Analysis of Channel Blockage of MNSR Reactor Using the System Thermal-Hydraulic Code RELAP5/MOD3.3

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ABSTRACT

The increased extensive use of research reactors and improved regulatory and operational safety requirements have increased the use of more realistic simulations of the plant phenomena involved during steady-state and transient conditions. The earlier adopted conservative model assumptions in the reactor safety analysis which were based on conservatism are now been replaced with best-estimate methodologies. The best-estimate approach aims at providing a detailed realistic description of postulated accident scenarios based on best-available modelling methodologies and numerical solution strategies sufficiently verified against experimental data from differently scaled separate and integral effect test facilities.

The core behaviour of Ghana Research Reactor one (GHARR-1) Miniature Neutron Source Reactor (MNSR) during the loss of flow has been investigated. Steady-state and transient analysis were done with best estimate code RELAP5/MOD3.3. The simulated transient characterizes a Loss-of- Flow-Accident (LOFA) type transient. The study forms part of the ongoing core conversion program that is currently ongoing at the facility to convert the reactor from highly-enriched uranium (HEU) to low-enriched uranium (LEU) fuel.

Partial and total blockage of coolant to the reactor core transients were performed to study the behaviour of the reactor. It was observed in the case of partial blockage that although boiling occurred in the blocked channels, which lead to increase in both coolant and cladding temperature, the reactor presented a safer steady again due to in-flow of coolant from adjacent channels to the blocked channels. The calculations showed that cladding and coolant temperatures of blocked channels are below the melting point of the assembly. For total blockage the calculations ended abruptly at about 70 s after the start of transient. Therefore we could not observe the whole transient but the observed phenomena indicate unsafe behaviour of the reactor.

1 INTRODUCTION

The continuous interest in commercialization of nuclear research reactors has led to consideration of their corresponding safety issues. Safety analysis of research reactors highlight on scenarios which can affect the safety of the plant. It is based on simulations of selected accident scenarios identified by the licensee through experience, from the vendor or events from similar reactors. The thermal-hydraulic analysis is considered as an essential aspect in the study of safety of nuclear reactors, since it can predict proper working conditions, steady-state and transient, thereby ensuring the safe operation of a nuclear reactor [1][2]. The International Atomic Energy Agency (IAEA) recommends that simulations are performed using validated nodalizations and internationally recognized verified and validated (V&V) codes. Thermal-hydraulic (TH) Best Estimate (BE) system codes are capable of providing more realistic information on the status of the plant, allowing better prediction of the real safety margins.

A number of System Thermal-Hydraulic (SYS-TH) codes, such as RELAP5, ATHLET or CATHARE, have been widely used in nuclear reactor safety analysis. The SYS-TH code RELAP5 was developed to simulate transient scenarios in nuclear reactors such as PWR, BWR, VVER, etc. Only a handful of work has been done on research reactors to access the applicability of the code on operating conditions[3][4].

SYS-TH code RELAP5 solves eight field equations for eight primary independent variables: pressure, two phases (i.e., one for liquid and one for vapour) specific internal energies, vapour volume (i.e., void) fraction, two phases velocities, non-condensable quality, and soluble component (e.g., boron) density [5].

The widespread research reactors in the world, and the safety concern of them, has necessitated the use of more realistic simulations of the phenomena involved during steady state and transient conditions and to be able to study the design and identify the safety requirements which needs to be relaxed or improved [6][7].

Removal of heart from the core of a reactor is very crucial phenomena that need much analysis. The inadvertent loss of coolant flow to remove the heat generated in the core may cause by factors such as pump or valve failure, core blockage or redistribution of coolant. Each of these phenomena can lead to unsustainable over heating of the fuel cladding and can eventually lead to core melting. The probability of occurrence of such accident (eg. core blockage) depends on the design of the reactor core coolant flow whether upward or downward flow.

The objective of this paper is to open a case for critical study of the behaviour of MNSR nuclear research reactor after a partial and total blockage of its coolant channels with best estimate code RELAP5 model. It is also to support the regulatory decision for approval of license for the ongoing safety analysis of reactor core conversion project.

2 DESCRIPTION OF THE EXPERIMENTAL FACILITY

2.1 Description of GHARR-1

The Ghana Research Reactor – 1 (GHARR-1) is a Chinese Miniature Neutron Source Reactor (MNSR). It is a low power research reactor with thermal power of approximately 30 kW.

The reactor is fuelled with U-Al alloy with a U-235 enrichment of 90.2%, and it has only one control rod. It is cooled by natural convection. As part of the ongoing global effort to convert research reactors from highly-enriched uranium (HEU) to low-enriched uranium (LEU) fuel, the GHARR-1 fuel will be replaced with a core consisting of uranium dioxide (UO_2 clad with Zircaloy-4 alloy) fuel pins with a nominal enrichment of 12.5 % and with maximum thermal power level of 34 kW to achieve the same flux for its utilization. Main characteristics of GHARR-1 are shown in Table 1 [8].

Table 1: The main parameters of GHARR-1 MNSR

Parameter	HEU	LEU
Reactor type, rated thermal power (kW)	Pool-tank, 30	Pool-tank, 34
U-235 enrichment (%)	90.2	12.5
Core shape	Cylindrical	Cylindrical
Core diameter (cm)	23	23
Core height (cm)	23	23
Fuel element shape	Thin rod	Thin rod
Fuel element number in the core	344	348
Fuel/Cladding material	U-Al alloy/ Alalloy (Al-303-1)	UO2/ Zircaloy-4
Fuel rod Diameter (mm)	4.3	4.3
Fuel Length (mm)	230.0	230.0
No. of fuel rod positions	350	350
No. of dummy elements/thickness (mm)	6/0.6	2/0.6
Dummy element material	Al	Zircaloy-4
Material of control rod guide tube	Al	Zircaloy-4

As shown in Figure 1, fuel pins are arranged in 10 multi-concentric circle layers with the pitch distance of 10.95 mm. Due to the physical core design characterized by under-moderated system, a large negative temperature feedback coefficient of reactivity is achieved. The core region of GHARR-1 is located 4.7 m under water, close to the bottom of a watertight reactor vessel. The quantity of water is 1.5 m^3 in the vessel, which serves the purpose of radiation shielding, moderation and as primary heat transfer medium. In addition, heat can be extracted from the water in the vessel by means of a water-cooling coil located near the top of the vessel. The water-filled reactor vessel is in turn immersed in a water-filled pool of 30 m^3 [9].

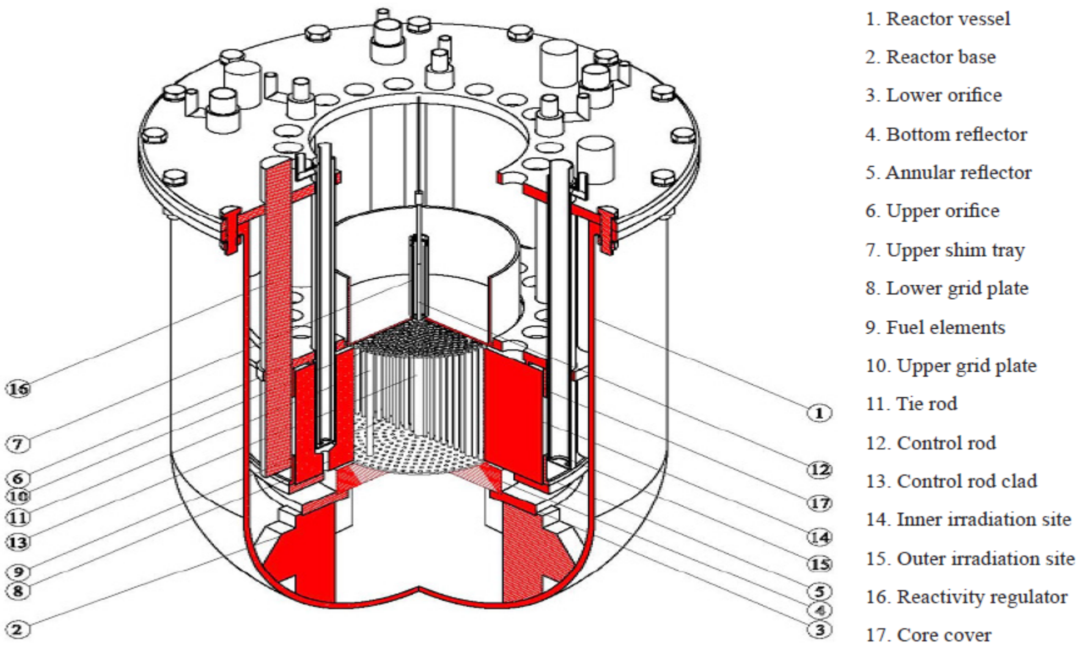

Figure 1: Schematic Diagram of GHARR-1

2.2 Heat Removal from the Core of GHARR-1

The heat generated by nuclear fission is conducted through the fuel meat to the fuel cladding, and transferred to the coolant by convection. Cold water is drawn through the inlet orifice by natural convection. The water flows between the fuel elements and comes out through the core outlet orifice. The hot water rises to mix with the large volume of water in the reactor vessel, and to the cooling coil. Heat is transferred through the walls of the container to the pool water. In Figure 2 it can be seen how heat is removed from the core and the heat transfer mechanism. The core inlet flow orifice slows down the natural circulation of fluid through the core.

2.3 Description of the Transient

The IAEA has described channel blockage as a transient event with different characteristics depending on direction of coolant flow. Downward cooling flow can lead to blockage due to objects dropping into the pool. Upward cooling flow can lead to blockage due to objects inside the primary cooling system piping being dragged into the core by the action of the pump [1]. In this paper two different transient events related to TH channel blockage have been investigated using RELAP5/MOD3.3 best estimate code. The core configuration of the GHARR-1 MNSR as depicted in Figure 2 has an upward flow. The probability of fuel swelling to block the channels in this case is higher than a material falling into the pool to block the channels [10].

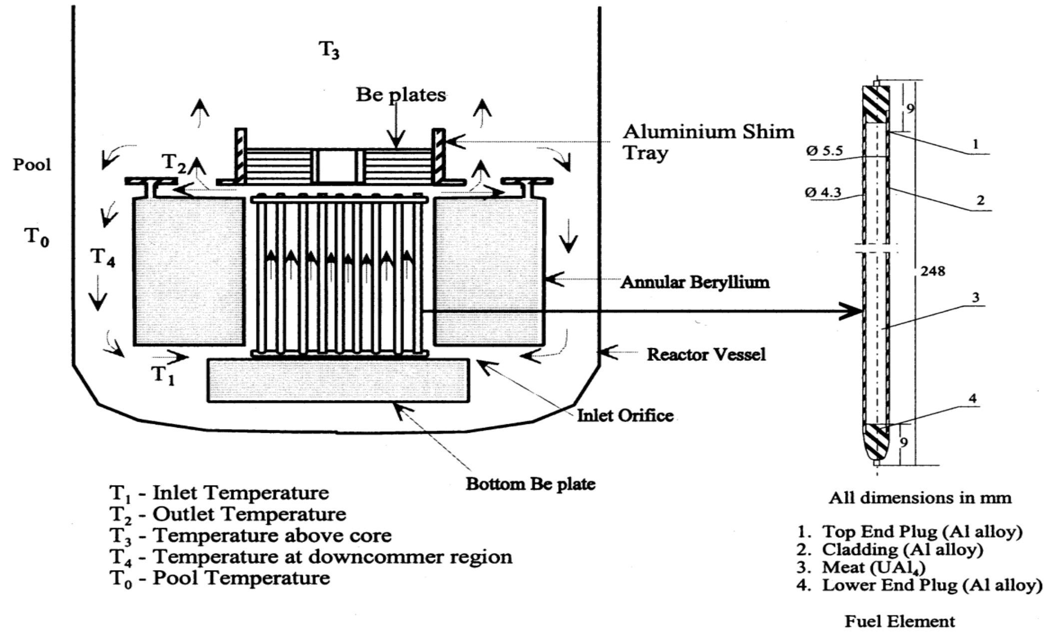

Figure 2: Schematic diagram of natural convective flow of coolant water within the reactor vessel and enlarged view of fuel element.

3 THE CODE AND NODALISATION

In this study, it has been assumed that the code has fulfilled the validation and qualification process and a “frozen” version of the code has been made available to the final user. This means that the code user did not modify or change the physical and numerical models of the code. In order to represent the GHARR-1 MNSR by the RELAP5 MOD3.3 code, the nodalisation developed by the Argonne National Laboratory was modified to include all the 10 coolant channels and their corresponding heat structures.

To perform this type of transient simulation, a qualified nodalisation of the MNSRP of GHARR-1 reactor was developed. The nodalisation includes the whole MNSR system: the core, the vessel, and the surrounding pool as well as the reflectors. Ten coolant channels that represent the whole core were modelled, as shown in Figure 3.

In order to simulate actual situation, cross flow was used since it provides coolant redistribution from adjacent channels after the channel blockage.

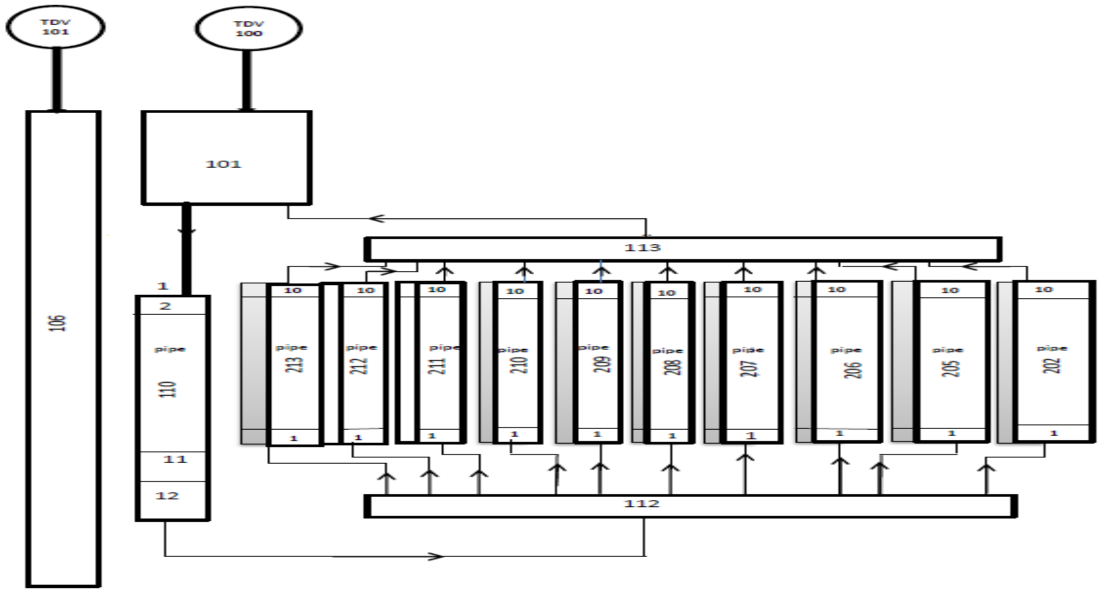

Figure 3: RELAP5 nodalisation of GHARR-1

4 ANALYSIS OF RESULTS

4.1 Steady-state Calculations

Steady-state calculations were performed to show that a stabilized input and boundary conditions for the transient calculations were achieved. It can be observed in Figure 4 that rod surface temperature (RST), fluid temperatures and the development of mass flow rate resulting from the natural circulation during operation at normal reactor power of 34 kW steady-state conditions are met. The simulation was considered at a constant nominal power of 34kW. The inlet pressure is kept constant by the Time Dependent volume (TDV) component modelling the condition of gas accumulator at the top of reactor vessel which is under atmospheric pressure.

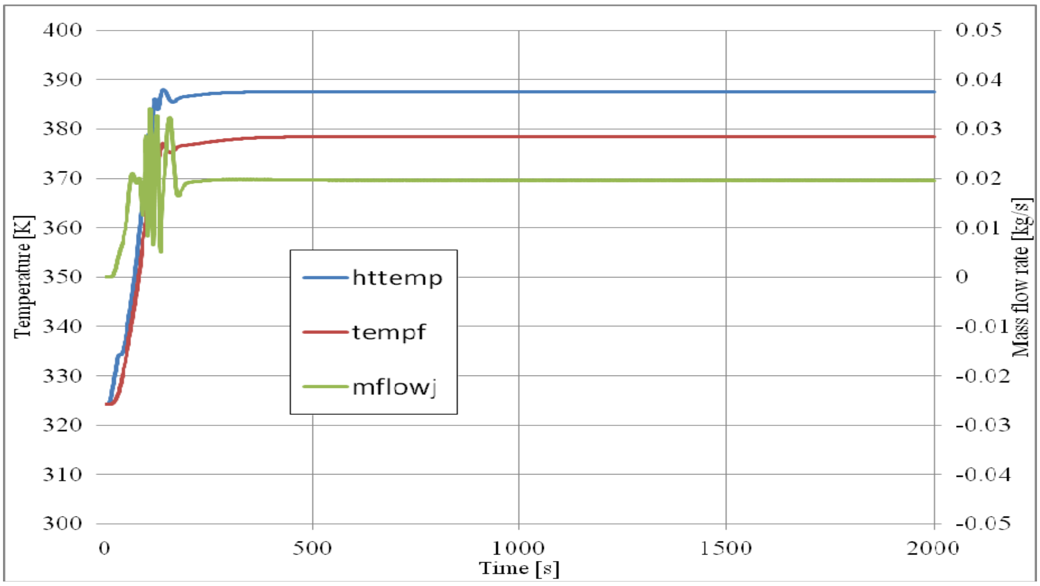

Figure 4: Steady State

4.2 Transient Analysis

The analyses of transient "core blockage" have been divided in two; partial and total blockage. As depicted in Figure 1 and Figure 3, ten coolant channels that represent the whole core were modelled. The first case was simulation of partial blockage where six channels representing 35% (first six channels 1-6) of normal flow area were blocked. The second was an extreme condition where a total blockage (i.e. 100% blockage of normal flow) was considered.

4.3 Partial Blockage

A summary of RELAP5 calculation results for blocking six channels representing 35% of normal flow area is given in Figure 5 - Figure 8. The transient began at 800 s of calculation time with blocking the mass flow in six selected channels, as it can be seen in the Figure 5. The core configuration is such that the ten coolant channels are interconnected hence fluid can flow through from adjacent channels. There was a mass flow from adjacent channels to the blocked channels which made the flow continue again. It was observed in the calculation that after around 50 s of the start of transient, situation has been stabilized at the new values.

The cladding temperatures in the blocked channels increased after the start of the transient because of the reduced coolant flow in the channels. At about 830 s of the calculation the cladding temperatures stabilizes till the end of the calculation, as shown in Figure 6. Furthermore, the reduced flow in the blocked channels causes an increase in the coolant temperature as depicted in Figure 7.

Figure 8 shows the void fraction of the upper volume of the obstructed channels with subsequent void production in the upper volumes of the blocked channels indicating boiling in the blocked channels.

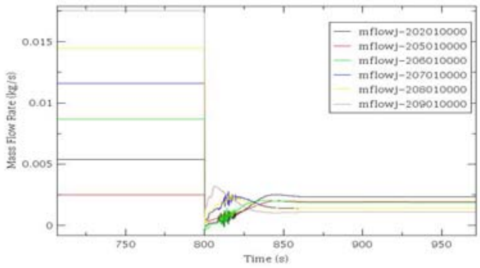

Figure 5: Mass flows in the blocked channels

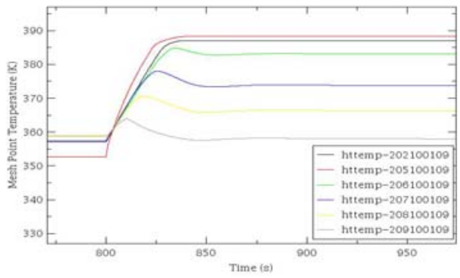

Figure 6: Cladding temperatures of the blocked channels

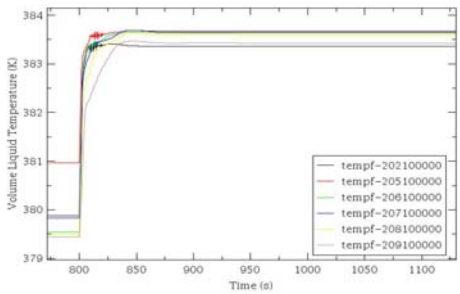

Figure 7: coolant temperature

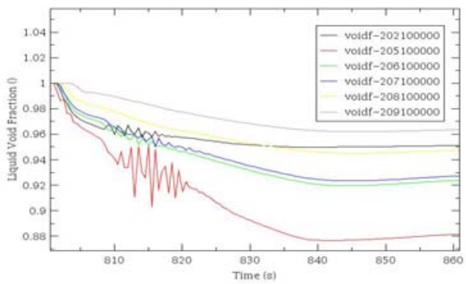

Figure 8: Void in the blocked channels

4.4 Total Blockage

For total blockage transient all ten cooling channels were blocked at 800 s of calculation time, after the steady-state conditions had been achieved. Calculation stopped at about 870 s since the pressure and temperature went out of the code calculating boundaries.

As shown in Figure 9, the mass flow rate, after the start of transient reduced to zero. This fast loss of flow through the core due to total blockage of all the channels lead to a rapid coolant temperature as shown Figure 10.

Figure 11 shows cladding temperature in which the temperature increases with oscillations. The calculations stopped at about 70s after the start of transient. These conditions resulted in production of vapuor in the core which can eventually cause dry out in the coolant channel as it can be seen in Figure 12.

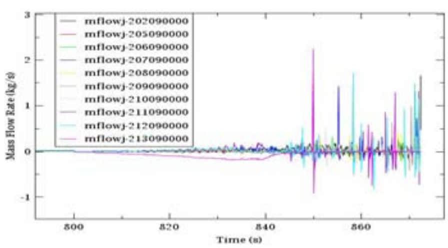

Figure 9: Mass flows in all channels

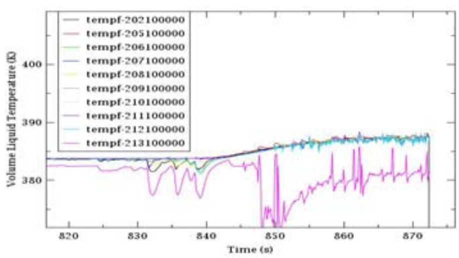

Figure 10: Coolant temperatures at top of the channels

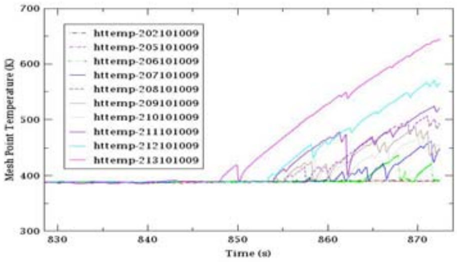

Figure 11: Cladding temperatures in all channels

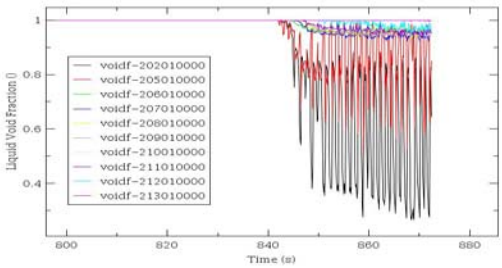

Figure 12: Void fractions at bottom of channels

5 CONCLUSION

This study investigated total and partial core coolant blockage transient of GHARR-1 with a validated SYS-TH code RELAP5/MOD3.3. The simulated transient characterizes a Loss-of- Flow-Accident (LOFA) type transient. For partial blockage transient, reduction in mass flow shortly after the start of transient causes an increase in coolant temperature in the blocked channels leading to void formation. Although boiling occurred in the blocked channels, the reactor conditions stabilized at values that are within required safety margins due to in-flow of coolant from adjacent channels into the blocked channels. The calculated cladding and coolant temperatures of blocked channels are below the melting point.

On the other hand, we could not observe the whole transient of total blockage due to stop of the calculation before the end. The observed phenomena during the total blockage, such as increase in temperature resulting in the coolant boiling in the core shortly after the beginning of the transient, indicate unsafe behaviour of the reactor.

Therefore, based on the calculation it can be concluded that natural circulation is capable of cooling the reactor at 35% blockage without posing any significant threat to the fuel material integrity from the temperature point of view. In conclusion channel blockage needs proper attention in safety evaluation of this RR and proper actions should be taken

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