Simulation of Steam Generator Tube Rupture Accident in a Pressurized Water Reactor

D. Shome[1, a)] M. A. R. Sarkar[2, b)]

[1]Bangladesh University of Engineering and Technology, Dhaka-1000,Bangladesh
[2]Professor, Department of Mechanical Engineering, Bangladesh University of Engineering and Technology, Dhaka-1000, Bangladesh

[a)]Corresponding author: dipen.me.buet@gmail.com
[b)]rashid@me.buet.ac.bd

Abstract:

The objective of this paper is to present and analyze the results of simulated tube rupture accident in VVER-1000 Nuclear Reactor in PCTRAN. In simulating the accident, 100% of one full tube rupture has been considered. The simulation result shows that the core pressure experience a rapid decrease from initial value of 155 bar (15.5 MPa) and stabilize around 80 bar (8 MPa) after the accident. This leads to stopping coolant leakage from primary circuit to secondary circuit due to absence of pressure differential between primary and secondary loop. After the initiation of tube rupture, the leak from affected Steam Generator 'A' is about 3000 t/h (833.33 kg/s) which is reduced to approximately 500 t/h(138.89kg/s) within 200s of the accident. The result also shows that the reactor power (both 'Thermal' and 'Nuclear Flux') collapses drastically following reactor trip. Both High Pressure Injection (HPI) pump is activated following "Reactor Scram" to prevent core damage. The average temperature of coolant at the reactor inlet decreases from 580K to 560K to facilitate cooling down of the primary coolant. The data obtained from the simulation are satisfactorily consistent with PSAR (Preliminary Safety Assessment Report) data regarding SGTR accident. These findings are expected to provide useful information in understanding and evaluating plants capability to mitigate the consequence of SGTR accident.

Keywords: *VVER-1000, Steam Generator Tube Rupture, PCTRAN, Nuclear Accident*

I. INTRODUCTION

Steam Generator is an integral part of Nuclear Steam Supply System (NSSS) performing an important role of producing steam by transferring heat from primary coolant system to secondary coolant system. In a typical PWR Steam Generator, the high pressure primary coolant exchanges heat with a low pressure secondary coolant and forms steam which drives the turbine. As a result, steam generator reliability and performance bear significant concerns in the safe operation of a PWR (Pressurized Water Reactor) power plant. During continuous and long term operation, these steam generators experience problems due to corrosion, flow induced vibration, stress, fatigue failure etc.[1] which degrades the structural integrity of U-tubes posing a great risk of radioactivity release into the atmosphere. To prevent the release of radionuclide, the steam generator tubing must be essentially free of cracks, perforations, and corrosion resistant. A number of Steam Generator Tube Rupture (SGTR) events have occurred in Pressurized Water Reactors (PWR), including those at Point Beach, Surry, Mihama, Oconee, Prairie Island and Gina [2]. SGTR causes a direct flow path between primary and secondary system and results in the release of radioactive materials into the environment. The study on tube rupture in Horizontal SG is however not as extensive in comparison to Vertical SG as most PWR plants are equipped with the latter one. In this paper, the simulation of tube rupture accident in VVER-1000 Horizontal SG is performed in PCTRAN [3], a PC based Nuclear Transient Accident Simulator developed by Micro-Simulation Technology and transient response of various plant parameter are presented.

II. VVER-1000 STEAM GENERATORS

The steam generators (Fig. 1) used in VVER-1000 NPP are horizontal shell-and-tube heat exchangers consisting of

- a Pressure Vessel (made of low-alloy pearlitic steel with stamped elliptical ends and stamped branch pipes and hatches welded together)
- a horizontal heat exchange tube bundle comprising of several thousand U-tube (tube arrangement: Triangular array)
- two vertical primary collectors ('Hot' and 'Cold' Leg)
- a feedwater piping system
- moisture separators
- Steam collector.

Primary coolant enters the steam generator through a vertical collector, travels through the horizontal U-shaped submerged stainless steel tubing, and exits through a second vertical collector. The tube ends penetrate the collector wall (which performs the same function as the tube sheet in a PWR steam generator) and are expanded using either a hydraulic or explosive expansion process and then welded at the collector inside wall surface. The collectors are made of low-alloy steel with higher tensile properties, lined with stainless steel. The vertical hot and cold primary coolant collectors penetrate the vessel near its mid-point. The feedwater is supplied to the top of the hot side of the tube bundle under a submerged perforated sheet. These tubes are designed to confine radioactivity to the primary coolant during normal operation. However, the primary pressure is higher than the secondary pressure, so rupture of the tube can result in radioactivity release to the environment through relief valves in the secondary system.

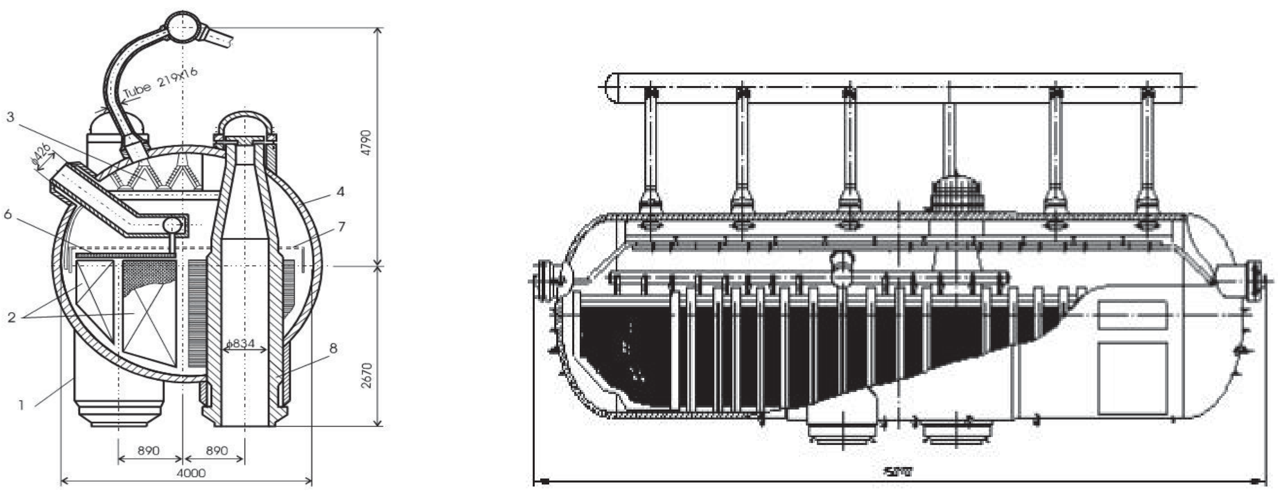

Fig. 1: PGV-1000MK Steam Generator [4]

1-Reactor coolant inlet header (Hot leg), 2-Tube bundles, 3-Moisture separator, 4-Steam generator shell, 5-Feed water inlet header, 6-Feed water distribution tubes, 7-Perforated plate, 8-Reactor coolant outlet header (Cold Leg)

III. SETTING UP INITIAL AND BOUNDARY CONDITIONS IN PCTRAN

Steady state condition of the plant before the accident are as follows:

- Reactor Power: 3000 MWth (100%)
- RC Pressure: 155 bar
- Core Average Temperature : 306.9 °C
- SG Pressure: 70 bar
- Time in Life: BOC (Beginning of Cycle)
- SG Water Level: 2.45 m
- Pressurizer Level: 6.96 m
- Max. Fuel Temperature: 788.9°C
- No interaction of operators during the accident.

For this analysis, 100% rupture of a single U-tube of steam generator has been taken into consideration. The duration between the initiation and complete rupture of the tube is 5 seconds. After setting up the initial and boundary conditions, they are used to provide a necessary conservatism of calculation results in terms of releases from the affected steam generator.

IV. ANALYSIS OF SGTR SIMULATION RESULT

The tube rupture is assumed to be initiated at 5 seconds in one of the two Steam Generators (in this case SG ‘A’) when the reactor is operated at full power(100%).

The simulation is performed for about 300 seconds (5 mins.) to study the various transient behavior of plant parameter during the accident.

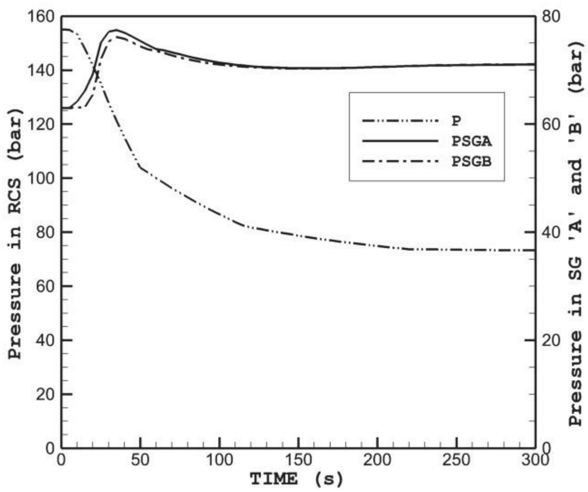

Fig. 2: Pressure in Reactor Coolant, Primary and Secondary System (bar)

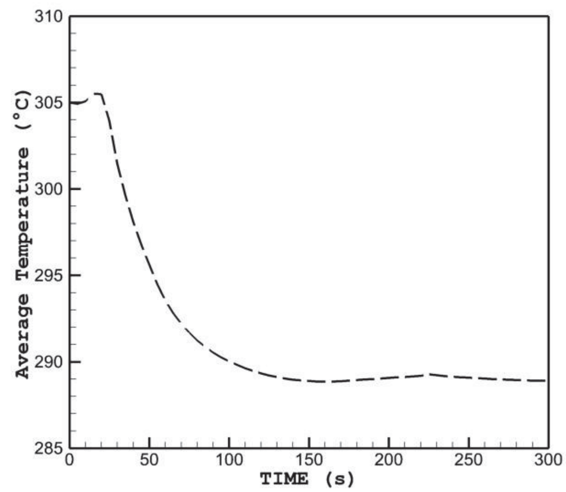

Fig. 3: Average Temperature (°C) in Reactor Core

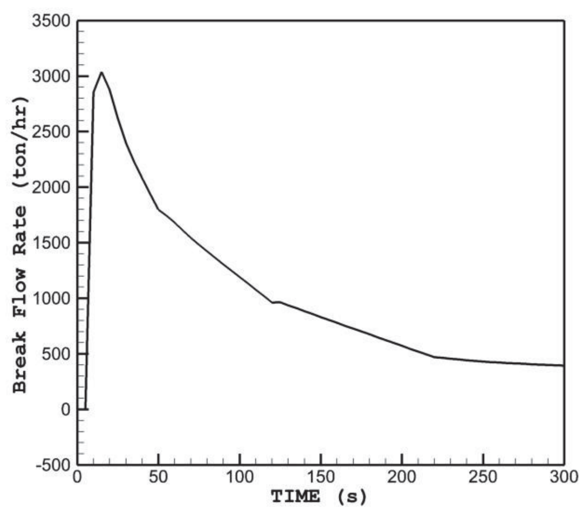

Fig. 4: Break Flow rate (t/h) in affected SG ‘A’

After the initiation of SGTR, the Primary Pressure drops instantaneously from 15.5 MPa to about 8 MPa due to opening and closing of Relief Valve. This rapid depressurization in the primary system was followed by a “Reactor Trip”.[5] The “Reactor Trip” tends to decrease “Average Temperature” (Fig. 3) of the reactor coolant while primary pressure drop is caused due to rupture of SG Tube. Fig. 2 shows the pressure behavior calculated during the transient. Pressurizer Heaters were turned on to increase the pressure on Primary side. [6] Pressurizer pressure and level (Fig. 5) continued to decrease. The pressure in both SG at first increased in the early phase of transient followed by closing of turbine throttle valve during "Turbine Trip" at 19.5 seconds. The pressure in SG 'A' was slightly higher than pressure in SG 'B' as shown in (Fig. 2). Since the "Primary Side" pressure was higher than "Secondary Side" pressure, there will be a considerable amount of coolant leakage as reactor coolant passes from "Hot Leg" to "Cold Leg" within the SG. Fig. 4 shows the break flow rate of SG 'A' after the accident. The primary coolant leakage peaks about 3000 t/h (833.33 kg/s) instantaneously following tube rupture at about 15 seconds before PORV (Pilot Operated Relief Valves) opened but after PORV is opened, leakage dropped to 500 t/h (138.89 kg/s) after 300 seconds and will be eventually dropped to zero due to equalized pressure between both system.

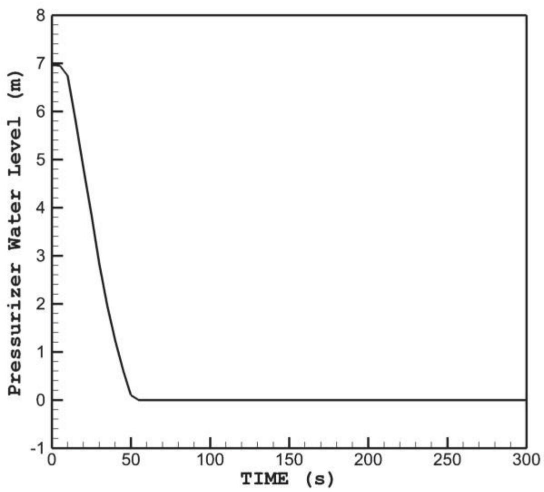

Fig. 5: Water Level in Pressurizer (m)

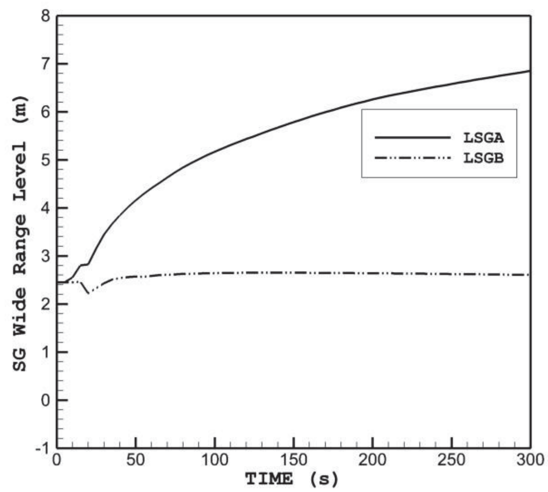

Fig. 6: Steam Generator Wide Range Level (m)

Fig. 5 & 6 shows, the water level in the pressurizer and both Steam Generator. As Fig. 5 shows, pressurizer water level drops rapidly from the initial value of 6.96 m and is emptied at about 50s following the transient. Both SG 'A' and 'B' mixture level drops instantaneously at the beginning of accident (Fig. 6). This is caused by collapsing of vapor bubble due to loss of heat source following reactor trip. SG 'A' level increased later due to rupture flow from the primary side (SG 'B' is isolated) and due to increase of secondary pressure, caused by turbine trip.

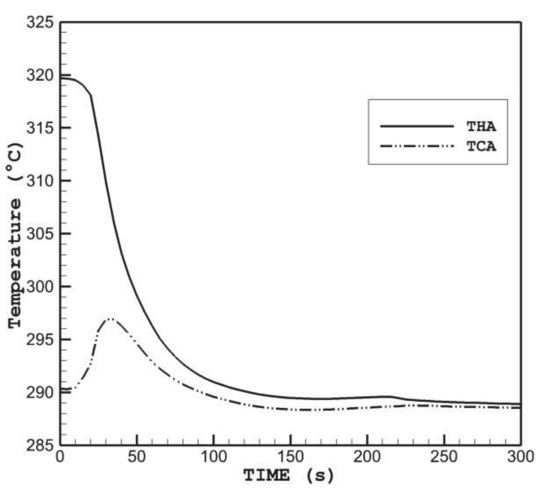

Fig. 7: Temperature in "Hot" and "Cold" Leg of SG 'A' (°C)

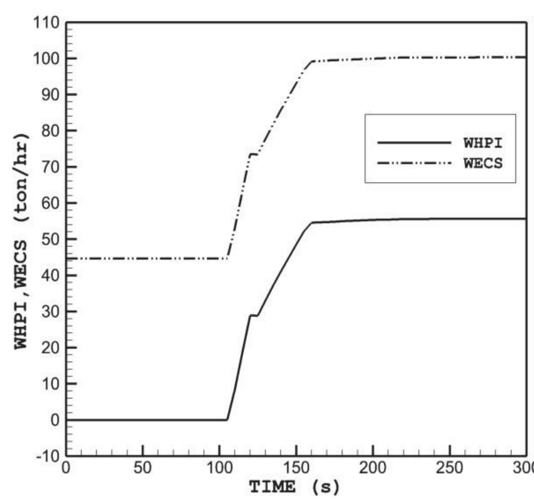

Fig. 8: HPI and ECCS Flow Rate (t/h)

The temperature in “Hot” and “Cold” leg of affected SG ‘A’ is shown in Fig. 7 which indicates the decrease of temperature at both leg although the temperature in ‘Cold’ increases at the initiation of the accident. ECCS, comprising of High Pressure Injection, Low Pressure Injection is provided after the pressure drops below a certain value. In this particular case, at about 100 seconds when the coolant pressure drops to approximately 80 bar (8 MPa) the ECCS is activated to prevent the core from being damaged due to excessive heat generation. This is due to the fact that following loss of coolant pressure, the primary coolant may get boiled at lower temperature and creates water vapor around the reactor core which has lower heat transfer capability in comparison to water.[7] As a result, the heat produced within the reactor core can’t be transferred to coolant resulting in the heating up the core.

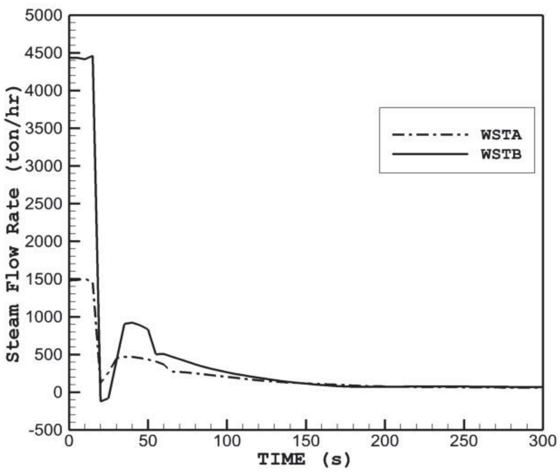

Fig. 9: Feed water Flow Rate in SG ‘A’ and ‘B’ (t/h)

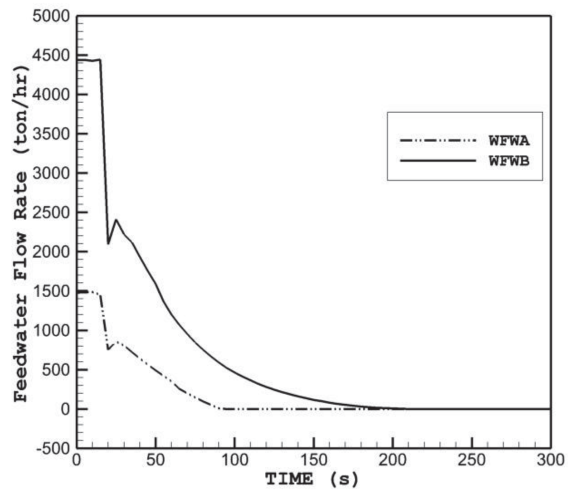

Fig. 10: Steam Flow Rate in SG ‘A’ and ‘B’ (t/h)

As shown in Fig. 9, the supply of feedwater into affected SG ‘A’ reduces rapidly and is eventually stopped within 200 seconds of the transient. The production of steam follows the same manner (Fig. 10). As the temperature of “Hot” leg approaches to “Cold” leg temperature, there will be no net produced steam. (Fig. 10)

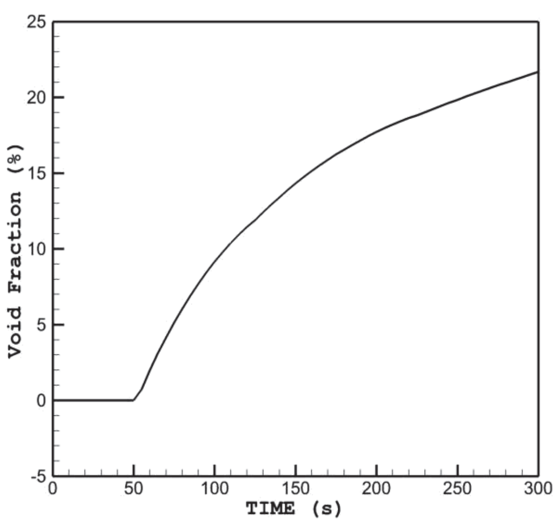

Fig. 11: Void Fraction in Reactor Coolant (%)

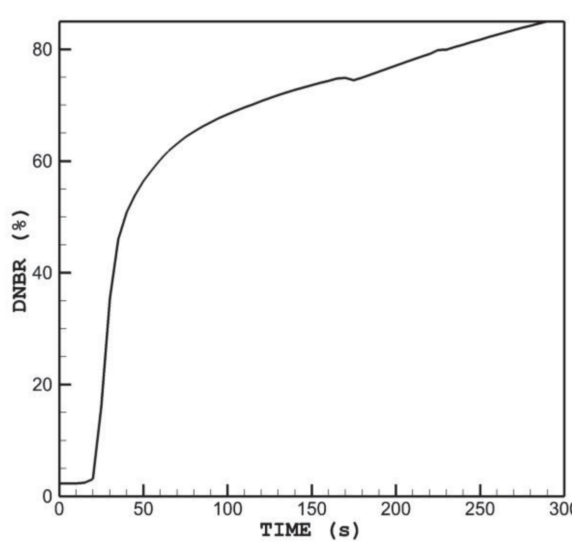

Fig. 12: Departure from Nuclear Boiling Ratio (%)

Void Fraction and DNB Ratio has been shown in Fig. 11 and 12 respectively on a percentage basis. Void fraction is the fraction of vapor present in a certain volume of two phase mixture (in this case "reactor coolant-vapor"). Fig. 11 indicates that the amount of vapor in the coolant started to increase at about 50 seconds after the transient. To prevent the leakage of coolant into the secondary system, primary system is depressurized in the early phase of the transient. As a result the coolant pressure is decreased so as its saturation temperature. This leads to boiling of the coolant to some extent forming vapor within the primary system. However, void formation can be controlled by restoring the primary pressure to its nominal value.

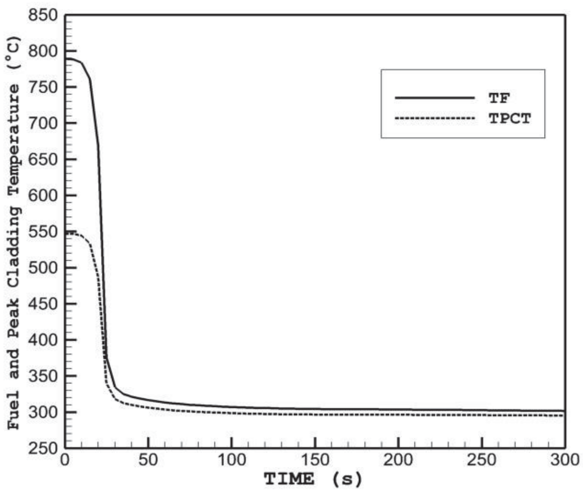

Fig. 13: Fuel and Cladding Temperature (°C)

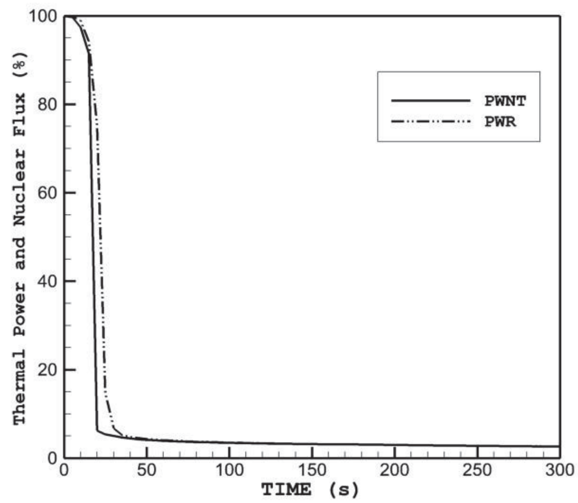

Fig. 14: Core Thermal Power and Nuclear Flux (%)

The fuel surface and cladding temperature are shown in Fig. 13. After reactor shut down, reactor power stays in nominal level with in 20 s from the beginning then in a short time it will decrease from nominal value to decay heat (Fig. 14). Since, the fuel temperature is proportional with heat generation in fuel surface unit, there for it will reduce with decrease of reactor power. As Fig. 13 shows, the fuel surface temperature reduces linearly. This behavior is due to the fact that heat generation of decay heat has the same linearly reduction in a day.

V. CONCLUSION

The thermal-hydraulic response of Pressurized Water Reactor during SGTR accident is presented and analyzed in this paper. A plant specific PCTRAN model (VVER-1000) was used for simulating the system behavior during the transient. The simulation was performed assuming single tube rupture in the secondary system. The results obtained from the calculation are in good agreement with experimental data conducted by [2, 6 and 7].

It is found from the study that, the plant safety system is capable to restore the plant in steady state condition without operator interactions and thus maintaining the acceptance criteria for any Design Basis Accidents (DBA) as prescribed in Safety Report Series [8] by IAEA. As it is seen from the simulation results:

- The pressure in the primary coolant and in the main steam system is maintained below a prescribed value (typically 110% for DBAs). This criterion ensures that the structural integrity of the reactor coolant boundary is maintained. The pressure in the primary coolant system decreases from 15.5 MPa to approximately 8 MPa during the whole transient. Thus, the ultimate value of coolant pressure of the primary side is not reached, and the acceptance criterion is met.
- The reactor is ensured to be filled with cooling water during the transient by activating ECCS. This protects the fuel rod from being uncovered and prevents melting of core.
- The peak fuel cladding temperature 788.9°C doesn't exceed the prescribed value of 1200°C which ensures that the melting and structural deformation of core integrity can be avoided.

It is expected to provide useful information in understanding the plant responses to SGTR event and evaluating the effectiveness of existing safety system and operator actions to mitigate the consequence of the transients.

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