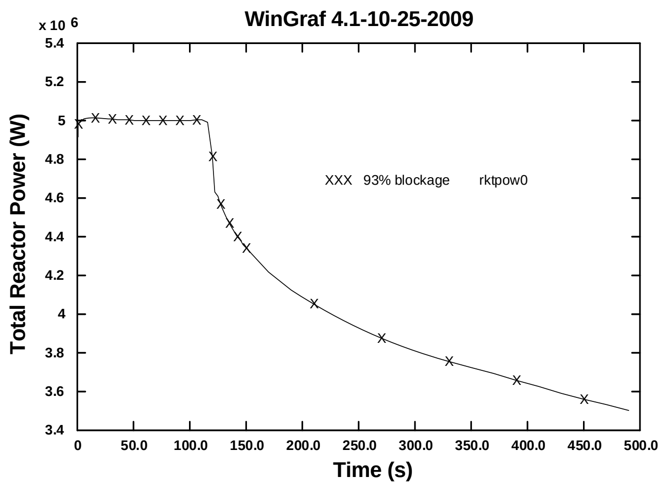

Fig. 9. Total power of T.R.R.

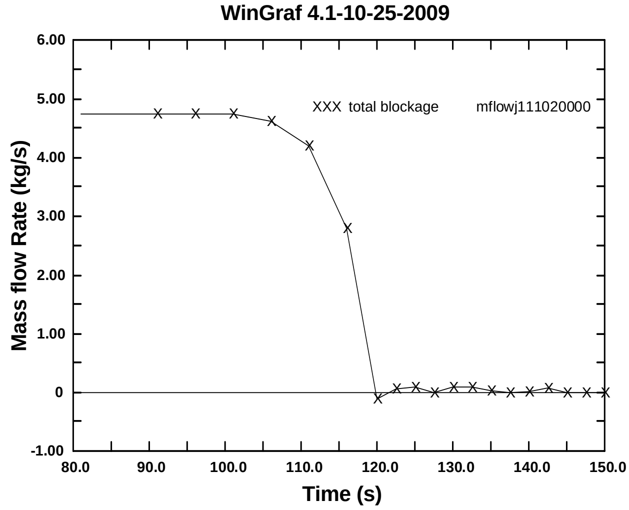

Fig. 10. Mass flow in the totally obstructed channel of T.R.R.

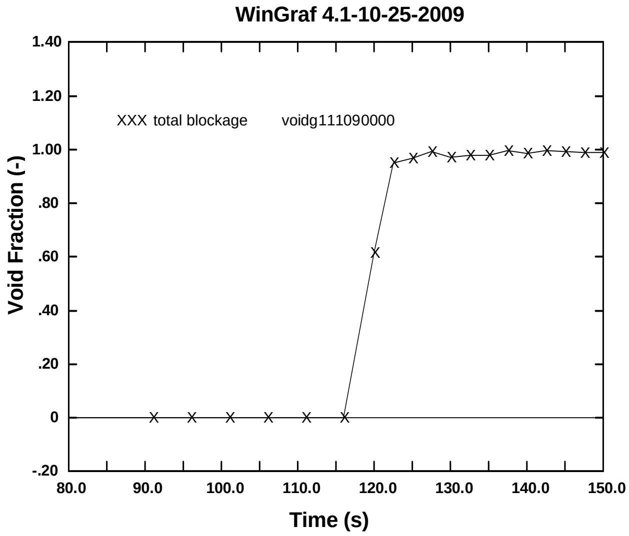

Fig. 11. Void fraction in the totally obstructed channel of T.R.R.

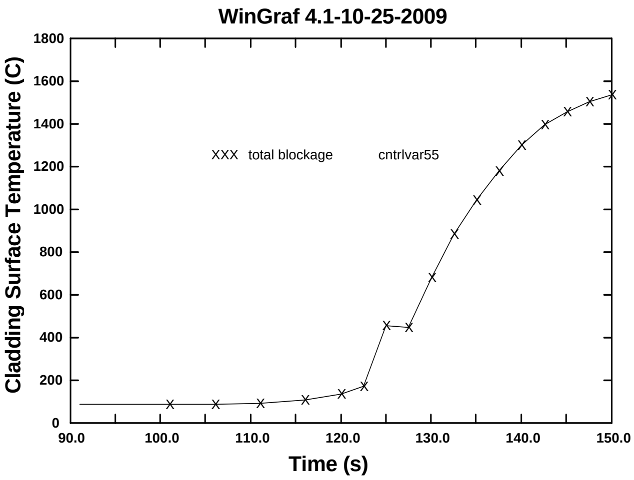

Fig. 12. Clad Temp. in the totally obstructed channel of T.R.R.

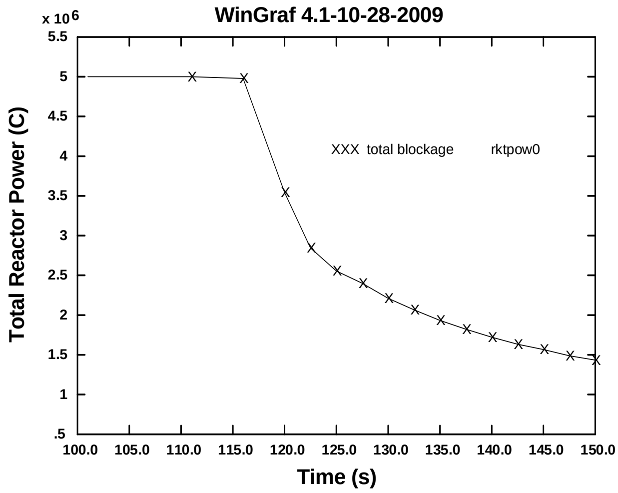

Fig. 13. Totally power of T.R.R.

7- Conclusions

Regarding our research we come to the conclusion chat the blockage of T.R.R. fuel element in two scenarios, a partial blockage of hot F.E. considering four different obstruction levels: 25%, 50%, 75% and 93% of nominal flow area, extreme scenario consisting of total blockage of the same F.E. which are analysed by RELAP5 system code with the following main results:

- In case of flow blockage under 93% in the nominal flow area of the hot F.E., only the increase of the coolant and clad temperatures are observed without any consequences for the integrity of the F.E. The mass flow rate remains sufficient for cooling the clad safely.
- In case of total obstruction for the nominal flow area, it has been observed that the transient turns out to be a severe accident due to the dry out conditions, which are reached shortly when melting the cladding occurs.
- The obtained results are comparable with available results in [5] presenting the flow blockage analysis of a single fuel element in the IAEA 10MW research reactor.

Nomenclature

T.R.R.	Tehran Research Reactor
F.E.	Fuel Element
S.F.E.	Standard Fuel Element
C.F.E.	Control Fuel Element
H.u.T.	Hold-up Tank
R.R.	Research Reactors

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