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Analysis of Steam Line Break Accident Using PCTRAN Model of VVER-1200 NPP

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Abstract: The investigation of thermal-hydraulic parameters during steam-line break (SLB) accidents is performed by applying the personal computer transient analyzer (PCTRAN) simulator model of the VVER-1200 nuclear power plant (NPP). Five cases, namely, 0.005 m^2 (Case-1), 0.01 m^2 break (Case-2), 0.02 m^2 break (Case-3), 0.04 m^2 (Case-4), and 0.08 m^2 (case-5) of SLB accident inside containment with the concurrent loss of AC power have been simulated. There was no variation in the timing of the trip of the reactor coolant pumps, the main feedwater pumps, or the turbine in any of the five SLB accidents. However, the reactor scram's onset time varies slightly between the five scenarios. Pressure and temperature in the reactor coolant system (RCS) quickly reached a peak following the start of the SLB accident, fell shortly after the reactor scram, and eventually stabilized in all cases. In comparison to the larger breaks in the SLB accident, the smaller breaks result in a higher RCS temperature and pressure. After the SLB accident, the pressurizer's liquid level rises and then quickly drops in all cases. The break mass flow rate from the steam line rapidly increases until the occurrence of the reactor scram and then decreases to a stabilized value. Steam generator A has a faster rate of heat removal rate than steam generator B, and its pressure and liquid level decrease more quickly than those of steam generator B. The thermal power of the reactor, peak cladding temperature, and fuel temperature showed a rapid drop after the initiation of the SLB accident. There was no increase in these parameters from the initial state of the simulation. The radiation in the air of the reactor building and steam line was very low during the simulation period. Therefore, there was no violation of the safety aspects of the SLB accident of the PCTRAN simulation of the VVER-1200 NPP model.

Keywords: PCTRAN; PWR; Steam line break accident; Steam generator, Thermal-hydraulics; VVER-1200.

1. INTRODUCTION

The steam generator plays a very important role in ensuring safety during the operational and accidental situations of a pressurized water reactor (PWR). The analysis of steam line break (SLB) accidents is requisite to be present in the final safety analysis report of a nuclear power plant (NPP) [1]. An SLB accident can occur inside or outside the containment of a PWR [2]. The SLB accident can occur due to thermal stress or cracking in a steam line pipe and this accident is categorized as asymmetric cooling behavior due to one of the steam lines breaking while others are intact [3]. The thermal-hydraulic effect of the excess heat removal in SLB accidents can cause the primary coolant inventory to decrease and the reactor coolant systems (RCS) to become depressurized [1]. This accident leads to overcooling of the RCS by introducing positive reactivity into the core [3]. Therefore, it is important to investigate how thermal-hydraulic parameters behaved during the SLB accident.

The VVER-1200 PWR has a thermal capacity of 3212 MWth, consists of a reactor pressure vessel, a pressurizer and four circulation loops consisting of a steam generator in each loop, a reactor coolant pump, a hot leg and a cold leg in each loop [4]. The steam generator is a horizontal single-vessel heat exchanger with an immersed heat transfer surface that transfers heat from the primary side to the secondary side of the VVER-1200 PWR core [5]. The heat exchanger tubes consist of a downwardly sloping U-shaped bundle, and the tubes are assembled by welding the ends to the inner surfaces of the main coolant inlet and outlet. Saturated steam generated by the steam generator passes through holes in the perforated plate immersed in the evaporating surface and the steam is dried, directed to the perforated diffuser plate in the upper part of the steam generator, and directed into the steam head, from where it enters the steam ducts [6].

The Personal Computer Transient Analyzer (PCTRAN), a product of Micro-Simulation Technology Inc., can be used to simulate nuclear power plant (NPP) accidents and transients [7]. Using a PCTRAN simulator, a number of hypothetical accidents of the VVER-1200 NPP model were studied, including the loss of coolant accident [8-12], the unintended withdrawal of the control rod [13], the anticipated transient without scram event [14], the steam generator tube rupture event [11,15-16], and the SLB accident [17].

The PCTRAN simulation of the VVER-1200 NPP model was performed for SLB accidents with the break size of 1000 cm^2 and the probability of contamination of the environment by radioactive material was examined [17]. However, thermal-hydraulics analysis of the different break size spectra of SLB accidents was not performed with the PCTRAN simulator. Thus, the primary aim of this study is to analyze the trend of thermal-hydraulic parameters during the SLB accident of VVER-1200 NPP by applying the PCTRAN simulator.

2. STEAM LINE BREAK ACCIDENT SIMULATION USING PCTRAN MODEL OF VVER-1200 NP

2.1 Initial Conditions

In this study, a demo version 1.2.0 of the PCTRAN model of the VVER-1200 NPP is used, which consists of two loops of RCS, with a reactor pressure vessel, a pressurizer, two steam generators, two hot legs, two cold legs, two reactor coolant pumps, emergency core coolant system, the low-pressure injection system, feed and bleed water system, accumulator, various safety valves and components of the NPP [18]. In regarding the thermal-hydraulic model in PCTRAN model, the lumped-loop approach with two-phase critical flow discharge, and non-equilibrium pressurizer for PWR are used [7]. The main advantages of the PCTRAN simulator are the Windows-based graphical interactive user interfaces and the direct manipulation of the graphical elements [19].

From the preloaded initial conditions of the PCTRAN simulator, 100% full power at the end of the cycle was selected as the initial condition of this study. Prior to the initiation of the SLB accident, the PCTRAN simulator was run for 10 seconds to reach steady state. At 10 seconds of the simulation of the PCTRAN model of VVER-1200 NPP, the value of the main thermal-hydraulic parameters namely, reactor thermal power, RCS pressure, steam generator pressure, cold leg temperature, hot leg temperature, cladding temperature, and fuel temperature was found as 3205.04 MWth, 161.98 bar, 73.99 bar, 298.02 °C, 329.03 °C, 611.27 °C, and 1802.30 °C respectively.

2.2 Transient Simulation

The PCTRAN model of the VVER-1200 NPP was used to simulate five hypothetical cases as the SLB accident inside the containment in steam generator A with the simultaneous occurrence of loss of AC power after the steady-state operation of 10 seconds. The range of break sizes, from small to large, is selected as the break spectrum. Five cases, namely, 0.005 m^2 (Case-1), 0.01 m^2 (Case-2), 0.02 m^2 (Case-3), 0.04 m^2 (Case-4), and 0.08 m^2 (Case-5) of SLB accident with the concurrent loss of AC power have been performed in this study. The SLB break size in Case-5 is 16 times the break size in Case-1. For each scenario, the transient calculation was run for 300 seconds. Figure 1 shows a snapshot of the SLB accident simulation for Case-3 using the PCTRAN model of the VVER-1200 NPP.

The main events of the SLB accident are shown in Table 1, where the unit of the occurring event time is seconds. Except for the occurrence of reactor scram and the beginning of high-pressure safety injection and containment spray, there is no variance in the timing of different events among the five cases. As shown in Table 1, the occurring time of the reactor scram does not show a linear trend according to the size of SLB. The high-pressure safety injection and the containment spray start earlier for larger break cases.

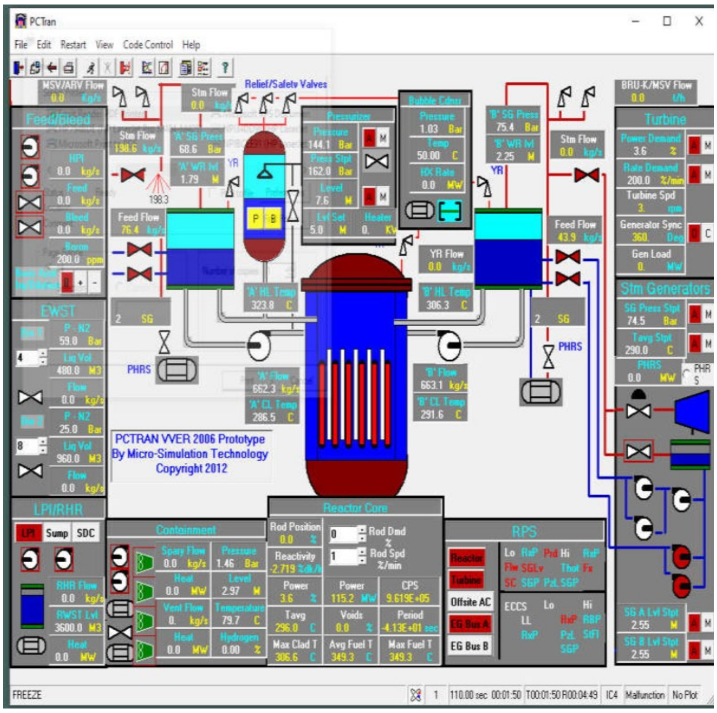

Figure 1. The snapshot of PCTRAN Simulation of the VVER-1200 SLB accident

Table 1. Sequence of events

Events	0.005 m^2 break (Case-1)	0.01 m^2 break (Case-2)	0.02 m^2 break (Case-3)	0.04 m^2 break (Case-4)	0.08 m^2 break (Case-5)
Initiation of SLBA with the concurrent loss of AC power	10.0	10.0	10.0	10.0	10.0
Trip of reactor coolant pumps	12.0	12.0	12.0	12.0	12.0
Trip of main feedwater pumps	12.0	12.0	12.0	12.0	12.0
Trip of turbine	12.5	12.5	12.5	12.5	12.5
Reactor scram	28.5	29.5	28.5	30.0	27.5
Starting of diesel generator A	71.0	71.0	71.0	71.0	71.0
Starting of turbine driven auxiliary feedwater pumps	71.5	71.5	71.5	71.5	71.5
Starting of high pressure safety injection and containment spray	270.0	140	74.5	43.5	27.0
End of Simulation	300	300	300	300	300

3. RESULTS AND ANALYSES

The pressure and temperature of the RCS are shown in Figures 2 and 3, respectively. According to Figure 2, the primary pressure increased rapidly after the start of the SLB accident, reaching a peak value of 169.60 bar, 168.26 bar, 167.54 bar, and 167.08 bar for Case-1, 2, 3, and 4 respectively, at 20 seconds, and it reaches to 166.97 bar at 30 seconds for Case-5. According to Figure 3, the average temperature of the RCS also peaks at 323.38 °C, 321.79 °C, 321.05 °C, and 320.61 °C for Case-1, 2, 3, and 4, respectively, at 20 seconds, and it reaches to 320.49 °C at 30 seconds for Case-5, which agrees well with the behavior of the RCS pressure. The pressure and temperature in the reactor peaked in all cases due to the simultaneous loss of AC power with the SLB accident and due to the occurrence of the trip of reactor coolant pumps, main feed-water pumps, and turbine before the reactor scram occurred. Following the reactor scram, these values decrease gradually until the end of the simulation. Compared to the other cases, the slope of the decreasing reactor pressure and temperature is higher in Case-4 and Case-5.

The pressure on the secondary side of steam generator A and steam generator B is shown in Figures 4 and 5, respectively. Immediately after triggering the SLB, the fluctuation of the pressure is observed in steam generators due to the repeated opening and closing of safety relief valves of the steam generators. The pressure of steam generator A is reached at 70.90 bar, 59.98 bar, 47.11 bar, 29.97 bar, and 16.80 bar for Case-1, 2, 3, 4, and 5 respectively, at 300 seconds while, the pressure of steam generator B is reached at 72.93 bar, 64.20 bar, 55.65 bar, 60.18 bar, and 64.96 bar for Case-1, 2, 3, 4, and 5 respectively, at 300 seconds. The pressure drop in steam generators is lower in smaller break cases due to the smaller inventory loss in comparison to the larger break cases due to larger inventory loss. The pressure drop rate of steam generator A is higher than the pressure drop rate of steam generator B. The reason for this asymmetric pressure behavior is the occurrence of a SLB accident in steam generator A while steam generator B is intact.

The collapsed liquid level of the pressurizer is shown in Figure 6. The liquid level of the pressurizer rises from its initial level of 11.20 m due to the liquid surge in the direction of the pressurizer after the initiation of the SLB accident. The pressurizer liquid level rises to a peak of 13.43 m at 20 seconds, 13.02 m at 25 seconds, 12.87 m at 25 seconds, 12.65 m at 20 seconds, and 12.67 m at 30 seconds for Case-1, 2, 3, 4, and 5 respectively due to the liquid surge towards the pressurizer after the initiation of the SLB accident. The pressurizer level is reached at the peak value after the reactor scram in Case-5, while in all other cases it occurs before the reactor scram. After reaching the peak value, the pressurizer liquid level begins to fall rapidly, reaching a value of 7.38 m, 6.64 m, 4.63 m, 2.58 m, and 0.45 m for Case-1, 2, 3, 4, and 5 respectively, at 300 seconds. At the end of the simulation, the collapsed liquid level of the pressurizer in Case-1 is 16.4 times that in Case-5, which is attributed to the SLB size in Case-5 being 16 times that in Case-5.

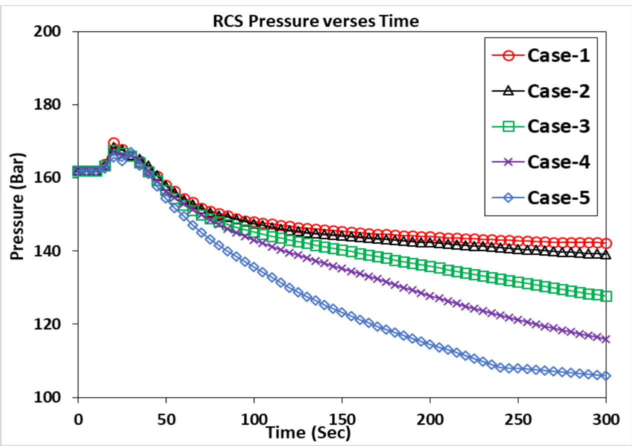

Figure 2. Pressure of RCS

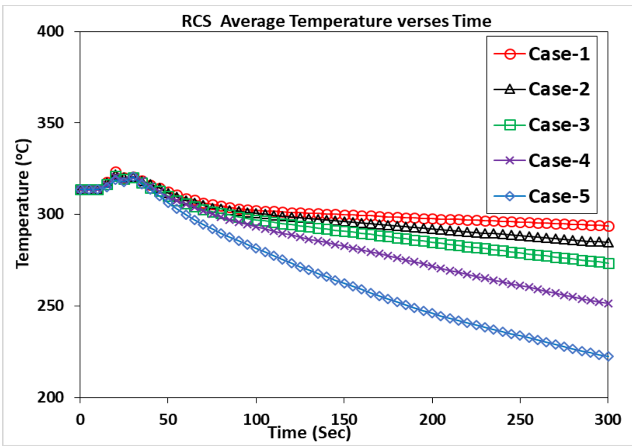

Figure 3. Average temperature of RCS

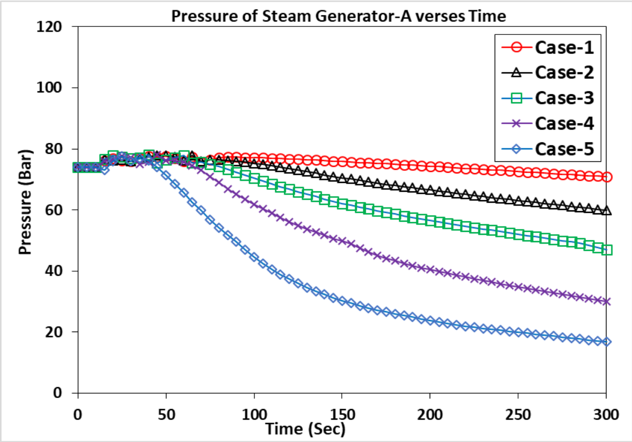

Figure 4. Secondary side pressure of steam generator A

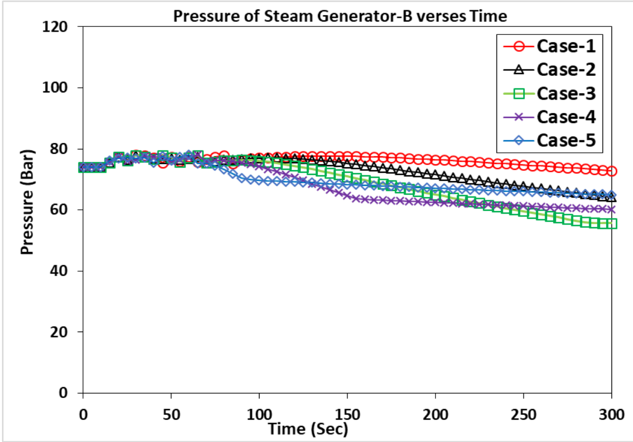

Figure 5. Secondary side pressure of steam generator B

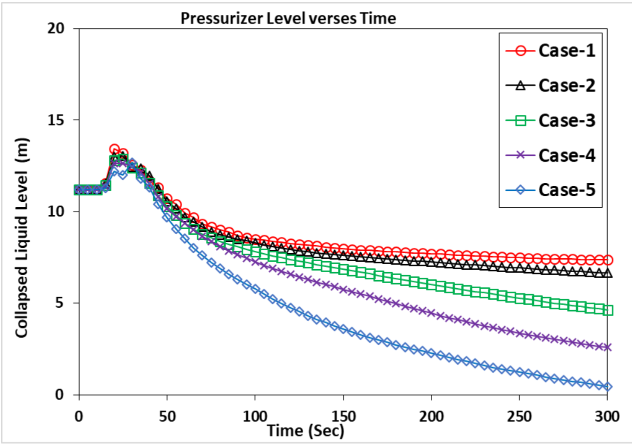

Figure 6. Collapsed liquid level of the pressurizer

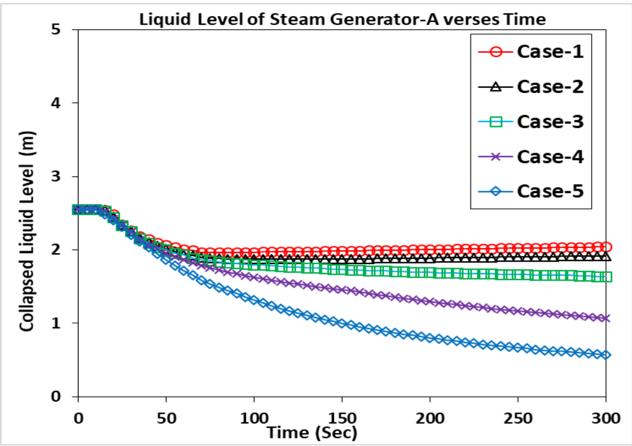

Figure 7. Collapsed liquid level of the steam generator A

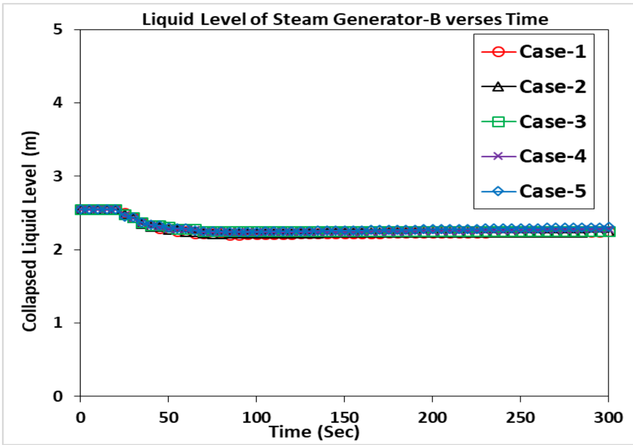

Figure 8. Collapsed liquid level of the steam generator B

The collapsed liquid level of steam generator A and steam generator B is shown in Figures 7 and 8, respectively. The liquid level of the steam generator A decreases to a value of 2.04 m, 1.92 m, 1.64 m, 1.07 m, and 0.57 m for Case-1, 2, 3, 4, and 5 respectively, at 300 seconds from the initial liquid level of 2.5 m. The liquid level of the steam generator B decreases to a value of 2.25 m, 2.23 m, 2.25 m, 2.28 m, and 2.31 m for Case-1, 2, 3, 4, and 5 respectively, at 300 seconds. The falling liquid level of steam generator A is higher than that of steam generator B. The reason for this deviation is the loss of the liquid inventory from steam generator A while steam generator B is intact.

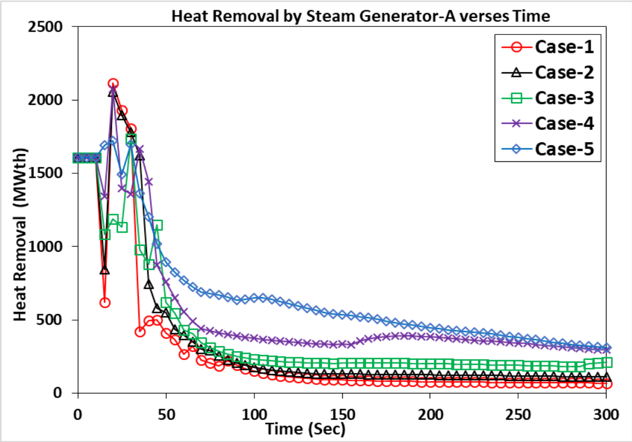

Figure 9. Heat removal by the steam generator A

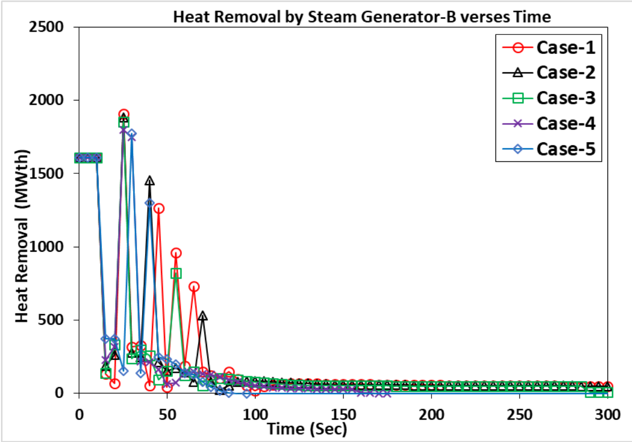

Figure 10. Heat removal by the steam generator B

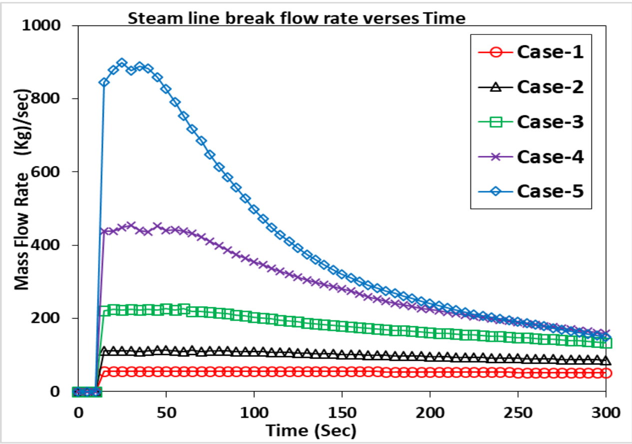

Figure 11. Break mass flow rate

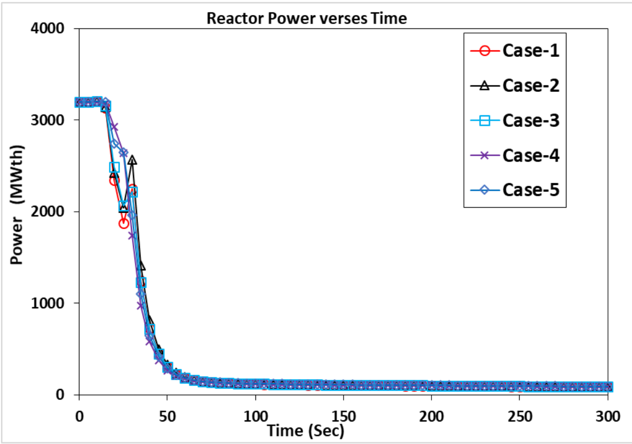

Figure 12. Reactor thermal power

The heat removal by steam generator A and steam generator B is shown in Figures 9 and 10, respectively. The generated heat is removed from each of the steam generators at 1608.83 MWth at the beginning of the simulation. In the case of heat removal by the steam generator A, there was a peak value of 2114.92 MWth at 20 seconds, 2057.52 MWth at 20 seconds, 1737.78 MWth at 30 seconds, 2078.47 MWth at 20 seconds, and 1721.95 MWth at 20 seconds for Case-1, 2, 3, 4, and 5 respectively. In the case of heat removal by the steam generator B, at 25 seconds, there was a peak value of 1906.41 MWth, 1883.41 MWth, 1850.37 MWth, and 1798.10 MWth, for Case-1, 2, 3, and 4 respectively at 25 seconds and this value is 1775.31 MWth for Case-5 at 30 seconds. During the opening and closing of the safety valves of the steam generators, there was a fluctuation in the rate of heat removal. The heat removal rate is higher for the larger SLB in steam generator A from around 50 seconds to 300 seconds. The heat removal by the steam generator A in Case-5 is almost 4.5 times that of Case-1 at the end of the simulation. It implies that the heat removal by steam generator does not show a linear trend according to the size of SLB. The heat removal rate is extremely low for steam generator B, about 100 seconds to 300 seconds in comparison to steam generator A.

The break mass flow rate of the SLB accident is shown in Figure 11. After the initiation of SLB, the break flow rate rapidly increased to a peak value of 56.31 kg/sec, 112.97 kg/sec, 226.80 Kg/sec, 454.82 kg/sec, and 899.10 kg/sec for Case-1, 2, 3, 4, and 5 respectively. The calculated peak break mass flow rate in Case-5 is almost 16 times that of Case-1, which agrees well with the SLB size in Case-5 is 16 times that of Case-1. The break mass flow rate decreases to a plateau value for all cases after 150 seconds. The break flow rate is reached a value of 51.29 kg/sec, 86.08 kg/sec, 132.57 kg/sec, 158.74 kg/sec, and 149.85 kg/sec, for Case-1, 2, 3, 4, and 5 respectively, at the end of the simulation.

The thermal power of the reactor is shown in Figure 12. After the SLB is initiated, the thermal power starts to decrease from its operating power of 3200 MWth as shown in Figure 12. The thermal power shows a rapid drop following the scram for all the cases as a result of the rapid decrease in reactivity in core by control rods insertion. It is mentioned that there is a fluctuation of thermal power during the occurrence of reactor scram for all the cases, which could be attributed to the frequent opening and closing of the safety relief valves of steam generators. After 50 sec, the thermal power is stabilized and it reaches a value of 86.56 MWth, 87.21 MWth, 87.61 MWth, 88.58 MWth, and 89.22 MWth for Case-1, 2, 3, 4, and 5 respectively, at 300 sec.

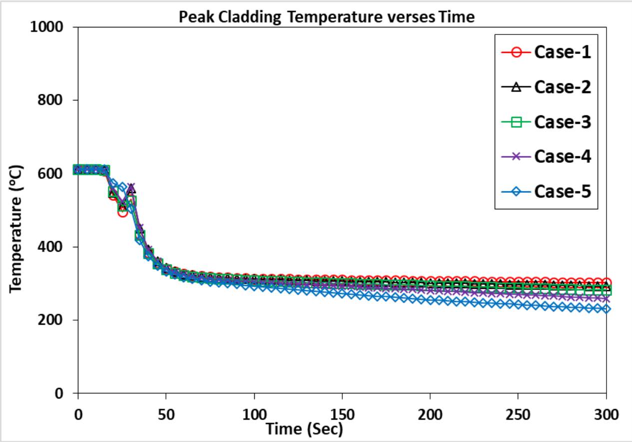

Figure 13. PCT Temperature

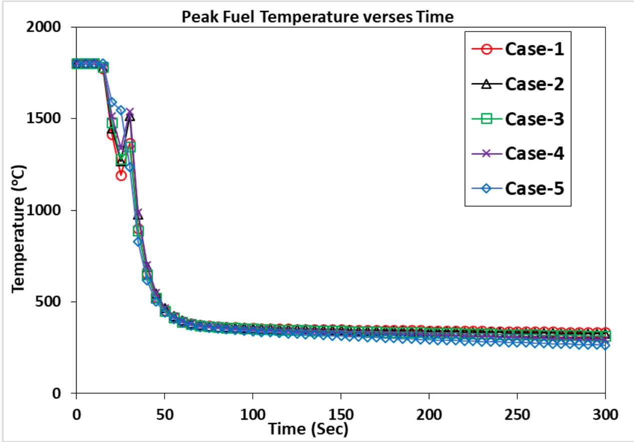

Figure 14. Peak fuel temperature

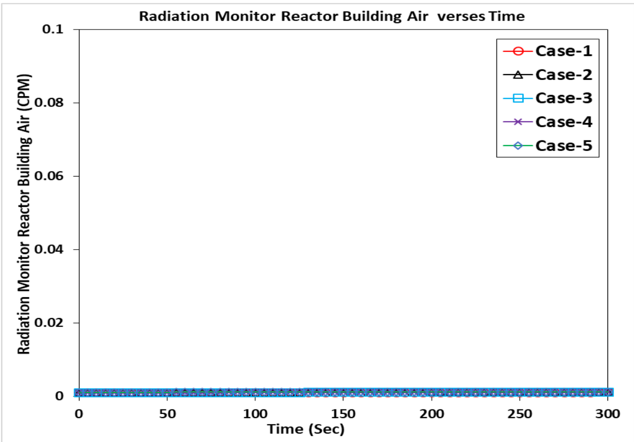

Figure 15. Reading in radiation monitor in the reactor building

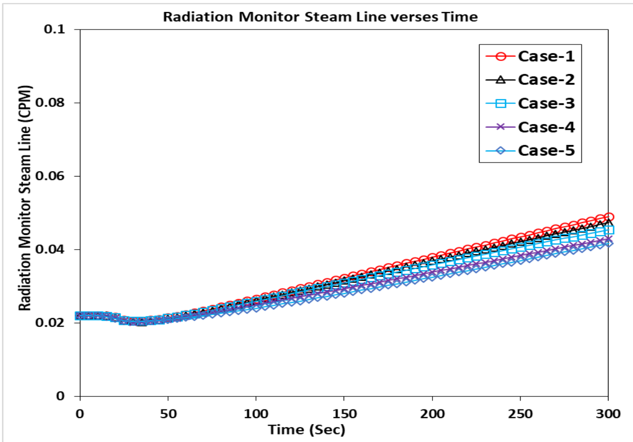

Figure 16. Reading in radiation monitor in the steam line

The peak cladding temperature (PCT) and the peak fuel temperature are shown in Figures 13 and 14 respectively. The peak cladding temperature (PCT) and fuel temperature begin to drop just after the initiation of the SLB from their initial values of 610.84 °C and 1800.00 °C, respectively due to the depressurization of the RCS. The PCT and peak fuel temperature fall rapidly to less than a value of 400 °C and 500 °C, respectively, just after the occurrence of reactor scram. The temperature dropped rapidly as a result of the rapid decrease of reactor power generation, which is ultimately occurred due to the rapid decrease in reactivity in core by control rods insertion. A fluctuation of the PCT and the peak fuel temperature occurred during the occurrence of reactor scram for all the cases, which could be attributed to the fluctuation of thermal power. After 50 sec, the values of PCT and peak fuel temperature are stabilized. The value of PCT reaches 301.87 °C, 292.75 °C, 281.74 °C, 259.63 °C, and 230.81 °C for Case-1, 2, 3, 4, and 5 respectively, at the end of the simulation. On the other hand, the value of peak fuel temperature reaches 334.04 °C, 325.15 °C, 314.29 °C, 292.75 °C, and 263.96 °C for Case-1, 2, 3, 4, and 5 respectively, at the end of the simulation. During the accident period, neither the PCT nor the peak fuel temperature shows any increase from the reactor's normal operating temperature.

The reading of radiation monitor in the reactor building and steam line is shown in Figures 15 and 16 respectively. As shown in Figures 15, there is no change of the radiation in the air of the reactor building after the initiation of the SLB from the initial value of 0.0011 counts per minute (CPM). However, the value of radiation in steam line is slightly increased from the initial value of 0.022 CPM to 0.049 CPM, 0.047 CPM, 0.045 CPM, 0.043 CPM, and 0.042 CPM for Case-1, 2, 3, 4, and 5 respectively, at the end of the simulation. Thus, the value of radiation monitor is low enough during the period of simulation.

The comparison of the results of steam generator tube rupture (SGTR) event of VVER-1200 by PCTRAN simulator [16] with the results of SLB events in this study. It was found that there was no variation of time between the SGTR [16] and SLB events for the occurrence of the trip of the reactor coolant pump, the main feedwater pump, and the turbine. However, the reactor scram occurred in a different time sequence between the SGTR [16] and SLB events. The primary pressure and temperature of the reactor rapidly rose to a peak value and then fell after the reactor scram in both cases of the SGTR [16] and SLB events. The secondary-side pressure of steam generator showed a rapid fluctuation during the transient calculation period in the SGTR event due to the frequent opening and closing of safety relief valves of the steam generators [16] while it does not show rapid fluctuation in case of SLB events. The trend of the liquid level of the pressurizer and steam generator showed a similar trend in both cases of the SGTR [16] and SLB events with dissimilar values. No visible peak in PCT and peak fuel temperature was observed during SGTR [16]. The PCT and the peak fuel temperature increased to a peak value during SLB events, but these values were lower than those of normal operating conditions. In general, there was a similarity in the behavior of various thermal-hydraulic parameters between the SGTR [16] and SLB events, as both events occurred due to coolant loss in the secondary sides of the steam generators.

The main steam line break accident was analyzed by using RELAP5-3D code for VVER-1000 PWR, when the accident initiated, the reactor power was 3120 MWth [20]. After the initiation of the accident, the reactor shut down rapidly, followed by the trip of the main circulating pump in the loop of affected steam generator. The power was not increased and there was no occurrence of the re-criticality during the simulation [20]. The steam line break accident was analyzed by using RELAP5/MOD 3.2 code at full power reactor for VVER-1000/V320 NPP [21]. The primary side experienced a decreasing pressure in coincidence with decreasing steam line pressure. The reactor scram occurred earlier than the reactor trip [21] while in this study the reactor scram occurred earlier than the reactor tip due to the simultaneous loss of AC power with the SLB accident.

The low likelihood of a boiling crisis in the core, the RCS pressure remaining below 110% of the design pressure, and the absence of fuel melting in the core are the acceptance criteria for a PWR accident [22]. These acceptance criteria were met in SLB accidents during the simulation period of 300 seconds. Thus, in this study, there was no violation of safety aspects during the PCTRAN simulation of the VVER-1200 NPP model for 300 seconds.

4. CONCLUSION

The thermal-hydraulic parameters of the SLB accident of the VVER-1200 NPP model are analyzed using the PCTRAN simulator. Five SLB incidents have been simulated for this study, including breaks of 0.005 m^2 (Case-1), a break of 0.01 m^2 (Case-2), a break of 0.02 m^2 (Case- 3), a break of 0.04 m^2 (Case-4), and a break of 0.08 m^2 (Case-5). Each of these SLB accidents occurred inside containment with a simultaneous loss of AC power. Among five cases of SLB accidents, there was no variation in time for the occurrence of the trip of the reactor coolant pumps, the main feedwater pumps, and the turbine and for the starting of the diesel generator and turbine-driven auxiliary feedwater pump. However, the occurring time of the reactor scram varies slightly among the five scenarios and it does not show a linear trend according to the size of SLB. The high-pressure safety injection and containment spray start earlier for larger break cases.

The RCS pressure and temperature peaked rapidly after the start of the SLB accident, dropped swiftly after reactor shutdown, and then gradually stabilized in all cases. The smaller break size results in higher RCS pressure and temperature for the period of the SLB accident due to the smaller coolant inventory loss. After the initiation of SLB accident, the pressurizer liquid level rises due to the liquid surge towards the pressurizer, and it drops quickly in all cases. The collapsed liquid level of the pressurizer was found to be 16.4 times higher in Case-1 than it was in Case-5 at the end of the simulation because the SLB size in Case-5 was 16 times that in Case-1.

The asymmetric behavior between the two steam generators was observed due to the initiation of an SLB accident in steam generator A while steam generator B was intact. Owing to asymmetrical behavior, the liquid level of steam generator A dropped quicker than that of steam generator B, and the pressure in steam generator A decreased at a faster rate than in steam generator B. Moreover, the heat removal rate is extremely low for steam generator B, in comparison to steam generator A at the end of the simulation. Neither the PCT nor the peak fuel temperature rose during the simulation period.

The break mass flow rate from the steam generator shows a rapid increase before the reactor scram, and it thereafter stabilized in all cases. The calculated peak break mass flow rate in Case-5 is almost 16 times that of Case-1, which agrees well with the SLB size in Case-5 is 16 times that of Case-1. Reactor thermal power, peak cladding temperature, and peak fuel temperature decreased rapidly after the beginning of the SLB accident. Neither the PCT nor the peak fuel temperature rose during the simulation period. There was no change of the radiation in the air of the reactor building during the simulation period. The value of radiation in steam line slightly increased to a value less than 0.05 CPM from the initial value 0.022 CPM for all the cases. The safety prospects of the SLB accident of the PCTRAN simulation of the NPP model VVER-1200 revealed no risk during the simulation period. The SLB accident scenario was analyzed for 300 seconds only, which is a limitation in this study.

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