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RESEARCH ARTICLE

A PCTRAN BASED ANALYSIS ON THE EFFECT OF BREAKSIZE AND COMPARATIVE STUDY BETWEEN HOT AND COLD LEG LOSS OF COOLANT ACCIDENTS IN VVER 1200 POWER REACTOR

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ABSTRACT

In this paper, a comparative analysis of loss of coolant accident (LOCA) in hot leg and cold leg of primary circuit in a VVER 1200 nuclear power plant is investigated. The effect of break size on the severity of the accident is observed. The break size was varied in the range 200-11350 cm^2. For all the accident scenarios, station blackout (SBO) condition is set up. Additionally, it is assumed that no ECCS (Emergency Core Cooling System) is available due to system malfunction. The whole scenario is simulated in PCTRAN (Personal Computer Transient Analyzer) software. Results reveal that with the increase in the size of the break area, the core uncovering time decreases sharply. However, for a break size of 2800 cm^2 or smaller, the water level in the core doesn't drop to zero, indicating that the core is partially uncovered throughout the accident scenario. In case of hot leg LOCA, the draining of the reactor vessel is observed to be more rapid compared to cold leg LOCA, while the core melting started earlier in case of cold leg.

KEYWORDS

Loss of Coolant Accident, Double Ended Guillotine Break, Station Blackout, Emergency Core Cooling System, VVER 1200

1. INTRODUCTION

If we talk about accidents in nuclear plants, loss of coolant accidents are of prominent types. Loss of coolant accident, as the name suggests, takes place when a part of the coolant is lost from the reactor vessel. Loss of coolant accidents can be classified according to the break size in the pipelines. It may be termed small break LOCA (SBLOCA) or large break LOCA (LBLOCA). Large break is considered usually when the break size is greater than a value of approximately 0.1 m^2 or 1000 cm^2 and small break is smaller than the mentioned value (Joyce, 2017). Accidents in nuclear plants maybe classified into two major types, design basis accident (DBA) and beyond design basis accident (BDBA). Design basis accidents or postulated accidents are those which are considered when a plant is designed and to mitigate its effects, there might be inherent, engineered, or other passive safety systems. As a result, a plant has the capability to withstand such kind of accidents.

On the other hand, beyond design basis are considered to be highly unlikely to happen and not considered fully in the plant design. According to IAEA, beyond design basis accidents have more severe consequences than design basis accidents having considerable core deterioration (International Atomic Energy Agency, 2019). After beyond design basis accidents like Chernobyl and Fukushima Daiichi, researchers are now taking deep interest in analyzing severe or beyond design basis accidents. A group researchers performed code crosswalk of Fukushima like simulations using MELCOR2.1/SNAP, TRACE/SNAP, PCTRAN and MAAP5.03 (Chiang et al., 2017). The authors analyzed the similarities, differences, and appropriate applications of these codes. A group researchers simulated Fukushima Daiichi accident using PCTRAN (Aliyu et al., 2018). Their obtained results were very similar to the findings of the Nuclear Accident Independent Investigation Committee (NAIIC) of the Fukushima Daiichi Nuclear Power Plant in Japan.

A group researchers analyzed thermal hydraulic parameters of VVER-1200 due to LOCA with a loss of offsite power (Nashiyat et al., 2019). The acquired results were consistent with all PSAR (Preliminary Safety Analysis Report) data on LOCA. A group researchers investigated the effect of inadvertent control rod withdrawal on the thermal-hydraulic parameters of a VVER-1200 nuclear power reactor using PCTRAN (Hossain and Islam, 2019). The authors identified the maximum limit of instantaneous positive reactivity insertion which is supported by the results from further simulations. A group researchers analyzed thermal hydraulic parameters of a nuclear reactor due to LOCA with and without ECCS (Dep et al., 2020). Their findings showed that, LOCA without Emergency Core Cooling System (ECCS) resulted in core meltdown along with a release of radioactivity after a specific time.

From the literature survey, it is evident that there have been numerous studies that investigated different accident scenarios in a NPP with PCTRAN. However, to the authors' knowledge, none of the studies considered LOCA with loss of ECCS, a BDBA. In this paper, both small break LOCA and large break LOCA was simulated, and the effect of break size was investigated. This was done both in the hot leg and the cold leg of the primary circuit to see if there is any differences between these two accidents. Finally, SBO condition was considered to further increase the severity of the accident. PCTRAN was used as the simulation software because of its reliable results.

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2. METHODOLOGY

In this work all the simulation data were generated using PCTRAN VVER 1200. PCTRAN has been chosen by the IAEA as the teaching platform for its annual IAEA Simulators Workshop since its launch in 1985 by Micro-Simulation Technology (Cliff Po, 1988). It is a pressurized water reactor simulator with two loops. The simulator's plant model is a generic design that employs a number of assumptions to simulate simplified plant conditions. Although the simulator is not designed for safety analysis, it can be utilized for research purposes as the general behavior of the plant responses during regular operations and transients is found to be quite accurate (Hossain and Islam, 2019; Cheng et al., 2012). Figure 1 presents the graphical user interface for PCTRAN VVER-1200. The normal operating conditions are also presented in the figure.PCTRAN is capable of simulating the plant in normal operating conditions, transient conditions and also accidental conditions. At first, a set of initial conditions is set. Then, as desired, normal, transient or accidental conditions, declared as "malfunctions" in the software, can be applied to the nuclear facility with a specific delay time and a fractional intensity. If required, multiple accidental conditions may be initiated simultaneously. For our current study, malfunctions 1and 2 were used. Malfunction 1 activates LOCA in hot leg whereas malfunction 2 activates LOCA in cold leg. It was desired to observe if there is any difference or similarities between hot leg LOCA and cold leg LOCA.

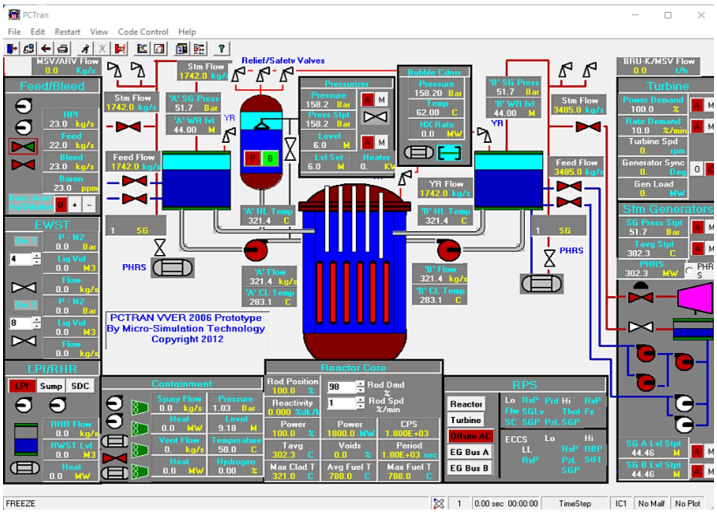


Figure 1: Graphical User Interface of PCTRAN (Main mimic view)

Additionally, loss of AC power, designated as malfunction 6, was also induced to simulate a station blackout. A delay time of 7 seconds was selected for initiating the accidents to give the reactor sufficient time to come to a steady state condition. For research purposes, all ECCS systems including the passive ones were made deactivated. The active ECCS were automatically deactivated due to malfunction 6. But the passive ECCS was deactivated manually. The simulations were run for a total of 300 seconds and the path of development of the accident was observed. Table 1 presents the initial steady-state plant conditions considered in the study.

Table 1: Initial steady-state plant conditions

Reactor power	100%
Reactor Core pressure	155 bar
Average temperature of the core	306.9° C
Steam Generator pressure	70 bars
Time in life	BOC (Beginning of Cycle)

For both hot leg and cold leg LOCA, five break sizes were used. Two of them were small break LOCA (SBLOCA) and three of them were large break LOCA (LBLOCA). Table 2 presents the break sizes considered in the study. The objective was to observe how the thermal hydraulic parameters as well as the core uncovering time vary with break sizes.

Table 2: LOCA type with break size

Break Size (cm^2)	Type of LOCA
200	Small break
500	Small break
2800	Large break
5675	Large break
11350	Large break

3. RESULTS AND DISCUSSION

Tables 3 and 4 represent the thermal hydraulic behavior of the reactor in case of LOCA in hot leg and cold leg respectively.

Table 3: Simulation data obtained for hot leg LOCA

Break size (cm^2)	Start of core uncovering time(s)	Time required for the water level in the core dropping to zero (s)	Average fuel temperature after 300 s (°C)	Maximum fuel temperature after 300 s (°C)	Maximum cladding temperature after 300 s (°C)	Pressure in the reactor after 300s(bar)
200	277.0	-	285.9	288.2	267.4	48.5
500	106.5	-	399.9	562.3	259.8	11
2800	27.5	60	1086.4	1251.1	372.2	5.2
5675	19.5	40	1687.9	1945.3	515.4	5.9
11350	15.5	25	2225.0	2225.6	581.3	6.9

Table 4: Simulation data obtained for cold leg LOCA

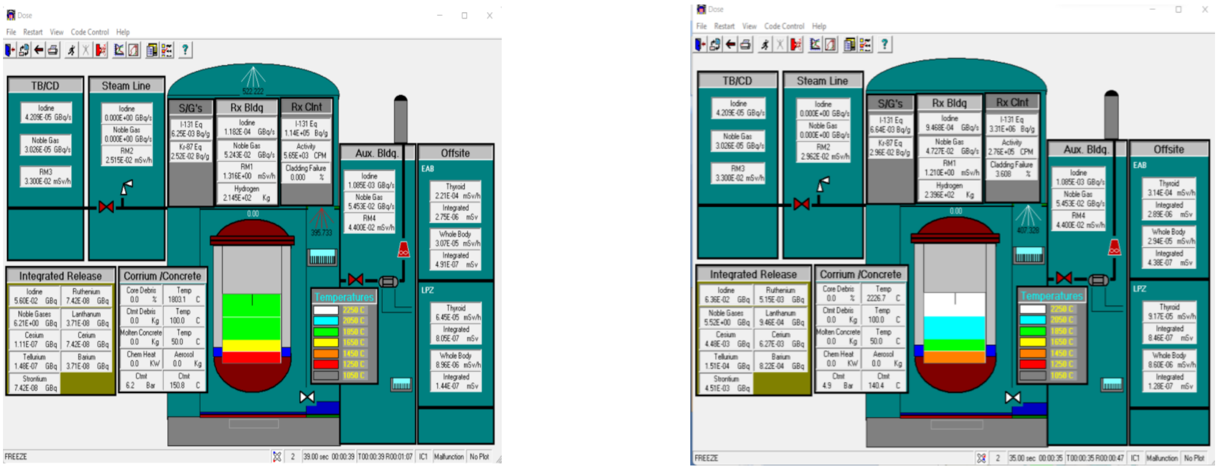
Break size (cm^2)	Start of core uncovering time(s)	Time required for the water level in the core dropping to zero (s)	Average fuel temperature after 300 s (°C)	Maximum fuel temperature after 300 s (°C)	Maximum cladding temperature after 300 s (°C)	Pressure in the reactor after 300s (bar)
200	277.0	-	285.8	288.4	267.5	48.4
500	106.5	-	400	562.4	259.8	11
2800	26.0	60	1077.99	1250.2	371.5	5.1
5675	20.5	45	1574.4	1918.2	505.7	5.2
11350	17.0	30	2225.8	2226.7	576.7	5.4

From Table 3 and 4, it can be observed that with the increment of break size, start of the core uncovering time as well as the time required for the water level dropping down to zero kept on decreasing, which is expected. But it should be mentioned that for break sizes smaller than or equal to 500 cm^2, full core uncovering does not occur within the 300-second simulation time. The pressure of the primary circuit decreased too, which was somewhat expected, although at 11350 cm^2 break size, pressure increased a bit. This happened probably due to buildup of steam inside the reactor. On the other hand, average and maximum fuel temperature kept on increasing. Maximum fuel cladding temperature followed similar trend with an exception at the break size of 500 cm^2 where the maximum cladding temperature was lower than that for 200 cm^2 break size. This may be due to the more rapid pressure drop in the coolant that facilitated localized boiling of the coolant and subsequent increase in the heat transfer coefficient for a short period of time.

The alarming observation from Table 3 and 4 is that if the break size is 2800 cm^2, core uncovering time is only 27.5 seconds and 26 seconds respectively for hot leg and cold leg. This reduces to 15.5 s and 17s in case of 11350 cm^2 break size which is equivalent to 200% of DEGB (Double Ended Guillotine Break) in the pipeline, and 5675 cm^2 can be considered a DEGB in a single pipeline of 850mm in diameter. In VVER 1200, where the passive ECCS safety system can cool the reactor for at least 72 hours, it is alarming to see that reactor core is lead to a melting condition only within a few seconds in absence of ECCS (Asmolov et al., 2017). Table 5 presents the melting behavior of the core under the simulated accidents. It may be observed that no meltdown was initiated up to the break size 2800 cm^2 within the simulation period. Melting of the core was only observed in case the 5675cm^2 and 11350 cm^2 breaks in both legs. Also, melting began a little earlier in the event of a cold leg break than in the case of a hot leg break.

Table 5: Melting behavior of the core

Name of the Leg	Break Size (cm^2)	Starting of Melting(s)
Hot	200	No meltdown
	500	No meltdown
	2800	No meltdown
	5675	55.5
	11350	39
Cold	200	No meltdown
	500	No meltdown
	2800	No meltdown
	5675	54
	11350	35



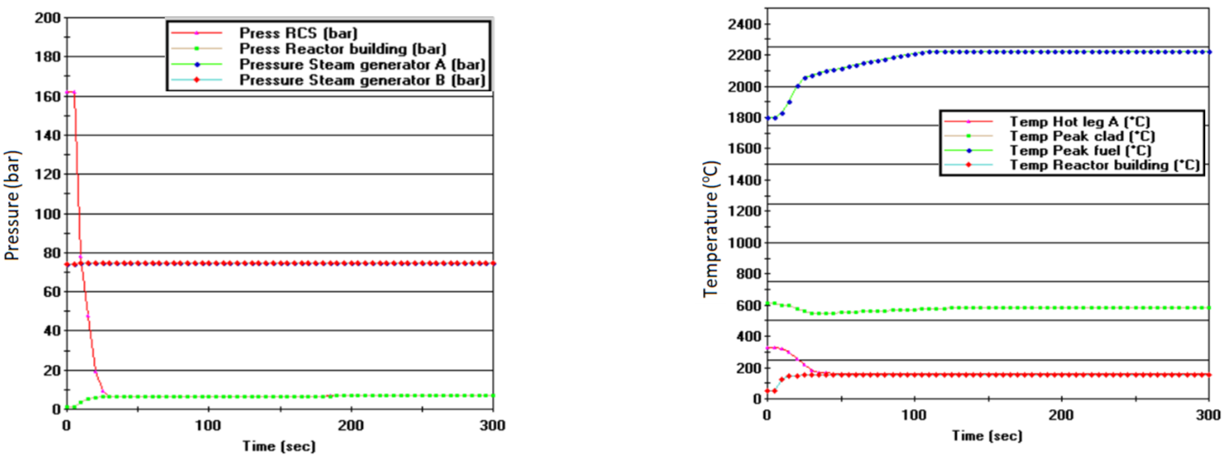
(a) (b)

Figure 2: Starting of core melting in case 11350 cm2 break in (a)hot leg (b)cold leg

Figure 2 presents the dose mimic view of the nuclear facility. This view can be used to determine when core melting began. Amount of radioactive release in the nuclear facility can also be known from this view.

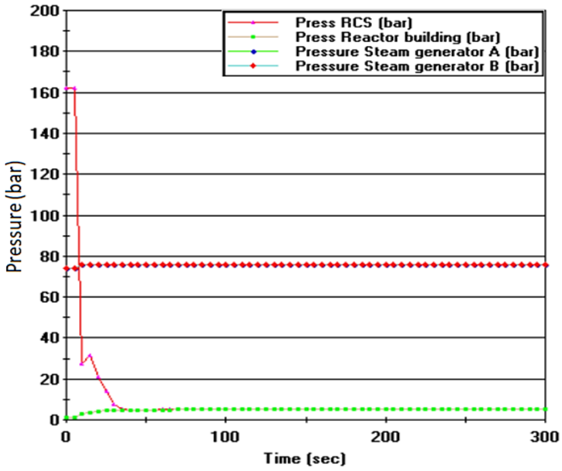
Curves in Figure 3 and 4 shows variation of some thermal hydraulic parameters in case of large LOCA with a break of 11350 cm^2. From figure 3(a) and 4(a), the pressure in the primary circuit drops down to the reactor building pressure at around 35 seconds as there is a break in the primary pipeline. But the steam generator pressure is constant throughout at around 74 bars reasonably having no breaks in the secondary pipelines.

Figure 3 (b) and 4 (b) shows rise of peak fuel temperature reaching around 2200^{o}C. In case of hot leg, the rise is a bit more gradual and requires over 100 seconds to reach whereas in case of cold leg, the rise is rather sharp requiring around 15 seconds. The peak cladding temperature reaches around 600^{o}C in both the legs. A sudden peak of around 800^{o}C was observed in case of the cold leg. The reactor building temperature rises to around 150^{o} C and merges with the hot leg temperature at around 25 seconds and cold leg temperature at around 50 seconds. The variations observed between hot and cold legs are mostly due to the pressure difference between the two.

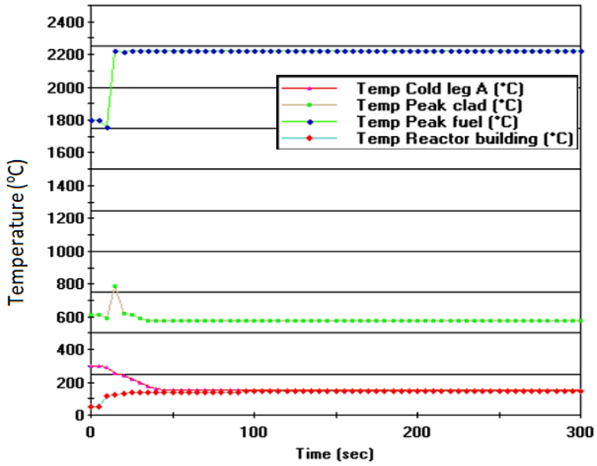


(a) (b)

Figure 3: Variation of parameters in case of 11350 cm2 break in hot leg



(a)



(b)

Figure 4: Variation of parameters in case of 11350 cm2 break in cold leg

4. Conclusion

'Loss of coolant accidents' in the hot and cold legs of the primary circuit were simulated in this study, along with station blackout, no ECCS, and various break sizes.This beyond design basis scenario demonstrated that in the case of significant fractures, such as a 5675 cm^2 crack, the core is uncovered extremely quickly (in 20 seconds) and partial core melting begins within a minute (under 40 seconds).Although the chances of such an accident are remote, the consequences are devastating, and we should always be prepared for them.There were no significant variations between hot leg LOCA and cold leg LOCA till the break size of 500 cm^2. However, there were slight variations when the break size was higher. This might be due to pressure differences between the hot and cold legs. This study was carried out for a VVER-1200 based NPP. The response of other Gen III+ NPPs to similar accident conditions may also be investigated. Also, this study considered only a few discrete break sizes. Other break sizes may be considered in further studies to derive correlation between break size and core meltdown probability. Finally, the use of machine learning and fuzzy logic may be considered.

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