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SAFETY ANALYSIS OF ADVANCED CANDU REACTOR-700 (ACR-700) DURING TRANSIENT AND EMERGENCY CONDITION USING ACR SIMULATOR

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Abstract

The development of nuclear technology leads to improvement in nuclear power plant design. The latest generation of nuclear reactor tries to rely more on passive system to minimize human intervention and increase the safety of the nuclear power plant itself. ACR-700 is designed to be able to cope with some transient's condition. This study tries to simulate the condition of ACR-700 during the transient condition loss of one of reactor coolant pump using ACR Simulator developed by IAEA. The ACR-700 safety system successfully identifies the malfunction and stop the malfunction to escalate. In addition, this paper also tries to simulate the previous transient condition with another malfunction in reactor setback and setback system, one of the safety systems of the ACR-700.

Keywords: ACR-700, transient condition, ACR simulator, pump loss

Introduction

The energy demand is increasing every year as the global technology increases. The depletion of non-renewable energy such as oil and coal and some pressure from the environmentalists to abandon non-renewable energy make the energy supply can't cope with the energy demand. The abundant resource and high energy yield from nuclear power makes the nuclear power one of the most prominent candidate to supply the world's energy [1]. The nuclear technology has been developed continuously from the first time nuclear power plant being operated in 1954 [2]. The efficiency and safety of the nuclear power plant has greatly increase due to improvement in power plant models. The latest generation of nuclear power plant relies more on the passive system in order to ensure the safety of the plant in emergency situation [3][4].

Unfortunately, the phobia toward nuclear power plant still exists. From the fear of weaponized nuclear technology such as the bombing of Hiroshima and Nagasaki in 1945 to nuclear power plant accident such as in Three Miles Island and Chernobyl and especially the latest accident in Fukushima has propelled the fear of the society toward nuclear technology [5]. This condition leads to some decommissioning of some of the nuclear reactors and halted progress in nuclear technology development [6]. Atomic Energy Canada Limited (AECL) developed Advanced Candu Reactor 700 (ACR-700). ACR-700 is one of the latest generations of Pressurized Heavy Water Reactor (PHWR).

It is developed from CANDU design. As improvement from the previous CANDU design, it could be built with lower capital cost and shorter construction time. But even though ACR-700 has lower capital cost, it still maintains high-capacity factor. The new ACR-700 design also has longer operating life. It uses much simpler component thus easier and cheaper to replace its component. This design leads to low operating cost. It also got enhanced safety features [7].

ACR-700 use slightly enhanced uranium as fuel (2.1% wt U-235). It uses light water as coolant and use heavy water as moderator as oppose to CANDU that use heavy water as both moderator and coolant. The new design thus has lower heavy water inventory which is one of the reasons for its lower capital cost. The lower heavy water inventory also led to even more compact design than the previous CANDU reactor.

As one of the latest generations of PHWR, ACR-700 is equipped with passive systems. This includes two independents shut down systems. It is also filled with low pressure and temperature moderator that could act as heat sink. Furthermore, it is surrounded by water shield tank. The reactor also equipped with emergency gravity supplied feedwater toward steam generators. The reactor is also contained in pre-stressed concrete to limit the exposure of radioactive material toward the environment.

ACR-700 is designed can cope with transient's condition. An experimental study is needed to verify improve the safety of the design. As computational technology development, computational simulation could be performed to simulate condition during transient condition. As it is easier to use computational simulation than a lab experiment. However, the use of computational simulation to verify the safety system of ACR-700 during transient or even emergency condition is still rarely done.

One of the computational simulation already done is to investigate how ACR-700 design would cope with transient condition of Small Break Loss of Coolant Accident (LOCA) [8]. The simulation is done using CATHENA 3.5d. This paper study the safety system cope with the transient condition especially loss in one of reactor coolant pump and reactor setback and step back both fails using computational simulation. ACR-700 safety system should be able to cope with both the transient and emergency condition without any human intervention. Furthermore, the author hope that the result from this study could be used to increase understanding of the safety system in nuclear reactor especially in ACR-700. A better understanding of nuclear safety itself is the only way to improve and create a better nuclear power plant design.

Method

This paper simulates ACR-700 during normal operation at 100% power, transient, and emergency condition using Advanced Candu Reactor Simulator (ACR Simulator). ACR Simulator is developed Cassiopeia Technologies Inc. in 2011 and is one of the IAEA simulator collections. ACR Simulator mainly designs to simulate ACR-700, but it could also be used to simulate other advanced nuclear heavy water reactors. The main window will appear once we open the ACR Simulator. The main window shows 3D design of the ACR-700 and panel on the left to choose the preferable Initial Condition (IC). ACR Simulator provide 5 loadable ICs including simulation of the reactor at 100% and 75% power. This paper would study ACR-700 at 100% power.

Figure 1. Main window of ACR-700

Once the user loads IC a new window with default setting of the loaded IC will pop out. On top of windows, user can see panel that serves as indicator light for the current reactor condition. While in bottom user can see panel for general condition of reactor. The infamous "reactor trip" and "turbine trip" buttons could be seen on bottom left of window. These buttons resemble real buttons in reactor control room to completely stop the nuclear reactor.

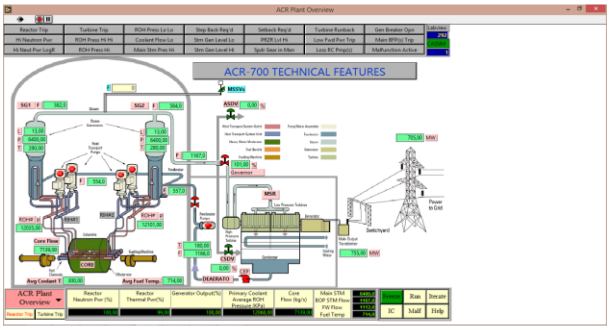

Figure 2. ACR plant overview window

The "run" and "freeze" buttons is used for starting and pausing the simulator respectively. It can be seen on bottom right of the window along with "malf" button. ACR Simulator is equipped with 20 malfunction that could be simulated directly without accessing any control panel.

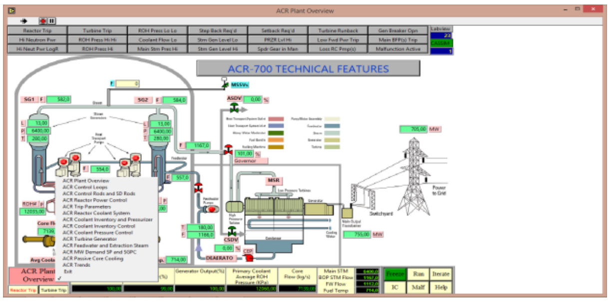

Figure 3. ACR Simulator navigation panel

Accesing the black arrow on the bottom left side of the window could navigate us through ACR Simulator displays screens (Fig. 3). ACR Simulator has 14 displays screen and each represents different aspect or control panel of the reactor such as control rod, coolant, and turbine control. Some display screens merely just a display of the current condition of the reactor. ACR Plant Overview window (Fig. 2) has no input toward the simulation and only show a 'line diagram' of the main plant system and parameters. But for some other display screens such as Reactor Coolant System (Fig. 4), user could make input for the simulation by interacting with the display screens. For example, user could manually disable pump by clicking the P1, P2, P3, or P4 button.

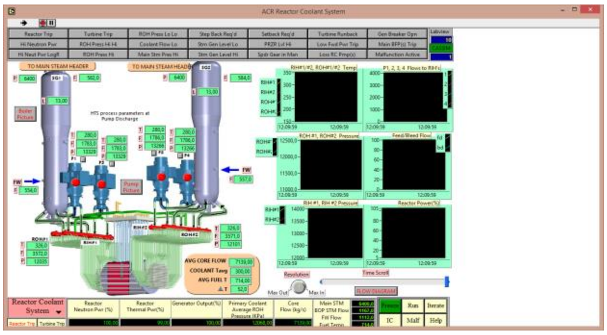

Figure 4. Reactor coolant System window

As previously mentioned, to run the simulator at 100% power, we just have to load the IC and observe the behavior of each of the components in the reactor. The 100% power condition already set up by the IC and there is nothing to tweaks by the user. For the second part of the simulation, this also simulate the transient condition. It could be simulated by first load the 100% power condition IC. Once the window screen pops out and all the reactor parameters set up to resemble the 100% power working condition, user have to manually input the transient condition by using “Malf” button and then choose the condition that is wanted to be simulated from the list. We add “Loss of one PHT Pump P1” as this paper tried to simulate the loss of one of the main pumps in the reactor.

To simulate the emergency condition, it is not much different from the transient condition. We have to load the 100% power condition IC. Then we add the malfunctions through the “Malf” button. In this paper we add “Reactor Setback/Stepback both fail” first since we want to simulate this condition first. Once the simulator simulates the first malfunction, we add the second malfunction “Loss of one PHT Pump P1” by using the “Malf” button. We then could compare the results from all of the simulations. The result from 100% power condition would be used as the benchmark for the simulation. As ACR-700 is equipped with passive system, ACR-700 should be able to handle the transient condition without any intervention.

Result and Discussion

The main parameters that will be observed in this paper are related to the coolant of the reactor as this paper focused on the loss of one of the main pump transient condition. Those parameters could be observed through “Reactor Coolant System” panel. Those parameters include the temperature and pressure of both RIHs and ROHs of the reactor and also coolant flow to ROHs.

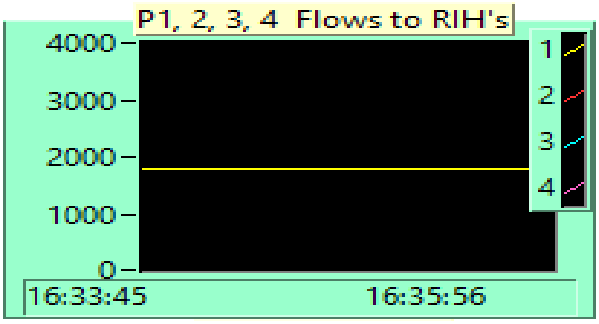

Figure 5. Coolant flows to RIHs from all the main pumps during 100% power operation

The 100% power as normal working condition shows a stable coolant flow from all the four pumps (Fig. 5). The steady water supply to the RIHs also followed by a steady temperature and pressure on both RIHs and ROHs as can be seen on Fig. 6 to Fig. 8

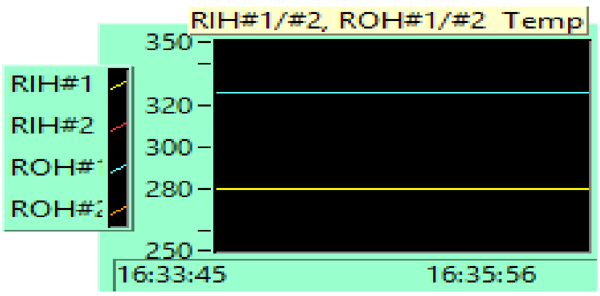

Figure 6. Temperature of RIHs and ROHs during 100% power operation

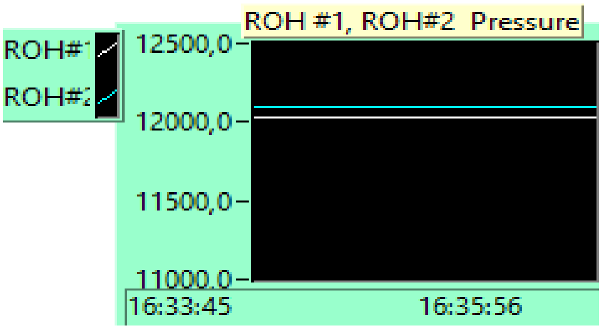

Figure 7. Pressure in ROHs during 100% power operation

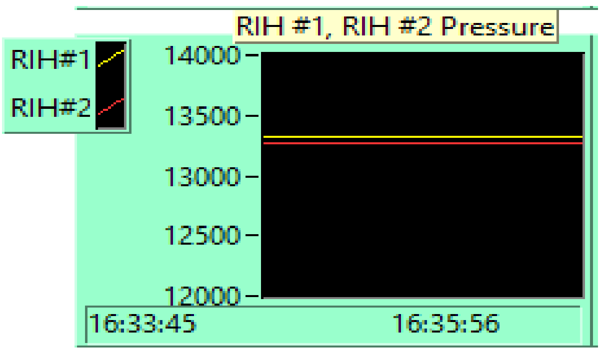

Figure 8. Pressure in RIHs during 100% power operation

We could notice a slight difference in temperature and pressure on each RIHs and ROHs even though technically they are supplied with similar pump. The slight difference is due to bleed and feed flow for Heat Transport Purification System. For the transient condition where Pump 1 is loss due to a malfunction, we can see that the coolant flow in Pump 1 is drastically down (Fig.9).

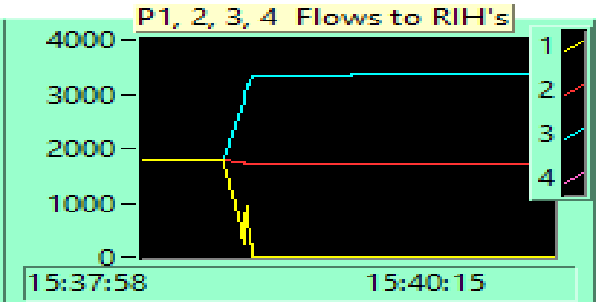

Figure 9. Coolant flows to RIHs from all main pumps during transient loss of one RC pump condition

Pump in the same loop to Pump 1 – Pump 3 – increase its coolant flow to compensate the loss of flow from Pump 1. A little jagged flow from Pump 1 before it finally touches 0 flow is due to most of its flows are directed toward Heat Transport Purification System through bleed flow. Similar to that case, the decrease flow in loop two is caused by the feed flow. The bleed and feed flow can be seen on Fig.10.

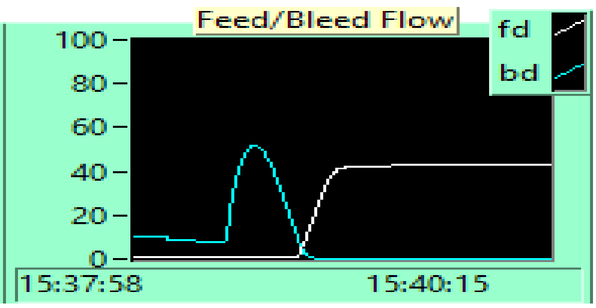

Figure 10. Feed and Bleed flow during transient loss of one RC pump condition

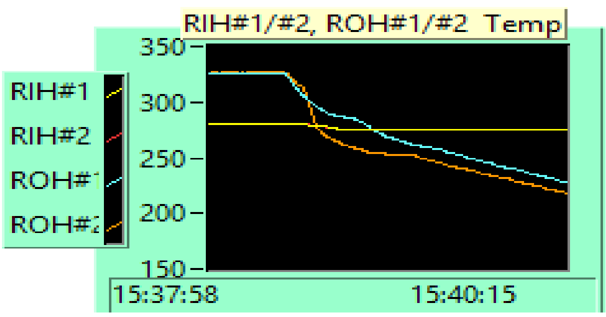

Figure 11. Temperature of RIHs and ROHs during transient loss of one RC pump condition

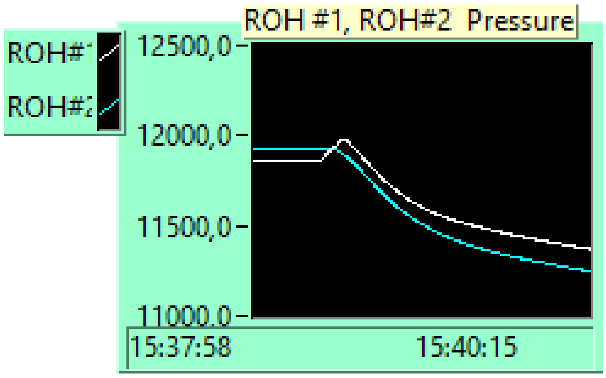

Figure 12. Pressure in ROHs during transient loss of one RC pump condition

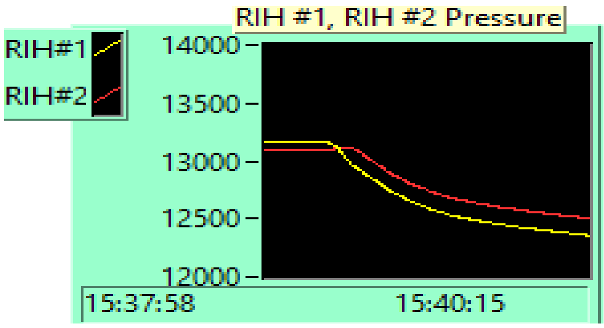

Figure 13. Pressure in RIHs during transient loss of one RC pump condition

The reactor setback is initiated right after the Pump 1 under malfunction as loss of one of the reactor coolant pumps is one of the causes for reactor setback. This leads to rapid decrease in reactor power, then leads to decrease in ROHs temperature as the reactor thermal power also decrease (Fig. 11). The decrease in temperature leads to more liquid phase in the pressurizer, thus also decreasing the coolant pressure (Fig. 12 and Fig. 13). The flow from each pump during this emergency condition is not much different from the coolant flow during the previous transient condition. The Pump 3 cover the loss of coolant flow the Pump 1 (Fig. 14). The main difference is that during this condition, reactor stepback failed to operate, thus leaving the reactor power at high level as can be seen on Fig. 15.

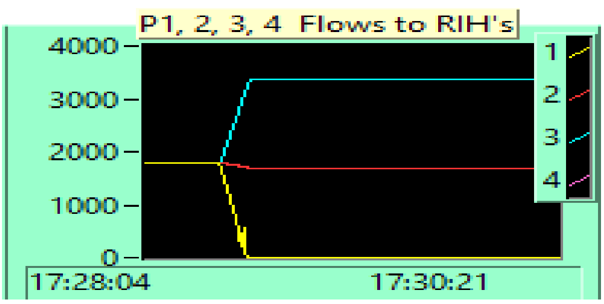

Figure 14. Coolant flows to RIH during emergency condition

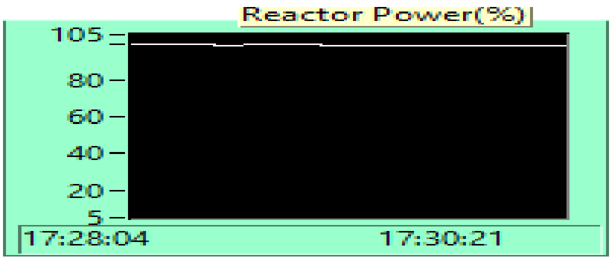

Figure 15. Reactor power during emergency condition

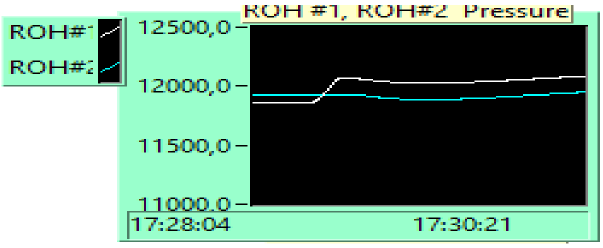

Figure 16. Pressure in ROHs during emergency condition

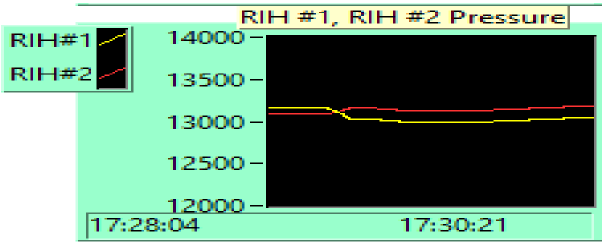

Figure 17. Pressure in RIHs during emergency condition

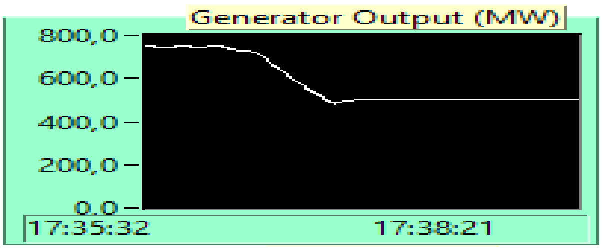

Figure 18. Generator output during emergency condition

This condition would lead to increase in pressure in steam generator. During this condition turbine bypass valve, consist of Condenser Steam Discharge Valves (CSDV) and Atmospheric Steam Discharge Valves (ASDV) would open to relieve steam pressure. This leads to drop in turbine steam and decrease turbine output as we can see on Fig 16 to Fig. 18. As first loop lose one of its pumps, there is a sudden increase in temperature of ROH 2 as pump lose some of its coolant flow (Fig. 19). This condition leads to SG 2 produce steam with higher pressure than usual. The increase in steam pressure leads to increased pressure in SG 1 can to keep up with the steam generation process.

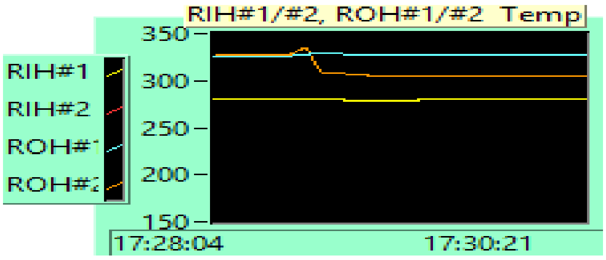

Figure 19. Temperature in RIHs and ROHs during emergency condition

As the steam pressure rises, the turbine bypass valves finally open to stabilize the pressure in SGs. Thus, the opening decreases the pressure of both ROHs. The pressure in both ROHs then started to build up more, and the valve would then open even bigger if there is no human intervention. The opening of turbine bypass valve also leads to decrease in temperature of ROH 2 while the temperature of ROH 1 is not much affected as the opening of the turbine bypass valve is determined by SG 1 in this situation. The reactor still be able to operate but as time passed, it become more riskier as the pressure build up in both steam generator. Human intervention is needed to stabilize the reactor back to normal condition. But the reactor safety systems could be able to cope with the emergency condition, giving time for the human intervention.

Conclusion

The results show that under normal working condition, the pressure in ROHs is stable at around 12.1 Mpa. While pressure in RIHs is stable around 13.2 Mpa. The temperature for RIHs and ROH is 280°C and 326°C respectively. This number is not greatly different from the ACR-700 unit data. The data then used as a benchmark for transient and emergency condition. The simulation of transient condition shows that the loss of one of the reactor coolant pumps would be covered by the pump in the same loop. Reactor setback is also operated automatically once the pump malfunction is detected. This condition result in rapid decrease in reactor power, thus prevent the malfunction to escalate and give chance for component maintenance. In case of reactor setback and step back failed to operate due to another malfunction, the reactor is still operating in full power. The other safety system started to operate to stabilize the reactor once certain parameter exceeded the tolerable criteria. In this simulation, the steam pressure trigger turbine bypass valve to open. The opening of turbine bypass valve caused decreased pressure in steam generator.

This condition makes the reactor still be able to function. But this condition still creates pressure being built up over the time. Human intervention is needed to fully stabilize the reactor back to normal working condition. In this simulation ACR-700 is able to endure a transient condition loss of a reactor coolant pump and emergency condition loss of a reactor coolant pump with reactor setback and step back failed to operate. Even though further human intervention is needed to fully stabilize the reactor during emergency condition.

Acknowledgment

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