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PCTRAN Generic Pressurized Water Reactor Simulator Exercise Handbook

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PCTRAN GENERIC PRESSURIZED WATER REACTOR SIMULATOR EXERCISE HANDBOOK

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FOREWORD

The IAEA assists Member States by providing education and training courses on the physics and technology involved in nuclear power plant operation through hands-on learning with nuclear reactor simulation computer programs. 'Basic principle simulators', including those available to Member States upon request, are an accessible resource for teaching and training on a wide range of topics such as nuclear power plant design, safety, technology, simulation and operations. The often simplified reactor designs of basic principle simulators allow professionals to grasp fundamental concepts without becoming overwhelmed by the details of a more complex, full scope simulator. The objective of these simulators is to provide insight into and a practical understanding of reactor operational characteristics and plant responses to perturbations and accident scenarios.

The IAEA regularly publishes reference material and holds training courses to assist professionals in Member States in understanding simulators and associated technologies. Education and training courses support an integrated approach that combines lectures with 'learning by doing' on the specifics of plant operation, including reactor physics, thermohydraulics and safety aspects, using basic principle simulators.

This exercise handbook is based on the IAEA's Personal Computer Transient Analyzer (PCTRAN) — a two loop pressurized water reactor simulator. The plant model used in the simulator is a generic design and uses various assumptions for simulations of simplified plant conditions. While the simulator is not intended to be used for any safety analysis purposes, it can be used for educational purposes, as the general behaviour of the plant responses during the normal operations and transients is accurate.

The handbook includes practical exercises to be included as part of education and training materials provided in IAEA training courses and workshops. It provides detailed, step by step instructions and explanations needed to run various normal reactor operations and transients with the PCTRAN simulator. The publication can be used independently for self-learning or for further training. From a general point of view, this publication acts as reference material to support human capacity building in Member States by refining expertise in education and training activities on pressurized water reactor technology.

The IAEA officers responsible for this publication were C. Batra and T. Jevremovic of the Division of Nuclear Power.

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1. INTRODUCTION

1.1. BACKGROUND

The PCTRAN simulator version 6.0.4, available from the IAEA, is based on a generic two-loop pressurized water reactor (PWR) with inverted U-bend steam generators (SGs) and dry containment. The PWR, as stated above, is generic and could represent a Westinghouse, Framatome or Kraftwerk Union design with thermal output of 1800 MW(th) (600 MW(e)). PCTRAN models two loops of the reactor coolant system (RCS), represented as loop A and loop B in the simulator. Figure 1 shows the plant main mimic display of PCTRAN which appears when the simulator is launched. It shows major systems and their components such as the RCS, the Emergency Core Cooling System (ECCS), the Turbine-generator system, and the Reactor Protection System (RPS), in a single unified display. Table 1 provides the details of the pumps and valves as represented in Figure 1. The components are numbered in order to properly describe them in the table and is intended to aid the user for better understanding of components. The components that exist in both loops are distinguished with A and B; for example, pumps 1A and 1B and valves 8A and 8B. Red coloured components indicate operating pumps and open valves, while white coloured components are idle pumps and closed valves. The system mimic is interactive, and the valves and pumps could be switched on/off by simply clicking on them. An alternative to the main mimic display is the dose mimic display shown in Figure 2. This view provides information on dose rates, radioactive releases, and core degradation for accident scenario simulations.

TABLE 1. PCTRAN SIMULATOR COMPONENTS

Number	Component
1A, 1B	Reactor Coolant Pumps (RCPs)
2	Pilot-Operated Relief Valves (PORV)
3	Pressurizer spray nozzle
4	Pressurizer auxiliary spray valve (from CVCS)
5	Pressurizer spray valve (from a cold leg)
6	Pressurizer relief tank
7A, 7B	Atmospheric Dump Valves (ADVs) or Main Steam Safety Valves (MSSVs)
8A, 8B	Main Steam Isolation Valves (MSIVs)
9A, 9B	Feed Water Isolation Valves (FWIVs)
10	Turbine governor valve
11	Turbine bypass valve
12	Condensate pumps
13	Feedwater pumps
14	Motor-driven auxiliary feedwater pumps
15	Turbine of the auxiliary feedwater pump
16	Turbine-driven auxiliary feedwater pump
17	Valve for steam to the turbine of the auxiliary feedwater pump
18	High pressure injection (HPI) pumps / Charging pump
19	Letdown valve
20	Accumulators
21	Low pressure injection (LPI) pumps
22	Reactor building spray pumps
23	Containment cooling fans
24	Containment vent valve

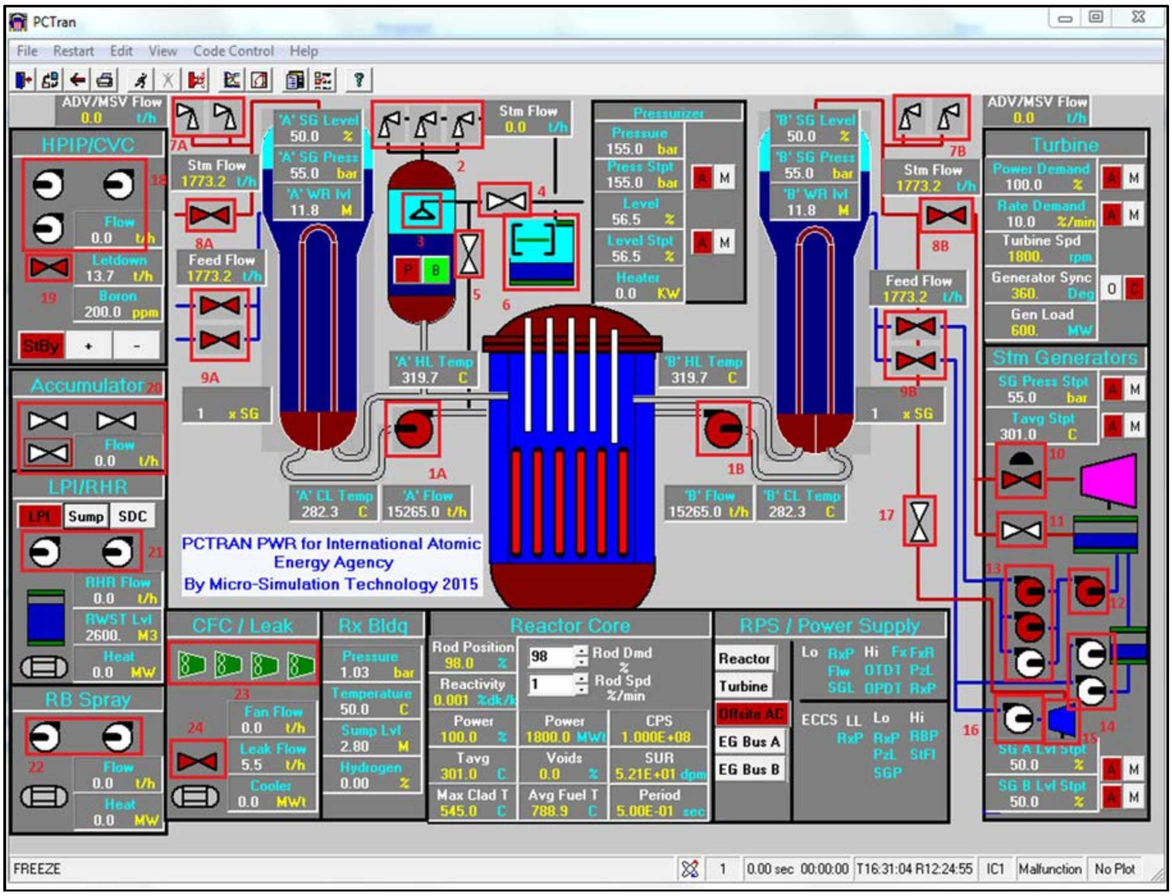

FIG. 1. Main system mimic screen of PCTRAN.

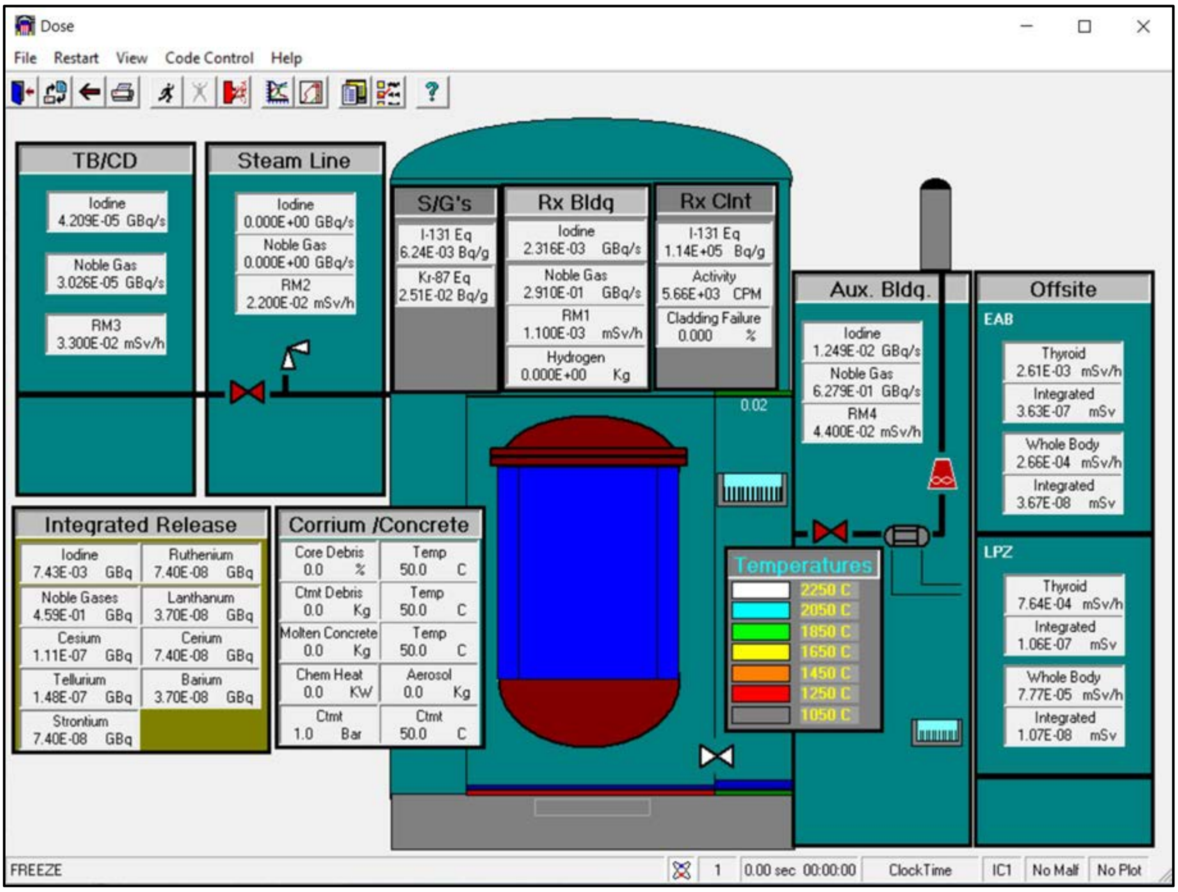

FIG. 2. Dose mimic screen of PCTRAN.

1.2. OBJECTIVE

The main objective of this publication is to provide a set of useful PCTRAN exercise scenarios to help the PCTRAN users with better understanding of various systems, operations, and transients of PWRs.

For detailed background information of PCTRAN, such as how to install the software and plant specifics, refer to the PCTRAN manual that comes with the simulator. Details on how to obtain the simulator are also available on the IAEA website [1]. In addition, the manual provides details of the physical models that have been used in the simulator [2].

1.3. STRUCTURE

This publication consists of 12 PCTRAN exercises divided into the following three sections:

(a) Normal operations;
(b) Transient and accident analysis;
(c) Severe accidents.

Each exercise consists of the following three subsections:

(a) Simulation description for a general explanation of each exercise and a table describing a simulation procedure with snapshots of the simulator, including classification of accident scenarios;
(b) Transient behaviour description, where the simulation results are analysed;
(c) Transient plot description, where relevant variables are plotted and explained.

The annex contains tabular information on PCTRAN component setpoints, basic data, initial conditions, malfunction scenarios, and plottable variables. This reference information is as it appears in the simulator and may include non-standard abbreviations, designations or units. These are explained to a limited extent within the annex and the abbreviations section; however, it is the responsibility of the user to further interpret and properly utilize this data.

1.4. BASIC INSTRUCTIONS AND LIMITATIONS

An upper and lower toolbar, shown in Figure 3, are included in the PCTRAN user interface. A simulation is started by clicking the ‘Run’ button, indicated by an image of a running figure on the upper toolbar. It is paused by clicking the ‘Freeze’ button, indicated by a standing figure, located immediately right of the ‘Run’ button. Alternately, a toggle button, located on the lower toolbar and indicated by an image of an atom, can be used to switch between ‘Run’ and ‘Freeze’. The simulator offers several ‘speeds’ for running a scenario, which can be changed by clicking on the number on the lower toolbar immediately right of the ‘Run’/’Freeze’ toggle button. Available are a ‘real time’ speed, and speeds running 2, 4, 8, and 16 times faster than the real time. These faster speeds could be used to run longer transients to see the results in relatively less time, however, it is better to use real time speed during most of the scenarios.

(a)

(b)

FIG. 3. PCTRAN (a) upper and (b) lower toolbars.

PCTRAN provides several transient plotting options which can be accessed through the ‘View Transient Plots’ button on the upper toolbar or through the ‘Transient Plot’ option in the ‘View’ tab of the menu bar. Upon opening, the user is prompted to select variables to plot as a function of time, followed by a prompt for plot settings. Available variables for plotting are listed under A.5. in the appendix, and multiple may be selected for simultaneous plotting. Variables and plot settings can be changed or adjusted from the menu bar in the transient plot window.

In some of the scenarios presented, the user is instructed to initiate malfunctions, either to the reactor system or to specific reactor components. Reactor system malfunctions can be initiated by the ‘Malfunctions’ buttons located on the upper and lower toolbars or through the ‘Code Control’ tab of the menu bar. Upon clicking one of these buttons, the user is presented with the 20 available reactor system malfunctions listed under A.4. in the appendix. To initiate a reactor component malfunction, the user right clicks on the desired component and is presented with component specific malfunction options. Components that are malfunctioning are indicated by either a change in colour or a red box outlining the component.

In some of the included accident scenarios, the user will need to look at the dose mimic display. Switching between the main mimic and dose mimic can be done by selecting either the ‘Main Mimic’ or ‘Dose Mimic’ options found under the view tab of the menu bar.

PCTRAN comes pre-loaded with 12 initial conditions (ICs), numbered 1 to 12 and listed under A.3. in the appendix. Many of the exercises described in this report use initial condition #1, ‘100% POWER EOC’, which means that the plant is operating at full power, and the fuel is at the ‘End of Cycle’ (EOC). This version of the simulator, which is distributed by the IAEA, has the following limitations:

1. The simulator run is limited to 1000 seconds. If a user wants to run it further than this, the state of the plant in the simulator at the end of this time can be saved as an initial condition and loaded again for another 1000 seconds of simulation. This can be repeated as many times as the transient simulations demands. Most of the scenarios explained in this book, however, will not need more than 1000 seconds of simulation time. By saving a new initial condition from number 13 will avoid overwriting the 12 pre-loaded initial conditions. In this way, one can prolong the simulation with repetitive initial-condition saving and loading. However, some minor variables are not saved. The effect that this loss of information has on a simulation has not been explored completely by this time. It is possible that the more times the simulator is run in this way, the loss of information would compound and become more significant. Table 2 provides a step-by-step procedure for extending the running time of a simulation scenario.
2. The visual display of the simulator does not always accurately represent the real plant behaviour. For example, in real nuclear power plants, several control rods change their position together for a reactor power manoeuvre. However, in PCTRAN, the control rod

icons move one by one to represent change of rods' position. On the other hand, the numerical results and plots generated by the simulator are realistic and therefore can be used to interpret the results.

All the figures in this publication are screen shots of the simulator display and plots generated by the simulator. These plots are y versus x graphs, where the x axis is the time, and the y axis is the variable(s) mentioned in the figure's caption in this report.

TABLE 2. SAVING AND LOADING INITIAL CONDITIONS FOR PROLONGED SIMULATIONS

STEP	PROCEDURE STEPS	EXPECTEDRESPONSE
1	Run the simulator for 1000 seconds.	
2	When a message box asking to exit the simulator, click the 'No' button.	Simulator stops.
3	In the menu bar, click the 'Restart' tab and click the 'Initial Conditions'.	The 'Initial conditions' window appears.
4	Click the 'Save New IC' button	The 'New Initial Condition' window appears.
5	Select an 'IC number' in which you want to save the current status of the simulation. Be cautious of overwriting the existing IC.	
6	Write a description for the simulation in the 'Description' textbox and click the 'Ok' button. It is necessary to enter a description because the IC will not be saved if no description is entered.	The current simulation status is saved as a new initial condition and appears in the initial conditions list.
7	Select the initial condition saved in the previous step and click the 'OK' button.	
8	When the message boxes asking if you want to save the transient report, the plot data, and the dose data, click 'Yes' or 'No', depending on whether you want to save intermediate data.	
9	Click the 'Run' button for another 1000 seconds of simulation.	The simulator resumes operation.
10	Continue from Step 1 in case another 1000 seconds of simulation is needed.	

2. NORMAL OPERATION

2.1. POWER REDUCTION FROM 100% TO 75% – TURBINE LEADING MODE (REACTOR FOLLOWING MODE)

In turbine leading mode (or reactor following mode), which is normally the preferred mode of operation in a PWR, the plant's operator sets the target electric power output and the target rate of change of turbine power. The change in electric power output results in a change in reactor power output by automatic control of control rods' position.

2.1.1. Simulation description

To perform this operation in the simulator, Table 3 provides a step-by-step procedure with corresponding values for relevant variables.

In Step 1, run the simulator by clicking the 'Run' button, and wait about 5 seconds for the plant to reach a steady-state. In Step 2, freeze the simulator to initiate the power reduction by turbine load decrease. In Step 3 and 4, the turbine power demand and the turbine rate demand are manually decreased to 75% and 5%/min, respectively. Figure 4 shows the pop-up window where the turbine demand can be adjusted. In Step 5, the simulation is resumed, and as a result, plant variables start to change according to the turbine load change. After running the simulator for 1000 seconds, the variables listed in Step 6 reach a stable state (the steady state might be achieved earlier as well). Figure 5 shows the simulator in a steady-state about 1000 seconds after the turbine output demand change.

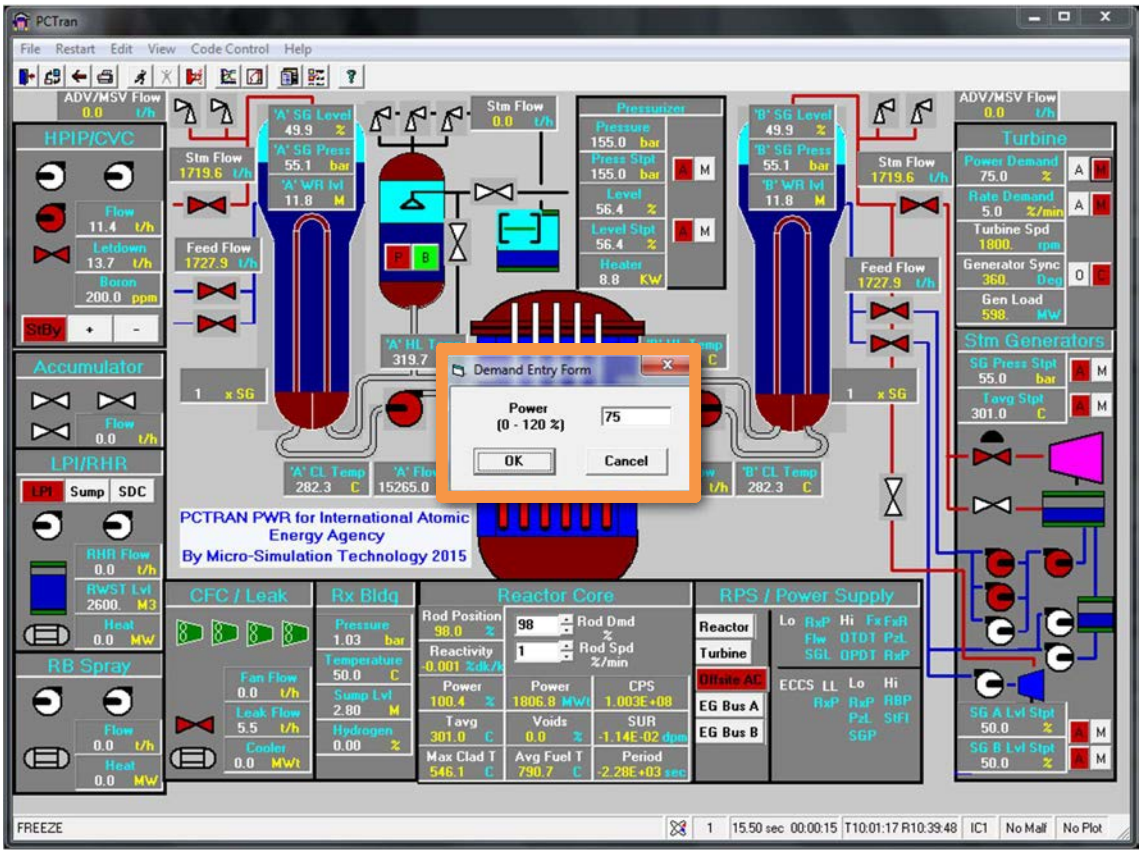

FIG. 4. Power reduction (turbine leading mode) from 100% to 75%.

TABLE 3. POWER REDUCTION FROM 100% to 75% – TURBINE LEADING MODE

STEP	PROCEDURE STEPS	EXPECTED RESPONSE	FIGURE
Setup	Load initial condition #1 100% POWER EOC, or #2 100% POWER MOC, or #3 100% POWER BOC. Initial condition #1 is used for this exercise.		
1	Run the simulator for about 5 seconds to achieve a steady-state condition.		
2	Freeze the simulator.	The simulator freezes.	
3	Reduce the power demand to 75%. (1) Click the 'M' button next to 'Power Demand' in 'Turbine' section. (2) Type 75 in the textbox and click 'OK'. (3) Check the power demand changed to 75%.	The turbine power demand is set to 75%.	Figure 4
4	Adjust the power rate demand. (1) Click the 'M' button next to 'Rate Demand' in 'Turbine' section. (2) Type the desired rate demand in the textbox and click 'OK'. 5%/min. is used in this exercise. (3) Check the rate demand changed to 5%/min.	The ramp rate of the power demand is set to 5%/min.	Figure 4
5	Resume the simulator (*speed up the simulator if needed*).	The simulator resumes operation.	
6	Observe that reactor power is stable at 75% (~1350MWt) at 1000 seconds (*could be before*). Check the following variables: (1) Rod position (%) (2) Reactor power (MWt) (3) T_{avg} (°C) (4) Pressurizer pressure (bar) (5) Pressurizer level (%) (6) Generator load (MW) (7) Steam generator pressure (bar) (8) Steam generator level (%)	All variables stable at: (1) ~90% (2) ~1351 MW(th) (3) ~296.8 °C (4) 155.0 bar (5) 48.3% (6) 450 MW(e) (7) ~57 bar (8) 50.0%	Figure 5

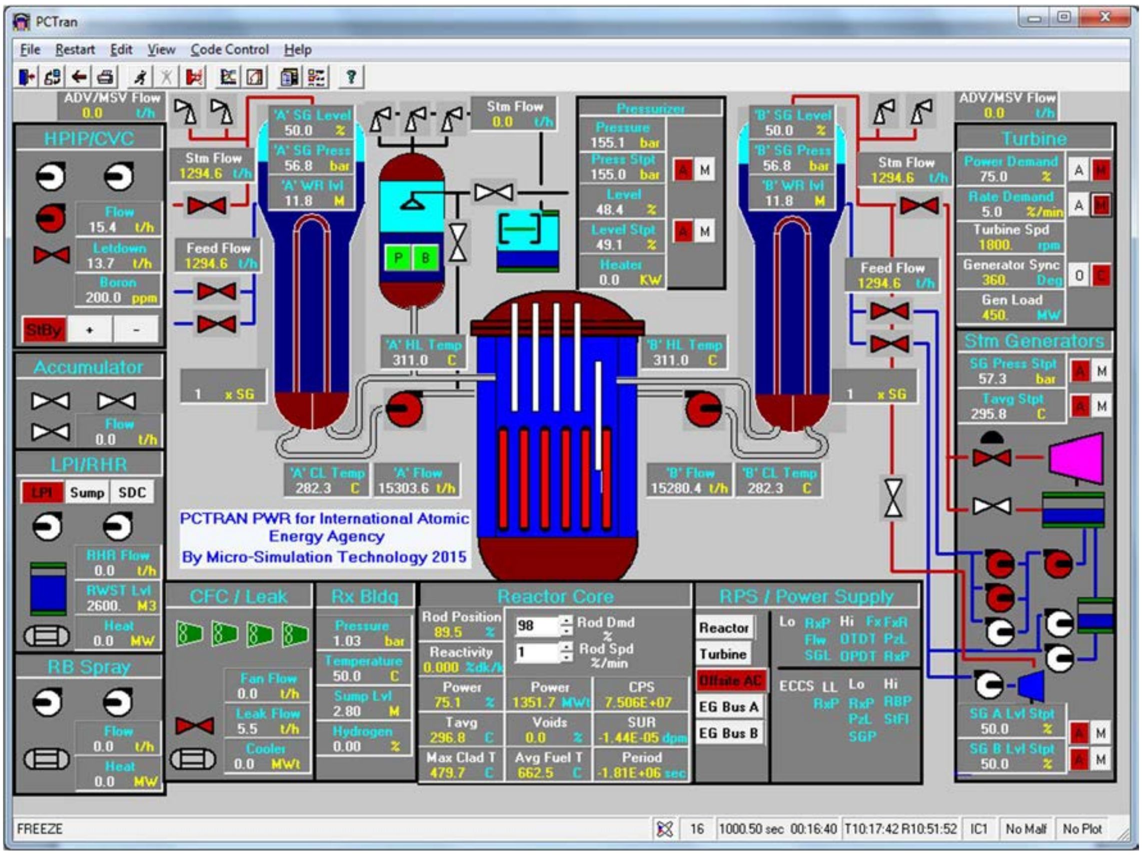

FIG. 5. 1000 seconds after the power reduction (turbine leading mode).

2.1.2. Transient behaviour description

The change in the turbine output demand alters the reactor thermal output with the actuation of the automatic control of the control rods.

As the turbine power demand decreases, the governor valve allows less steam into the turbine. The plant takes a little over 300 seconds to reduce the turbine power to 75%. During this period, reduced steam flow results in increases in steam generator pressure and water level, and the addition of negative reactivity due to control rod insertion lowers the reactor power, which results in decreased reactor coolant average temperature (T_{avg}) and primary pressure.

In a typical Westinghouse PWR, the automatic rod control system consists of two error signal channels which are summed to generate a rod speed demand signal. The two data points used to generate the total error signal are:

1. The deviation between the actual reactor coolant system average temperature (T_{avg}) and the programmed average temperature (T_{ref}). The latter temperature is programmed as a function of turbine load.
2. The rate of change of the mismatch between nuclear power and turbine power. Since a mismatch of power produced by the core to power used by the turbine will result in a changing T_{avg}, the rate of change of the mismatch is used as an anticipatory signal.

During the scenario of reduction of turbine power to 75%, the actual T_{avg} is greater than T_{ref} for a given load, so the rod control system inserts the control rods to add negative reactivity to the reactor.

$T_{avg} - T_{ref} > 0$ => insert control rods

$T_{avg} - T_{ref} < 0$ => withdraw control rods

$T_{avg} - T_{ref} = 0$ => no control rods movement needed

Once reactor power is stable at 75%, the primary and the secondary system pressures and the steam generator (SG) water level are steady.

2.1.3. Transient plot description

In this transient, the control rods are inserted so the reactor power follows the turbine load (Figure 6). Since T_{avg} is reduced, there is also a positive insertion of reactivity due to the negative moderator temperature coefficient.

The rod control system is programmed to compare the reactor power with the turbine load, and also accounts for T_{avg}. According to this programme, the rate of insertion of the rods is set to follow the load decrease.

The turbine load stops decreasing once it reaches 75% at about 300 seconds and the rod control system reduces the rate of insertion of the rods. As a result of this insertion, the reactor thermal output drops slightly below 75%. This undershooting of the thermal power could be minimized if the rate or power demand is reduced.

The rod control system accounts for the difference between reactor power and turbine load, based upon the calculation of T_{avg} and T_{ref}, and the positive reactivity due to the reduction in T_{avg} to adjust the rate of insertion of the rods so the reactor power returns to 75%.

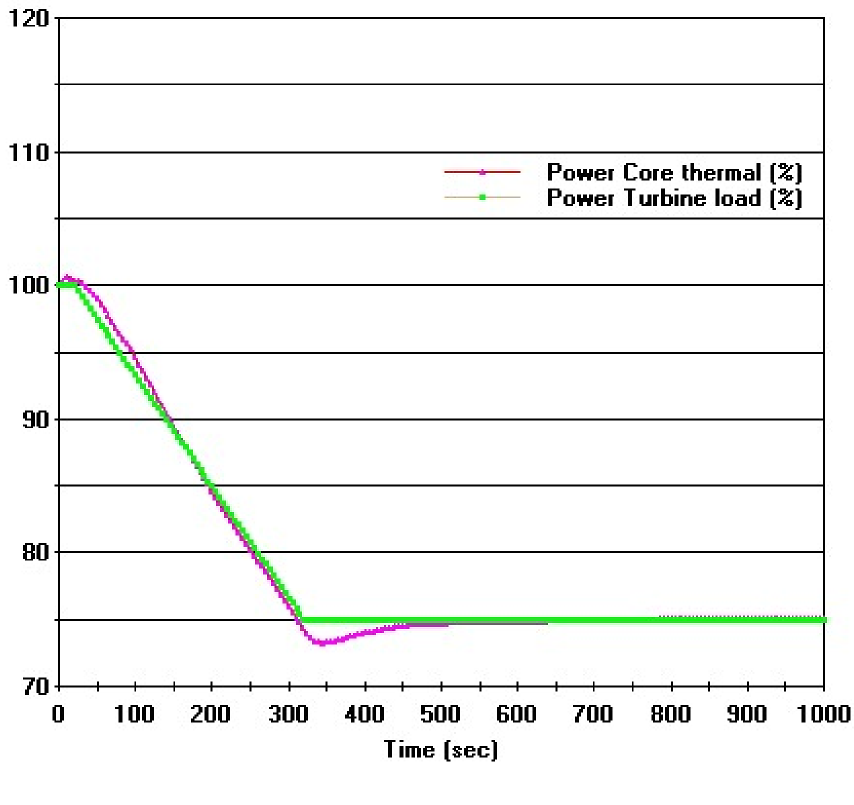

FIG. 6. Turbine load and reactor power output (%).

Figure 7 shows the steam and feedwater flow rates. The former rate decreases according to the turbine load change for the first 300 seconds, and then it stabilizes once the turbine power reaches 75% power level. The feedwater flow rate follows the steam flow; however, it keeps on decreasing for about 100 seconds and then stabilizes with the help of SG water level control system. This system then adjusts the feedwater flow rate until it matches the steam flow rate.

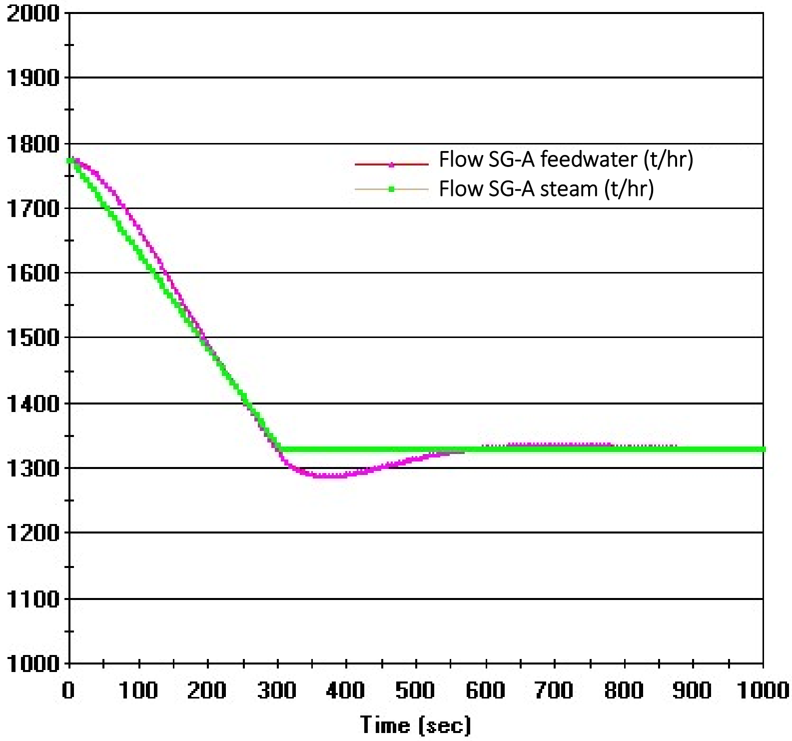

FIG. 7. Steam and feedwater flow rates (t/hr).

Figure 8 shows an initial increase of the pressure of the SG-A due to the reduced steam flow rate to the turbine until the turbine power reaches 75% at about 300 seconds.

Subsequently, the pressure decreases because the reactor power becomes less than 75% before stabilizing when the reactor power matches the turbine power. The steam generator pressure settles at a slightly higher value at this lower power.

Figure 9 shows the water level of the SG-A. The SG water level increases as there is a mismatch between the steam flow and the feedwater flow. This is evident in the first 300 seconds as the feedwater flow rate is larger than the steam flow rate.

After the turbine power reaches 75% at about 300 seconds, the feedwater flow tries to match the steam flow necessary at this power level, as was also seen in Figure 7. Then, the water level follows the changes in feedwater flow until it reaches a steady-state value.

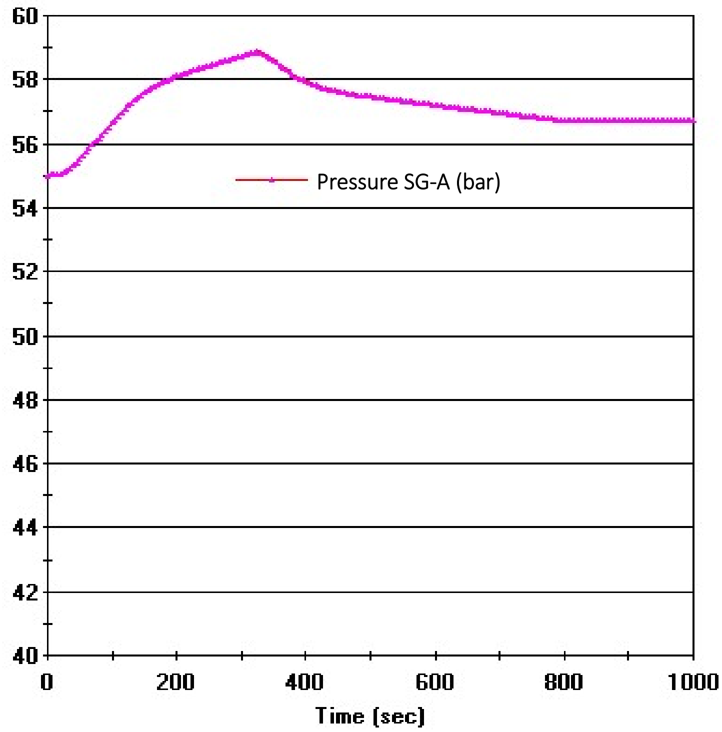

FIG. 8. SG-A pressure (bar).

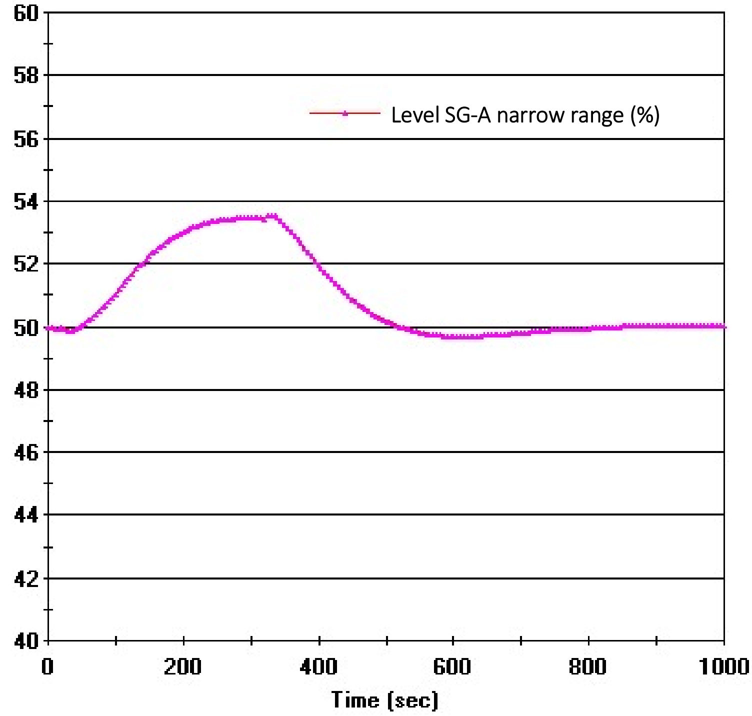

FIG. 9. SG-A water level (%).

The reactor coolant system (RCS) pressure changes as the coolant temperature changes. Coolant volume expands when heated, causing an increase in the RCS pressure as well as the pressurizer water level (slightly). Similarly, when the coolant's temperature decreases, the RCS pressure and the pressurizer level are reduced.

Accordingly, Figure 10 shows first a slight increase due to coolant heat up and then the RCS pressure decreases during the initial cool down due to the control rod insertion (until about 300 seconds). Then this pressure is restored to the setpoint of 155 bar by the actuation of the pressurizer's electric heater as the reactor power reaches a steady-state corresponding to the 75% turbine power. The RCS pressure is maintained at a constant value at all operating conditions.

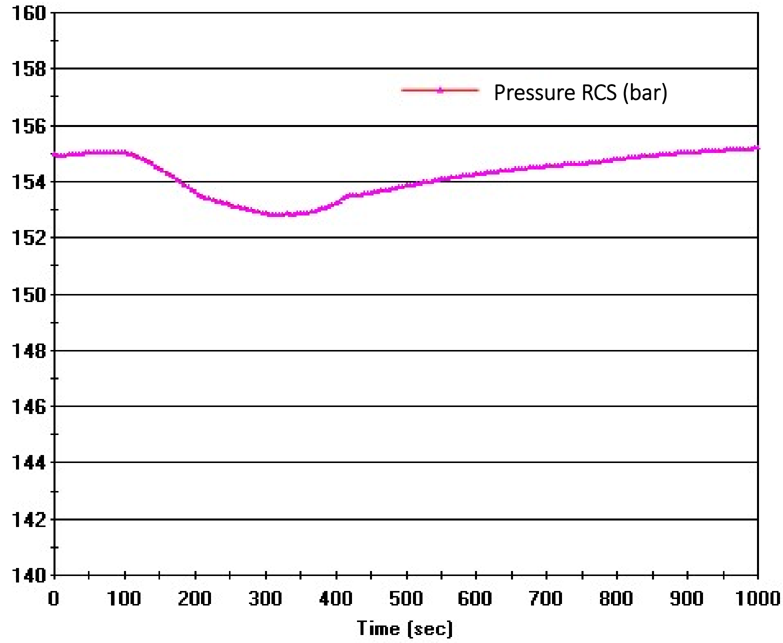

FIG. 10. Reactor coolant system pressure (bar).

2.2. POWER INCREASE FROM 75% TO 100% – TURBINE LEADING MODE (REACTOR FOLLOWING MODE)

This operation is similar to the one explained in previous exercise, however, in reverse direction. In this exercise, the user is expected to increase the power output from 75% to 100% in turbine leading mode. The system behavior is also expected to be opposite of the previous power reduction exercise.

2.2.1. Simulation description

To perform this operation in the simulator, Table 4 provides a step-by-step procedure for simulation with corresponding values for relevant variables.

To set up the simulation, one of the three pre-loaded 75% power output initial conditions is loaded. The state reached in the previous exercise could also be stored as a separate IC and used as a starting IC for this exercise. In Steps 3 and 4, the power demand and power rate demand is set to 100% and 5%/min, respectively (Figure 11)

After the simulator is resumed in Step 5, all variables listed in Step 6 will start to change according to the power demand increase and finally, the plant reaches a steady-state. In this exercise the variables are recorded at 1000 seconds, however the steady state could be reached earlier (Figure 12).

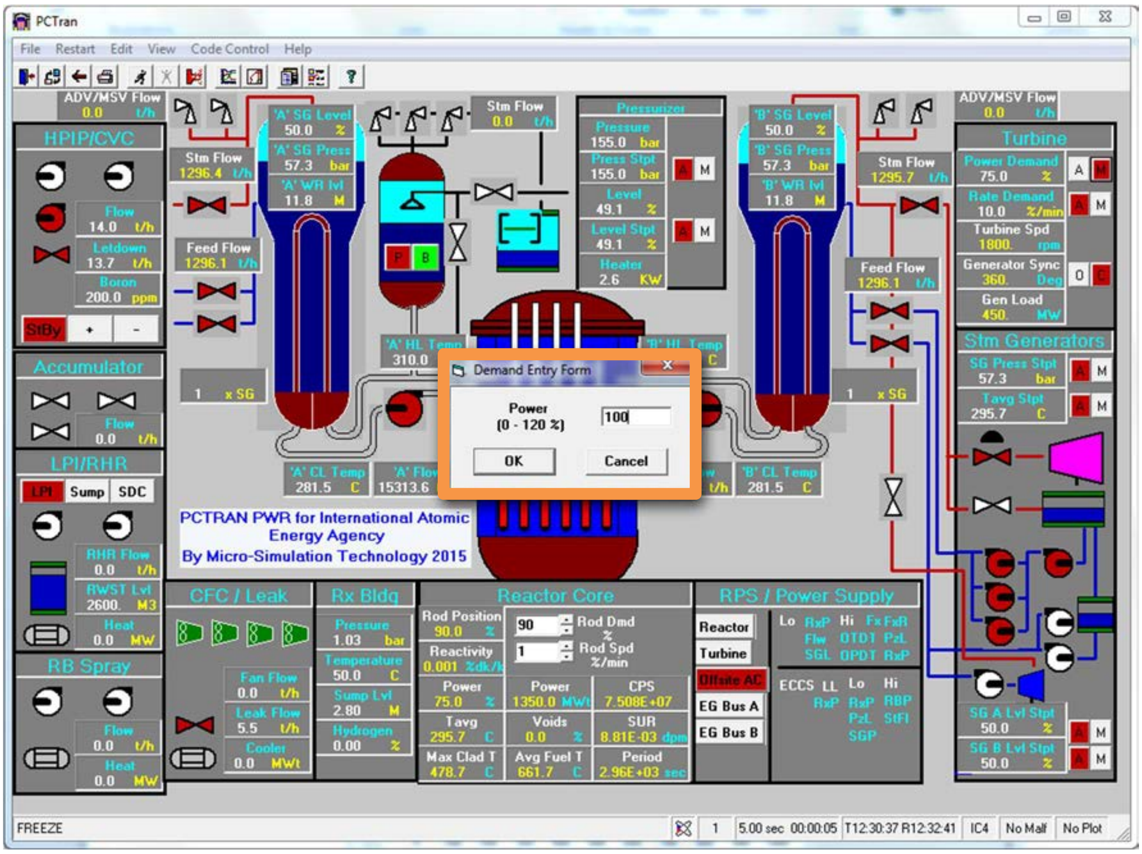

FIG. 11. Power increase from 75% to 100% (turbine leading mode).

TABLE 4. POWER INCREASE FROM 75% to 100% – TURBINE LEADING MODE

STEP	PROCEDURE STEPS	EXPECTED RESPONSE	FIGURE
Setup	Load initial condition #4 75% POWER EOC, or #5 75% POWER MOC, or #6 75% POWER BOC. Initial condition #4 is used for this exercise.		
1	Run the simulator for about 5 seconds to achieve a steady-state condition.		
2	Freeze the simulator.	The simulator freezes.	
3	Increase the power demand to 100%. (1) Click the 'M' button next to 'Power Demand' in 'Turbine' section. (2) Type 100 in the textbox and click 'OK'. (3) Check the power demand change to 100%.	The power demand is set to 100%.	Figure 11
4	Adjust the power rate demand. (1) Click the 'M' button next to 'Rate Demand' in 'Turbine' section. (2) Type the desired rate demand in the textbox and click 'OK'. The value around 5%/min is a usual rate demand for PWRs. (3) Check the rate demand changed to 5%/min.	The power rate demand is set to 5%/min.	Figure 11
5	Resume the simulator (*speed up the simulator if needed*).	The simulator resumes operation.	
6	Observe that reactor power is stable at about 100%. At about 1000 seconds (*could reach before*) Check the following variables: (1) Rod position (%) (2) Reactor power (MW(th)) (3) T_{avg} (°C) (4) Pressurizer pressure (bar) (5) Pressurizer level (%) (6) Generator load (MW) (7) Steam generator pressure (bar) (8) Steam generator level (%)	All variables stable at: (1) ~100% (2) ~1805 MW(th) (3) ~300.0 °C (4) 155.0 bar (5) 57.0% (6) 600 MW (7) 55.7 bar (8) 50.0%	Figure 12

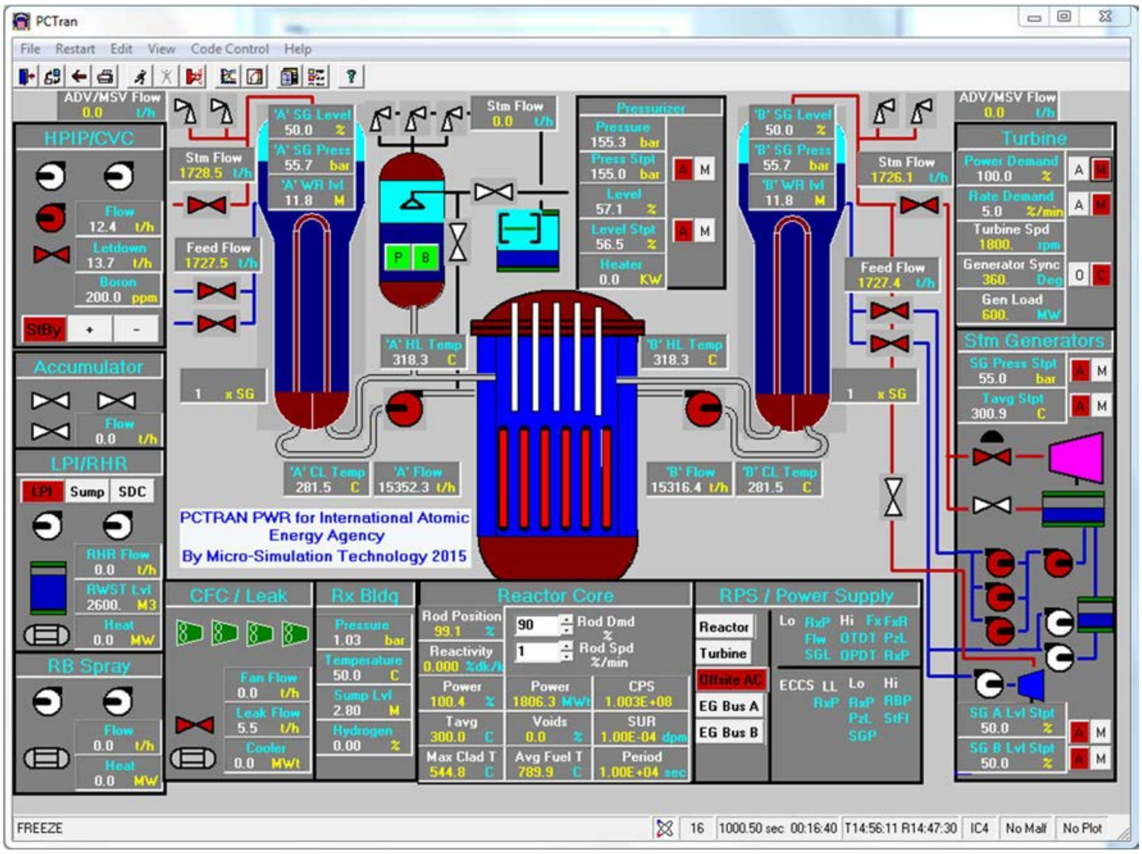

FIG. 12. 1000 seconds after the power increase (turbine leading mode).

2.2.2. Transient behaviour description

With the increase in turbine demand, the governor valve is automatically opened at the rate specified by the user. As more steam flows into the turbine, the SG pressure as well as the SG water level decrease. With reduced SG pressure, a new SG pressure setpoint and a corresponding T_{avg} setpoint are determined.

The previous exercise briefly described the logic of the automatic rod control system. During the scenario of increasing turbine power from 75% to 100%, the actual T_{avg} is less than T_{ref} for a given load as the turbine load is increased, and therefore, the control rods are withdrawn to add positive reactivity, thus increasing the reactor power.

$T_{avg} – T_{ref} > 0$ => insert control rods

$T_{avg} – T_{ref} < 0$ => withdraw control rods

$T_{avg} – T_{ref} = 0$ => no control rods movement needed

Once reactor power is stable at 100%, the primary and the secondary system pressures as well as the SG water level stabilize.

2.2.3. Transient plot description

Figure 13 shows that the turbine load increases linearly from 75% to 100%, and the reactor power follows the turbine load by withdrawing control rods.

During power increase, the rod control system compares the turbine load with the reactor power and also takes T_{avg} into account to determine the rate of extraction of control rods.

The turbine load stops increasing once it reaches 100% at about 300 seconds and the rod control system reduces the rate of control rods extraction. As a result of this extraction, the reactor power reaches slightly above 100%. This overshoot could be minimized by reducing the rate demand of the power increase. In order to check that, use a rate demand of 1% and repeat the exercise.

The rod control system balances the difference between the reactor power and turbine load with the negative reactivity added due to the increase in T_{avg}. It does this by adjusting the rate of withdrawal of control rods so the reactor power returns to 100%.

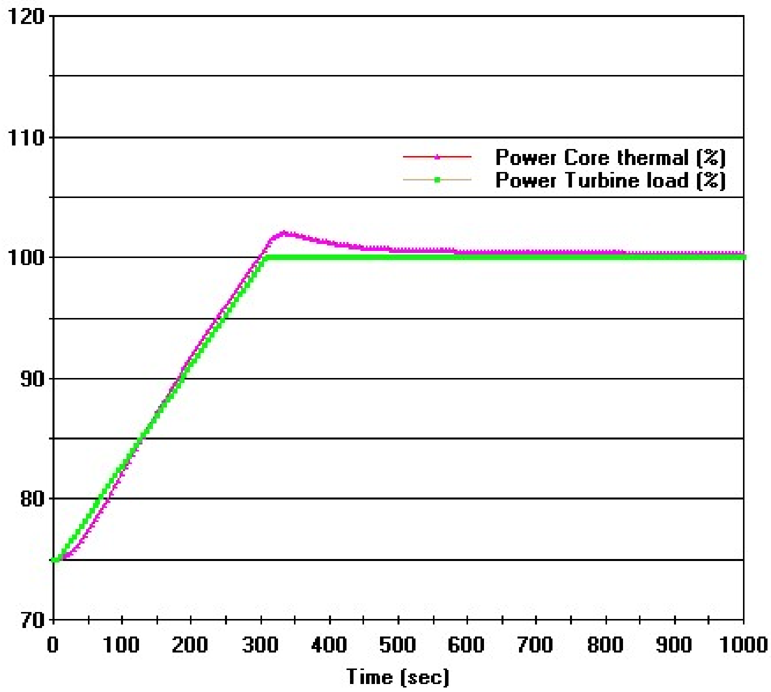

FIG. 13. Turbine load and reactor power output (%).

Figure 14 shows the trends of feedwater flow rate and steam flow rate in the secondary system. As the steam flow rate is directly proportional to the turbine load, it increases linearly until the turbine load reaches 100%.

Figure 15 shows that as the turbine governor valve lets more steam flow into the turbine, the SG pressure continuously reduces until the turbine power reaches 100% at about 300 seconds. Subsequently, the pressure increases and stabilizes at a lower value setpoint for 100% power level.

While the steam flow rate increases, the feedwater flow rate follows the steam flow rate with a noticeable lag which is the reason for the SG water level change shown in Figure 16. At about

300 seconds, the steam flow rate stabilizes at the 100% power level. The feedwater flow rate, however, keeps on increasing for about 100 seconds more and then stabilizes with the help of SG water level control system. This system then adjusts the feedwater flow rate until it matches the steam flow rate.

The difference between the steam and the feedwater flow rates results in changes in the SG water level. When the steam flow rate exceeds the feed water flow rate during the power increase, the water level goes down as shown in Figure 16.

Conversely, a little after 300 seconds when the turbine power reaches 100%, the feedwater flow rate exceeds the steam flow rate, and the water level goes up until it eventually stabilizes as the feedwater flow rate matches the steam flow rate.

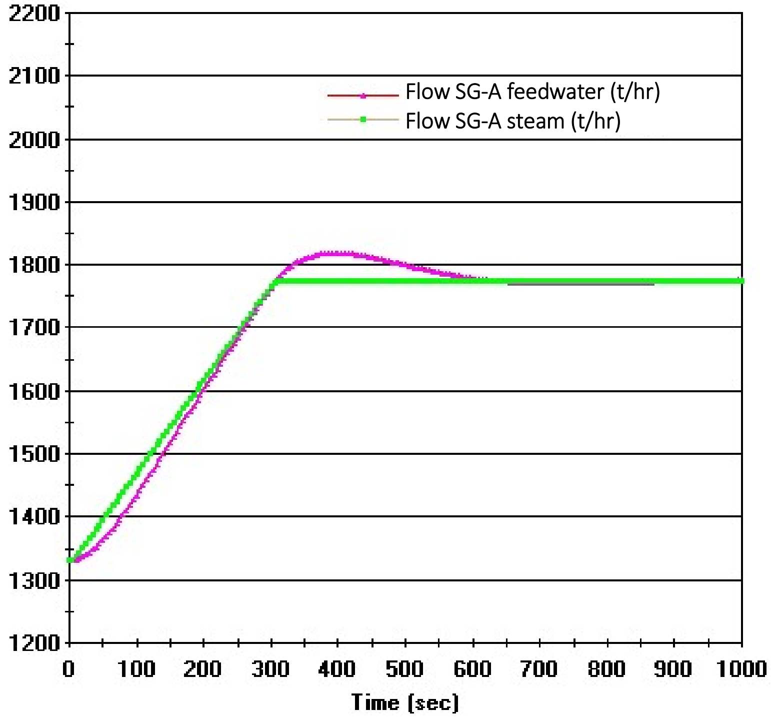

FIG. 14. Steam and feedwater flow rates (SG-A) (t/hr).

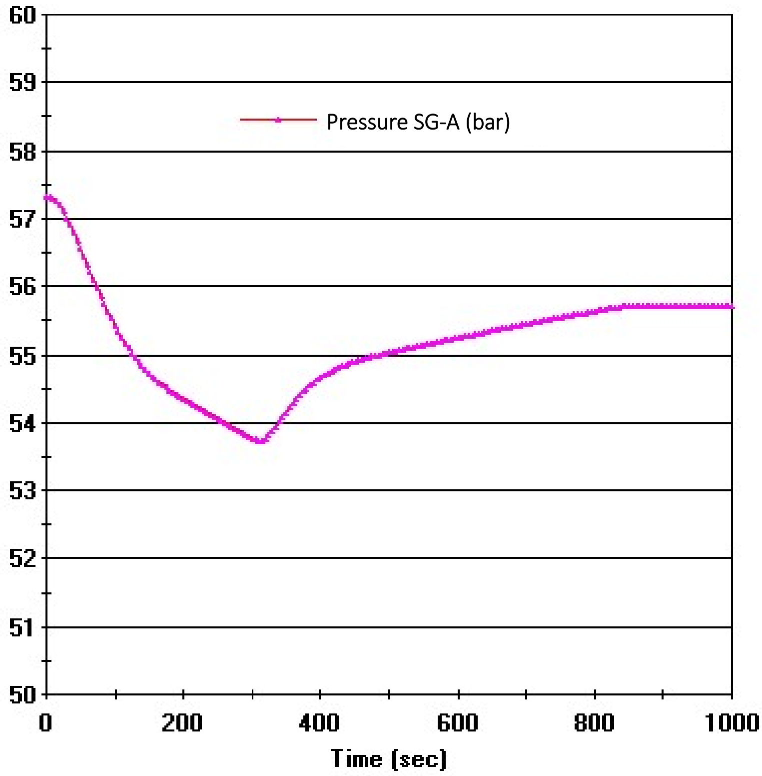

FIG. 15. SG-A pressure (bar).

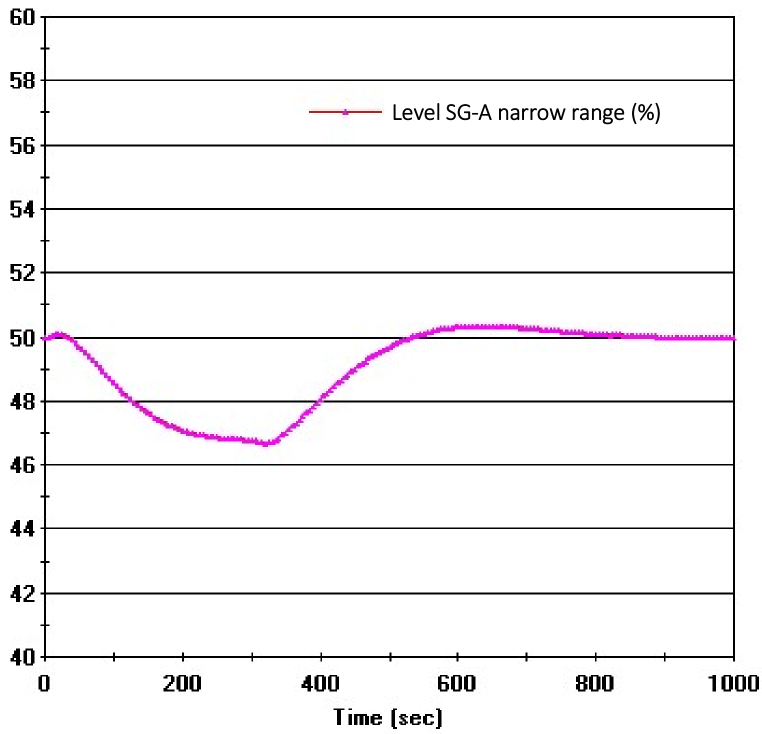

FIG. 16. SG-A water level (%).

2.3. POWER REDUCTION/INCREASE – REACTOR LEADING MODE (TURBINE FOLLOWING MODE)

In reactor leading mode (or turbine following mode), the plant's operator specifies the reactor power output target (control rods demand) and the turbine power output changes accordingly. In this exercise, the user will change the rod position and rod speed and understand how they affect the reactor and turbine power outputs. However, the reactor leading mode is usually used in Boiling Water Reactors (BWRs) and not in PWRs, except during the reactor start-up operations.

2.3.1. Simulation description

To perform this operation in the simulator, Table 5 provides a step-by-step procedure for simulation with corresponding values of relevant variables. In Steps 3 and 4, new rod demand and rod speed are introduced to change reactor power output (Figure 17). In Step 6, the primary and secondary systems change due to these modifications and find a new equilibrium. (Figure 18).

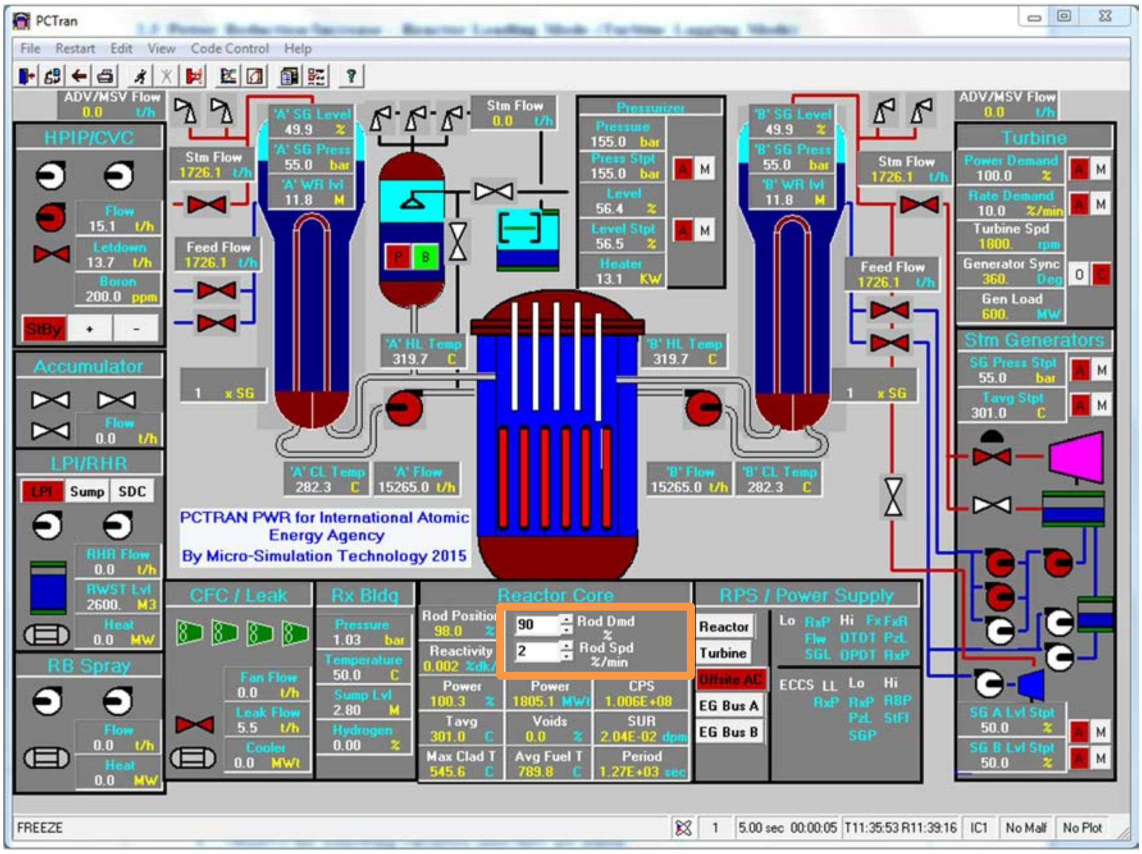

FIG. 17. *Power decrease settings of rod demand and rod speed (reactor leading mode).*

TABLE 5. POWER REDUCTION/INCREASE – REACTOR LEADING MODE (TURBINE LAGGING MODE)

STEP	PROCEDURE STEPS	EXPECTED RESPONSE	FIGURE
Setup	Load initial condition #1 100% POWER EOC, or #2 100% POWER MOC, or #3 100% POWER BOC. Initial condition #1 is used for this exercise.		
1	Run the simulator for about 5 seconds to achieve a steady-state condition.		
2	Freeze the simulator.	The simulator freezes.	
3	Manually decrease the rod demand in 'Reactor Core' section. (1) Click the down arrow button to decrease the demand to 90%. (2) Alternatively, users can type 90% in the textbox.	The rod demand is set to 90%.	Figure 17
4	Adjust the rod speed in 'Reactor Core' section. (1) Click the up-arrow button to increase the speed to 2%/min. (2) Alternatively, users can type 2% in the textbox.	The rod speed is set to 2%/min.	Figure 17
5	Resume the simulator (*speed up the simulation if needed*).	The simulator starts operating.	
6	Observe the following variables until they are stable (1000 seconds or before): (1) Rod position (%) (2) Reactor power (MW(th)) (3) T_{avg} (°C) (4) Pressurizer pressure (bar) (5) Pressurizer level (%) (6) Generator load (MW) (7) Steam generator pressure (bar) (8) Steam generator level (%)	All variables stable at: (1) ~90% (2) 75.2%, 1353MW(th) (3) ~296.8 °C (4) 155.0 bar (5) 49% (6) ~450 MW (7) 56.8 bar (8) 50.0%	Figure 18

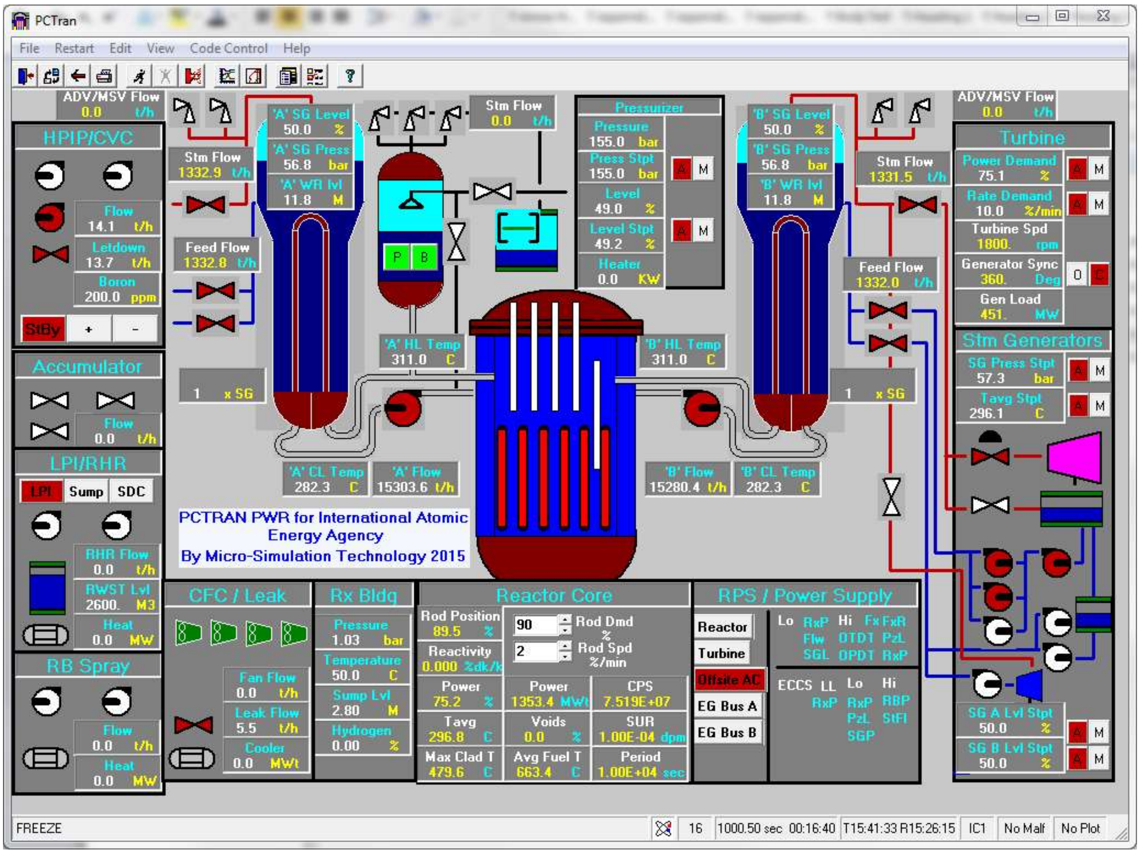

FIG. 18. Reactor state 1000 seconds after the rod demand and speed change (reactor leading mode).

2.3.2. Transient behaviour description

In reactor leading mode, the rod control system constantly compares the current rod position and the specified rod demand. When there is a difference between them, this system generates a signal which moves the control rods at the operator's specified rate to introduce positive or negative reactivity.

Furthermore, change in thermal power alters the heat transfer in the SGs, which changes the amount of steam generation in the secondary side of the SGs. This, in turn, increases or decreases the pressure of the SGs.

2.3.3. Transient plots description

Figure 19 shows the reactor thermal output and the turbine load change during the simulation. The turbine load closely follows the decrease in core thermal power. When the rod position reaches the value specified by the user, the turbine load stabilizes at about 240 seconds. Subsequently, the core thermal power slightly oscillates mainly due to the reactivity feedback of the fuel and coolant temperature. The reactor thermal output and the turbine load stabilize at about 75% at around 500 seconds

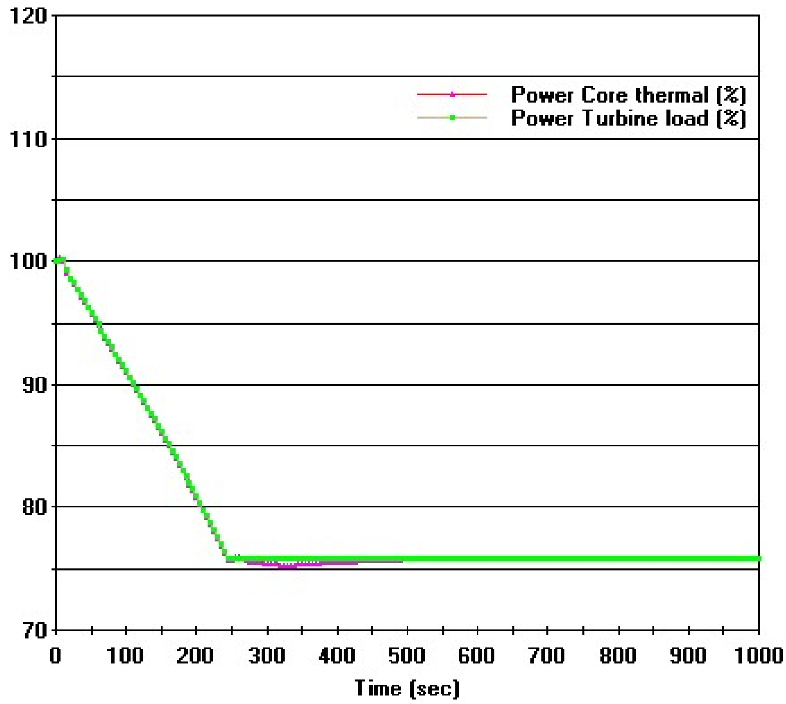

FIG. 19. Turbine load and reactor power output (%).

In Figure 20, as the steam flow rate is directly proportional to the turbine load, it decreases linearly until the rods are inserted as set by the user, and the reactor power reaches about 75%.

The feedwater flow rate, on the other hand, keeps decreasing for about 100 seconds because the SG water level control system has a delay in its processing. This system then adjusts the feedwater flow rate until it matches the steam flow rate at about 550 seconds

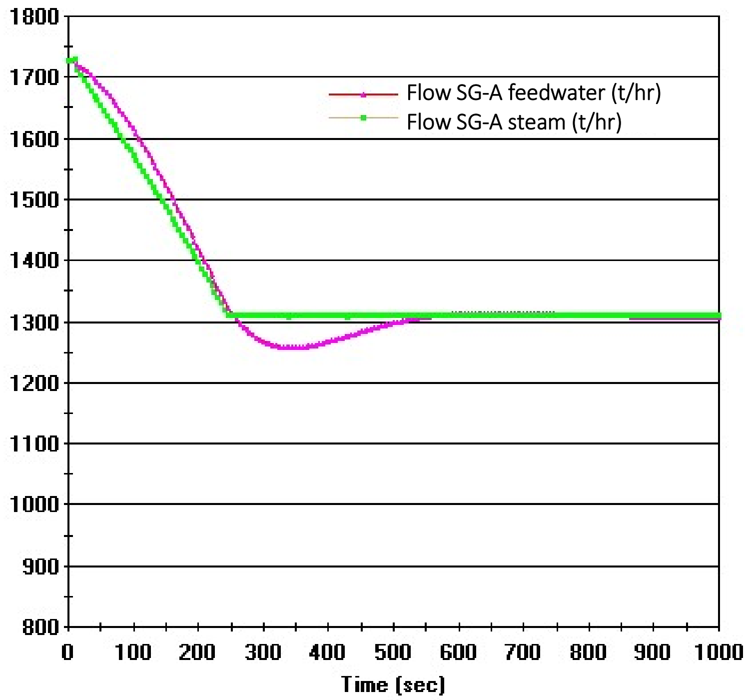

FIG. 20. Steam and feedwater flow rates (SG-A) (t/hr).

As shown in Figure 21, the SG pressure increases as the steam flow rate decreases with the turbine governor valve closing until about 250 seconds. Subsequently, the SG pressure decreases because of the reduced core power, and then slowly reaches a steady-state by 550 seconds.

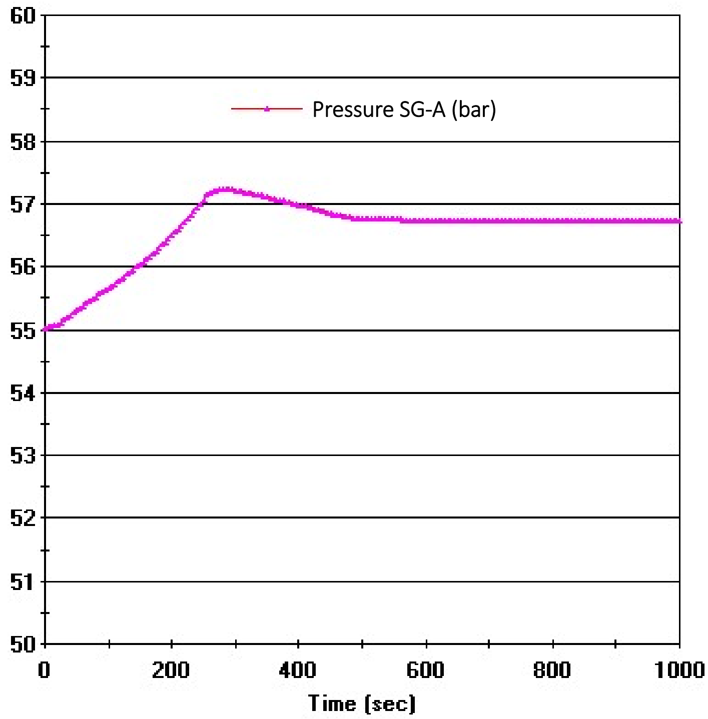

FIG. 21. SG-A pressure (bar).

The pressurizer water level is directly dependent upon the coolant temperature as the coolant expands or contracts with the temperature changes.

The pressurizer level change during the simulation is shown in Figure 22. As the reactor power is initially reduced, the primary coolant temperature and pressurizer water level are reduced until the insertion of rods is completed at about 250 seconds. The pressurizer water level then slightly increases due to the automatic actuation of the pressurizer heaters to reach the new setpoint and are about 49% by the end of the 1000 second simulation.

Figure 23 shows the reactor coolant pressure. Due to the decreasing primary coolant temperature, the reactor coolant system pressure drops until the reactor power stops decreasing at about 250 seconds.

Afterwards, by automatic actuation of the pressurizer heaters, the primary pressure slowly increases and returns to the 155.0 bar setpoint by 900 seconds.

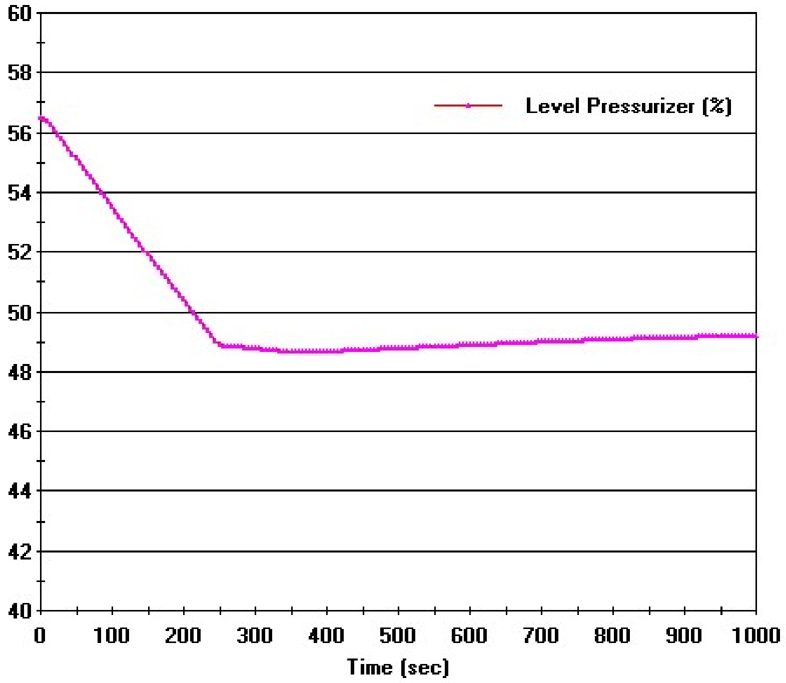

FIG. 22. Pressurizer level (%).

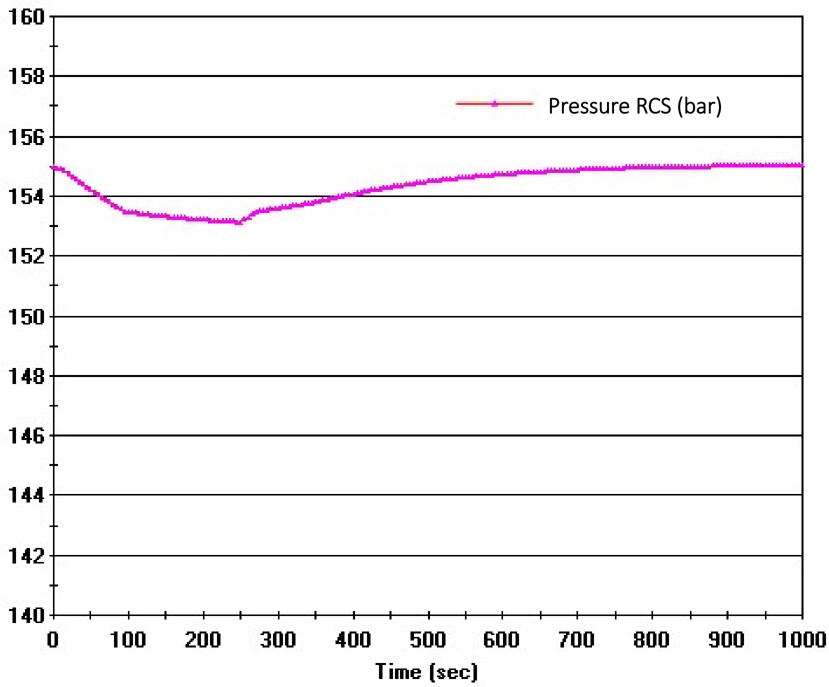

FIG. 23. Reactor coolant system pressure (bar).

2.4. NORMAL REACTOR TRIP

The Reactor Protection System (RPS) shuts down a PWR power plant when certain safety system settings, or setpoints, are reached or when commanded by the operator. Some of the crucial parameters, such as pressure inside the pressurizer, reactor coolant flow rate, and steam generator water level etc., are continuously compared to specified safe operation limits, and when any parameter exceeds its limit, the RPS automatically shuts down the reactor. This is called a ‘reactor trip’ or ‘scram’. With a reactor trip signal, all the control rods are inserted rapidly to absorb neutrons in the reactor and thus, to cease the nuclear fission chain reaction. In this exercise, the user will manually trip the reactor and observe how the nuclear power plant manages the decay heat from the reactor core after the trip.

2.4.1. Simulation description

To perform this operation in the simulator, Table 6 provides a step-by-step procedure for simulation with corresponding values of relevant variables. In Step 3, a reactor trip signal is manually generated. Figure 24, which corresponds to Step 5, shows the reactor and turbine trip because of the automatic initiation of turbine trip signal in case of reactor trip. In Step 6, the main feedwater is isolated due to the low primary system coolant temperature (Figure 25).

An extended simulation run of 4000 seconds was carried out to explore the behaviour of some variables during this time. In particular, in Step 7, the auxiliary feedwater system (AFW) is automatically actuated on low SG water level (Figure 26) to provide feedwater to the SGs.

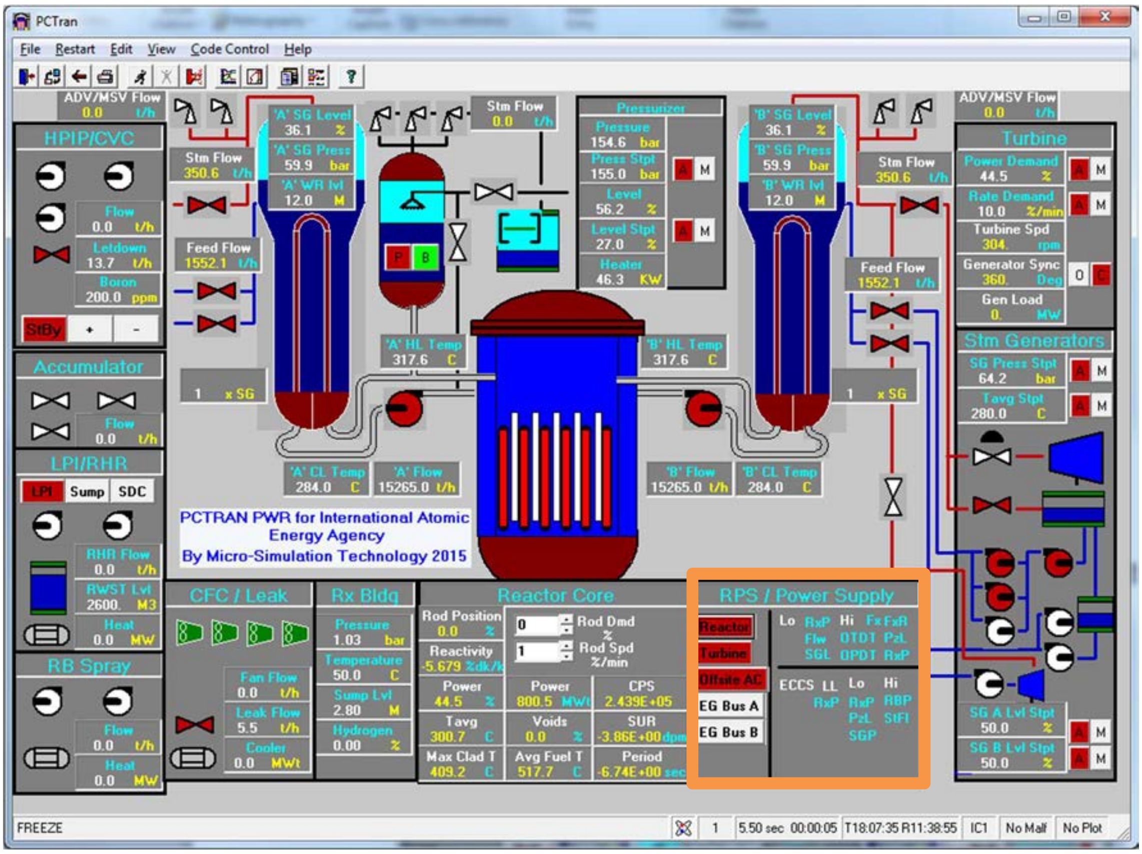

FIG. 24. Manual reactor trip followed by turbine trip.

TABLE 6. NORMAL REACTOR TRIP

STEP	PROCEDURE STEPS	EXPECTED RESPONSE	FIGURE
Setup	Load initial condition #1 100% POWER EOC, or #2 100% POWER MOC, or #3 100% POWER BOC. Initial condition #1 is used for this exercise.		
1	Run the simulator for about 5 seconds to achieve a steady-state condition.		
2	Freeze the simulation.	The simulator freezes.	
3	Manually trip the reactor by clicking the 'Reactor' button in the Reactor Protection System (RPS) section.	The colour of the 'Reactor' button turns from grey to red.	
4	Resume the simulation (*speed up the simulator if needed*).	The simulator resumes operation.	
5	Observe the reactor trip and the subsequent turbine trip.	All the control rods are inserted into the core. The turbine governor valve closes, and the turbine bypass valve opens to let the steam flow to the condenser. The colour of the 'Turbine' button in RPS section also changes from grey to red due to the turbine trip.	Figure 24
6	Observe the closure of the Main Feedwater Isolation Valves (FWIVs) on low T_{avg} (281°C) at about 45 seconds.	The FWIVs colour changes from red to white, indicating closed condition.	Figure 25
7	Observe the automatic actuation of the auxiliary feedwater system on low steam generator water level (17%) at about 1750 seconds.	Two motor-driven auxiliary feedwater pumps and one turbine-driven auxiliary feedwater pump start operating to remove the residual heat generated in the core.	Figure 26
8	Observe the following variables changing: (1) Rod position (%) (2) Reactor power (MWt) (3) T_{avg} (°C) (4) Pressurizer pressure (bar) (5) Pressurizer level (%) (6) Steam generator pressure (bar) (7) Steam generator level (%)	The values vary with the time passing since the decay heat is reduced over time.	

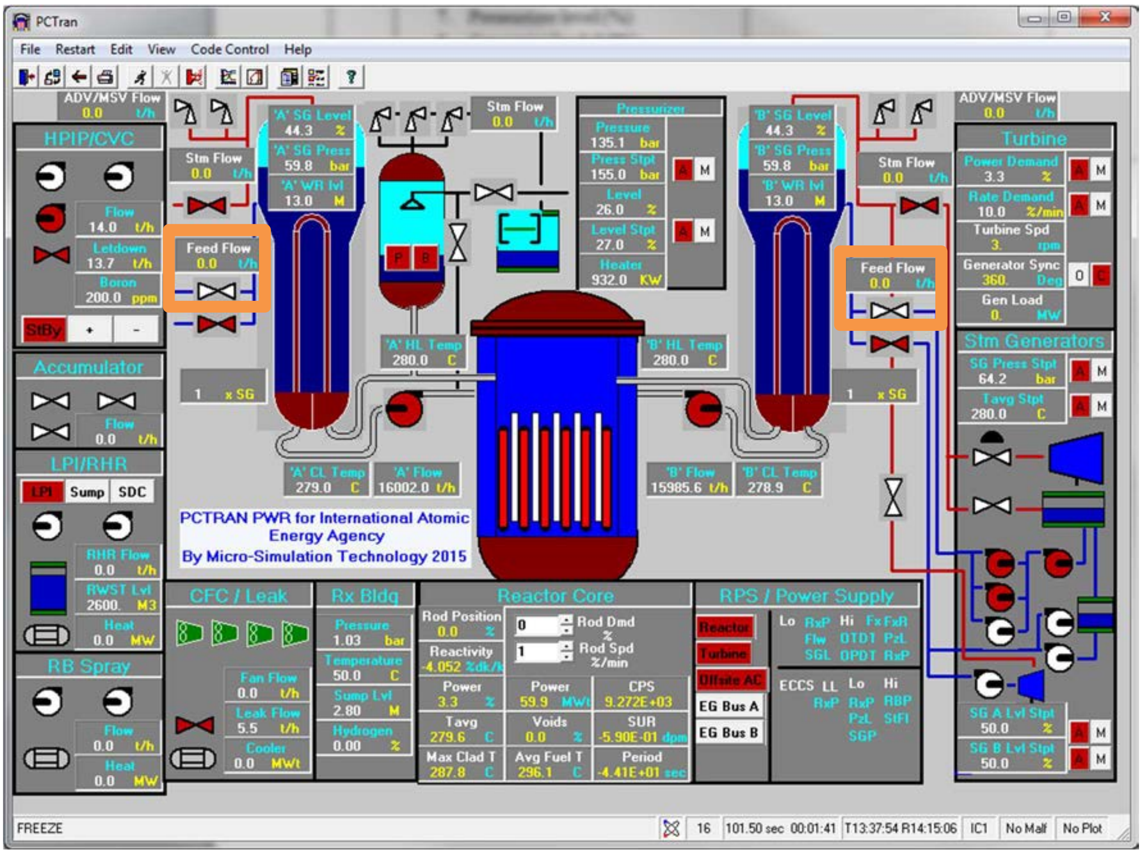

FIG. 25. Closure of the feedwater isolation valves.

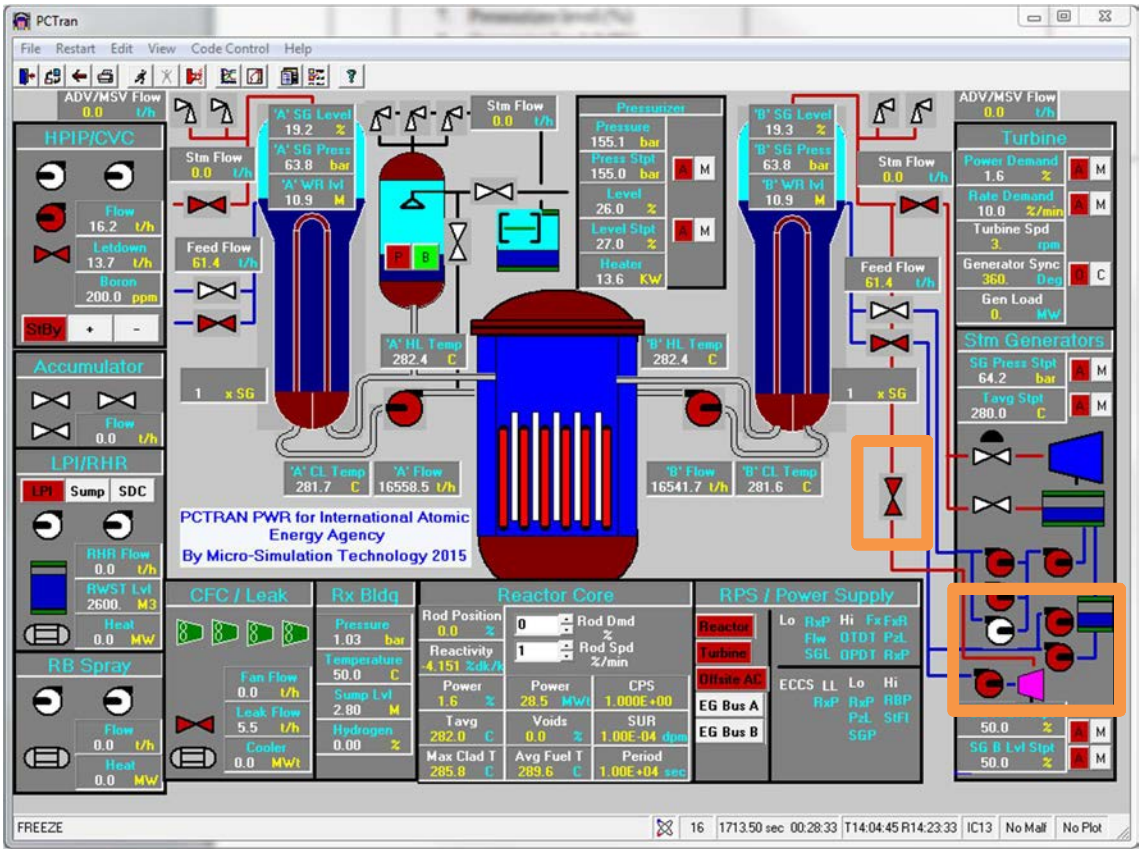

FIG. 26. Auxiliary feedwater system actuated.

2.4.2. Transient behaviour description

A reactor trip (or scram) is the result of operator action in the control room or automatic actuation of the Reactor Protection System (RPS). When the RPS is actuated, control rods are rapidly inserted into the core within a few seconds to absorb neutrons, thus stopping the fission chain reaction. Due to the quick control rod insertion, the reactor core power rapidly drops to less than 5% (see Figure 27). However, the core still generates residual heat due to the radioactive decay of the fission products that have accumulated during the reactor operation.

Accordingly, to remove the residual heat in the core after the turbine trip, the turbine bypass valve opens to let the steam flow directly to the condenser. When the SG water level is lower than a pre-set value (17% narrow range), the Auxiliary Feedwater System (AFW), consisting of one turbine-driven and two motor-driven auxiliary pumps, is automatically actuated to provide feedwater to the SGs and maintain a heat sink.

2.4.3. Transient plot description

Figure 27 shows the reactor thermal output and the turbine load during the simulation. When the reactor trips due to the insertion of control rods, the reactor power rapidly decreases; however, it does not immediately reach 0% due to the decay heat from fission fragments. On the other hand, the turbine load drops to 0% due to the turbine trip, which happens quickly after the reactor trip.

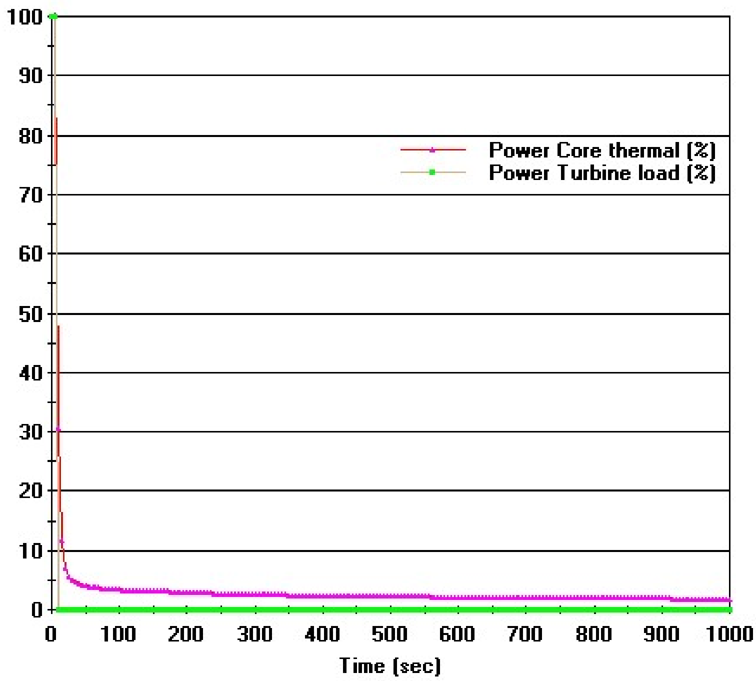

FIG. 27. Reactor power output and turbine load (%).

The sudden rod insertion introduces large negative reactivity which is enough to shut down the reactor within a few seconds. The initial negative reactivity peak illustrated in Figure 28 is due to the quick rod insertion. The sudden reduction of fuel and coolant temperatures then introduces positive reactivity due to the negative fuel and moderator temperature coefficients (MTC). However, the negative reactivity of the control rods then maintains the core sub-critical with a substantial margin of around -4% dk/k.

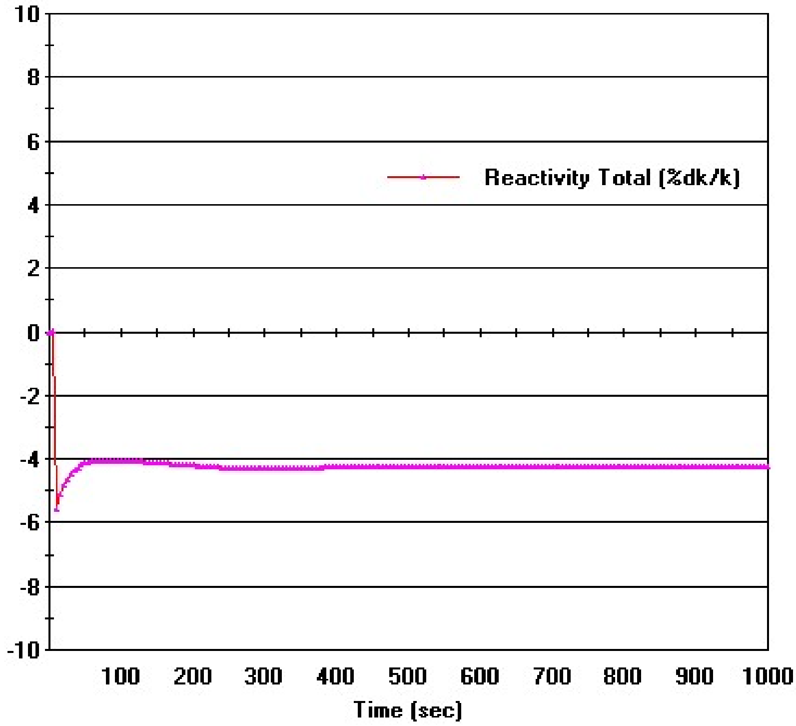

FIG. 28. Total reactivity (% dk/k).

Figure 29 shows the hot- and cold-leg temperatures of loop A, as well as the RCS average temperature. The hot-leg temperature rapidly decreases after the reactor trip for about the first 100 seconds, and then increases for about 150 seconds due to the decay heat and reduction in heat transfer to the secondary side. Finally, it follows the slow reduction of decay heat.

The cold-leg temperature initially increases due to closure of the governor valve of the main turbine, which implies no heat transfer to the secondary side. It then decreases after the turbine trip due to the rapid reduction of core power during the first 100 seconds. Similar to the behaviour of the hot-leg temperature after the first 100 seconds, the cold-leg temperature increases for about 150 seconds due to the decay heat and reduction in the heat transfer to the secondary side, and then it follows the slow reduction of decay heat.

As the reactor coolant system pressure directly depends upon the coolant temperature, the rapid decrease in average reactor coolant temperature after the reactor trip causes the reduction of RCS pressure during the first 100 seconds shown in Figure 30.

After this reduction, the RCS pressure increases to the setpoint of 155.0 bar by the automatic actuation of the pressurizer electric heater.

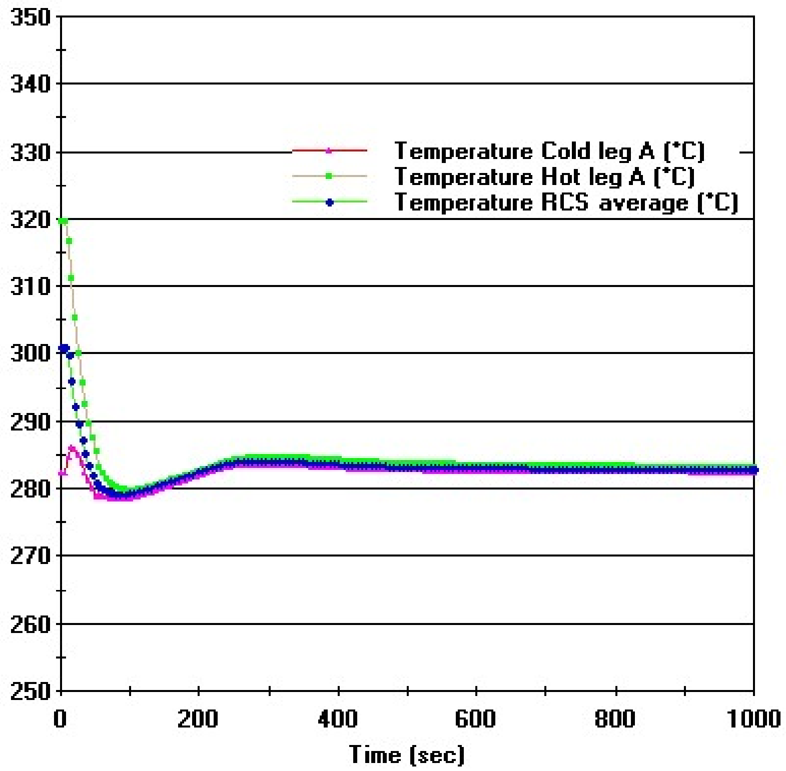

FIG. 29. Reactor coolant temperature (°C).

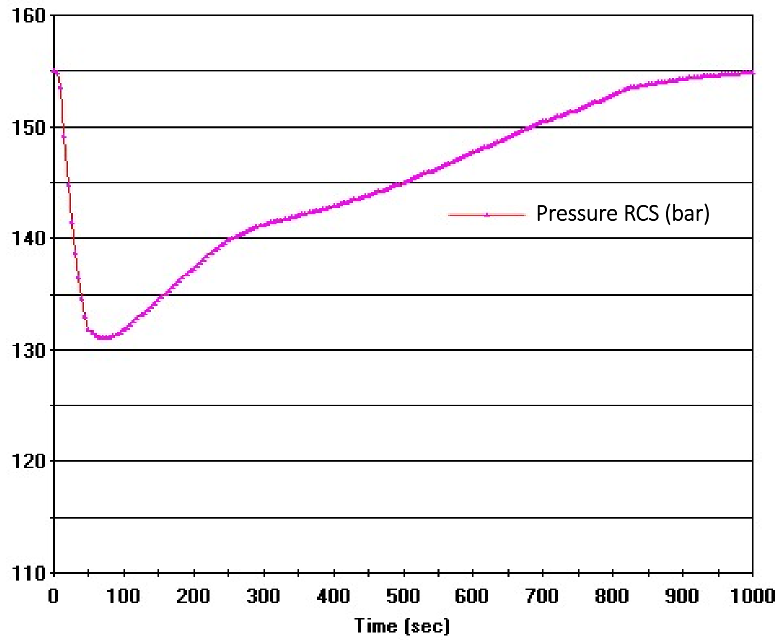

FIG. 30. Reactor coolant system pressure (bar).

Figures 31 and 32 show some results of an extended simulation run of 4000 seconds that was carried out to explore the behaviour of some variables during this time. Figure 31 shows the rapid decrease in steam and feedwater flow rates after the reactor and turbine trips. A surge in steam flow occurs at about 200 seconds due to decay heat, and the turbine- and motor-driven auxiliary feedwater pumps automatically operate on low SG level at about 1700 seconds the feedwater flow rate roughly matches the steam flow rate from about 3200 seconds.

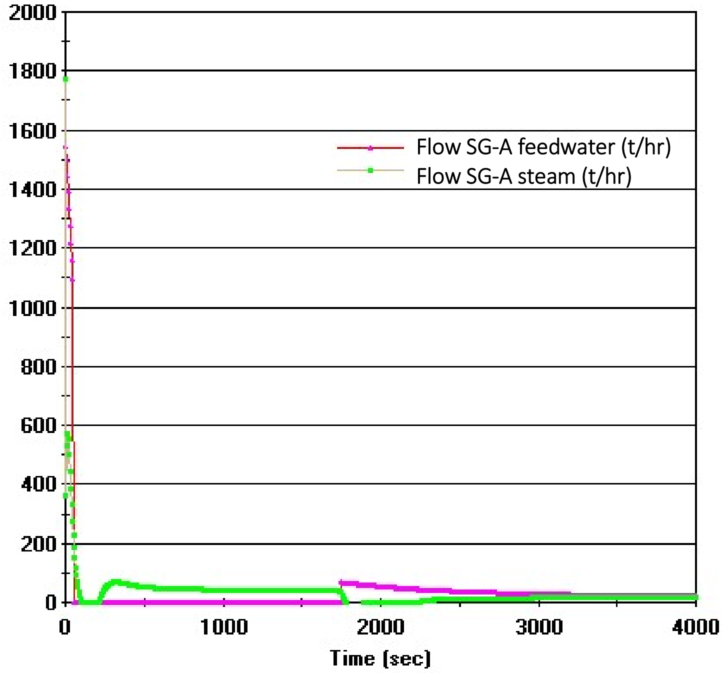

FIG. 31. Steam and feedwater flow rates (A-side) (t/hr).

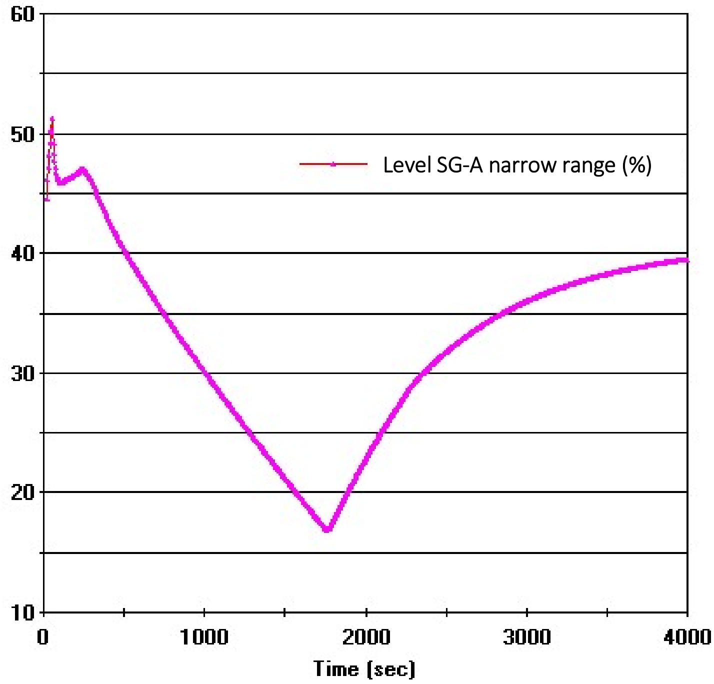

FIG. 32. SG-A water level (%).

Figure 32 presents the water level of a SG. After the reactor trips, the level quickly and sharply increases and shows a peak in the graph due to closure of the governor valve of the main turbine. At about 100 seconds, the slight increase is due to the expansion of the liquid on the secondary side. Eventually the water level drops and decreases linearly due to cooling, as there is less heat coming from the primary side and therefore the liquid contracts and water level goes down. The level linearly decreases due to lack of feedwater flow until it reaches the point that generates an actuation signal for the AFW system at about 1700 seconds as mentioned in the description of the previous figure, this system then provides feedwater to the SGs.

3. TRANSIENT AND ACCIDENT ANALYSES

3.1. PARTIAL LOSS OF REACTOR COOLANT FLOW – TRIP OF ONE RCP

Reactor coolant pumps (RCPs) provide the pressure head necessary for forced circulation of the reactor coolant in the primary system. Under normal operation condition, trip of one or more RCPs causes a rapid increase in fuel temperature due to the reduced coolant flow rate. Without prompt reactor trip, this may bring about departure from nucleate boiling (DNB) and serious fuel degradation. Thus, a reactor trip signal is generated when the flow rate reaches below a predefined setpoint. This transient scenario may be classified as an anticipated operational occurrence (AOO) [3].

3.1.1. Simulation description

To simulate this transient in the simulator, Table 7 provides a step-by-step procedure with corresponding values of relevant variables. In Step 3, RCP-B is manually tripped by the user (Figure 33). After the simulation is resumed in Step 5, the reactor power is reduced by about 30% without control rod insertion due to decrease in the moderator – reduced moderator flow in the core; and the effect of the negative moderator temperature coefficient – as the coolant's temperature increases. As the RCP-B motor speed is continuously reduced, the reactor protection system automatically generates a reactor trip in Step 6 (Figure 34), and a consequent turbine trip happens shortly afterwards (turbine trip always follows reactor trip). In Step 7, after the reactor and turbine trips, negative flow rate is observed in RCP-B indicating the flow direction in loop-B is reversed (Figure 35).

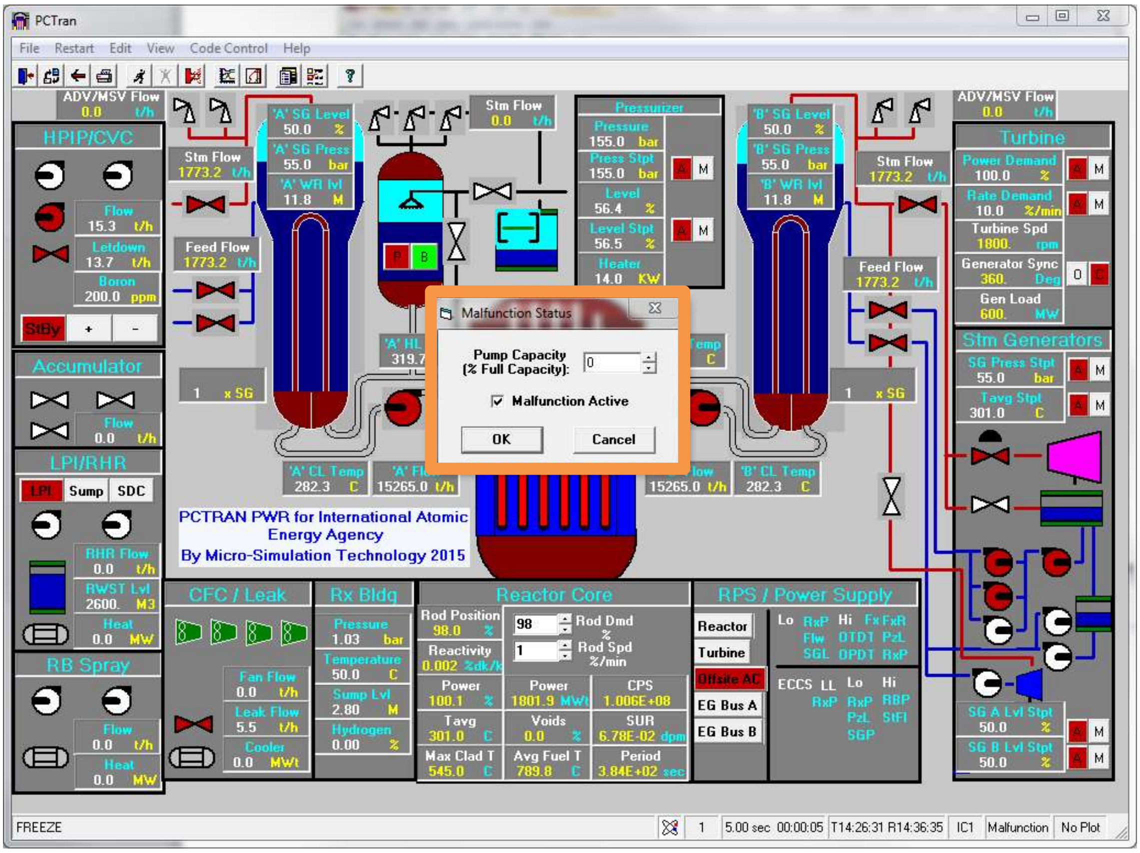

FIG. 33. Manual RCP trip (pump B).

TABLE 7. LOSS OF REACTOR COOLANT FLOW – RCPs FAILURE

STEP	PROCEDURE STEPS	EXPECTED RESPONSE	FIGURE
Setup	Load initial condition #1 100% POWER EOC, or #2 100% POWER MOC, or #3 100% POWER BOC. Initial condition #1 is used for this exercise.		
1	Run the simulator for about 5 seconds to achieve a steady-state condition.		
2	Freeze the simulator.	The simulator freezes.	
3	Trip one of the RCPs by clicking the RCP icons. It can be also tripped by right clicking the RCP icon(s) and insert 0 for the pump capacity and check 'malfunction active' box. RCP-B is tripped for this exercise.	The pump icon colour will change from red to white, indicating inoperable condition.	Figure 33
4	Resume the simulator (*speed up the simulator if needed*).	The simulator resumes operation.	
5	Check the following variables change: (1) T_{avg} (°C) (2) Reactor power (%) (3) Pressurizer water level (%) (4) Pressurizer pressure (bar)	 (1) Increases by 3–4 °C (2) Decreases by 30% (3) Increases by 10% (4) Increases by 3 bar	
6	Observe the reactor trip followed by turbine trip at about 15 seconds due to the low flow rate of primary coolant.	All the control rods are inserted into the core, the reactor trips followed by the turbine trip. The turbine governor valve closes, and the turbine bypass valve opens to let the steam flow directly to the condenser. The colour of the 'Reactor' and 'Turbine' buttons in RPS section changes from grey to red.	Figure 34
7	Check the coolant flow rate in both A- and B-loops.	Negative flow rate in loop-B is observed at about 40 seconds indicating backflow.	Figure 35

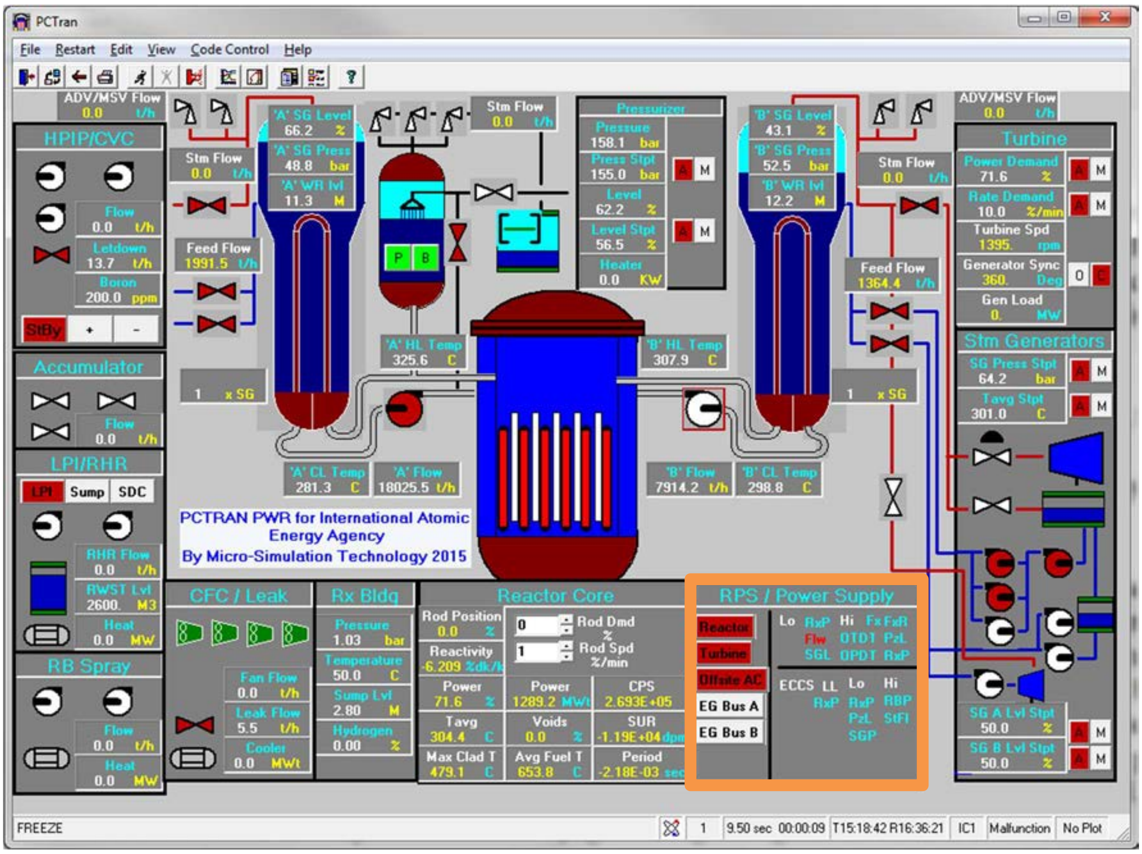

FIG. 34. Reactor trip due to the low flow rate.

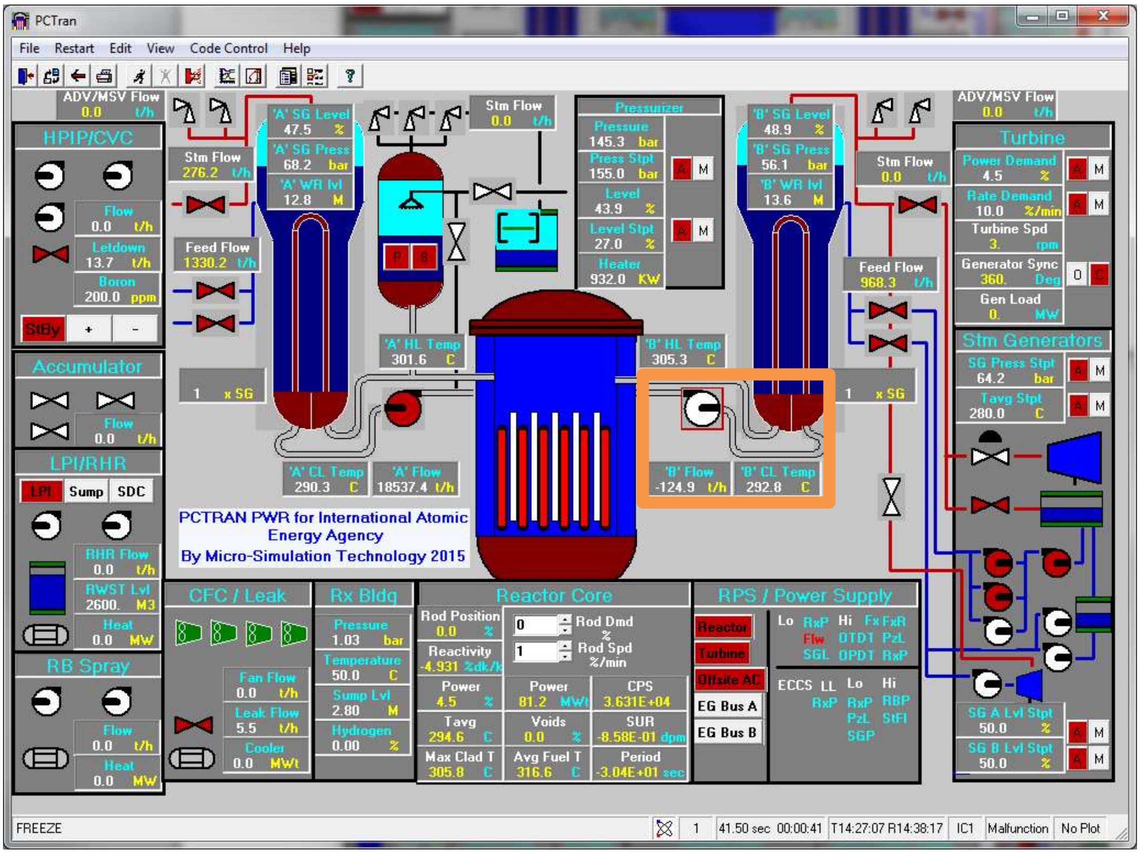

FIG. 35. Back flow observed in loop-B.

3.1.2. Transient behaviour description

After the RCP-B is manually tripped, T_{avg} rises due to reduction in forced circulation of coolant in loop-B – as there is less coolant flow to remove the heat from the core. The reactor power also decreases by about 30% during approximately the first 10 seconds after the RCP trip without any movement of control rods. This reduction of power is due to a combination of the reduced moderation caused by the reduced coolant flow in the core, and the negative reactivity added by the negative moderator temperature coefficient as the T_{avg} increases.

In about 10 seconds, the reactor trip signal is generated due to low coolant flow rate. This can be observed in the RPS/Power Supply block on the main system mimic. After the reactor trip, the turbine also trips and eventually natural circulation is established in loop-B that helps to remove the residual heat in the core. At about 50 seconds, the flow rate in loop-B becomes negative because the coolant flow has been reversed, and the loop-B cold-leg temperature is greater than the hot-leg temperature. The flow in the intact loop becomes elevated and the flow from the loop-A pump is distributed at the inlet of the RPV; the majority of flow goes through the core and a minor part contributes to the backward flow through loop-B. Heat removal mainly occurs in the intact loop and heat transfer via the second loop is low due to the newly established conditions.

3.1.3. Transient plot description

Figure 36 shows the reactor coolant flow rate in loop-A and loop-B. After the RCP-B is tripped, the RCP-A motor speed increases due to pressure head loss in loop-B. As only one RCP is in operation, the coolant in loop-B flows backwards as explained in section 3.1.2. This can be observed by the negative coolant flow rate in loop-B.

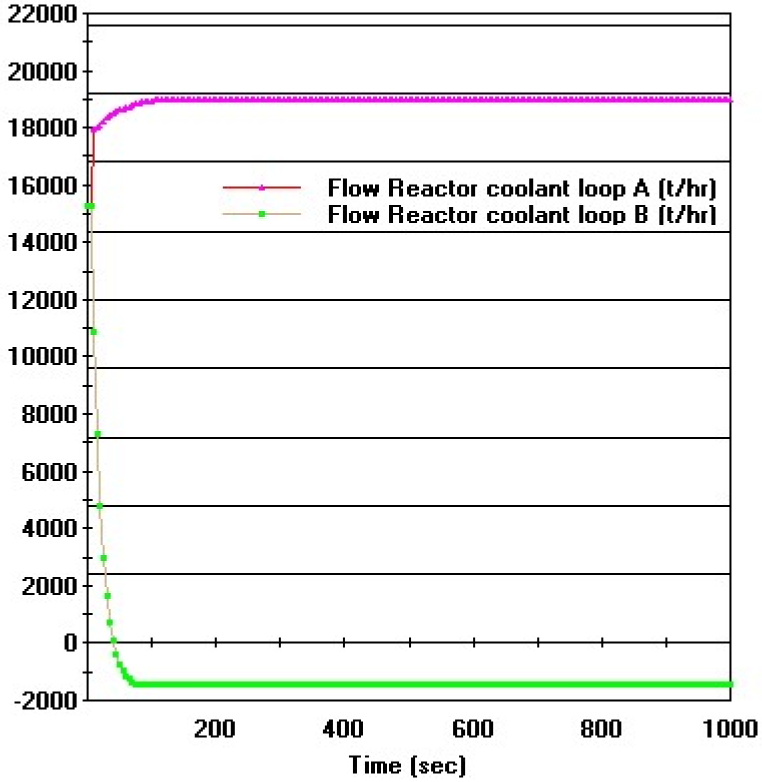

FIG. 36. Reactor coolant flow rate in loop-A and B (t/hr).

Figure 37 shows the cold-leg and hot-leg coolant temperatures in loop-B, in which the RCP was tripped. The initial sudden decrease in hot-leg coolant temperature is due to the reduced core power, and the increase in cold-leg temperature is attributed to the reduced coolant flow rate in this leg.

At about 10 seconds, the loop-B cold-leg coolant temperature rapidly falls as a result of the reactor trip. At about 30 seconds, the hot-leg temperature slightly increases due to the decay heat, and at about 40 seconds it rapidly falls as backflow in this loop begins to happen. At about 60 seconds, the cold-leg coolant temperature becomes greater than the hot-leg coolant temperature due to the backflow in this loop.

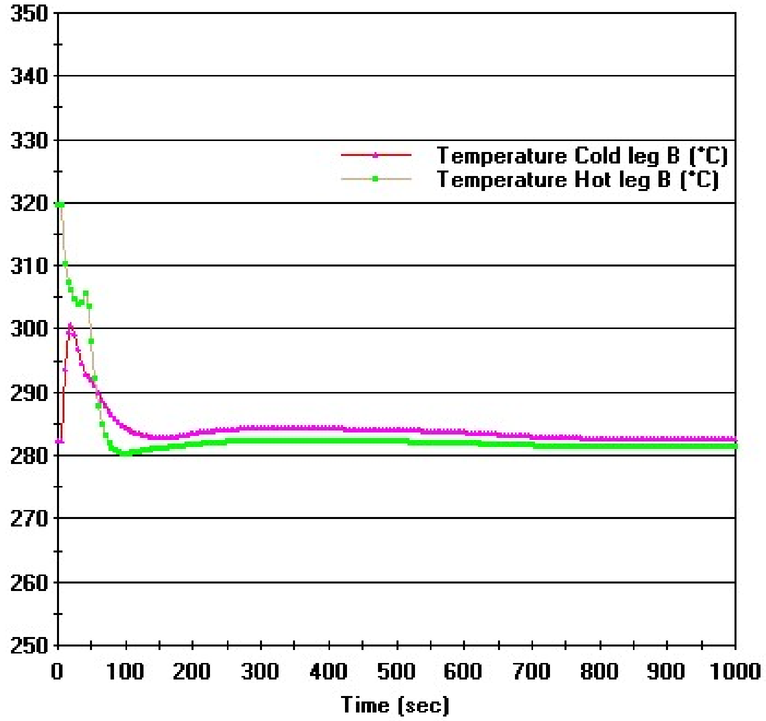

FIG. 37. Loop-B cold leg and hot leg coolant temperatures (°C).

Figure 38 shows the heat removal rate of each SG. As soon as the RCP-B is tripped, the SG-B heat removal rate rapidly decreases due to the loss of forced circulation. Conversely, the SG-A heat removal rate sharply increases with the increased loop-A coolant flow rate.

After the reactor trip, the heat removal rate in SG-A also falls rapidly due to the substantial reduction of reactor power. However, the SG-A heat removal rate is still greater than the SG-B heat removal rate due to the forced circulation.

The SG water level is different in each SG, as presented in Figure 39, since in this exercise only one RCP is tripped. In loop-B, the loop with the RCP that was tripped, the level initially decreases due to the cooldown resulting from the reactor trip. Subsequently, the level increases and reaches a steady state because the SG-B does not produce significant steam because of the loss of forced circulation in this loop.

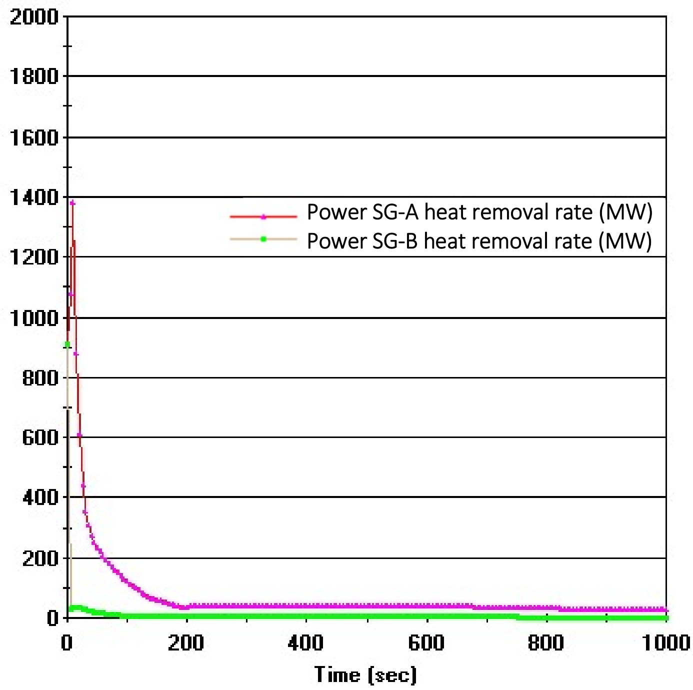

FIG. 38. Heat removal rate of SGs (MW).

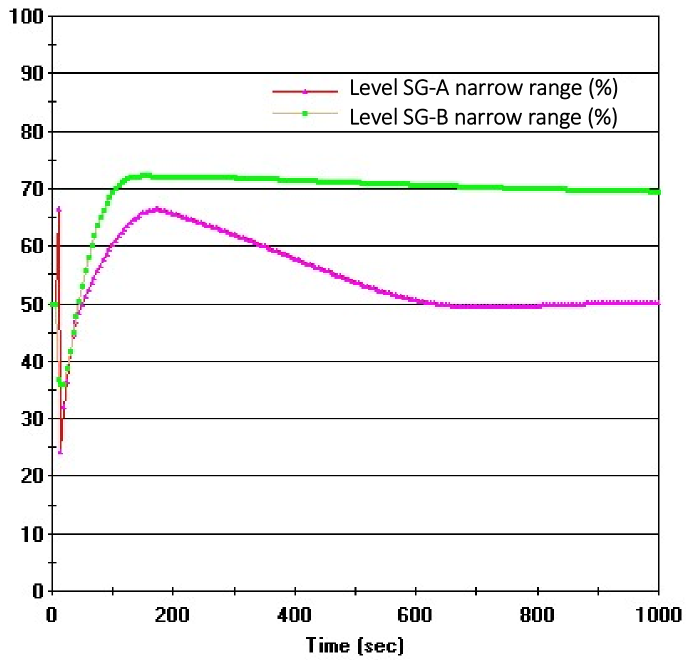

FIG. 39. SG water level (%).

For the loop-A, the SG water level initially increases due to the reduction of core power, and then sharply decreases due to the reactor trip and consequent reduction of feedwater flow. After increasing for approximately 200 seconds, the SG water level control system modifies the speed of the feedwater pumps so that the SG-A water level returns to the setpoint of 50%.

3.2. LOSS OF NORMAL FEEDWATER FLOW

The main feedwater pumps provide the necessary flow to the secondary side. The proper functioning of feedwater pumps is necessary to transfer the heat from the primary side to the secondary side. When loss of normal feedwater flow happens, the reactor trips automatically due to the low water level in the SGs. Subsequently, the Auxiliary Feedwater System (AFW) is automatically actuated due to low water level in the SGs. Loss of normal feedwater flow is a postulated initiating event for design basis accidents (DBAs) [3].

3.2.1. Simulation description

In this exercise, the user will manually trip the Main Feedwater Pumps (MFWPs) and observe how the primary and the secondary systems respond. Table 8 provides a step-by-step procedure for simulation with corresponding values of relevant variables. In Step 3, all three MFWPs are manually turned off. In Step 5, upon resuming the simulator, the condensate pump is automatically tripped due to the trip of the MFWPs (Figure 40). In Step 6, rapid decrease in SG water level is observed because of the loss of feedwater flow. In Step 7, an automatic reactor trip signal is generated due to low water level in the SGs (Figure 41). In Step 8, the AFW is automatically actuated due to low SG level (Figure 41). After the actuation of AFW, the SG water level increases slowly (Figure 42).

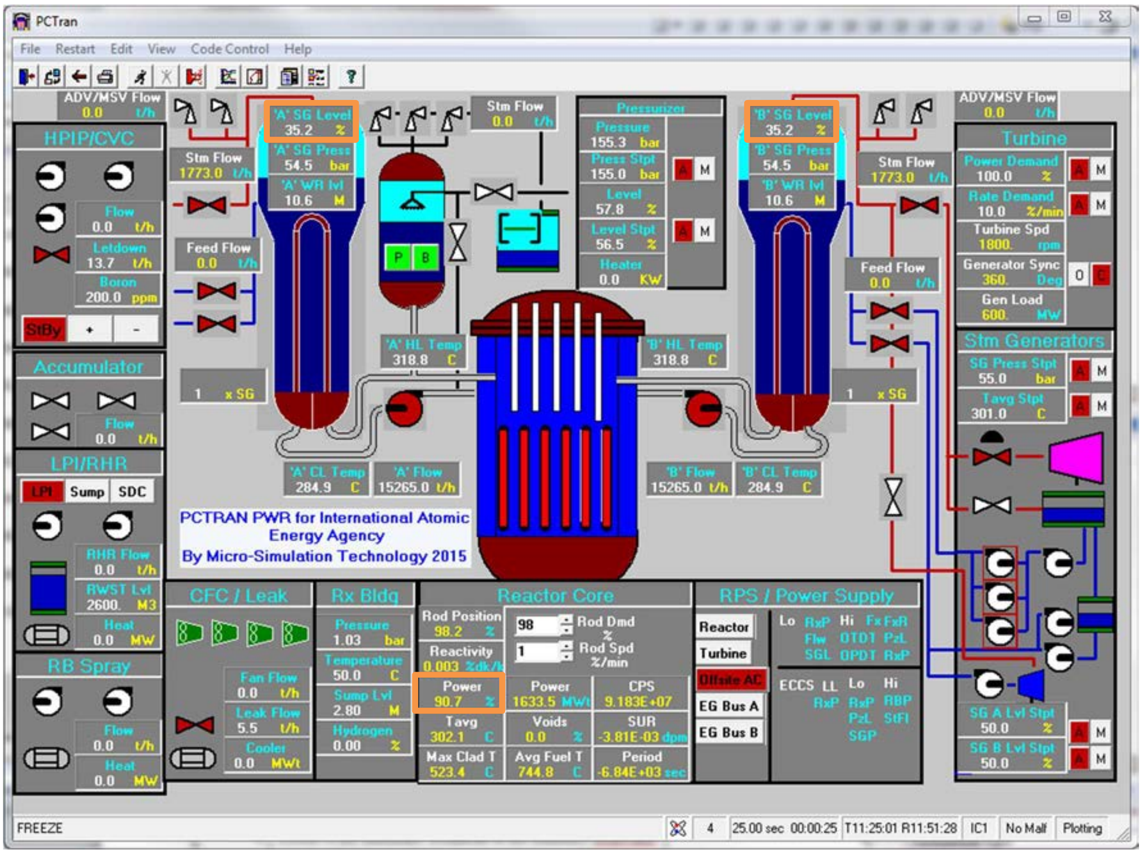

FIG. 40. Reactor power drops to about 90% and the SG water level decreases.

TABLE 8. LOSS OF NORMAL FEEDWATER FLOW

STEP	PROCEDURE STEPS	EXPECTED RESPONSE	FIGURE
Setup	Load initial condition #1 100% POWER EOC, or #2 100% POWER MOC, or #3 100% POWER BOC. Initial condition #1 is used for this exercise.		
1	Run the simulator for about 5 seconds to achieve a steady-state condition.		
2	Freeze the simulator.	The simulator freezes.	
3	Manually trip all three main feed water pumps by right click the MFWP icons, located in the 'Steam Generators' block of the main system mimic, and insert 0 for the pump capacity and check the 'malfunction active' box.	The pump icon colour changes from red to white, indicating inoperable condition.	
4	Resume the simulator (*speed up the simulator if needed*).	The simulator resumes operation.	
5	Observe the automatic condensate pump trip immediately after the MFWPs trip.	The condensate pump icon colour changes from red to white, indicating inoperable condition.	Figure 40
6	Check that there is no feedwater flow and how the following variables change before the reactor trip: (1) Steam flow rate (t/hr) (2) Steam generator level (%) (3) Reactor Power (%, MWth)	(1) ~1773 t/hr (2) Decreases (3) Drops to ~90%	Figure 40
7	Observe the reactor followed by turbine trip at about 50 seconds due to low steam generator level. The trip reason can be observed in the RPS/Power Supply block of the main system mimic.	All the control rods are inserted into the core. The turbine governor valve closes, and the turbine bypass valve opens to allow steam flow. The colour of the 'Turbine' button in RPS section also changes from grey to red due to the turbine trip.	Figure 41
8	Observe the automatic actuation of the auxiliary feedwater system when the reactor trips.	Two motor-driven auxiliary feedwater pumps and one turbine-driven auxiliary feedwater pump start operating to remove residual heat from the SGs.	Figure 41
9	Observe the steam generator water level is restored by the AFW system.		Figure 42

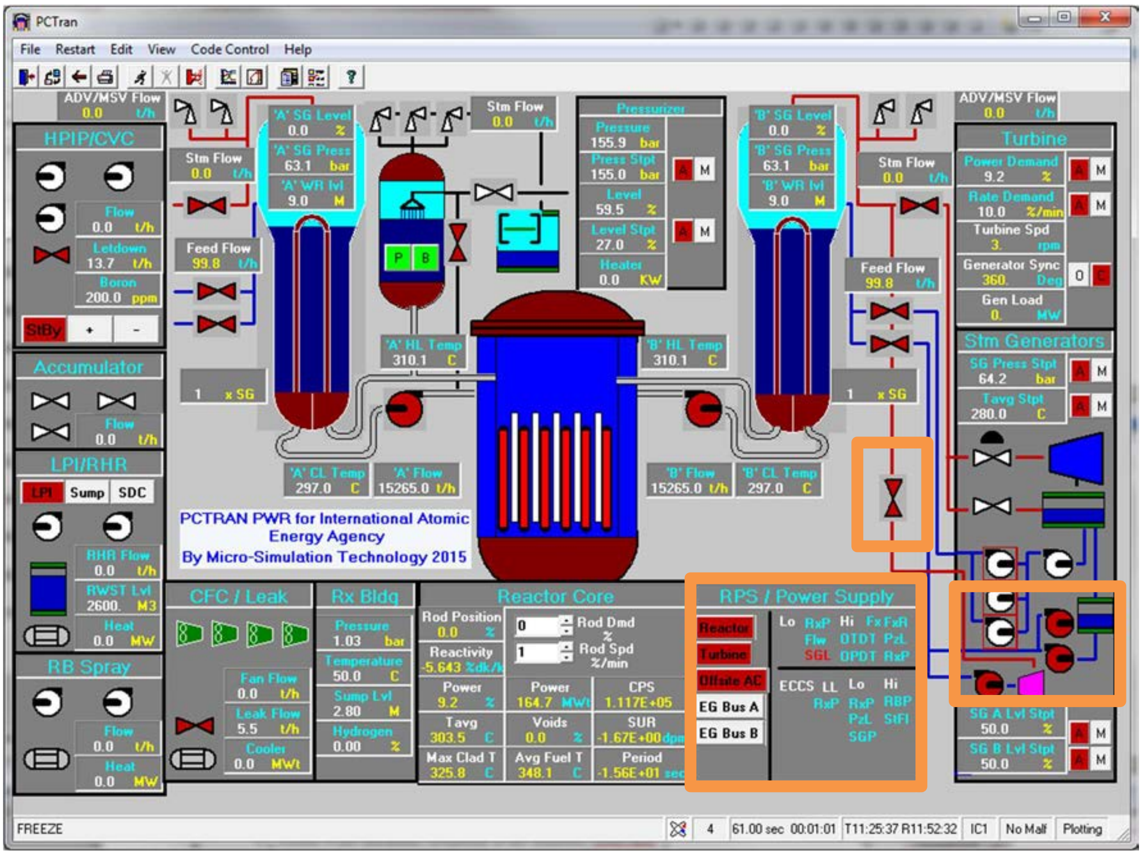

FIG. 41. The reactor trip and the actuation of AFW system.

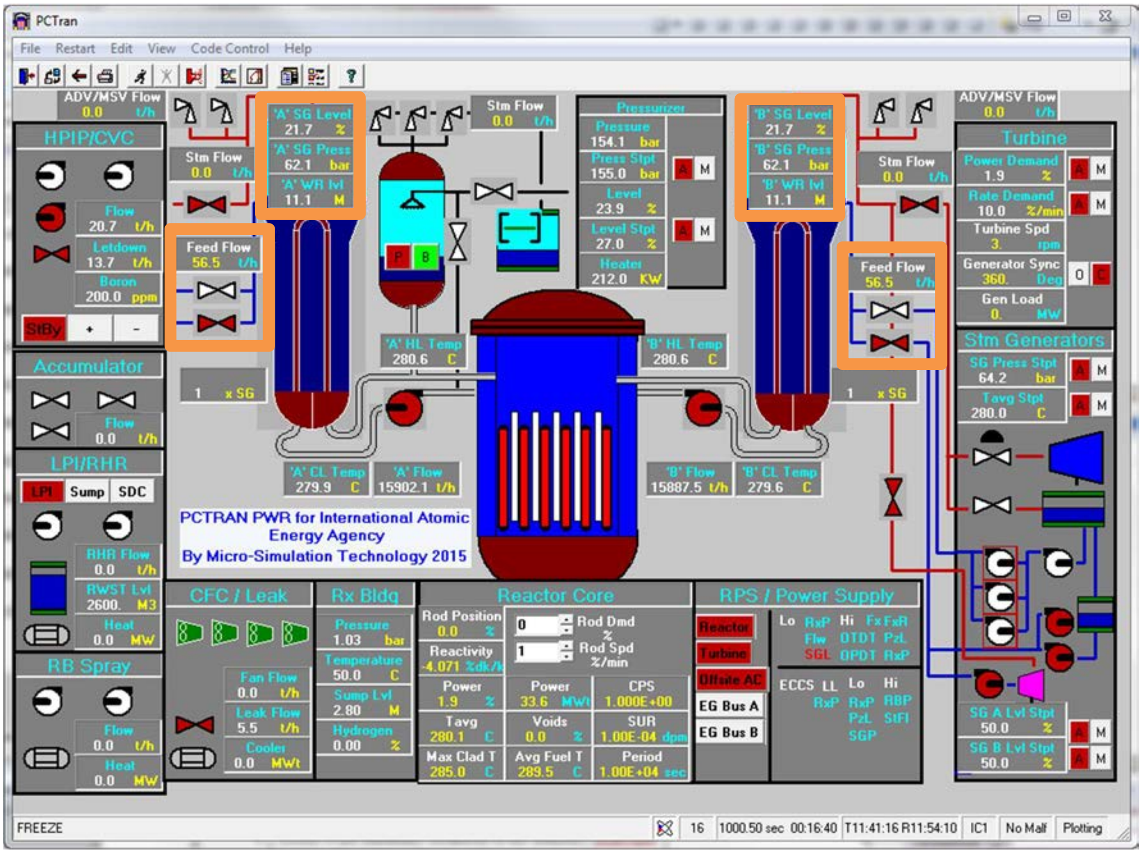

FIG. 42. 1000 seconds after the MFWPs trip.

3.2.2. Transient behaviour description

Under normal operation condition, steam from the turbines is condensed, and the MFWPs send the feedwater back into the SGs. When MFWPs are manually tripped, the water level of the steam generator decreases and therefore, the SGs lose some of their heat removal capacity. As a result of the imbalance between the heat generated in the core and the heat removal capacity of the SGs, the reactor coolant temperature increases. Thus, the reactor coolant volume expands, and the neutrons in the reactor core are less moderated due to the lower density of the coolant. Consequently, the reactor power decreases during about the first 15 seconds without control rods movement.

Loss of feedwater reduces the SG water level to the setpoint for reactor trip due to low SG level. After the reactor trip followed by turbine trip, Auxiliary Feedwater System (AFW), consisting of two motor-driven pumps and one turbine-driven pump, is automatically actuated. After the actuation of AFW, feedwater is provided to the SGs, thus supporting residual heat removal by restoring the heat sink.

3.2.3. Transient plot description

The feedwater and steam flow rates during 1000 seconds of the simulation is shown in Figure 43. After the MFWPs are manually tripped, the feedwater flow rate drops to zero. However, the steam flow rate remains the same due to boiling of the water inventory within the SGs until the reactor and turbine trips at about 60 seconds.

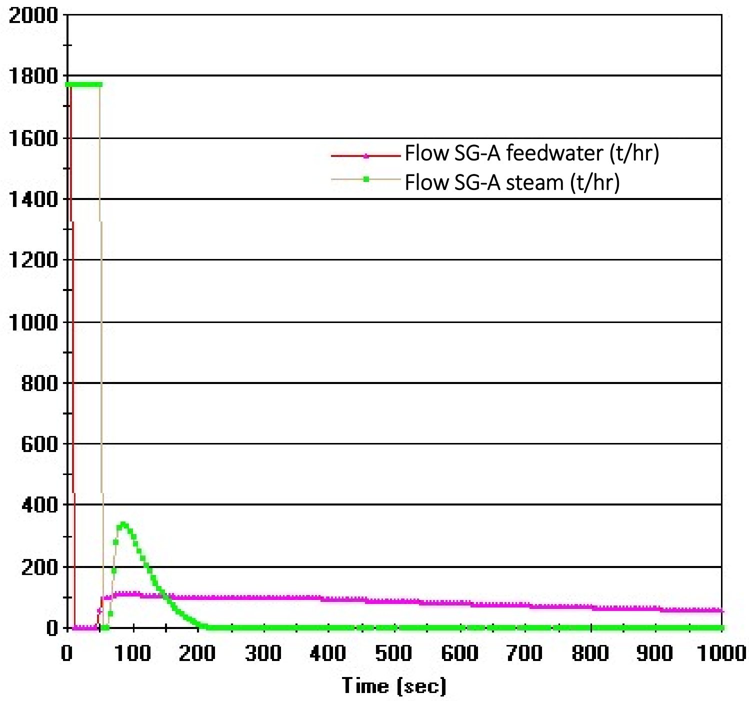

FIG. 43. Feedwater and steam flow rate (t/hr).

The auxiliary feedwater system also automatically actuates on low SG level at about 50 seconds, and sends the feedwater back to the SGs to restore their water level. For a brief moment, the steam flow rapidly increases due to the water provided by the AFW. Following this, a significant reduction in steam flow is observed. This is due to the relatively low amount of decay heat generation compared to the heat generated by the core during power operation. The steam production is reduced due to the reactor trip. It can be observed, in long run, that the steam flow increases as soon as enough steam is reaccumulated in the steam generators.

The SG level, shown in Figure 44, changes according to the difference between the steam flow rate and the feedwater flow rate (Figure 43).

At first it decreases because the MFWPs stop sending the feedwater, while steam is still being generated from the water inventory. After the AFW system starts operating, the SG water level starts to increase.

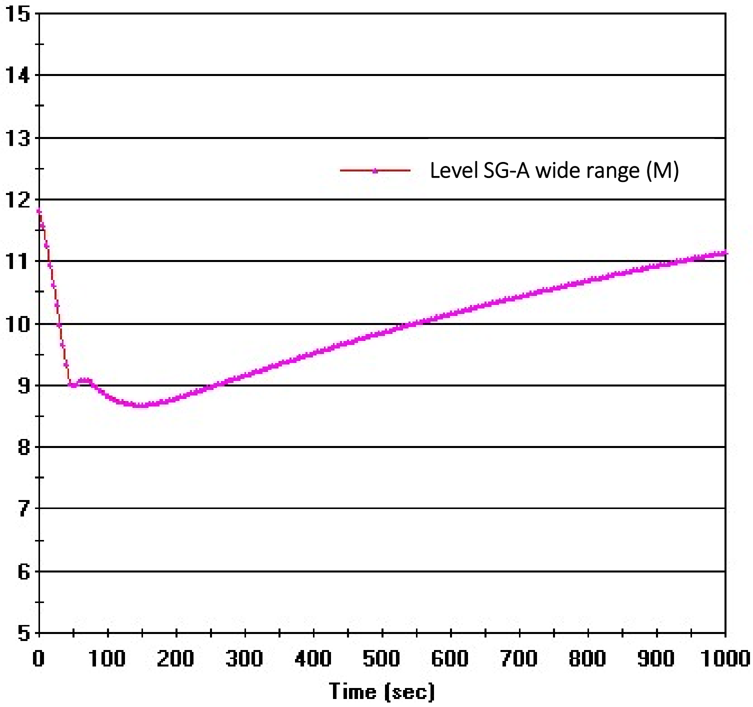

FIG. 44. SG-A water level – wide range (m).

After the MFWPs trip, the power decreases to about 90% without control rods movement until the reactor trip. This phenomenon is attributed to the negative moderator temperature coefficient (MTC) of the reactor coolant illustrated in Figure 45 and as also explained in section 3.2.2.

After the main feedwater flow stops, the SG water level drops and its heat removal capacity decreases. This, in turn, causes an increase in the primary coolant temperature due to the reduced heat transfer in the SGs. As the primary coolant temperature rises, negative reactivity is introduced due to the negative MTC, causing a reduction of reactor power (Figure 46). As a result of this power reduction, the fuel's temperature decreases, thus introducing positive reactivity due to Doppler broadening. The net effect of these positive and negative reactivities is a net negative reactivity during about the first 15 seconds. Subsequently, the positive and negative reactivities are of similar absolute magnitudes, so the total reactivity remains at about 0% dk/k until the reactor trips.

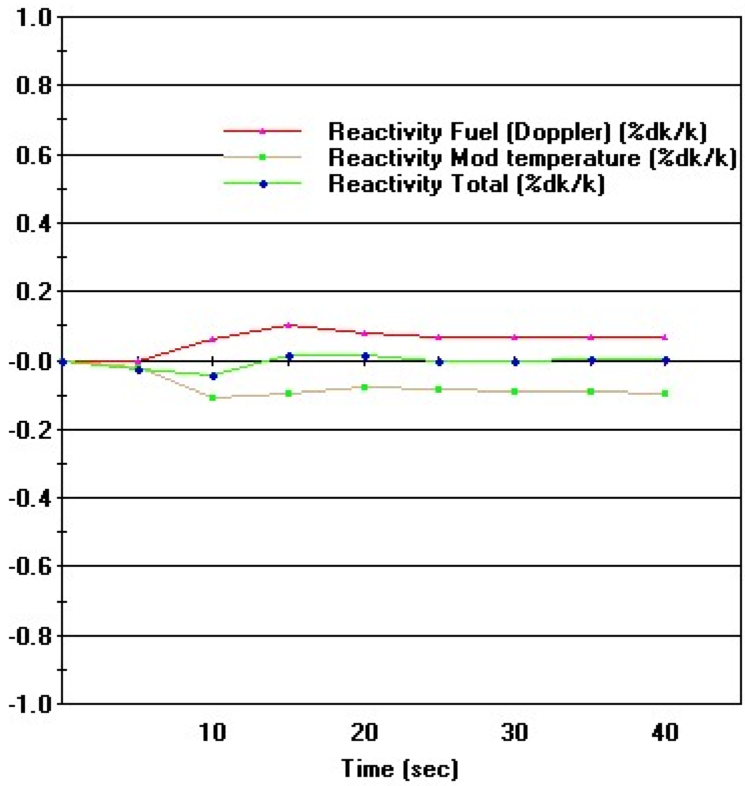

FIG. 45. Fuel temperature reactivity, moderator temperature reactivity, and total reactivity during the first 50 seconds of the simulation (% dk/k).

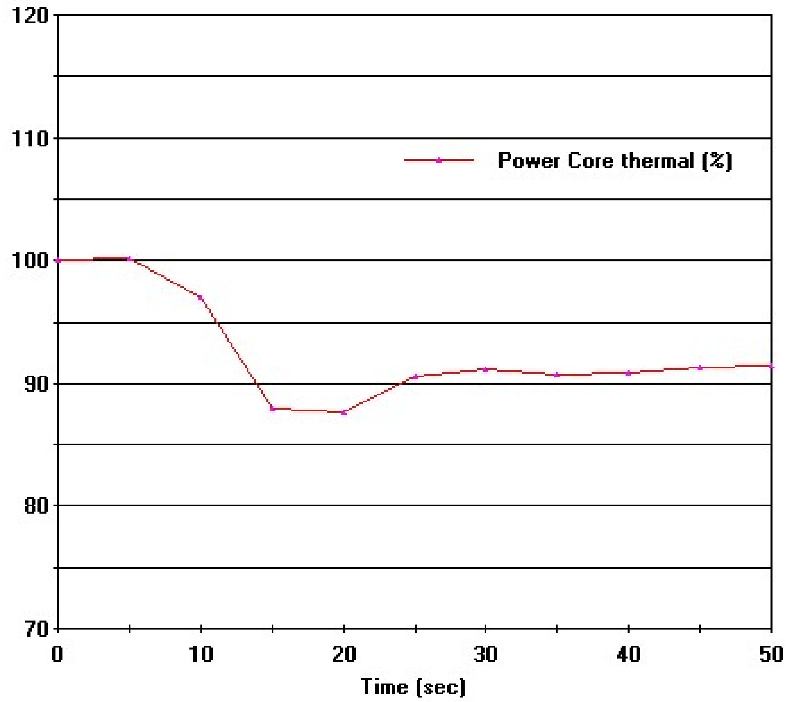

FIG. 46. Reactor power during the first 50 seconds of the simulation (%).

3.3.STEAM GENERATOR TUBE RUPTURE

There are usually around 3000 to 16 000 tubes in a pressurized water reactor's steam generator. The water inside the tubes, which is the primary side water, heats non-radioactive secondary side water on the outside, and produces steam. This steam is then used to run the turbine and generate electricity. The steam then condenses in the condenser and is returned back to the steam generators with the main feedwater pumps.

Steam generator tubes are subjected to a number of degradation processes that can lead to tube cracking, thinning, and possible leakage or rupture. This rupture can then lead to leakage of radionuclides from primary to secondary side. Therefore, it is important to keep the tubes intact. In this exercise, a double-ended steam generator tube rupture is simulated in the SG-A to understand the phenomenon properly. This is a postulated initiating event for DBAs [3].

3.3.1. Simulation description

To simulate the transient, Table 9 provides a step-by-step procedure for simulation with corresponding values of relevant variables. In Step 3, a double-ended tube rupture is initiated with the pre-defined malfunction #10. Upon resuming the simulator in Step 4, a graphical rupture icon appears in the SG-A showing the reactor coolant loss rate (Figure 47). With the reactor coolant moving to the SG secondary side through the rupture, the variables listed in Step 5 change accordingly (Figure 48). As a result, in Step 6, the reactor coolant system pressure is continuously reduced and reaches the reactor trip setpoint, and the ECCS actuation setpoint (Figure 49).

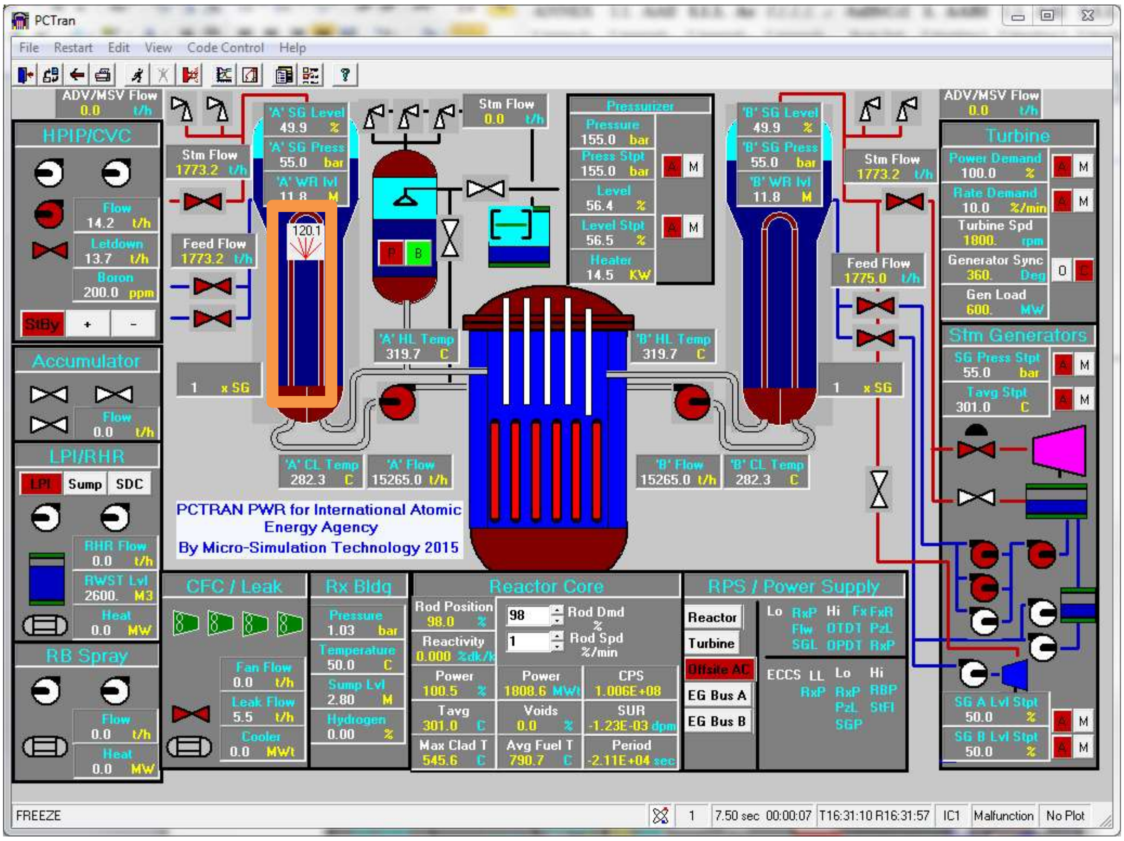

FIG. 47. SG-A tube rupture.

TABLE 9. STEAM GENERATOR TUBE RUPTURE

STEP	PROCEDURE STEPS	EXPECTED RESPONSE	FIGURE
Setup	Load initial condition #1 100% POWER EOC or #2 100% POWER MOC or #3 100% POWER BOC. Initial condition #1 is used for the exercise.		
1	Run the simulator for about 5 seconds to achieve a steady-state condition.		
2	Freeze the simulator.	The simulator freezes.	
3	Initiate a steam generator tube rupture event by selecting ‘Malfunction #10’. To simulate a double-ended tube rupture, put 100 for failure fraction and check the ‘active’ checkbox.		
4	Resume the simulator (*speed up the simulator if needed*).	Rupture icon will appear in the SG-A with the reactor coolant loss rate (t/hr).	Figure 47
5	Observe the following variables changing: (1) Pressurizer pressure (bar) (2) Pressurizer level (%) (3) Steam generator level (%) (4) Steam generator tube leak rate (t/hr)	 (1) Decreases (2) Decreases (3) Increases in SG-A (4) Decreases	Figure 48
6	Observe the reactor trip at about 350 seconds due to low pressurizer pressure (132.2 bar). The low pressurizer pressure (129.7 bar) actuates EECS at about 360 seconds to compensate for the reactor coolant loss. This actuation then causes the closure of the main FWIVs and the automatic operation of the AFW system.	All the control rods are inserted into the core. The turbine governor valve closes, and the turbine bypass valve opens to let the steam flow.	Figure 49

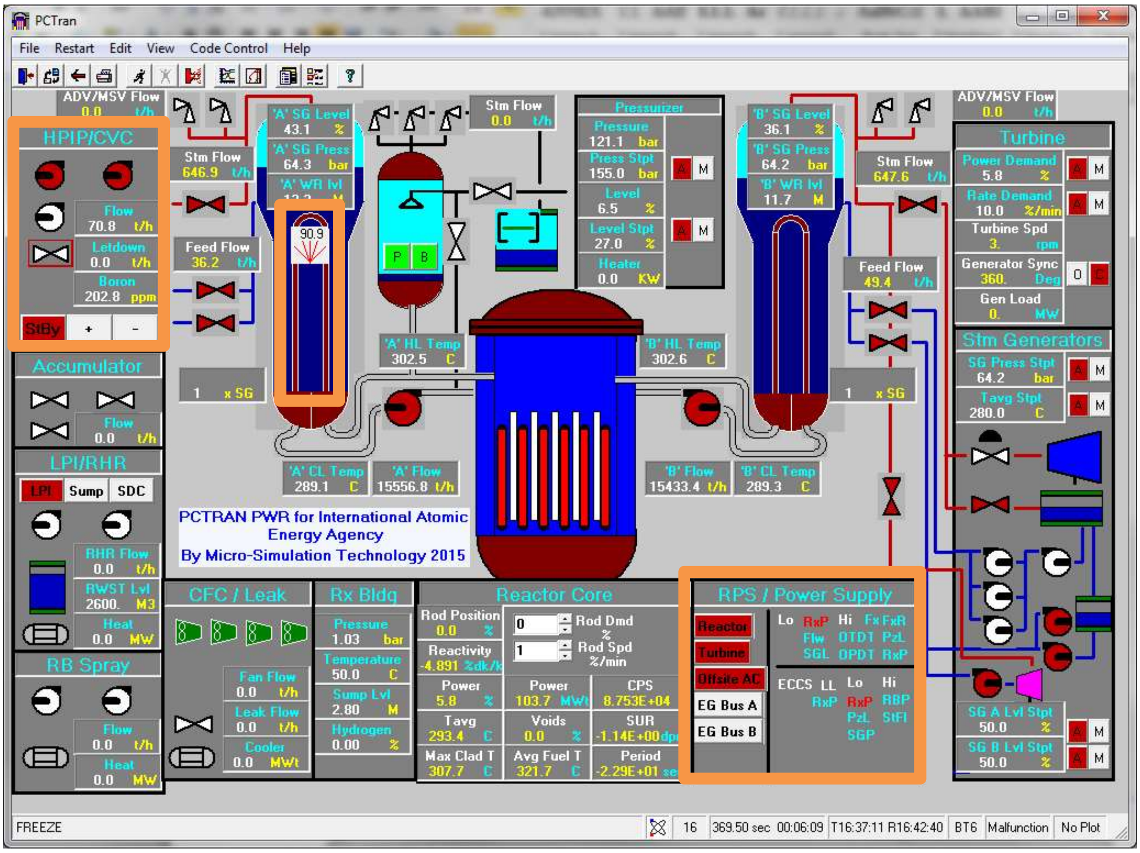

FIG. 48. Reactor trip and ECCS actuation.

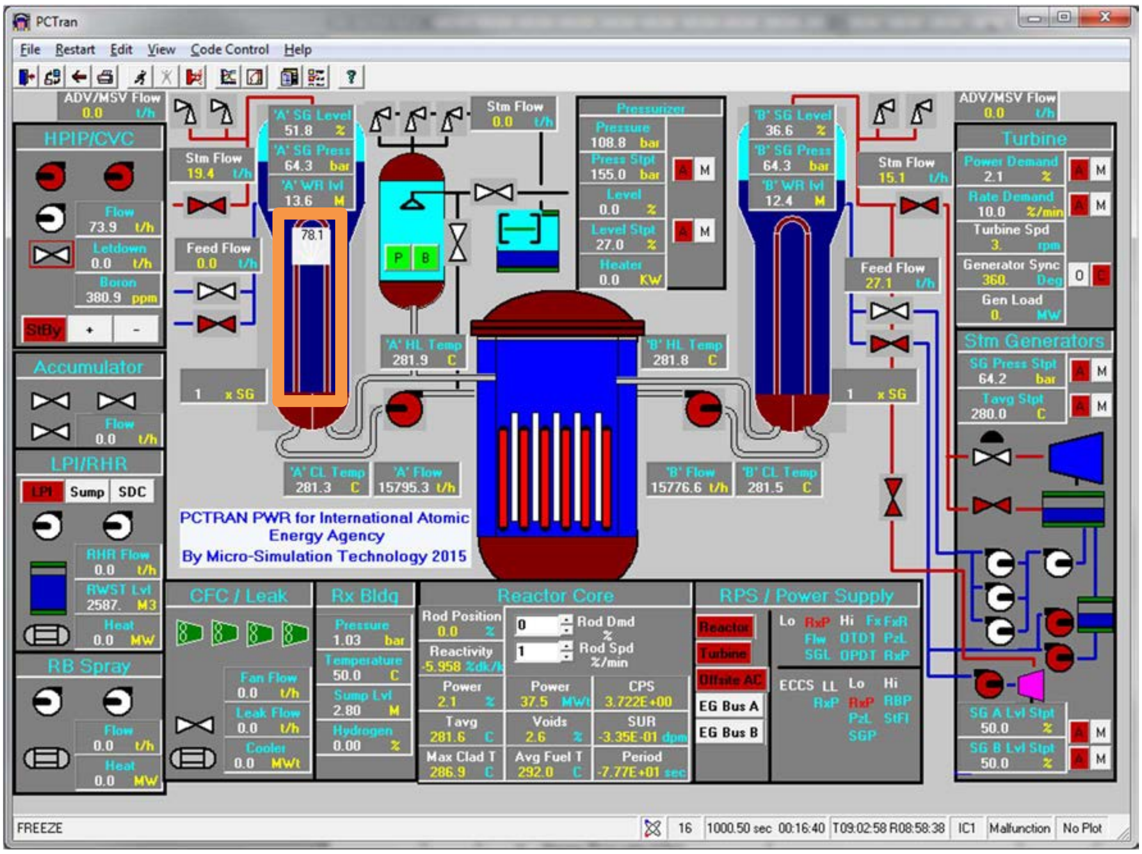

FIG. 49. 1000 seconds after the steam generator tube rupture.

3.3.2. Transient behaviour description

The pressure difference between the SG and the reactor coolant system, which is about 100 bar at the beginning of this accident, is the driving force of the reactor coolant flowing into the SG. This causes the contaminated water to be transferred to the secondary loop from the primary loop. With the transfer of reactor coolant through the rupture, the SG water level rises and the reactor coolant volume lowers. Consequently, the pressurizer pressure drops until it reaches the pressure setpoint for a reactor trip and actuation of the High Pressure Injection System (HPIS).

During normal operation, the primary coolant may contain some radioactive fission products released by faulty fuel elements, while the feedwater is not contaminated by radioactive material. Therefore, when some primary coolant reaches the secondary side of the SG, the activity in the steam line starts to rise.

3.3.3. Transient plot description

Figure 50 shows the reactor coolant volume change. The reactor coolant escapes the reactor coolant system through the ruptured tube with an initial flow rate of about 120 t/hr until the reactor trips at about 350 seconds.

The HPIS is actuated on low pressurizer pressure to compensate for the coolant loss and the reactor coolant volume is stabilized at about 450 seconds.

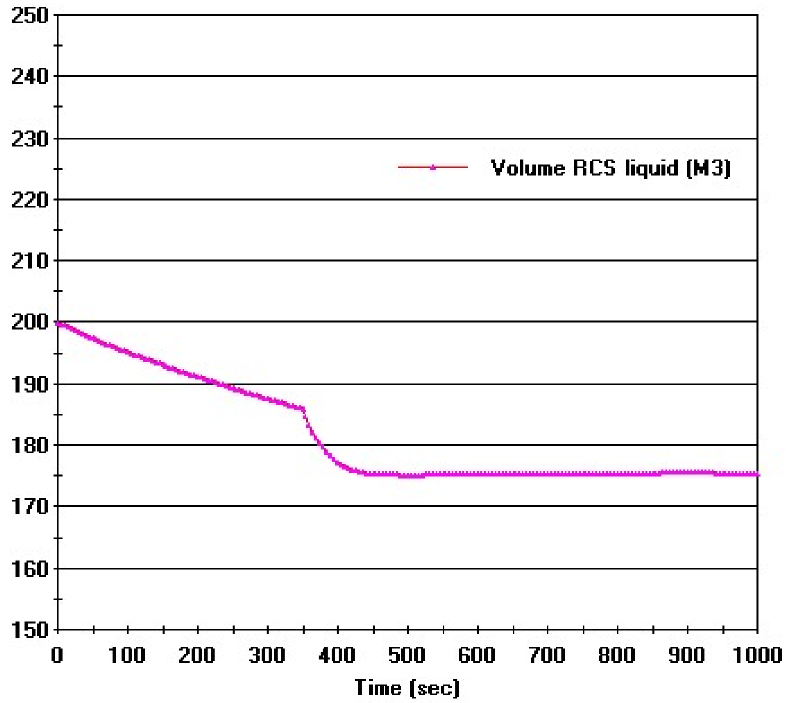

FIG. 50. Reactor coolant volume (m^3).

The loss of reactor coolant through the ruptured tube causes steady decreases in the reactor coolant system pressure shown in Figure 51 until the reactor trips at about 350 seconds, with the consequent pressure drop. Following the trip, the RCS pressure quickly reaches a steady state due to the HPIS operation.

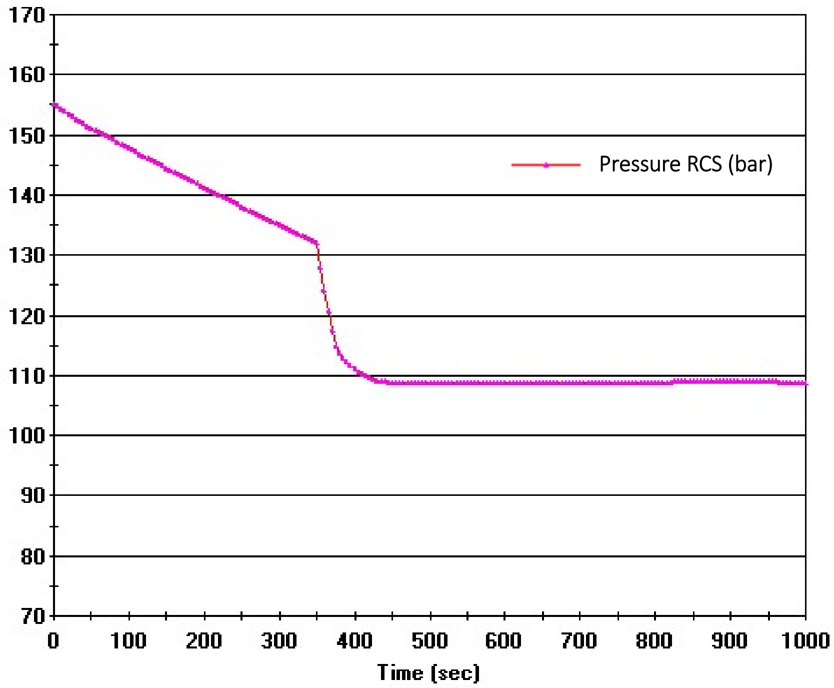

FIG. 51. Reactor coolant system pressure (bar).

Figures 52 and 53 show the water level of both SGs. Due to the primary coolant flowing into the SG through the ruptured tube, the SG-A water level is greater than that of SG-B. The water level of the both SGs sharply but briefly decreases because of the reactor trip on low RCS pressure at about 350 seconds. The level then drops in both SGs due to the quick conversion of feedwater into steam for about 100 seconds Finally, the level in both SGs steadily increases due to the continued operation of the AFW system. The narrow range level is shown to follow the same behaviour.

Increased levels of radioactivity in a steam line is one of the indications of a steam generator tube rupture. Figure 54 shows the monitored radiation level in the steam line A. Upon the steam generator tube rupture, the radioactive primary coolant starts to migrate into the SG. As a result, the radiation level in the SG increases and is detected by the radiation monitor in the steam line, which steadily increases.

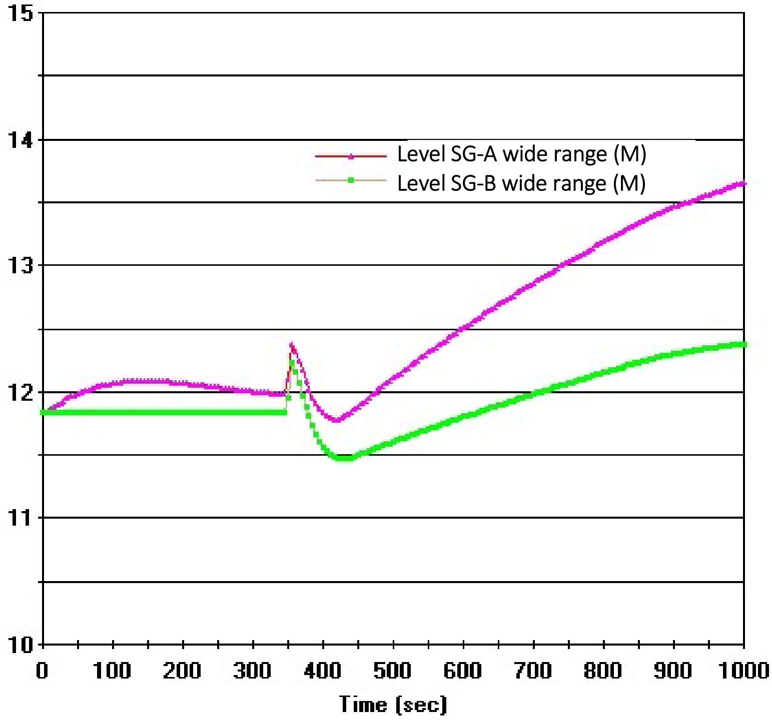

FIG. 52. SG-A and SG-B water level – wide range (m).

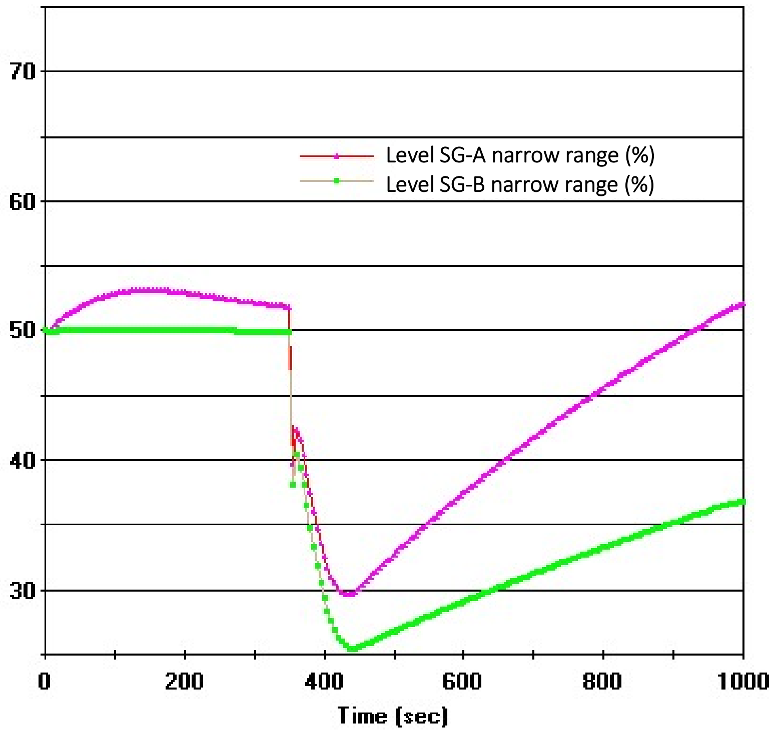

FIG. 53. SG-A and SG-B water level – narrow range (%).

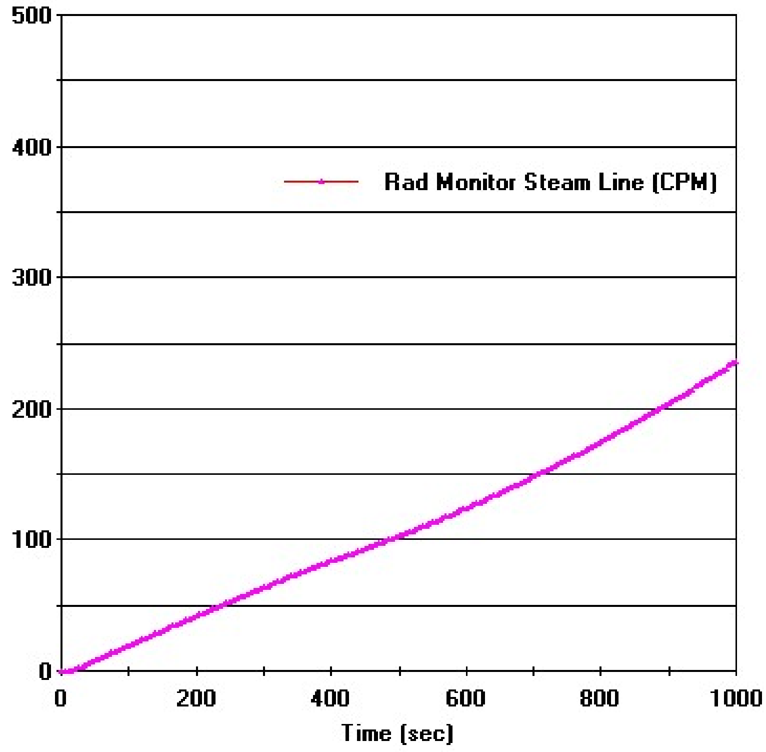

FIG. 54. Steam line monitored activity (Counts/min).

3.4.SMALL BREAK LOSS OF COOLANT ACCIDENT – COLD LEG SBLOCA

Small Break Loss of Coolant Accident (SBLOCA) has been of particular interest since the Three Mile Island (TMI) Unit-2 accident, which will be discussed in more detail later in this exercise book. The hypothetical SBLOCA is a classical DBA for PWRs [3]. The most serious impact of a Loss of Coolant Accident (LOCA) occurs when the break is located in a cold leg where the safety injection system is connected because part of the water makeup provided by this system is lost through the break. In this exercise, a 182.4 cm^2 (6-inch diameter) break in a cold leg is simulated.

3.4.1. Simulation description

Table 10 provides a step-by-step procedure for simulation with corresponding values for relevant variables. In Step 3, a 182.4 cm^2 (6-inch diameter) SBLOCA is introduced with the pre-defined Malfunction #2. Additionally, in Step 4, before resuming the simulator, both RCPs are manually tripped. This step is implemented to minimize the coolant loss through the break; this action may not be completely unrealistic because the plant operators may manually trip the RCPs after confirming that a LOCA has happened. After resuming the simulator, a graphical rupture icon appears in the cold leg-A (Figure 55) and the variables listed in Step 6 change accordingly. In Step 7, a reactor trip signal is generated on low pressurizer pressure (Figure 56). In Step 8, safety injection also starts automatically on low pressurizer pressure. Initially, the HPIS is actuated to inject borated water into the reactor coolant system. With the continuous decrease in pressure due to the break, the accumulators and the Low Pressure Injection System (LPIS) are actuated (Figures 57 and 58). Initially, the LPIS uses the water in the refuelling water storage tank (RWST). When the RWST is emptied (Step 10), this system is reconfigured to draw water that has accumulated in the containment building recirculation sump (Figure 59).

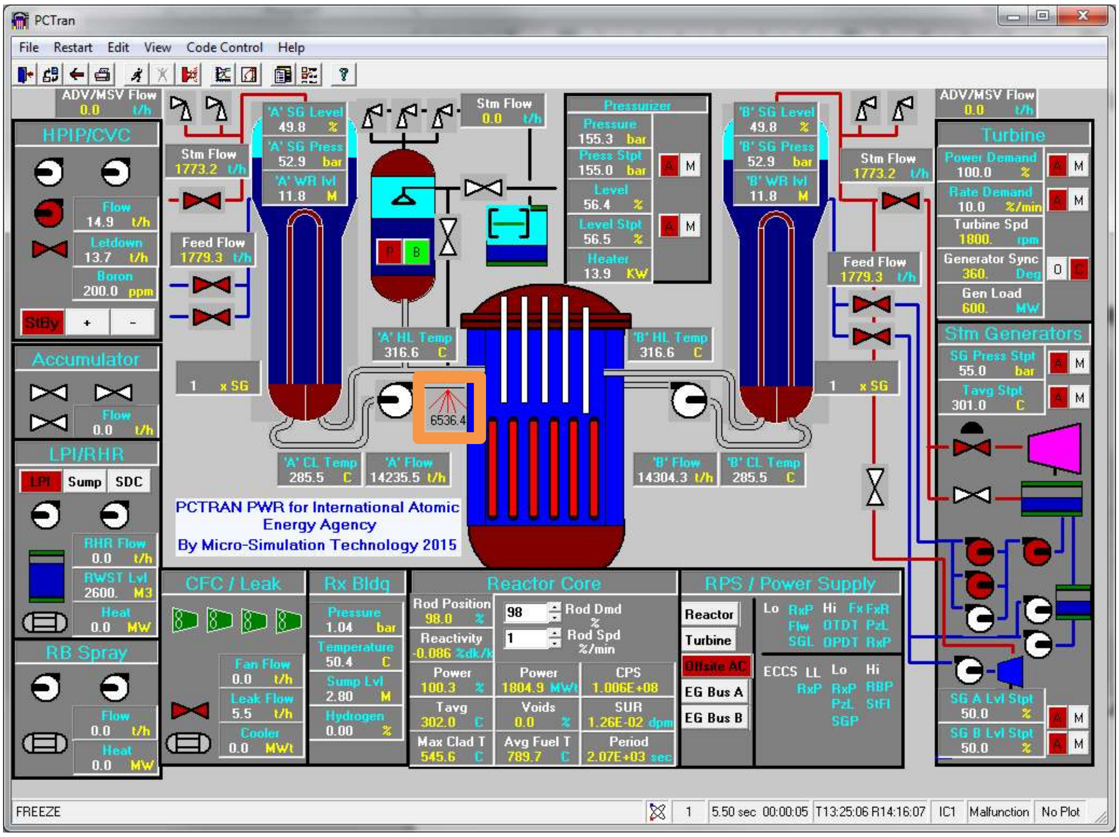

FIG. 55. Initiation of 182.4 cm^2 (6-inch diameter) cold leg SBLOCA and trip of RCPs.

TABLE 10. SMALL BREAK LOSS OF COOLANT ACCIDENT – COLD LEG SBLOCA

STEP	PROCEDURE STEPS	EXPECTED RESPONSE	FIGURE
Setup	Load initial condition #1 100% POWER EOC or #2 100% POWER MOC or #3 100% POWER BOC. Initial condition #1 is used for this exercise		
1	Run the simulator for about 5 seconds to achieve a steady-state condition.		
2	Freeze the simulator.	The simulator freezes.	
3	Initiate a cold leg SBLOCA by selecting 'Malfunction #2'. To simulate a 182.4 cm^2 (6-inch diameter) break, put 182.4 in the failure fraction box and check the 'active' checkbox.		
4	Manually trip both RCPs. This will minimize the coolant loss through the break.		
5	Resume the simulator (*speed up the simulator if needed*).	A rupture icon appears in cold leg-A with the primary coolant loss rate (t/hr).	Figure 55
6	Check the following variables changing: (1) Pressurizer level (2) RCS pressure (3) Reactor building pressure (4) Reactor building air temperature	 (1) Decreases (2) Decreases (3) Increases (4) Increases	
7	Observe the reactor trip due to low primary coolant flow rate, followed by turbine trip.	All the control rods are inserted into the core. The turbine governor valve closes, and the turbine bypass valve opens to let the steam flow. The colour of the 'Reactor' and 'Turbine' buttons in RPS section also changes from grey to red due to the reactor and turbine trips.	Figure 56
8	Observe the ECCS actuation: (1) The HPIS pumps automatically start at about 15 seconds when the RCS pressure is lower than the actuation setpoint of 129.6 bar. (2) The accumulators inject at about 180 seconds when the RCS pressure is lower than the nitrogen pressure in accumulator tanks (44.32 bar). (3) LPIS system operates the last when the RCS pressure reaches 11.3 bar.	Although the HPIS alone is not enough to compensate for the primary coolant loss, the accumulators and LPIS provide borated water to prevent core uncovery. Reactor building spray is also actuated at about 210 seconds when the reactor building pressure reaches the setpoint of 1.3 bar.	Figure 56 Figure 57 Figure 58
9	Observe the following variables change: (1) Refuelling water storage tank (RWST) volume (m^3) (2) Reactor building sump level (m)	(1) Decreases (2) Increases	
10	Check that after the LPIS uses up the water in the RWST at about 4000 seconds, the water in the reactor sump is pumped through a heat exchanger and injected into the reactor vessel for continuous cooling.	When the RWST is used up, the 'Sump' icon in the LPI/RHR section turns red indicating the sump recirculation mode is on.	Figure 59

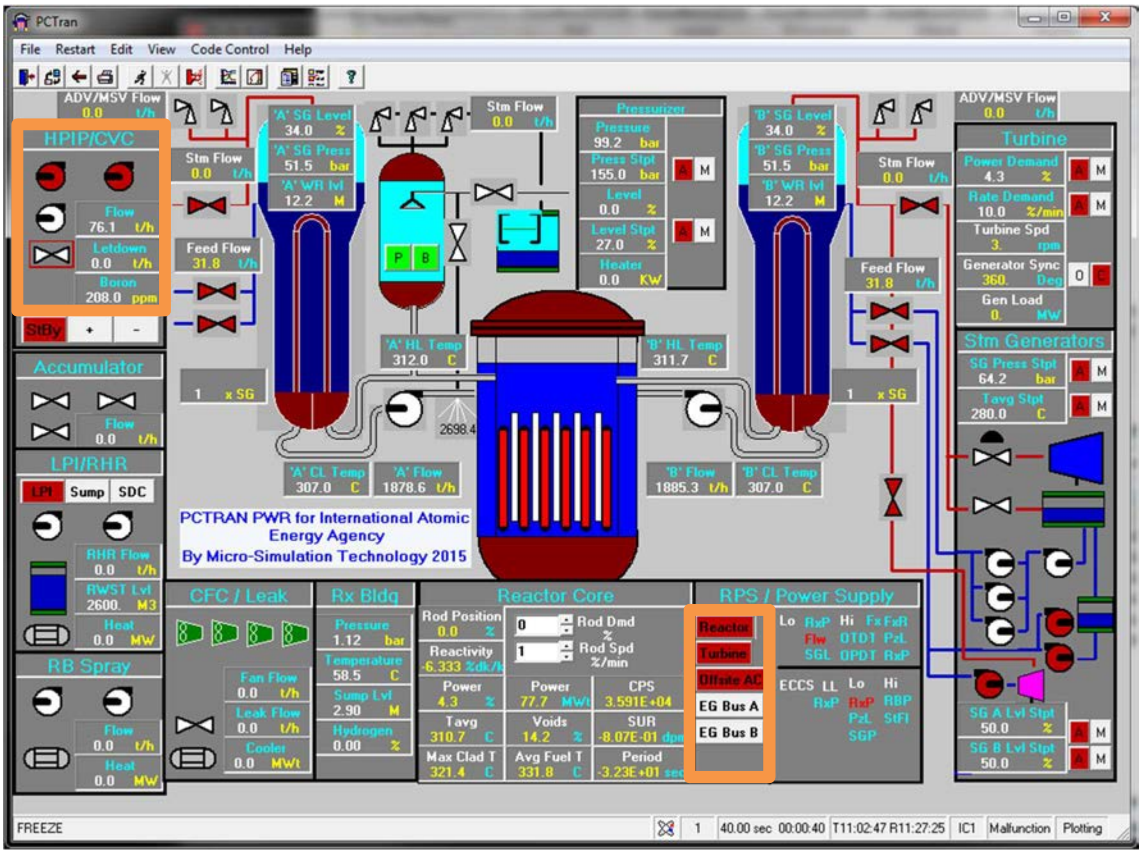

FIG. 56. Reactor trip and actuation of HPIS.

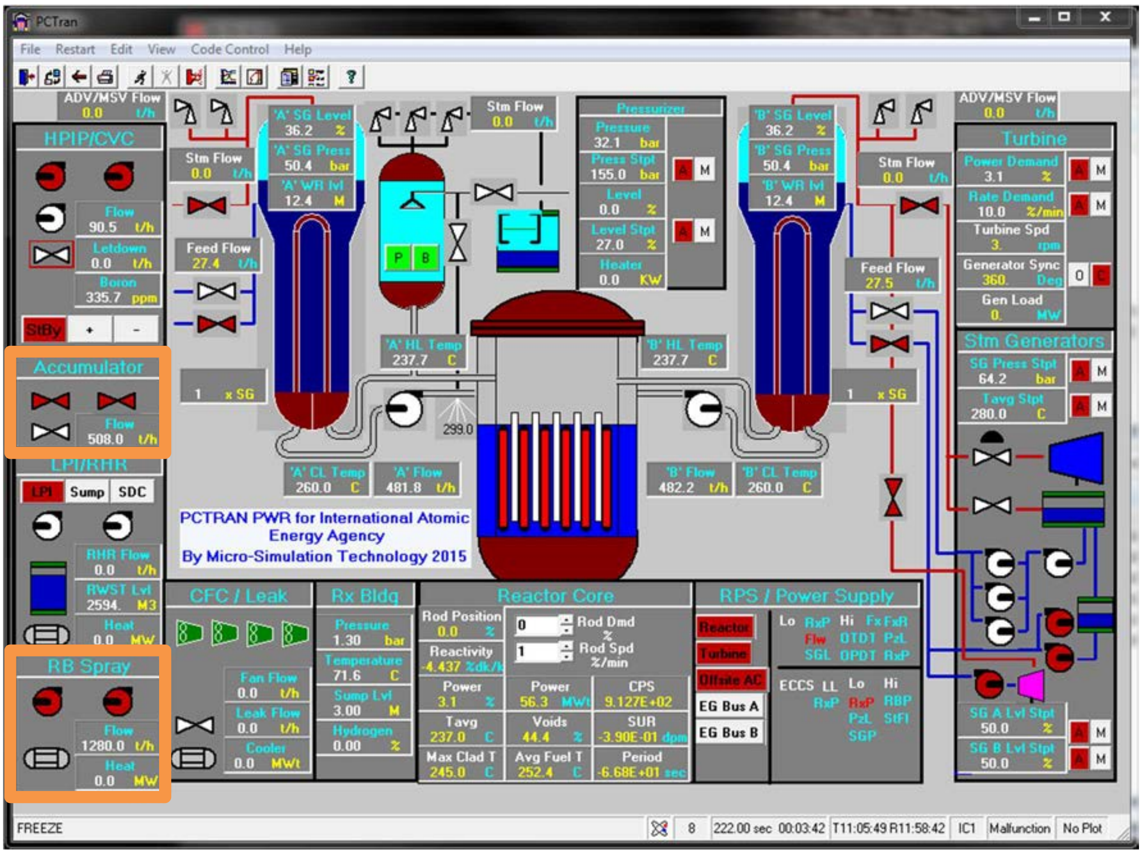

FIG. 57. Accumulator and reactor building spray operating.

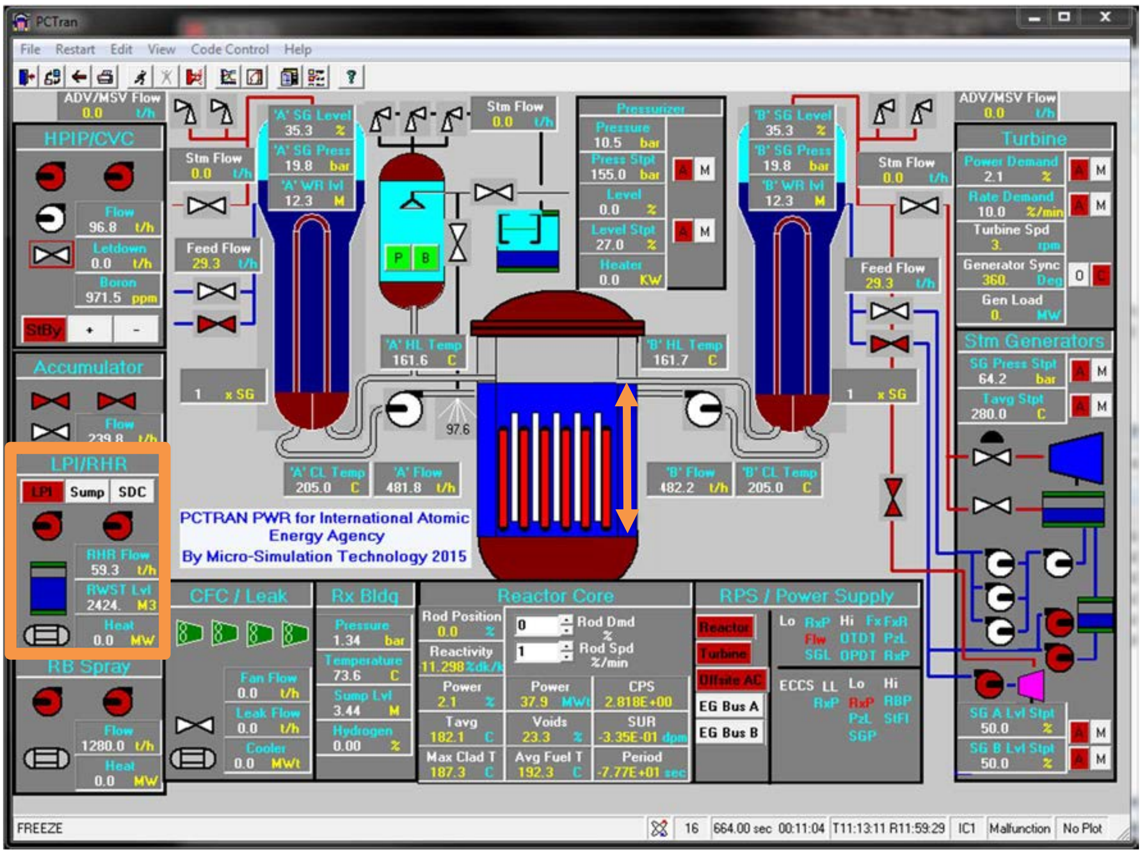

FIG. 58. LPIS operating and recovery of the reactor vessel level.

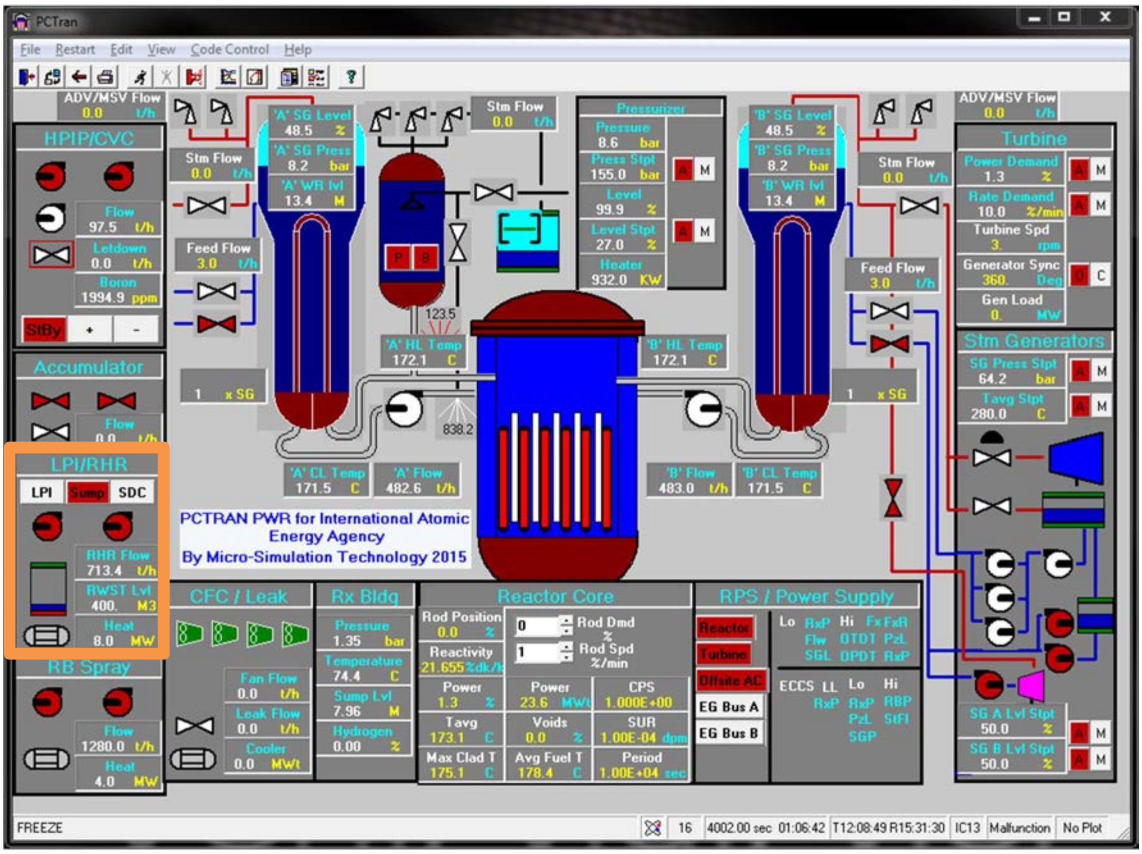

FIG. 59. LPIS sump recirculation mode for long term cooling.

3.4.2. Transient behaviour description

Following the initiation of the small break in the cold leg, the RCS depressurization starts and the reactor trips due to low pressurizer pressure setpoint. When the RCS pressure reaches 129.6 bar, the HPIS is automatically actuated and injects borated water into the reactor vessel. However, the HPIS alone is not enough to compensate for coolant loss through the break. Compared to a large break LOCA, the depressurization occurs relatively slowly in a small break LOCA. Thus, the pressure does not reach the accumulator actuation setpoint until about 180 seconds. Upon actuation of the accumulators, as shown in Figure 57, the ECCS flow starts to exceed the coolant discharge rate and the reactor vessel water level slowly rises again.

With the continuous decrease in reactor coolant system pressure, the LPIS is actuated. The LPIS provides enough water makeup to maintain the core submerged. The LPIS is responsible for long term cooling of the residual heat in the reactor. After the HPIS and LPIS use up all the water in the RWST, the LPIS switches over to pump the water accumulated in the reactor building sump and inject it through a heat exchanger to the reactor vessel for long term cooling (Figure 59).

3.4.3. Transient plot description

The reactor coolant system is depressurized upon initiation of the SBLOCA as illustrated in Figure 60. As the pressure reaches the ECCS actuation setpoint, the rate of depressurization decreases due to injection by this system.

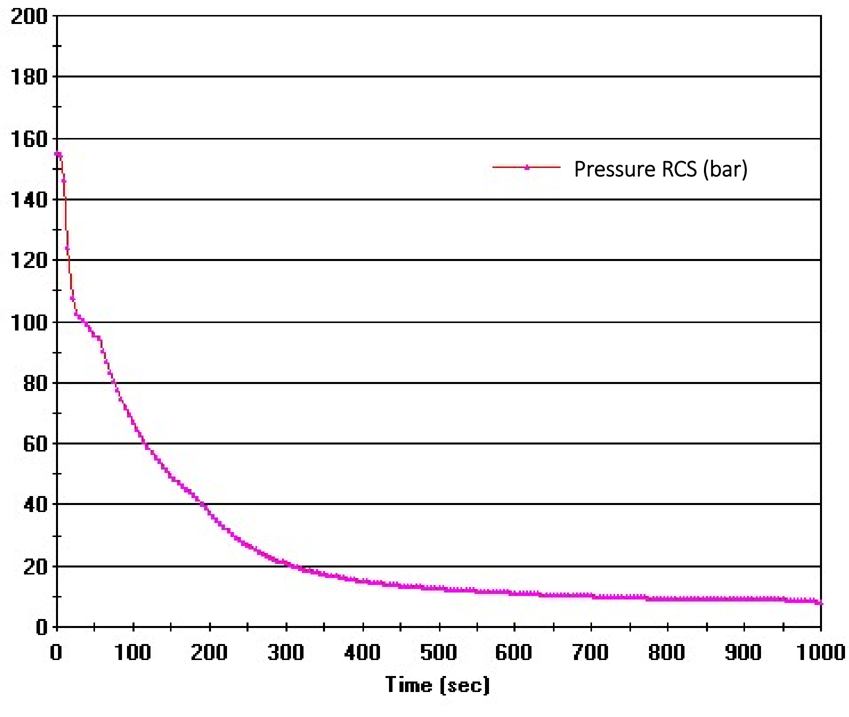

FIG. 60. RCS pressure (bar).

Figure 61 shows the reactor coolant leak rate and the total ECCS water injection rate. The reactor coolant leak rate depends on the pressure of the reactor coolant system, and on the reactor coolant level in the reactor vessel. As the RCS is depressurized rapidly, the reactor coolant leak rate decreases accordingly. When the pressurizer is being emptied during about the first 30 seconds, the reactor coolant leak rate rapidly decreases as the pressure at the elevation of the break changes fast. As the reactor coolant in the vessel is discharged, leak rate change slows until 100 seconds. When the coolant level reaches the elevation of the break, the coolant leak rate drops quickly again. Then the leak rate slowly decreases as the RCS pressure decreases (see Figure 60). When the coolant level reaches the break again, the coolant leak rate fluctuates representing the coolant leaking directly through the break. As seen in Figure 60, the pressure setpoint for the LPIS, which is around 11 bars is reached at around 950 seconds and therefore the ECCS flow suddenly increases.

Upon initiation of the SBLOCA, the reactor coolant is released through the break with a high flow rate and thus, the reactor coolant volume drops rapidly as illustrated in Figure 62. However, with the HPIS actuation and the depressurization of the primary system, the coolant loss rate decreases.

At about 180 seconds, when the accumulator starts operating, the coolant volume begins to increase steadily, and the reactor vessel is refilled. When the water level reaches the break again, the coolant volume stops increasing and is stabilized although the LPIS is actuated at 950 seconds. When the flow rate of LPIS rapidly increases at about 950 seconds, the RCS coolant volume starts to increase until the RCS is filled at about 1300 seconds.

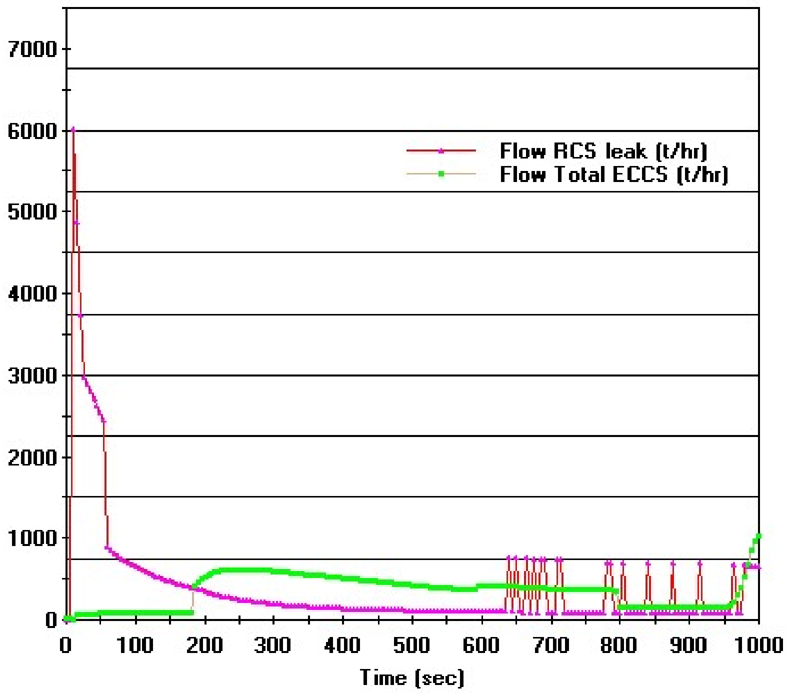

FIG. 61. Reactor coolant leaking rate and total ECCS flow rate (t/hr).

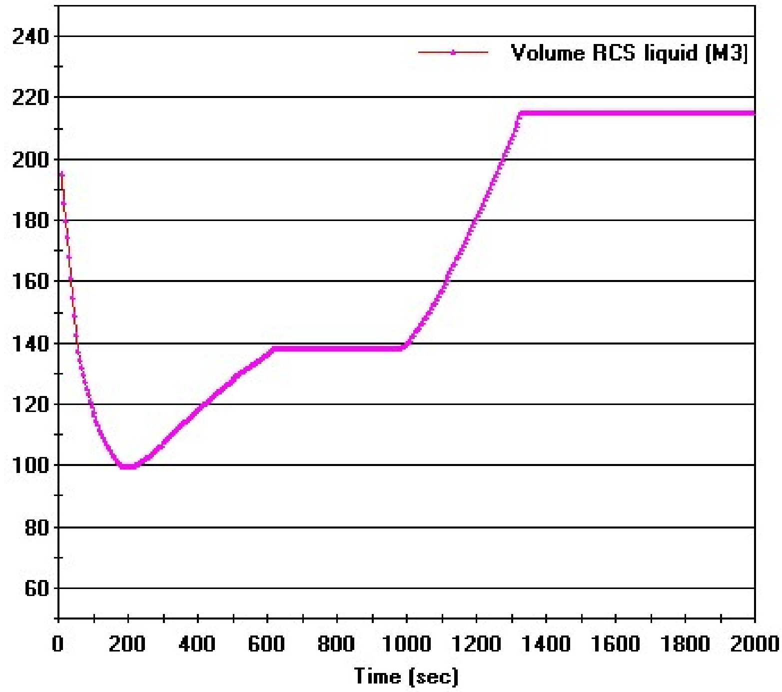

FIG. 62. Reactor coolant volume (m^3).

Figure 63 shows the peak fuel temperature and the peak cladding temperature during the simulation. The initial rapid drop is attributed to the sudden cease of nuclear chain reaction following the reactor trip at about 8 seconds. Due to the ECCS water injection, the reactor core remains submerged during the transient so therefore fuel and cladding temperatures regularly decrease and never reach the critical value.

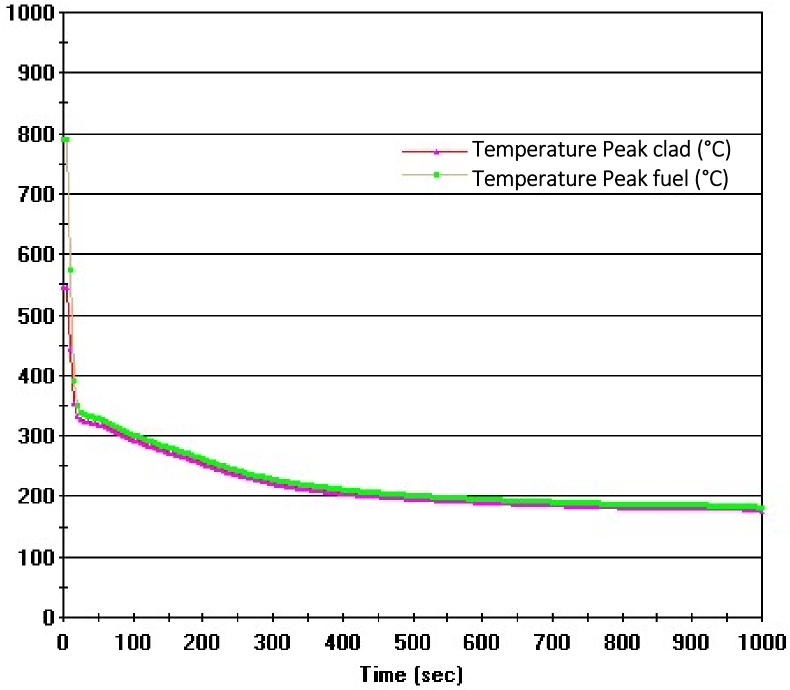

FIG. 63. Peak clad and fuel temperature (°C).

3.5. LARGE BREAK LOSS OF COOLANT ACCIDENT – COLD LEG LBLOCA

Large Break Loss of Coolant Accident (LBLOCA) is a DBA for a PWR [3]. In this accident, one of the cold legs (for a cold leg break) from the RCPs to the reactor vessel is completely broken and disconnected to allow free discharge of the reactor coolant. This break is called ‘double-ended guillotine break’. It is assumed that the leak flow is 200%, i.e. from both the ends of the break.

3.5.1. Simulation description

Table 11 provides a step-by-step procedure for simulation with corresponding values for relevant variables. In Step 3, a double-ended guillotine break is initiated with ‘Malfunction #2’. Next, in Step 4, before resuming the simulator, two RCPs are manually tripped. This step is implemented to minimize the coolant loss through the break; this action may not be completely unrealistic because the plant operators may manually trip the RCPs after confirming that a LOCA has happened. Upon resuming the simulator in Step 5, the reactor coolant system is rapidly depressurized with huge coolant loss rate. In LBLOCA, due to its large break size, depressurization occurs rapidly compared to SBLOCA. Thus, the reactor trip and the actuation of the ECCS happen in an early stage of the accident (Step 7 and Step 8) (Figures 64 and 65). After the core water level reaches the level of the cold legs in Step 10, the ECCS water injected leaks through the break (Figure 66). Finally, in Step 11, when the ECCS uses up all the water in the refuelling water storage tank (RWST), the LPIS starts to pump the water accumulated in the containment sump and inject it into the system.

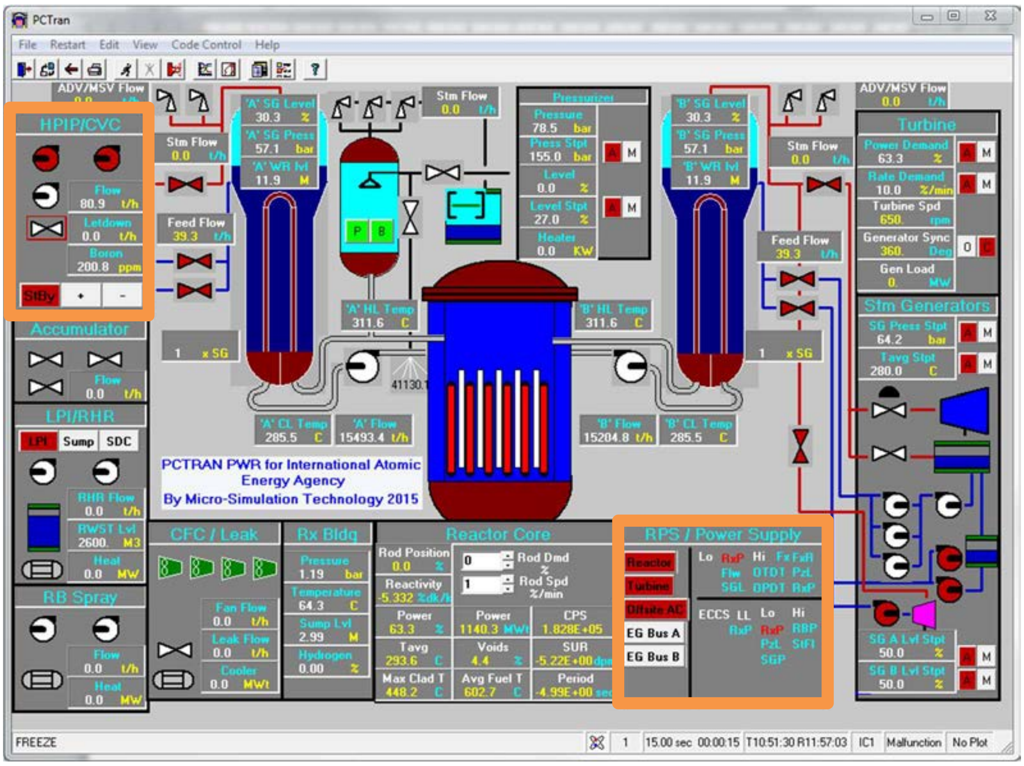

FIG. 64. HPIS actuation and reactor trip.

TABLE 11. LARGE BREAK LOSS OF COOLANT ACCIDENT – COLD LEG LBLOCA

STEP	PROCEDURE STEPS	EXPECTED RESPONSE	FIGURE
Setup	Load initial condition #1 100% POWER EOC, or #2 100% POWER MOC, or #3 100% POWER BOC. Initial condition #1 is used for this exercise.		
1	Run the simulator for about 5 seconds to achieve a steady-state condition.		
2	Freeze the simulator.	The simulator freezes.	
3	Initiate a cold leg LBLOCA by selecting ‘Malfunction #2’. To simulate 2800 cm^2 break, insert 2800 in the failure fraction box and check the ‘active’ checkbox.		
4	Manually trip the RCPs. This will minimize the coolant loss through the break.		
5	Resume the simulator (*speed up the simulator if needed*).	A rupture icon appears in cold leg-A with the primary coolant loss rate (t/hr).	
6	Check the following variables for changes: (1) Decrease in pressurizer level due to the coolant loss. (2) Decrease in RCS pressure. (3) Increase in reactor building pressure. (4) Increase in reactor building air temperature.	Rapid decrease in the reactor coolant volume and RCS pressure is observed. This is ‘Blowdown phase’, the first phase of LOCA which occurs for the first 20 seconds of the accident.	
7	Observe the reactor trip due to the low pressurizer pressure at about 7 seconds, followed by turbine trip.		
8	Observe the ECCS actuation between 7 seconds and 40 seconds: (1) The HPI pumps automatically start when the RCS pressure is lower than the actuation setpoint of 129.6 bar. (2) Continuous decrease in RCS pressure starts the accumulator when the pressure is lower than the nitrogen pressure in accumulator tank (44.32 bar). (3) LPI system operates as it reaches the setpoint of 11.3 bar.	Although the HPIS alone is not enough to compensate for the coolant loss in the reactor, the accumulator and LPIS successfully refill the reactor with the borated water. This is ‘refill phase’ and ‘reflood phase’: the second and the third phase of LOCA, respectively. Also observe that the RBS (Reactor building spray) are turned on when the reactor building pressure reaches the setpoint of 1.3 bar.	Figure 64 Figure 65
9	Observe the following variables: (1) Refuelling water storage tank water volume (m^3) (2) Reactor voids (water level) (%) (3) Reactor building sump level (m)	(1) Decreases (2) Initially decreases and then increases (3) Increases	
10	Check the safety injection flow rate makes the balance with the rate loss of coolant through the break at about 250 seconds.	The core water level is stable.	Figure 66
11	Check after the ECCS uses up the water in the refuelling water storage tank (RWST), the water in the reactor sump is pumped to the heat exchanger and injected to the reactor coolant system for continuous core cooling.	When the RWST is used up, the ‘Sump’ icon turns to red indicating sump recirculation mode is on.	

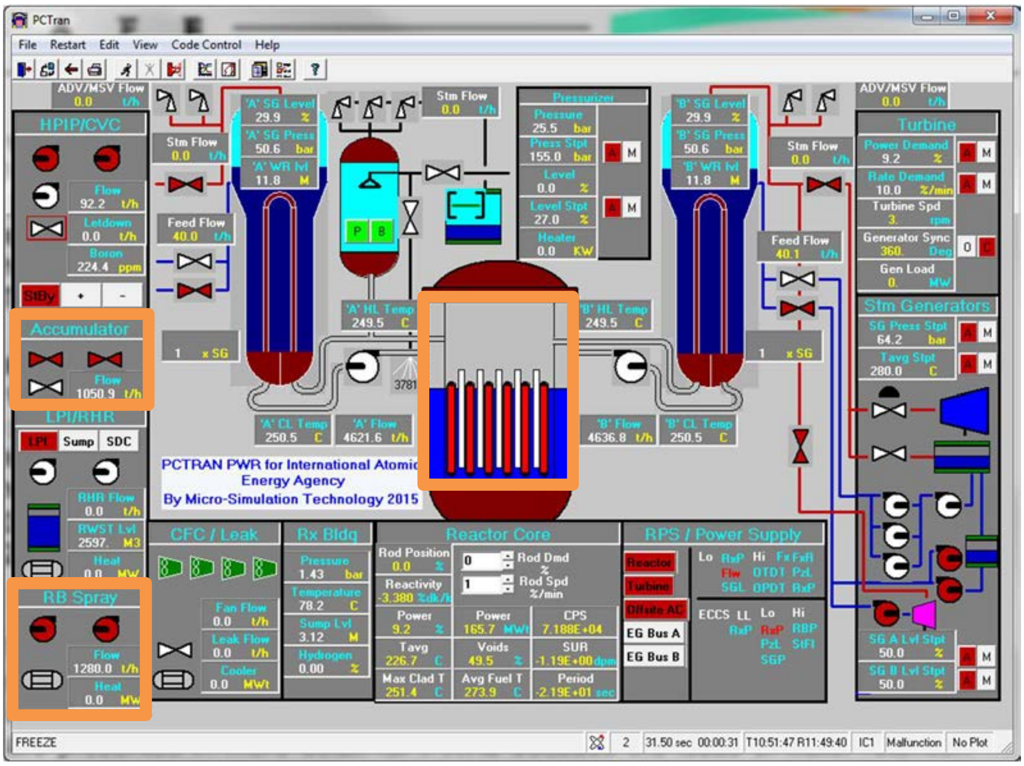

FIG. 65. Reactor core partly uncovered and accumulators, RBS actuated.

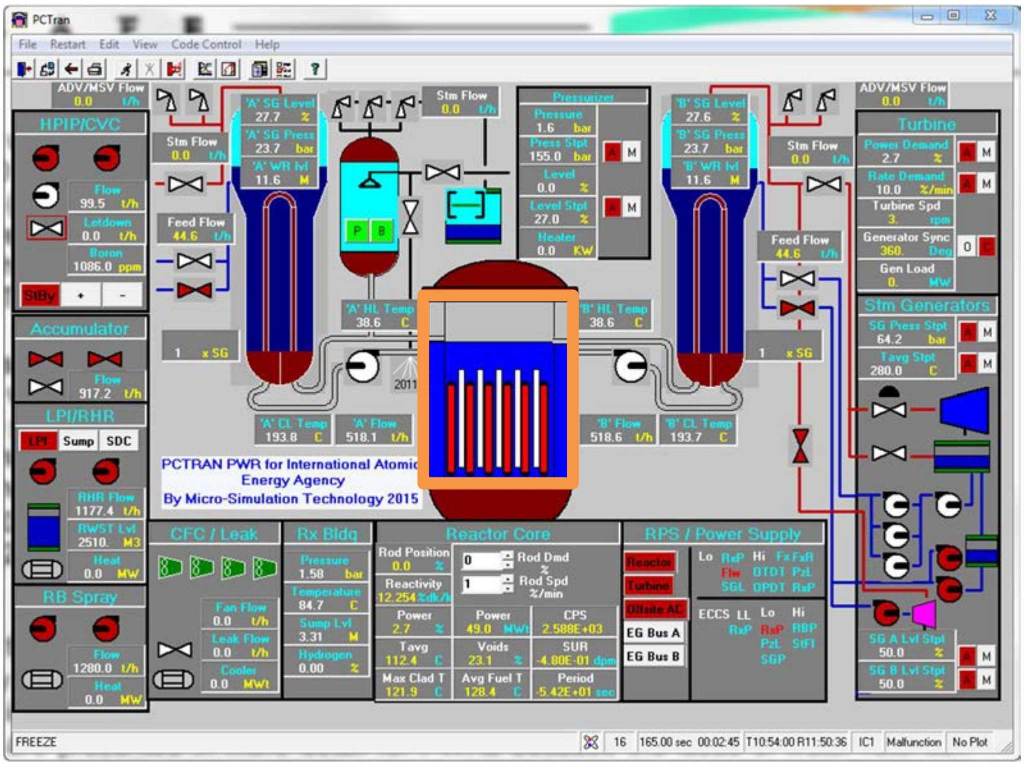

FIG. 66. The core reflooded by the ECCS water injection.

3.5.2. Transient behaviour description

The main difference of the LBLOCA form the SBLOCA is the faster depressurization of the RCS and reactor coolant loss rate. Following the initiation of the double-ended guillotine break, the RCS pressure reaches the pressure setpoint of reactor trip and HPIS actuation within 3 seconds. However, due to the high reactor coolant loss rate, even after the actuation of the accumulators at about 10 seconds, the reactor core water level lowers and the core uncovery occurs at about 20 seconds. This is the first phase of the LBLOCA called 'Blowdown phase' where the content of the reactor coolant system and primary loops are blown down through the break.

In a PWR LBLOCA, after the blowdown phase, the injected ECCS water is blocked by the upward flow of steam in the downcomer annulus through which the cooling water normally flows. This blocked water fails to reach the vessel below the core and bypasses the upper part of the inlet annulus. This is the bypass phase, the second phase of the LBLOCA. In PCTRAN, this phase is not observed and the injected ECCS water is continuously accumulated in the lower vessel region.

Upon core uncovery, the fuel and clad temperature starts to increase. Once the LPIS is actuated at about 40 seconds, the core water level begins to rise and when the core is submerged again, the fuel temperature stops increasing. When the core water level is increased enough to reach the cold leg level, further increase in the water level ceases and the water level is stabilized.

In a real LOCA, a refill phase and a reflood phase come after the bypass phase. In the refill phase, further depressurization occurs and the ECCS water starts to fill the lower plenum. Once the lower plenum is filled with liquid, the reflood phase begins. In the reflood phase, fuel elements are flooded by ECCS water from the bottom upward until the core water reaches the cold legs level. In PCTRAN, however, only the top of the core is uncovered and upon the actuation of the LPIS, the core is reflooded. Thus, the refill phase and reflood phase are not clearly distinguished.

Once the core water overflows through the break and the core water level is steady, the long term cooling phase begins. The injected ECCS water drives though the core by natural circulation. Once the RWST is emptied, the overflown water collected in the reactor building sump is pumped by the LPIS and reinjected into the core. During this phase, the fuel element temperature remains steady.

3.5.3. Transient plot description

Compared to SBLOCA, the depressurization of the RCS is much faster in a LBLOCA. As shown in Figure 67, within 100 seconds, the pressure reaches the LPIS setpoint of 11.3 bar. After the rapid decrease in RCS pressure, it will be stabilized around the reactor building pressure.

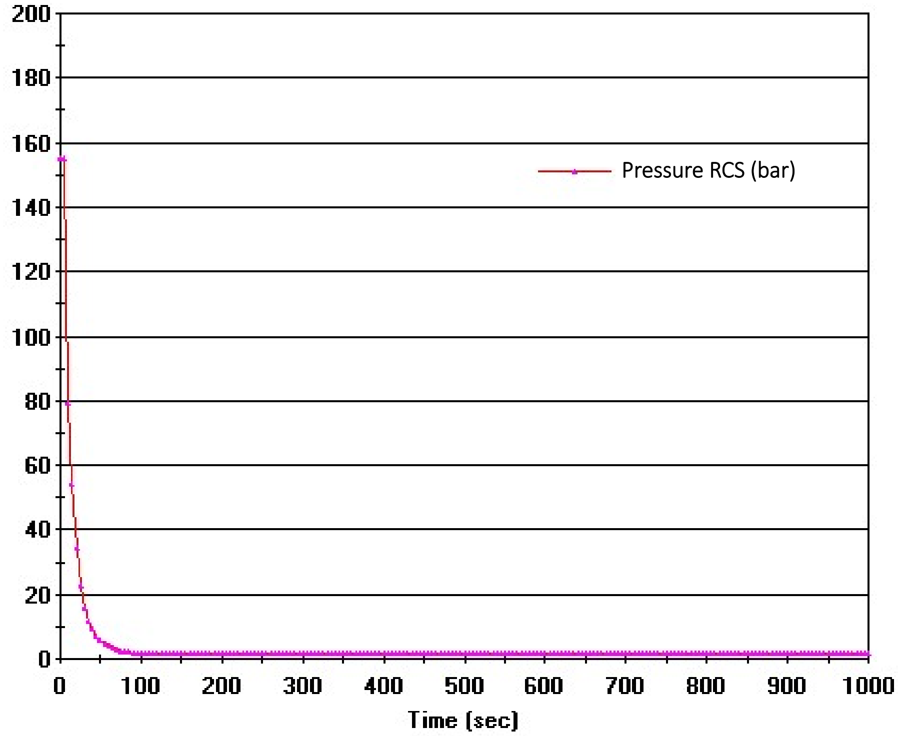

FIG. 67. Reactor coolant system pressure (bar).

Figure 68 shows the reactor coolant leaking rate and the total ECCS water injection rate. Starting from very high coolant leak rate, the leakage rate decreases rapidly as most of the reactor coolant is lost and the system is depressurized. However, when the ECCS fills the RCS and the reactor vessel water reaches the break again, the leakage rate increases (at about 200 seconds).

Figure 69 shows the peak fuel temperature and the peak cladding temperature during the simulation. With sudden loss of the primary coolant, the core is uncovered and the peak fuel temperature and cladding temperature increase as a result.

However, before core damage occurs, the water injected by the ECCS will submerge the core and peak fuel and cladding temperature decrease.

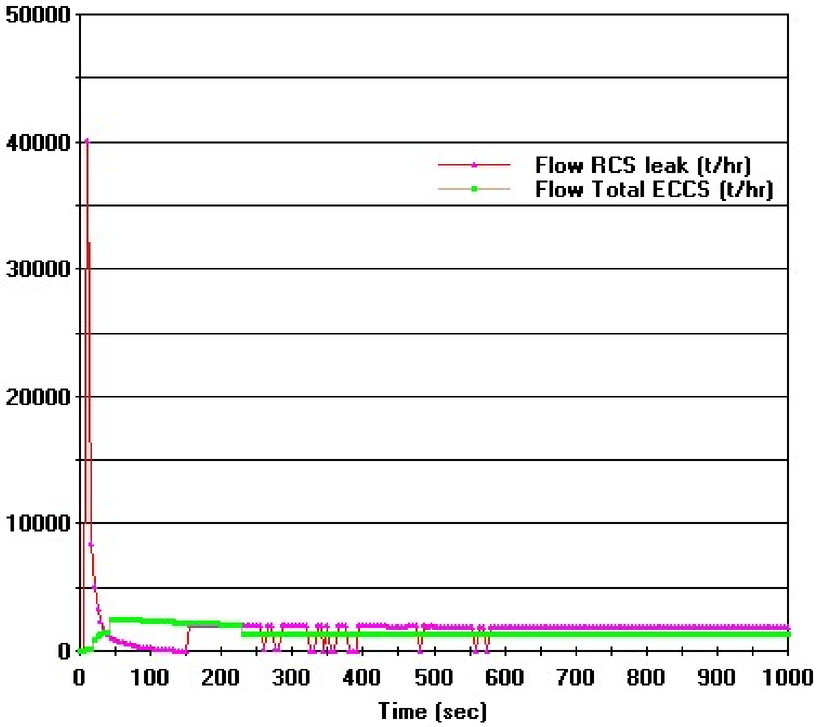

FIG. 68. Reactor coolant leaking rate and total ECCS flow rate (t/hr).

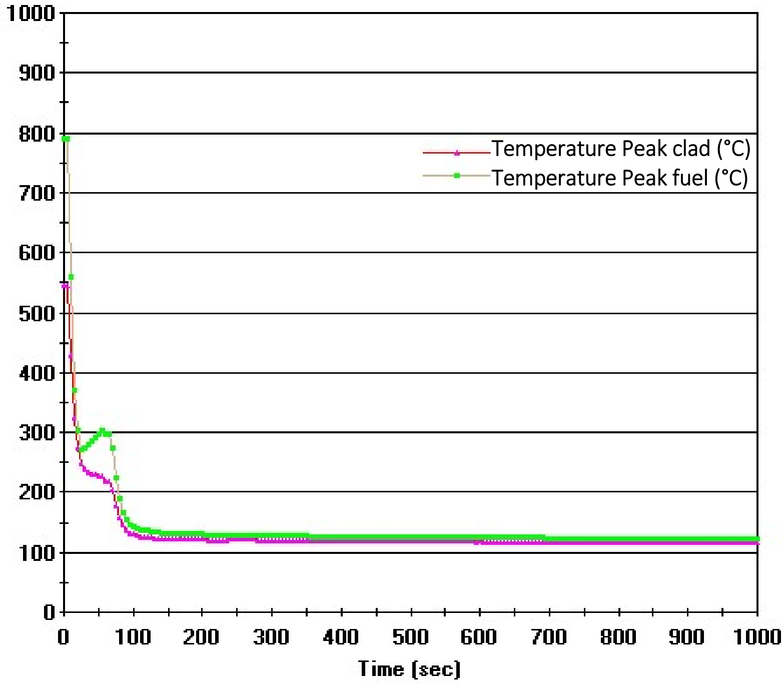

FIG. 69. Peak fuel and cladding temperature (°C).

4. SEVERE ACCIDENTS

4.1. TMI UNIT 2 TYPE ACCIDENT

4.1.1. Simulation description

The Three Mile Island (TMI) power station is near Harrisburg, Pennsylvania in the USA. It had two pressurized water reactors. Unit 1 was of 800MW(e) and connected to the grid in 1974. Unit 2, where the accident took place, was of 906MW(e) [4]. The accident happened at 4 am on 28 March 1979. The reactor was operating at 97% power. The initial cause of the accident started with a failure in the secondary system which prevented the condensate pumps and the MFWPs from sending feedwater to the SGs that remove heat in the RCS. This, in turn, triggered an increase in reactor coolant system pressure and the reactor trip. Normally, the auxiliary feedwater system comes on after about 30 seconds. However, the valves between the AFW system and SGs were closed due to a maintenance error.

In order to control the pressure, the pilot-operated relief valve (PORV) opened and started to discharge steam. The valve was supposed to close after the pressure dropped to the PORV closure setpoint. However, due to a mechanical malfunction, it was stuck open. Through the stuck-open valve, the discharge of the reactor coolant continued. With further depressurization, the HPIS was actuated automatically. However, as the pressurizer level started to increase between the first and approximately sixth minute, the operator manually turned off the HPIS believing that the PORV was closed. Consequently, the stuck-open valve allowed the reactor coolant to escape from the reactor coolant system and was the principal cause of the core disintegration that followed.

In this exercise, the TMI-2 accident is simulated in the same way the real TMI-2 accident was developed. However, PCTRAN only shows the general scenario of the accident and the simulator variables changing during the simulation do not accurately represent the variables during the real accident. TMI was a three loop PWR, whereas this simulator is a 2-loop PWR type design. This scenario is a design extension condition (DEC) with significant fuel degradation [3].

Table 12 provides a step-by-step procedure for simulation with corresponding values of relevant variables.

In Step 3, the auxiliary feedwater isolating valves are manually closed to simulate the pre-accident conditions. In Step 4, the condensate pump is tripped to initiate the event (Figure 70). Upon resuming the simulator, all MFWPs are automatically tripped. As the main feedwater is isolated, in Step 6 the SG water level starts to lower, triggering the reactor and the turbine trips (Figure 71). Simultaneously, the AFW system comes on but no feedwater flow occurs due to the closure of the valves in Step 3. As a result of the stopped feedwater flow, in Step 7, the SGs are dried out. As the reactor coolant system loses its heat sink, the coolant temperature as well as the reactor coolant system pressure starts to increase. In Step 8, when the pressure reaches the setpoint for PORV opening (165 bar), promptly change the valve position manually before the valve is closed so that the valve stays open (Figure 72). This setup cannot be done pre-accident simulation, otherwise the PORV will be open even under normal conditions. As the reactor coolant escapes from the reactor coolant system, depressurization begins and the HPIS is actuated (Step 9). However, the HPIS is manually disabled (Figure 73). Also, the RCPs are manually turned off to simulate manual RCPs trip for high vibration. In Step 10, as hot pressurized steam is discharged into the reactor building, the reactor building pressure rises and the reactor building sprays (RBS) start operating. In Step 11, with the continuous release of reactor coolant through the stuck-open PORV, the reactor core starts to uncover (Figure 74). Upon core uncovery, rapid increase in fuel elements temperature is observed. In Step 12, when the peak cladding temperature rises to 1000°C, Zircaloy cladding exothermically (Equation 1) interacts with nearby steam to form hydrogen and the hydrogen concentration in the reactor building starts to rise. In Step 13, core melt is observed in the dose mimic (Figure 75).

$$\mathrm{Zr} + 2\,\mathrm{H_2O(steam)} = 2\mathrm{H_2} + \mathrm{ZrO_2} + 6700\ \mathrm{J/g(Zr)} \tag{1}$$

TABLE 12. TMI UNIT 2 ACCIDENT

STEP	PROCEDURE STEPS	EXPECTED RESPONSE	FIGURE
Setup	Load initial condition #1 100% POWER EOC or #2 100% POWER MOC or #3 100% POWER BOC. Initial condition #1 is used for the exercise.		
1	Run the simulator for about 5 seconds to achieve a steady-state condition.		
2	Freeze the simulator.	The simulation freezes.	
3	Manually close the auxiliary feedwater isolation valves on both sides (1) Right click the auxiliary feedwater isolation valve icon. (2) Type '0' in the text box for valve position. (3) Check the malfunction active checkbox. (4) Click 'OK'.	Red box will appear around the valve indicating inoperable condition.	Figure 70
4	Manually trip the condensate pump by clicking the pump icon.	The pump icon colour will turn white.	Figure 70
5	Resume the simulator (*speed up the simulator if needed*).	The simulator resumes operating.	
6	Observe the reactor trip due to the low steam generator level, followed by turbine trip. The AFW system is actuated simultaneously.	The auxiliary feedwater system will be actuated. However, since the auxiliary feedwater isolation valve is closed, no feedwater will flow into the SG.	Figure 71
7	Observe the SG dry out at about 1500 seconds.		Figure 72
8	When the reactor coolant system pressure reaches 165 bar, which is the pilot operated pressure valve (PORV) opening setpoint, observe the PORV opening and manually open the PORV by following procedure: (1) Right click the PORV. (2) Type '100' in the text box for valve position. (3) Check the malfunction active checkbox. (4) Click 'OK'.	The coolant will escape through the open PORV in the form of steam. As the reactor coolant system relieves steam, the pressure will decrease rapidly.	Figure 72
9	Observe the RCS depressurization and the actuation of ECCS on low RCS pressure (129.7 bar). After its actuation, manually turn off the HPI pumps. At the same time manually trip RCPs due to high vibration.		Figure 73
10	Observe the Reactor Building Spray (RBS) pumps start operating due to the increased containment pressure. The RBS actuation setpoint is 1.3 bar.	The reactor building pressure will start to decrease.	

11	Observe core uncovery at about 4500 seconds.	As the core is uncovered, the peak fuel temperature as well as the peak cladding temperature will start to soar up.	Figure 74
12	When the peak cladding temperature is close to 1000°C, verify a sudden increase in hydrogen concentration in the reactor building.	When the hydrogen concentration is more than 5%, users can burn the hydrogen with ‘Malfunction 5’.	
13	Go to ‘View’ → ‘Dose Mimic’ and observe the core melt.	Black part in the middle of the core, representing the core cavity, appears.	Figure 75

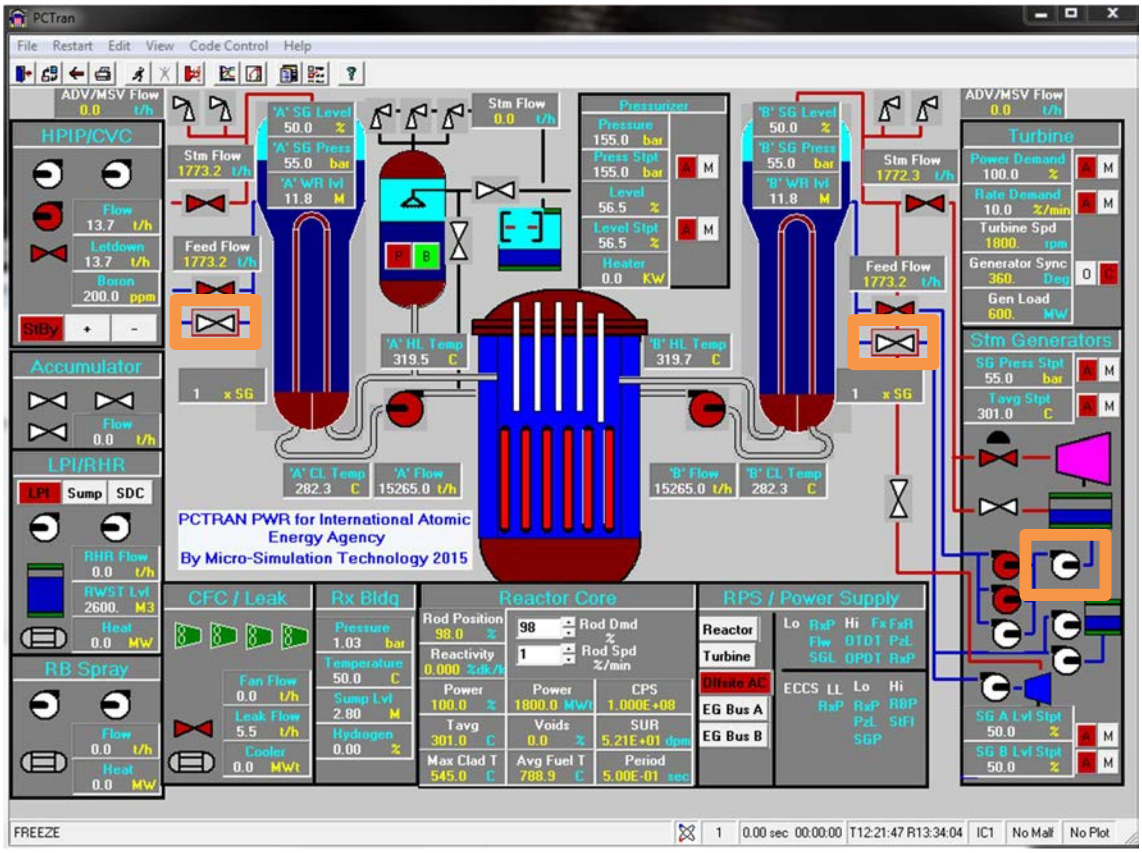

FIG. 70. AFW isolation valves closed and condensate pump trip.

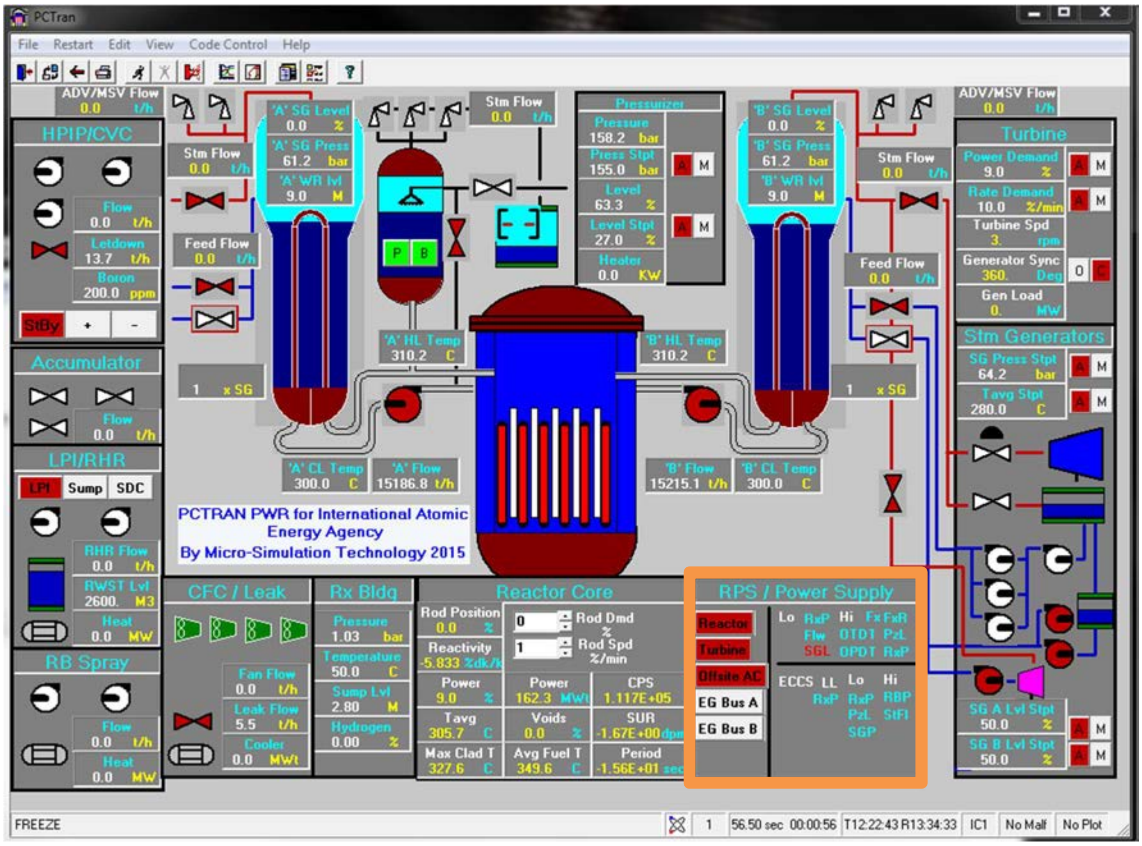

FIG. 71. Reactor trip on low SG level.

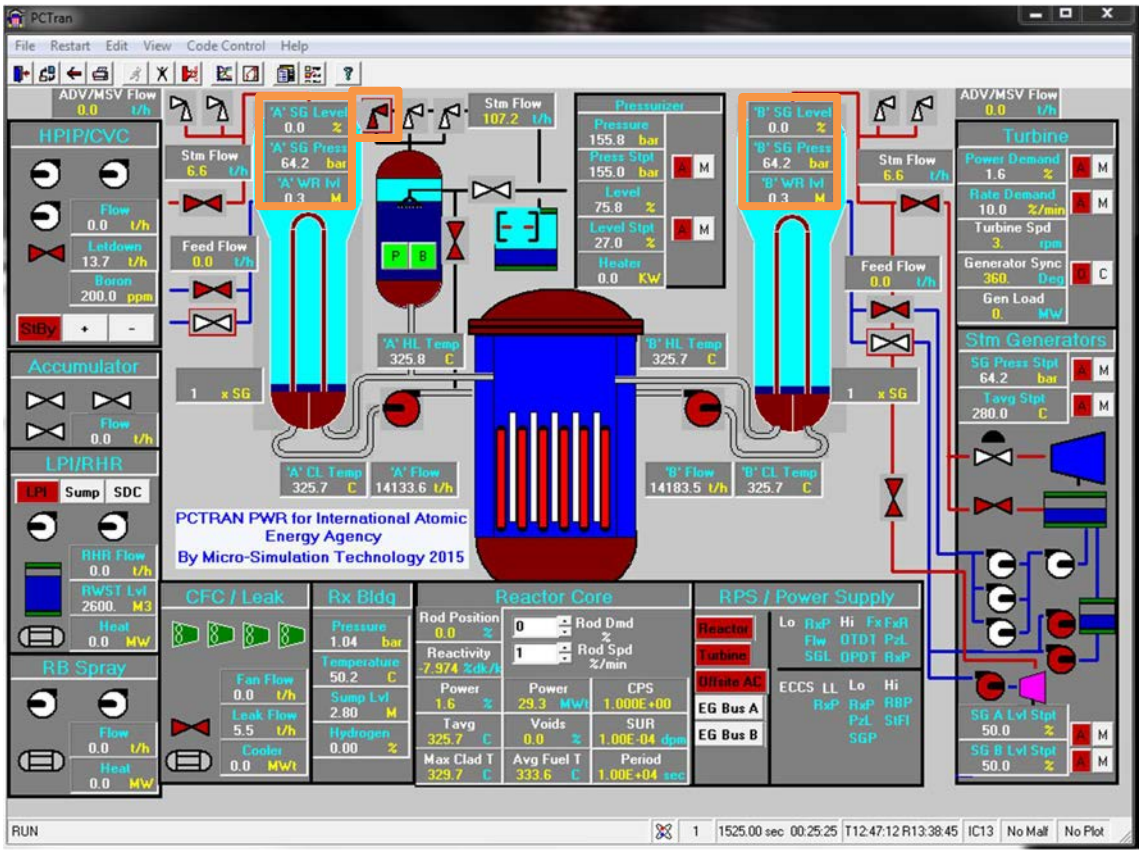

FIG. 72. SG dried up and PORV stuck-open.

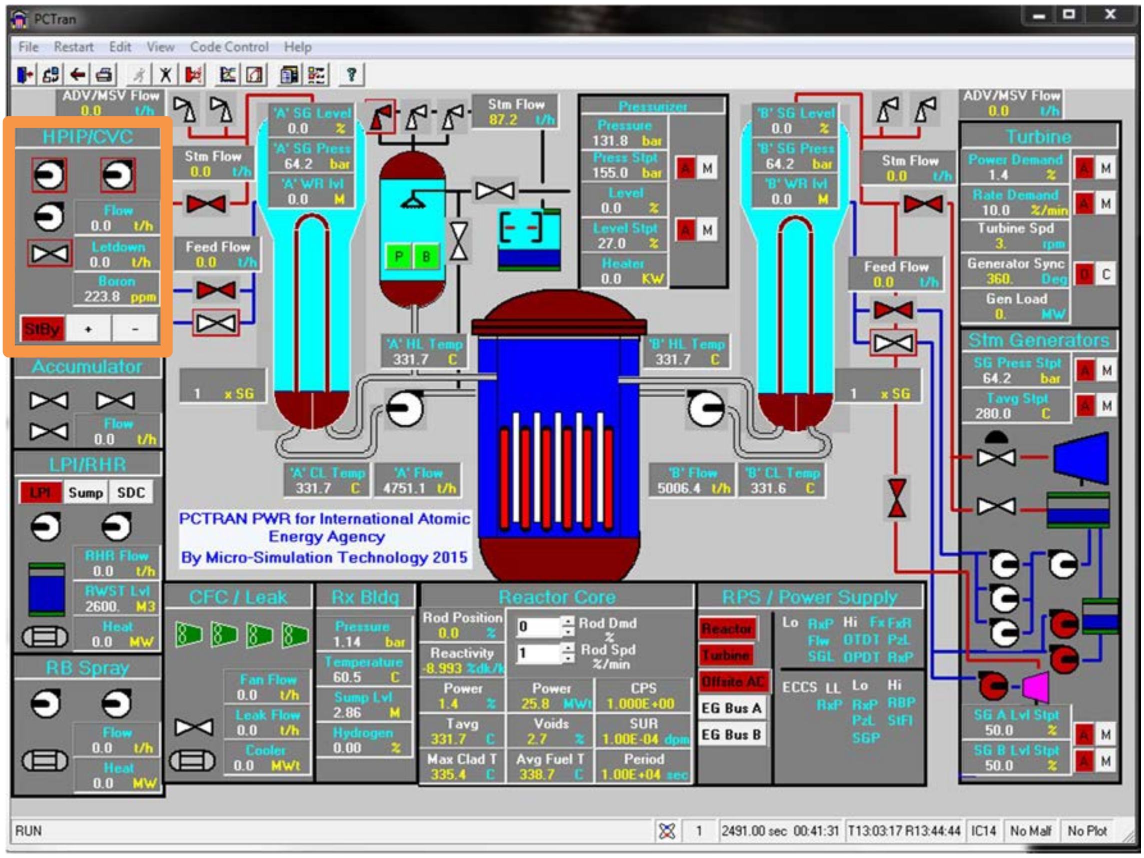

FIG. 73. HPIPs manually turned off.

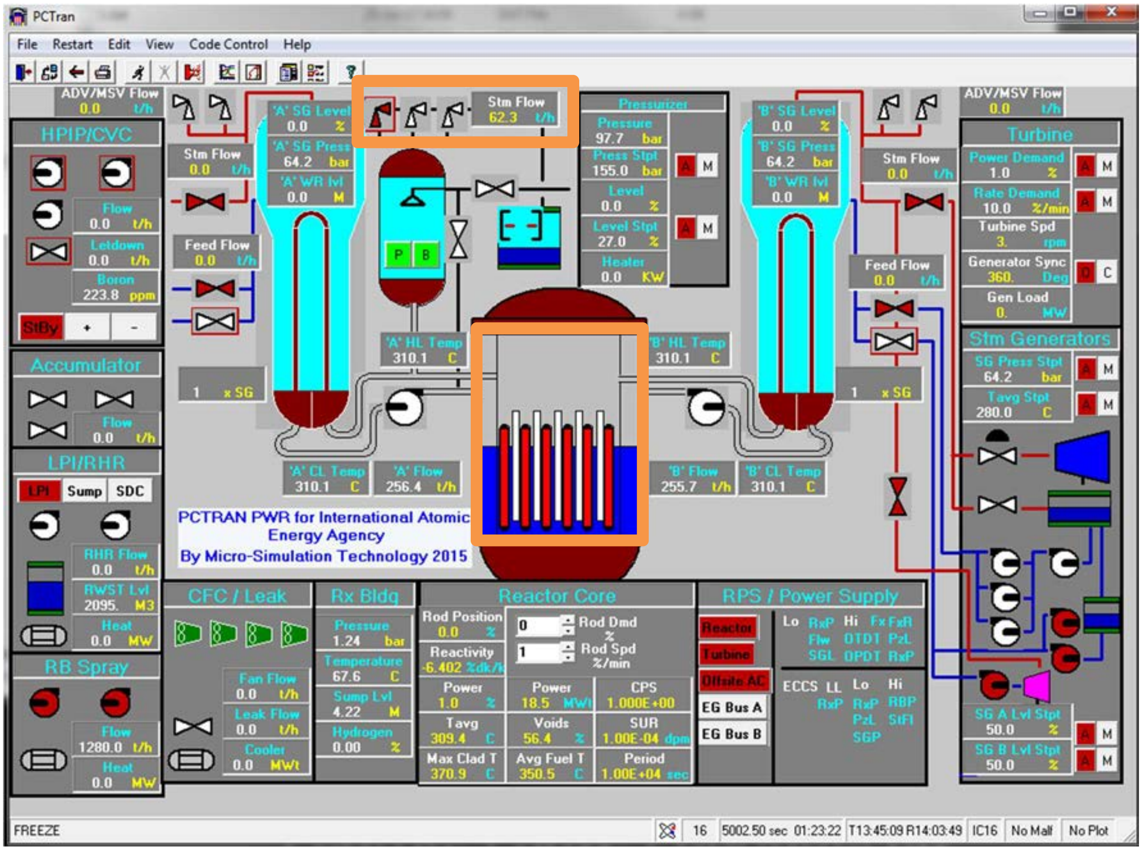

FIG. 74. Continuous coolant loss and core uncovery.

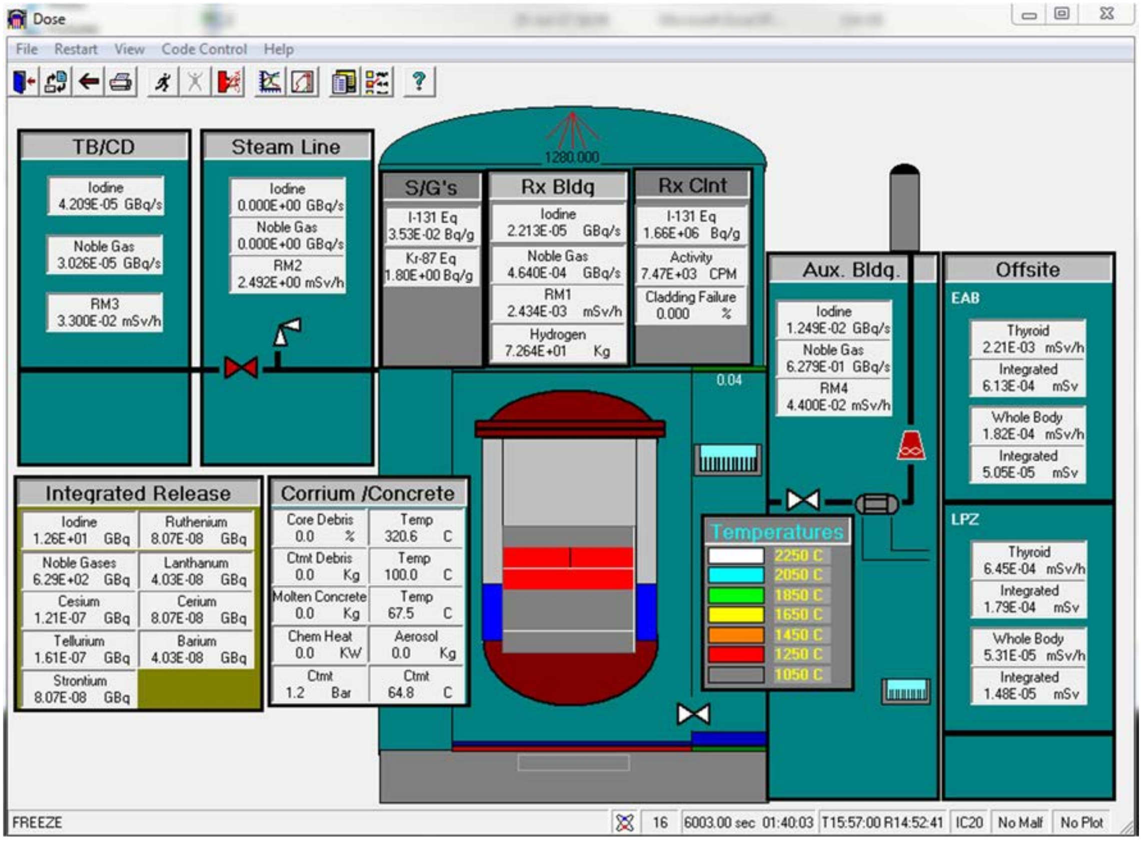

FIG. 75. The core melting starts (dose mimic).

4.1.2. Transient behaviour description

In this exercise, the TMI-2 type accident simulation is analysed up to 6000 seconds. Upon the condensate pump and the MFWPs trip, which is the beginning of the event, the SG water level starts to lower until it reaches the reactor trip setpoint at about 40 seconds. Due to the reduced heat sink, both reactor coolant temperature and pressure increase until the reactor trip. When the reactor trips, a sudden drop in reactor coolant system liquid volume and pressure is observed. However, due to the decay heat generation in the core and continuous loss of the heat sink, the RCS pressure rises again despite the operation of the pressurizer spray. When the PORV is stuck open, steam is released through the valve and the reactor coolant system is rapidly depressurized.

Through the stuck-open PORV, the reactor coolant continuously escapes in the form of steam and the reactor coolant system liquid volume lowers. As the HPIS is manually turned off, and the reactor coolant water boils, the core is uncovered and the fuel temperature starts to rise. Eventually, when the peak cladding temperature is high enough to interact with the steam and generate hydrogen, the hydrogen concentration in the reactor building begins to rise.

4.1.3. Transient plot description

Figure 76 presents the reactor coolant system pressure change during 6000 seconds of the simulation. After the initial pressure drop, which is attributed to the reactor trip, the pressure increases until it reaches the setpoint for PORV opening at about 1500 seconds.

Once opened, the PORV is stuck-open due to its mechanical malfunction, even after the pressure reaches the PORV closing setpoint. As the coolant is continuously escaping through the stuck-open PORV, the pressure decreases until all coolant dries up.

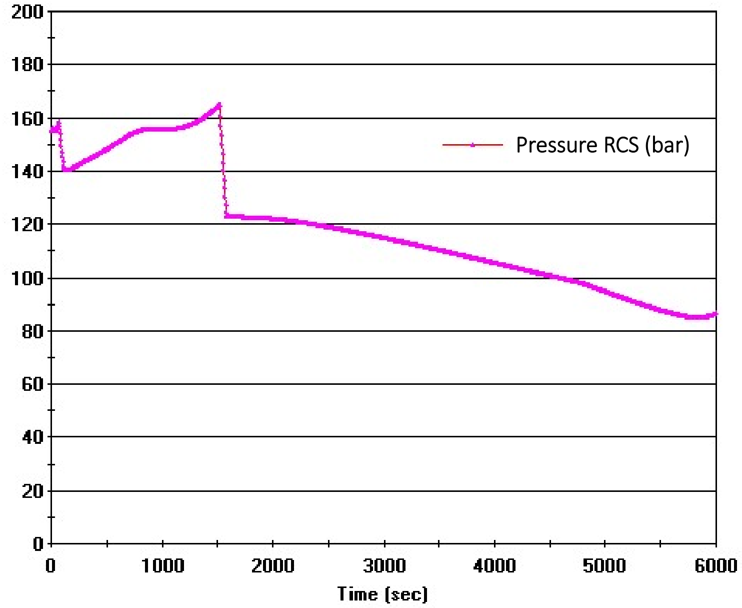

FIG. 76. Reactor coolant system pressure (bar).

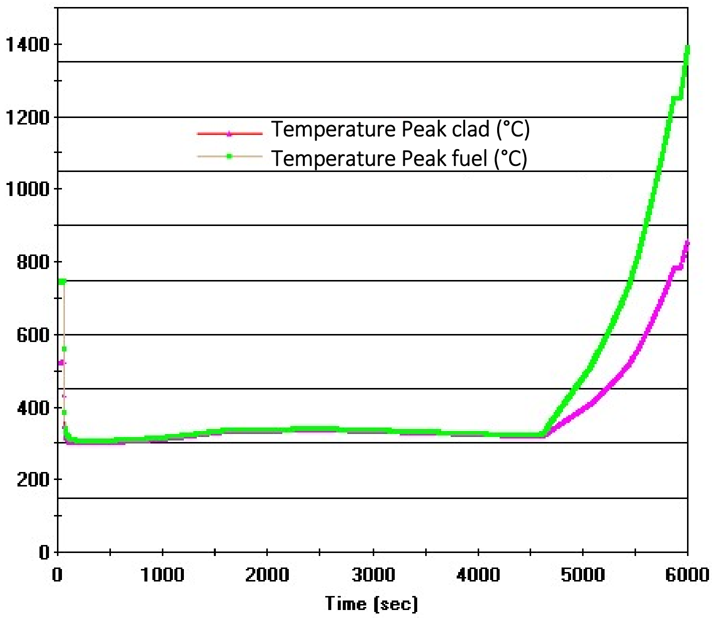

FIG. 77. Peak cladding and fuel temperatures (°C).

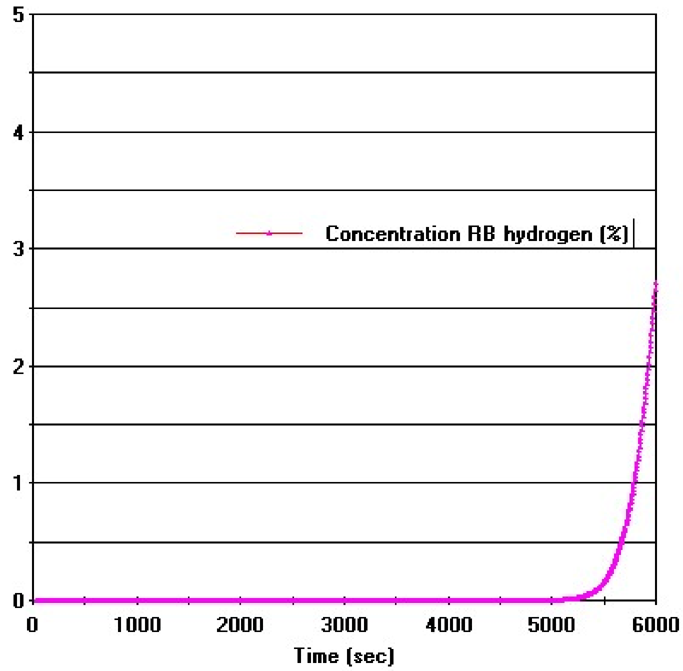

FIG. 78. Reactor building hydrogen concentration (%).

Following the initial sharp decrease in fuel and clad temperature shown in Figure 77, which is the result of reactor trip, submerged fuel elements are successfully maintained a constant temperature around 300°C.

However, upon the core uncovery (~4500 seconds), the peak fuel temperature as well as the peak clad temperature start to soar and the hydrogen generation reaction is accelerated, threatening the fuel integrity.

After the Zircaloy cladding temperature rapidly increases, the hydrogen generation reaction between zirconium and water is accelerated and the hydrogen concentration in the reactor building increases (Figure 78). The light hydrogen gas is accumulated at the top of the reactor building and when its concentration exceeds 5%, it may explode from even a small spark.

4.2.COLD LEG LBLOCA WITHOUT ECCS

4.2.1. Simulation description

In PWRs, in case of LOCA accident, the ECCS is automatically actuated to keep the reactor core submerged and cooled. In this exercise, a LBLOCA in a cold leg is simulated with disabled ECCS. While the reactor coolant is discharged through a break, there is no cooling water injection. The steps performed are similar to exercise 3.5, with the difference of ECCS unavailability. This, in turn, leads to complete core uncovery and eventually a severe accident with core meltdown. This scenario consists of a design extension condition (DEC) with significant fuel degradation [3].

Table 13 provides a step-by-step procedure for simulation with corresponding values of relevant variables. In Step 3, all HPIS, LPIS, and accumulators are manually disabled. Please note that accumulators are passive devices, however, for this hypothetical exercise it is assumed that they are also not available. In Step 4, a double-ended guillotine break is initiated in a cold leg with ‘Malfunction #2’. Upon resuming the simulator, massive amounts of reactor coolant are discharged through the cold leg break and the reactor coolant system is rapidly depressurized (Figure 79). However, due to the disabled ECCS, the reactor coolant system inventory is ever decreasing without refilling the system. In Step 8, the core is uncovered. In Step 9 and 10, users can observe the core melting and vessel failure in the dose mimic view. Finally, in Step 10, after the vessel failure, the molten core–concrete interaction (MCCI) is observed.

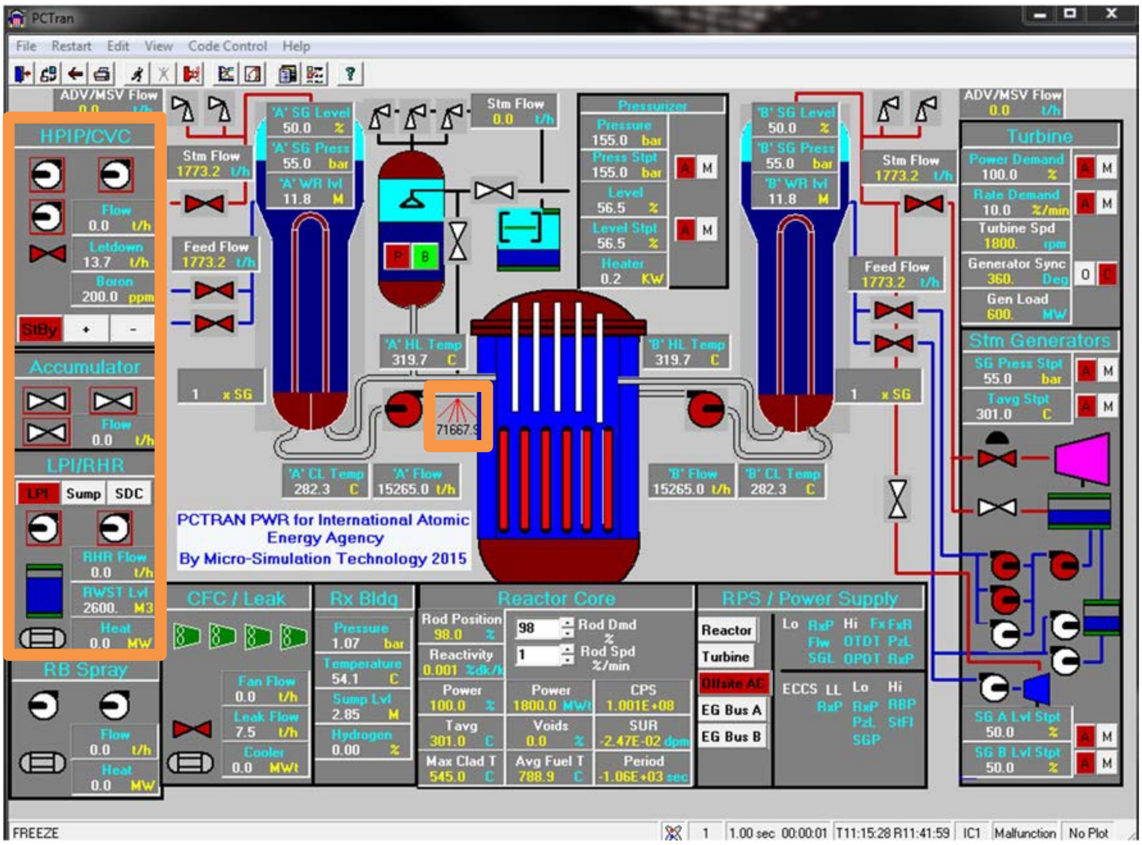

FIG. 79. The disabled ECCS and a cold leg LBLOCA initiated.

TABLE 13. COLD LEG LBLOCA WITHOUT ECCS

STEP	PROCEDURE STEPS	EXPECTED RESPONSE	FIGURE
Setup	Load initial condition #1 100% POWER EOC or #2 100% POWER MOC or #3 100% POWER BOC. Initial condition #1 is used for the demonstration.		
1	Run the simulator for about 5 seconds to achieve a steady-state condition.		
2	Freeze the simulator.	The simulation freezes.	
3	Manually disable all HPIPs, LPIPs, and accumulator valves.	Red boxes will appear around the pumps and valves indicating inoperable condition.	
4	Initiate a cold leg LBLOCA by selecting 'Malfunction #2'. To simulate 2800 cm^2 (~3ft^2) break, insert 2800 in the failure fraction box and check the 'active' checkbox.		
5	Resume the simulator (*speed up the simulator if needed*).	A rupture icon appears in cold leg-A with the primary coolant loss rate (t/hr).	Figure 79
6	Observe the reactor trip on low reactor pressure, followed by turbine trip.	The auxiliary feedwater system starts operating.	Figure 80
7	Check the following variables changing: (1) Decrease in pressurizer level due to the coolant loss (%) (2) Decrease in RCS pressure (bar) (3) Increase in reactor building pressure (bar) (4) Increase in reactor building air temperature (°C) (5) Increase in reactor building sump level (m)		
8	Observe the core uncover at about 25 seconds. Check the following variables changing: (1) Average fuel temperature (°C) (2) Maximum cladding temperature (°C) (3) Hydrogen concentration in the reactor building (%)	 (1) Increases (2) Increases (3) Increases	Figure 81
9	Go to 'View'→ 'Dose Mimic'. Observe the core melt starting in the mid part of the core.		Figure 82
10	Continue the simulation and observe the vessel penetration and the Molten Core Concrete Interaction (MCCI) in the dose mimic view at about 4000 seconds.	Molten core icon appears in the bottom of the reactor and after the vessel penetration, in the bottom of the reactor building. Upon the vessel penetration, a red rectangular appears in the bottom of the reactor building indicating the MCCI.	Figure 83 Figure 84

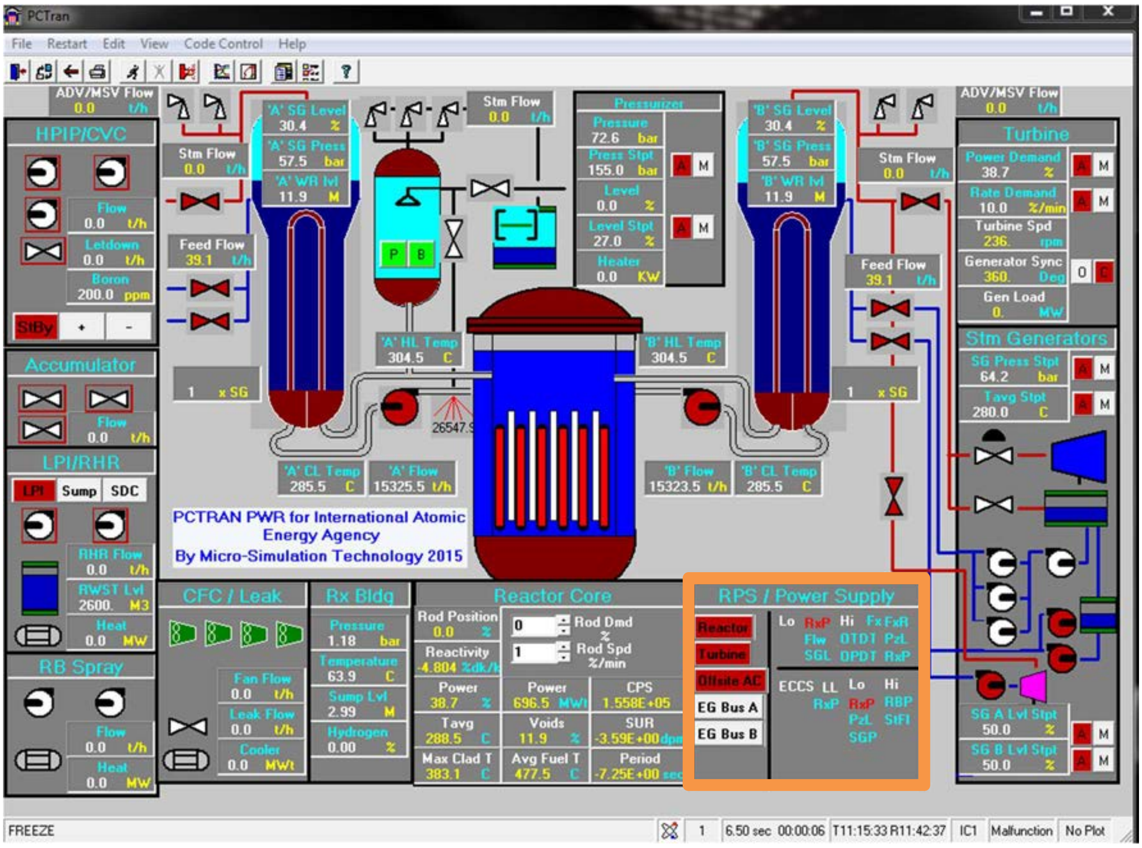

FIG. 80. Reactor trip on low reactor pressure.

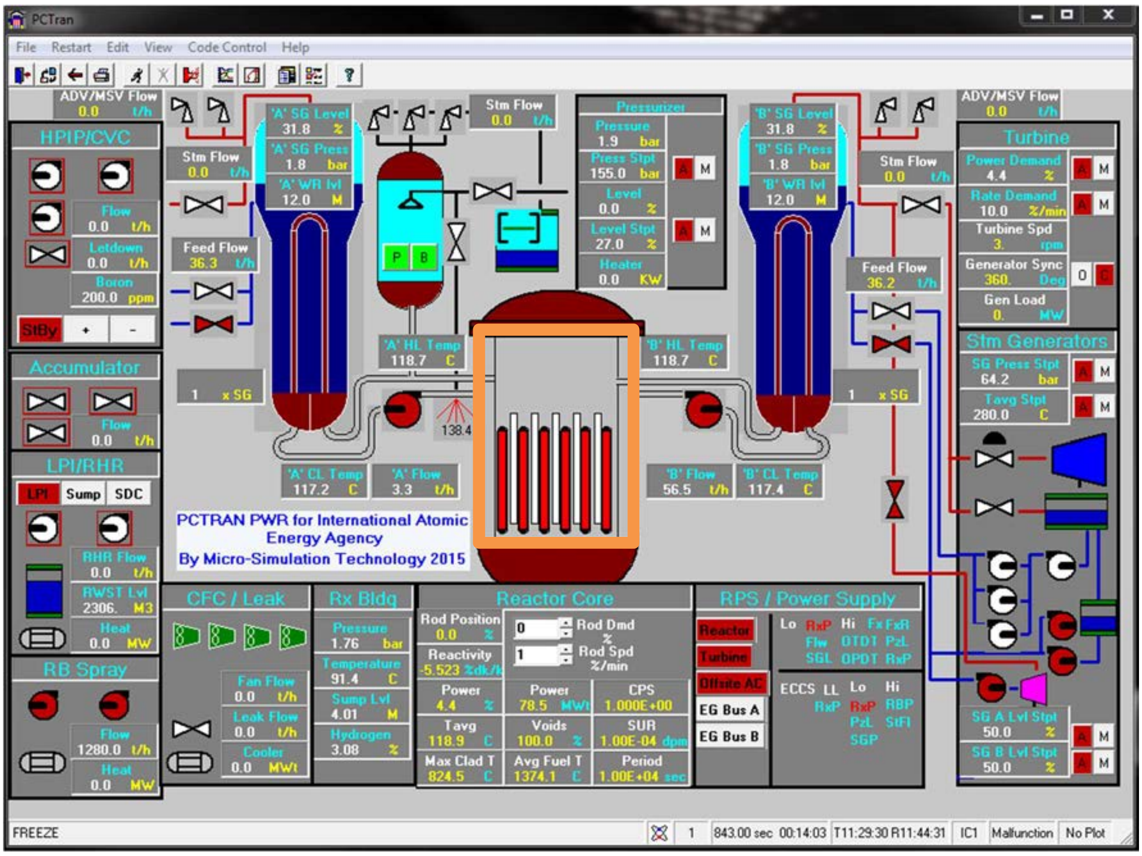

FIG. 81. Complete core uncovery.

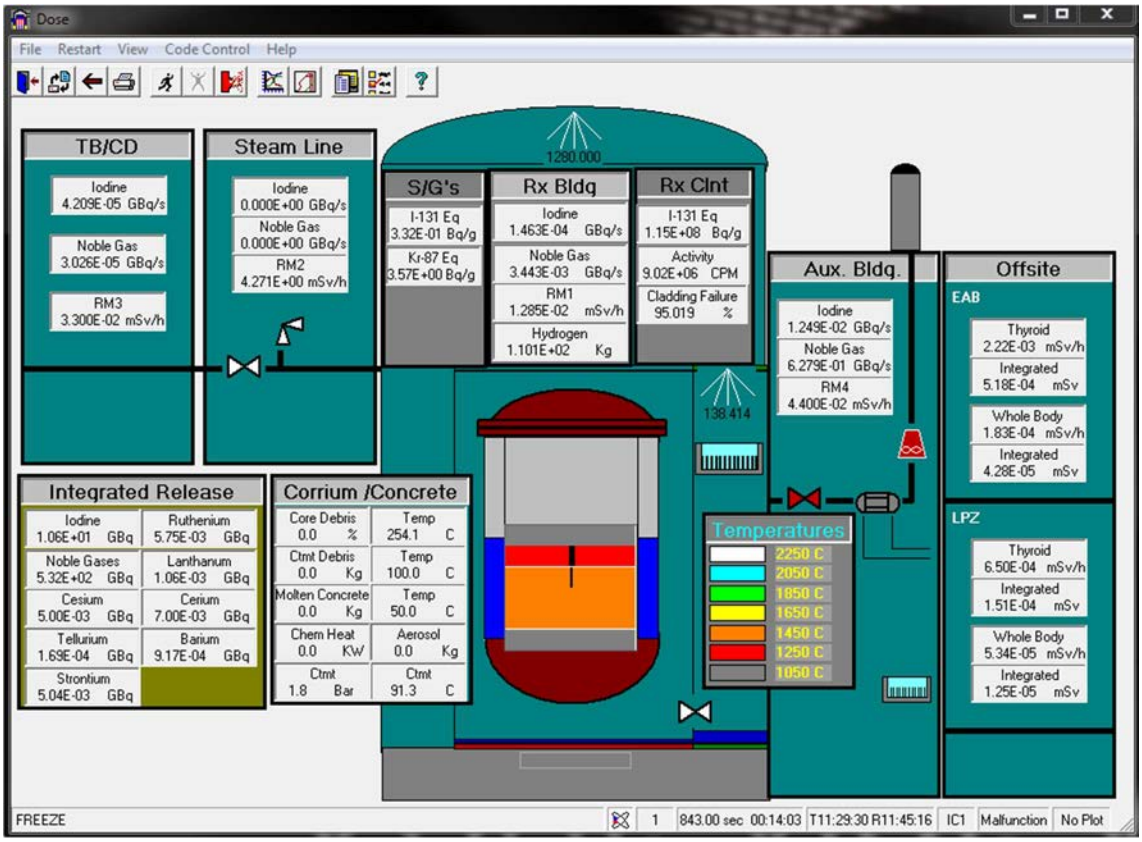

FIG. 82. The beginning of core melt (dose mimic).

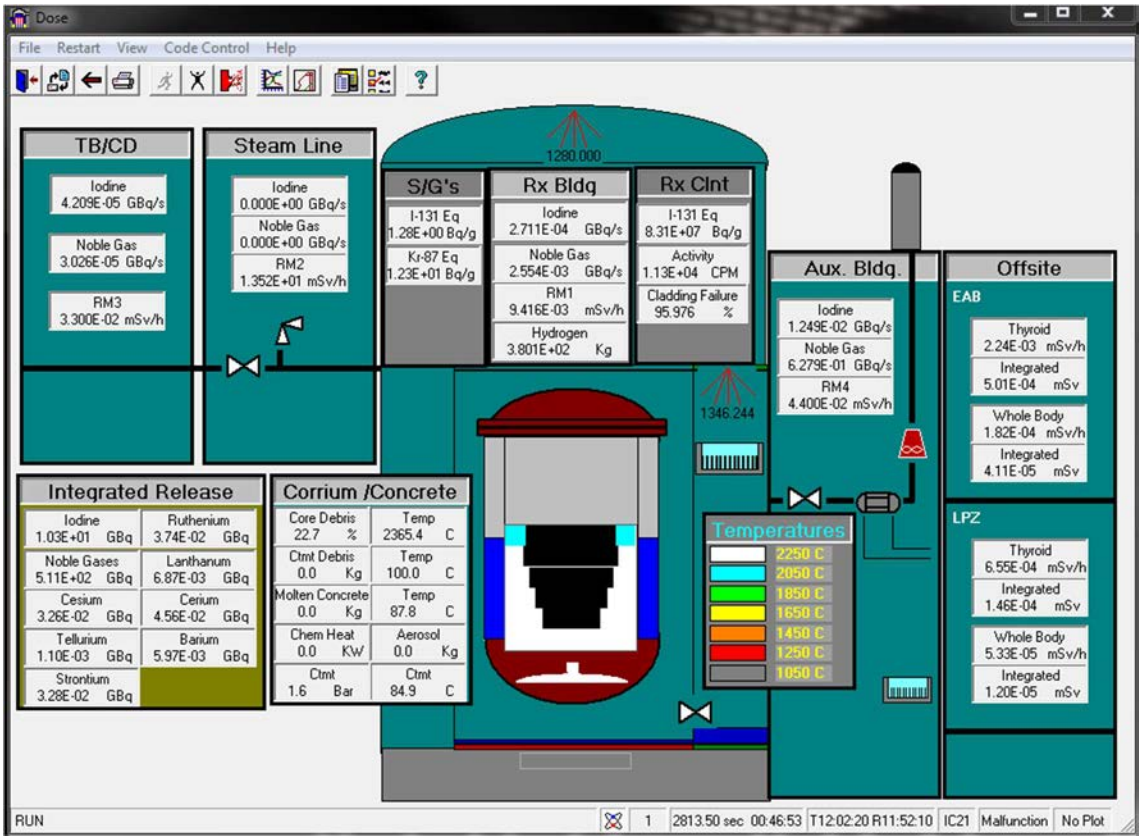

FIG. 83. Molten corium at the bottom of the core (dose mimic).

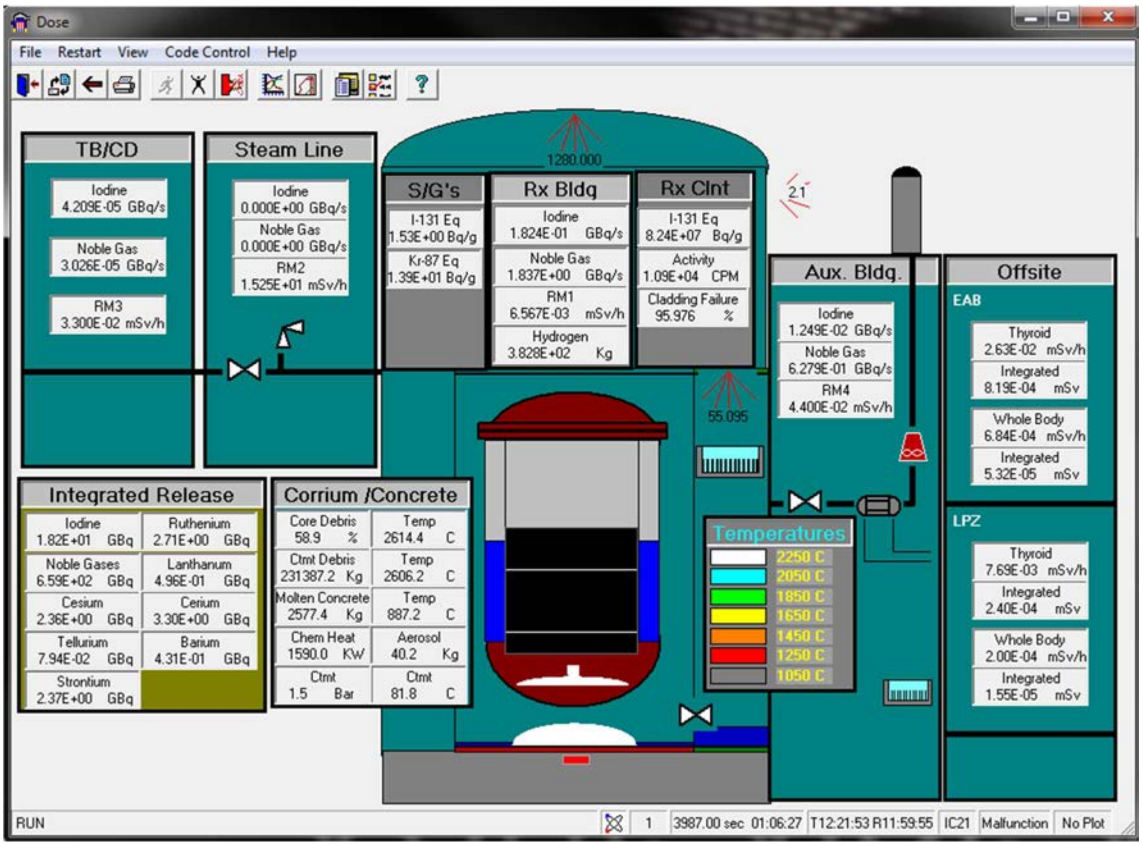

FIG. 84. The vessel has been penetrated and the Molten Core–Concrete Interaction (MCCI) has started (dose mimic).

4.2.2. Transient behaviour description

Following the initiation of LOCA accident, the reactor trips instantaneously and the core is uncovered within 20 seconds due to very rapid depressurization. Since no ECCS is available in this scenario, the fuel temperature continuously increases. With increased cladding temperature, the interaction between the cladding and steam is accelerated. Shortly after the core uncovery, the hydrogen concentration in the reactor building exceeds 5%. At this concentration, the hydrogen accumulated at the top of the reactor building may explode from an initiating spark and damage the reactor building. Users can simulate the hydrogen explosion with 'Malfunction 5, Spark Presence for Hydrogen Burn' (optional).

With further increase in fuel temperature, the reactor core starts to melt. Core melting can be seen in 'View'→ 'Dose Mimic'. Molten fuel may collapse into the bottom of the vessel. The vessel lower head may then heat up to the melting point, too. The molten debris may drop into the containment cavity floor. During the fuel damage process, first the fission gas in the cladding may leak out. Later, if the fuel and cladding continue their degradation, radionuclides will also be released. In addition to iodine and noble gases, there are alkali metals, tellurium, barium, cerium, lanthanides, etc. The elevated concentration of these radionuclides would find their ways through the vessel break, relief valves, and containment leakage into the environment.

In the event that the molten core heats up the vessel bottom and melts through it, the debris, called corium, falls into the reactor cavity (drywell pedestal for BWR). The molten metal interacts with concrete and forms a slump. At lower temperatures, degassing of concrete occurs and both steam and carbon dioxide can be released. At higher temperatures concrete can also be molten and mixed with metals.

4.2.3. Transient plot description

Due to the 2800 cm^2 break in the cold leg, the reactor coolant system pressure decreases rapidly until it reaches the reactor building pressure.

Steam generation stops with the loss of coolant due to the lack of heat transfer in the SGs, causing the secondary system pressure to decrease as well (Figure 85).

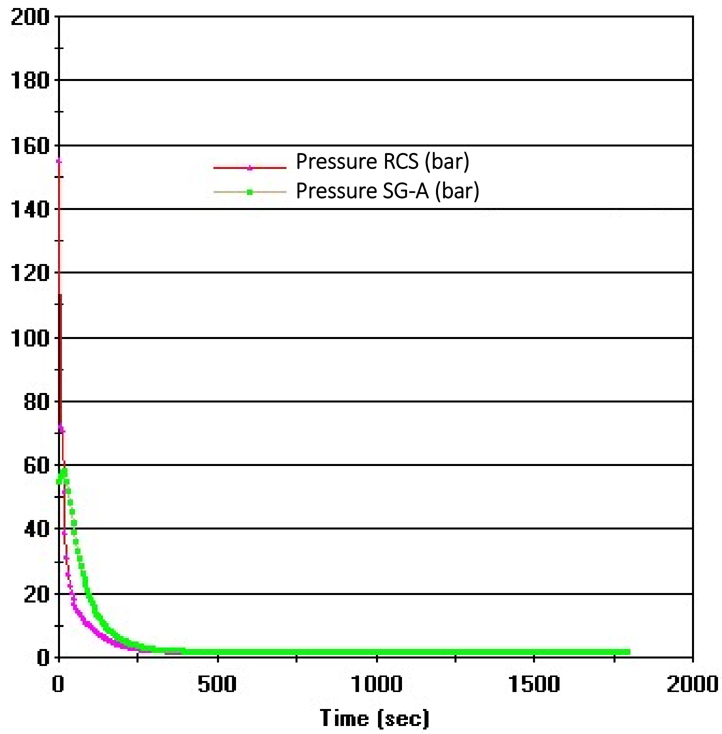

FIG. 85. Reactor coolant system and SG-A pressure (bar).

With the core uncovery, fuel and the cladding temperature increase rapidly (Figure 86) and reach the melting points of UO_2 fuel and zircaloy cladding.

After they reach the core melting temperature, core cavity will be visualized in the dose mimic view as shown in Figure 82.

Figure 87 shows the hydrogen concentration in the reactor building and the fraction of Zircaloy cladding oxidation. Zircaloy cladding and steam at high temperature interact and oxidise zirconium. As a result of this interaction, hydrogen is produced and accumulates in the reactor building. Furthermore, as the zirconium oxidation interaction (equation 1) is an exothermic reaction, it contributes to the increases in the maximum cladding temperature.

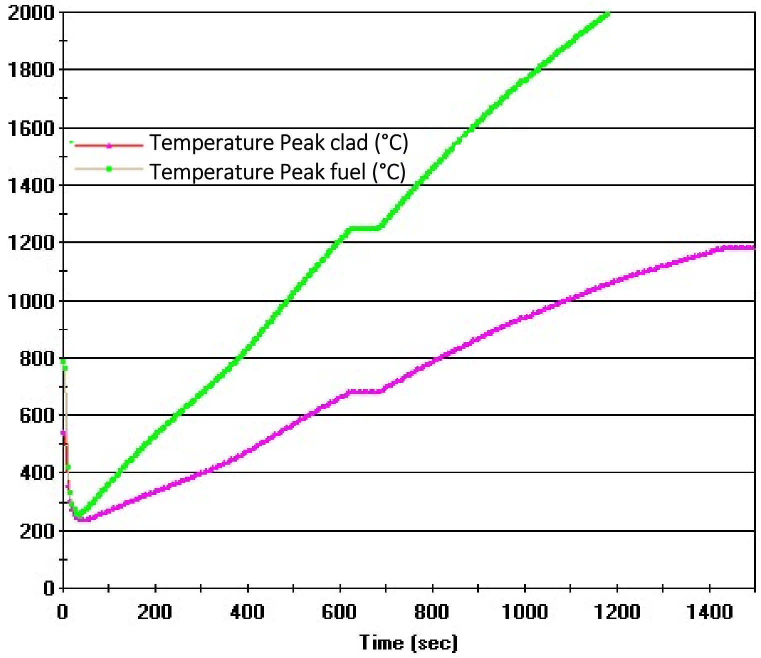

FIG. 86. Fuel and clad peak temperature (°C).

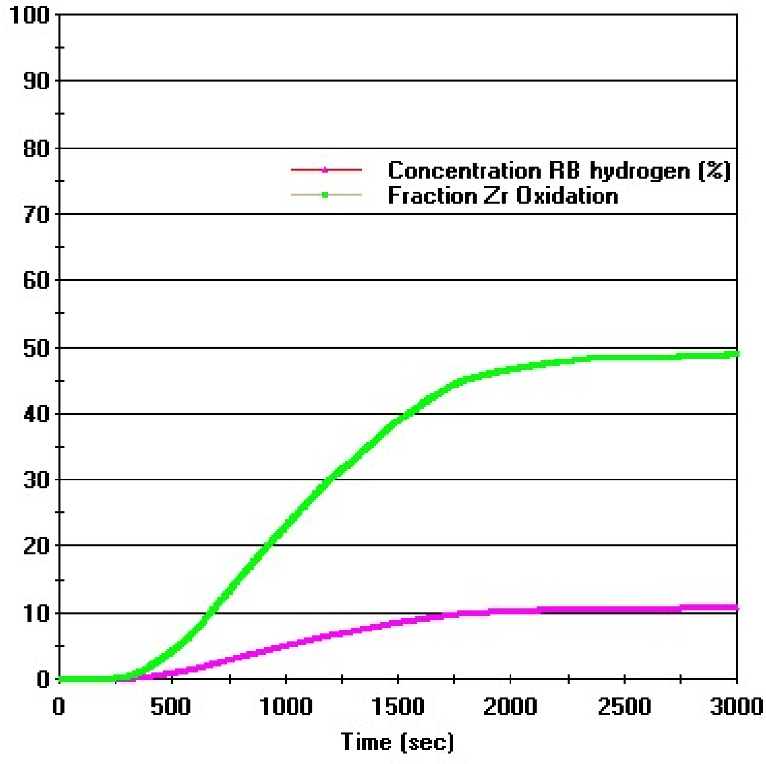

FIG. 87. Reactor building H_2 concentration and zirconium oxidation (%).

Since pressurized coolant leaves the primary system through the break in the cold leg, it increases the reactor building pressure as well as the reactor building temperature as shown in Figure 88.

When the containment building pressure reaches the reactor building spray setpoint of 1.3 bar, the spray at the top of the containment is actuated, and keeps the containment building from being overheated.

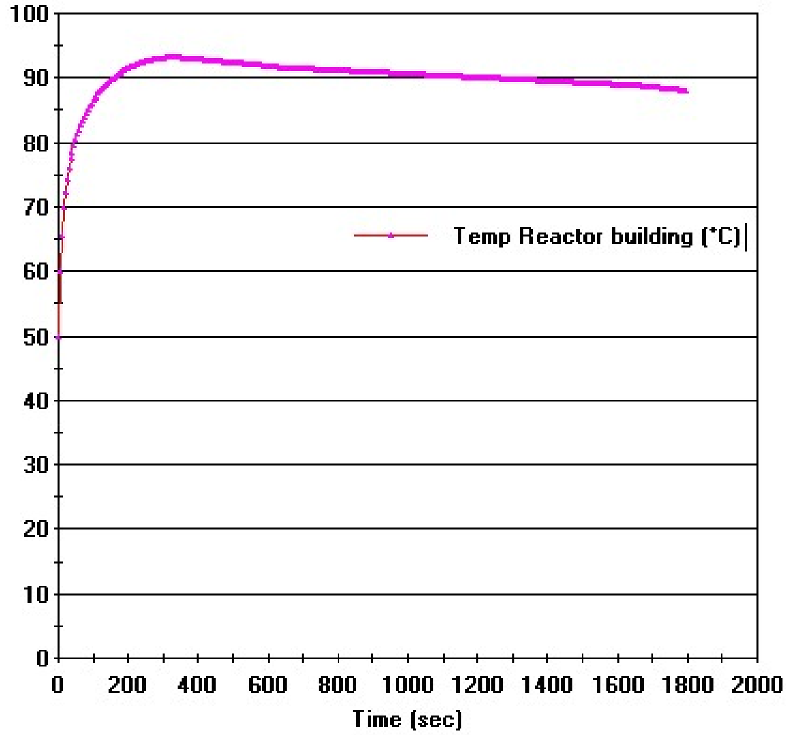

FIG. 88. Reactor building air temperature (°C).

4.3. PROLONGED STATION BLACKOUT WITHOUT AFW

4.3.1. Simulation description

The term 'station blackout' (SBO) refers to the complete loss of alternating current (AC) electrical power to a nuclear power plant. Not only is the offsite AC lost but the onsite emergency system is lost as well. Since many safety systems for reactor core residual heat removal and containment heat removal rely on AC power, prolonged station blackout might eventually lead to a severe accident in this design of PWR. In this exercise, the turbine driven auxiliary feedwater pump is assumed to be unavailable as well. This scenario would progress to a design extension condition (DEC) with significant fuel degradation [3].

Table 14 provides a step-by-step procedure for simulation with corresponding values for relevant variables. In Step 3, the offsite AC power is manually cut off. In Step 4, the turbine driven auxiliary feedwater pump is manually disabled. In Step 6, due to the disabled RCPs, the reactor trip signal is generated on low reactor coolant flow rate. In Step 8, increased steam pressure opens the steam generator atmospheric dump valve through which the secondary side steam is discharged. Similarly, in Step 9, the reactor coolant escapes from the reactor coolant system through the pressurizer relief valve. In Step 11, the reactor core is uncovered, and the fuel temperature starts to increase.

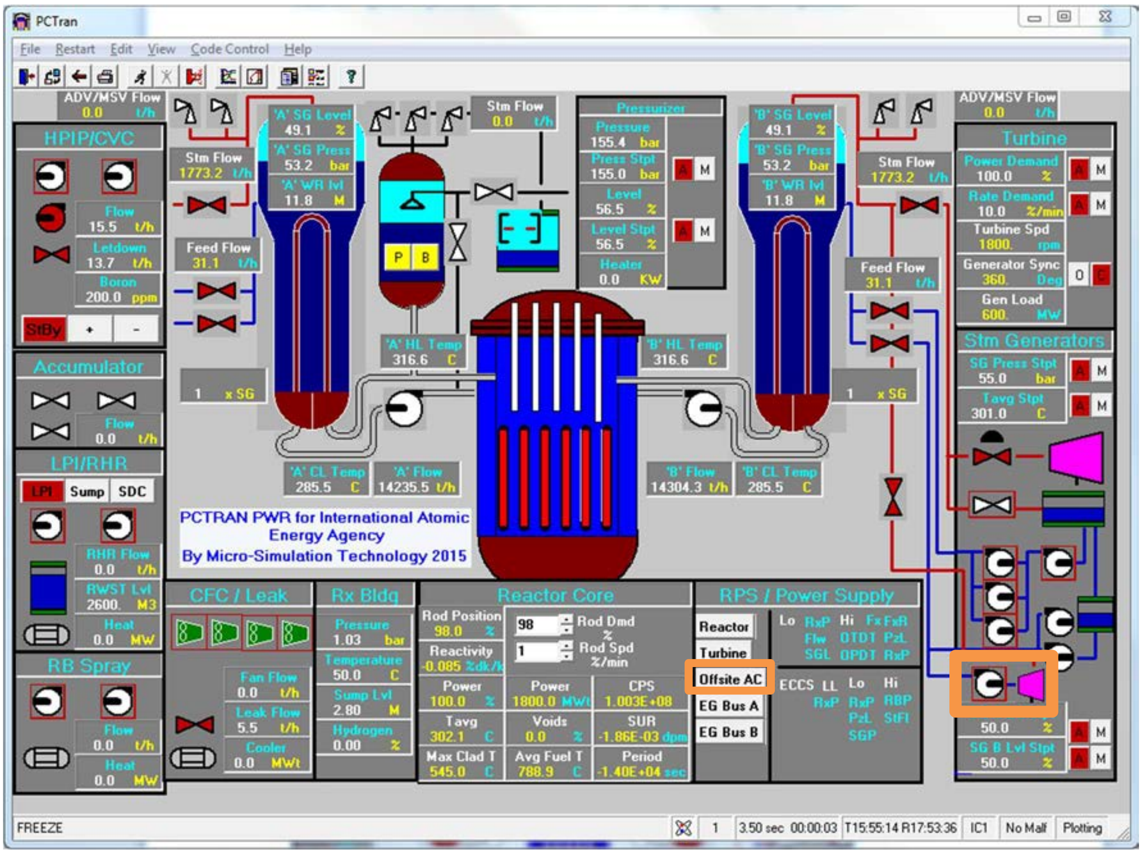

FIG. 89. Station blackout condition.

TABLE 14. PROLONGED STATION BLACKOUT WITHOUT AFW

STEP	PROCEDURE STEPS	EXPECTED RESPONSE	FIGURE
Setup	Load initial condition #1 100% POWER EOC or #2 100% POWER MOC or #3 100% POWER BOC. Initial condition #1 is used for the exercise.		
1	Run the simulator for about 5 seconds to achieve a steady-state condition.		
2	Freeze the simulator.	The simulator freezes.	
3	Click the 'Offsite AC' icon in the Reactor Protection System (RPS) section to cut off the offsite AC power.	Red boxes will appear around the HPIPs, LPIPs, RB spray, MFWPs, containment fans and the steam bypass valve icons indicating inoperable condition.	Figure 89
4	Manually disable the turbine driven auxiliary feedwater pump.	Red box will appear around the turbine driven auxiliary feedwater pump indicating inoperable condition.	Figure 89
5	Resume the simulator (*speed up the simulator if needed*).		
6	Observe the reactor and the turbine trip on low flow rate.		Figure 90
7	Check the following variables changing: (1) Steam generator level (%) (2) Steam generator pressure (bar) (3) Pressurizer pressure (bar) (4) Pressurizer level (%)	(1) Increases (2) Increases (3) Increases (4) Increases	
8	Observe the atmospheric dump valve opening when the steam generator pressure reaches 81.8 bar.		
9	Observe the pressurizer relief opening when it reaches the opening setpoint of 165 bar.	The reactor coolant is discharged through the relief valve.	Figure 91
10	Observe the steam generators boiled dry.		Figure 92
11	Observe core uncovery.		Figure 92

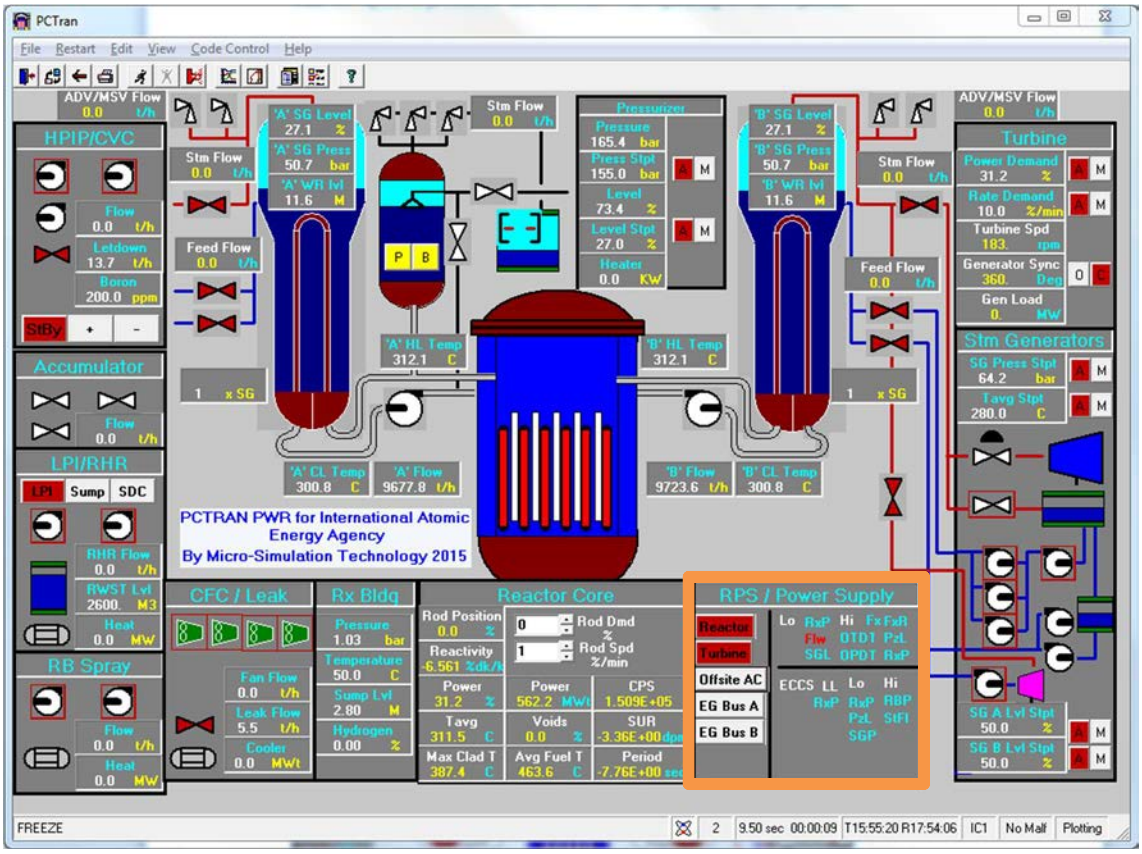

FIG. 90. Reactor trip on low flow rate.

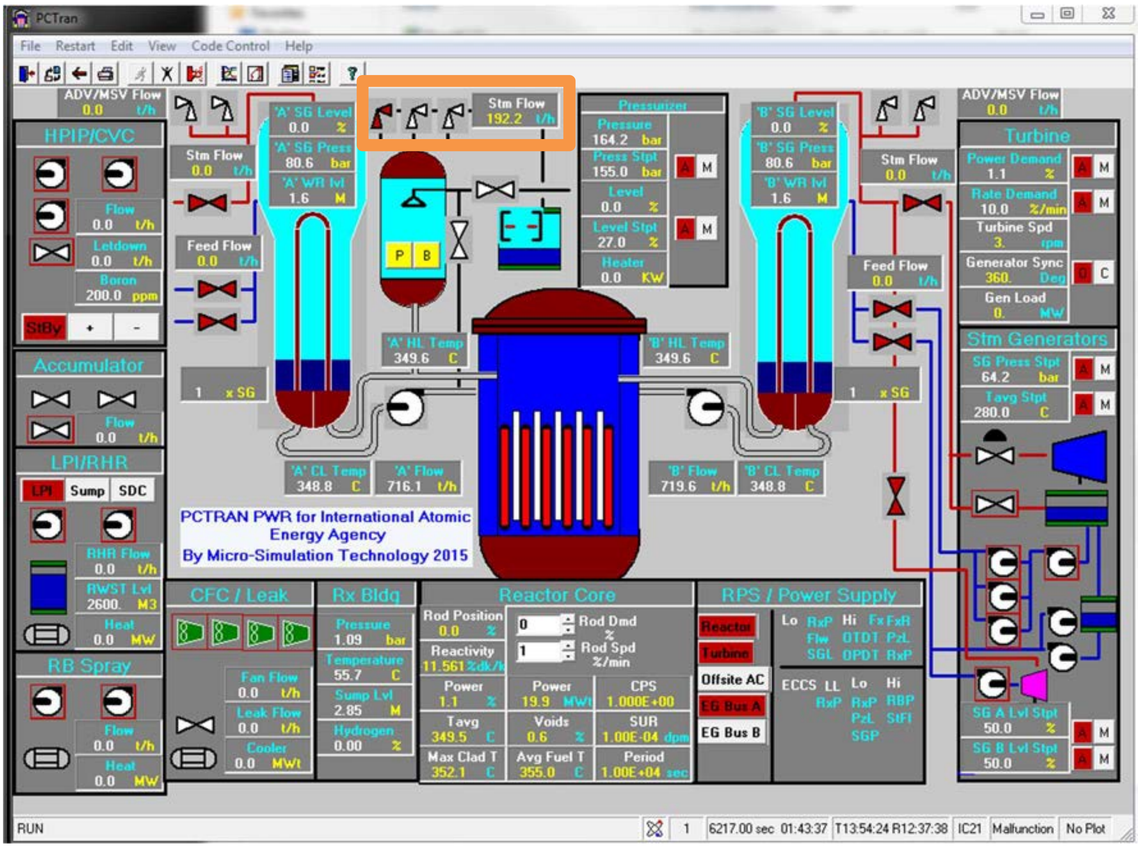

FIG. 91. The coolant loss through the relief valves.

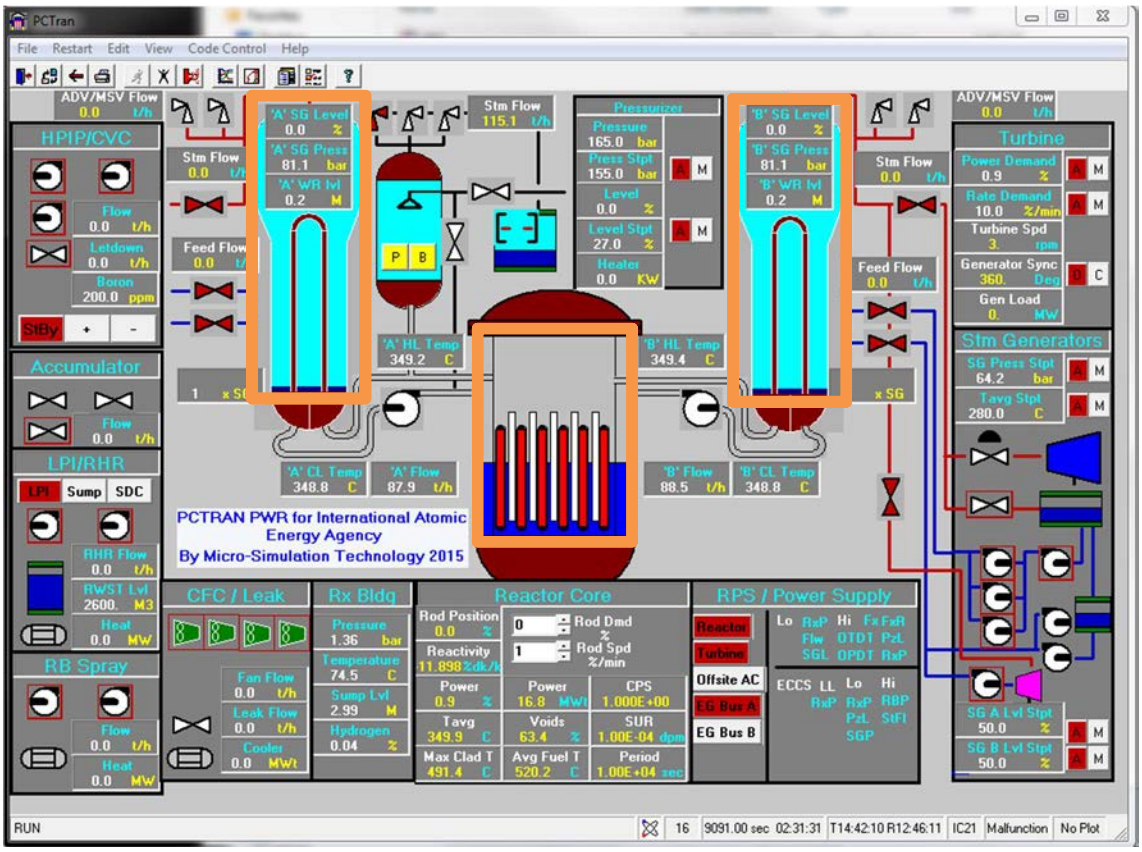

FIG. 92. The SG water exhausted, and the core is uncovered.

4.3.2. Transient behaviour description

Within a few seconds after the initiation of SBO, the reactor trip signal is generated due to low reactor coolant flow rate. At first, the RCS pressure increases up to about 165 bars since the RCP trip causes an increase in reactor coolant temperature. In the secondary side of SGs, since feedwater and steam flow do not exist, evaporated SG water pressurizes the SGs and is slowly discharged through the MSSVs (main steam safety valves).

As the heatsink is lost, the decay heat in the core is not well-transferred to the secondary side and thus, the RCS pressure rises until the relief valve is open to release the steam inside the pressurizer (3000–4000 seconds). Even after the opening of the relief valve the reactor coolant is continuously heated up and the RCS pressure is maintained high enough to keep the relief valve open. As a result, RCS liquid volume decreases, and the reactor core is uncovered at about 8000 seconds.

4.3.3. Transient plot description

Figure 93 shows the RCS and SG-A pressure during the 10000 seconds of the event. First, due to the RCPs trip, the RCS pressure slightly increases from 155.5 bar to about 160 bar. After the RCPs trip, natural circulation is established in the RCS and the decay heat in the core is transferred to the SGs. Until about 2800 seconds, the natural circulation and water left in the SG are sufficient to remove the decay heat in the RCS and the RCS pressure decreases. However, as the turbine has been tripped and the FWPs and AFW system are unavailable, both steam and feedwater flows stop and heated up SG steam is discharged through MSSVs. As the heat sink is lost, the heat removal rate of the SGs decreases (see Figure 96). Consequently, the RCS temperature and pressure rises. When the RCS pressure reaches the PORV opening pressure (165.2 bar) at about 4000 seconds, reactor coolant is discharged and pressure is maintained around 165 bar with little fluctuation (due to the constant PORV opening and closing). The sudden rise at the end could be attributed to limitation of the pressurizer relief tank to take any more steam.

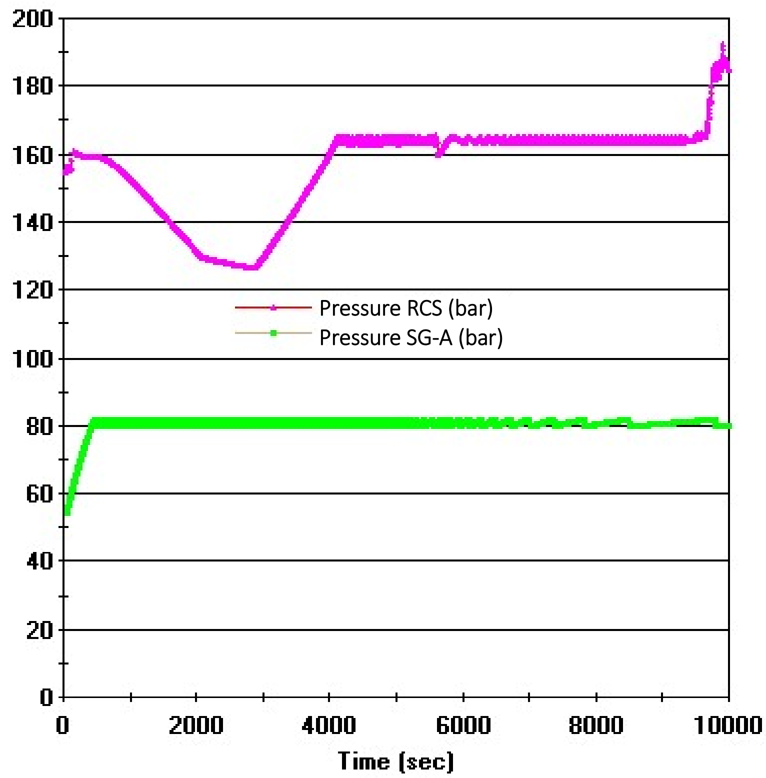

FIG. 93. Reactor coolant system and SG-A pressure (bar).

Figure 94 shows the SG-A water level during the simulation. During the normal operation before the SBO, the water level is 11.8 m. Shortly after the initiation of SBO, the water level slightly drops due to the reactor trip. As the feedwater pumps are not working, the SG pressure keeps on increasing until it reaches the MSSVs opening setpoint of 81.8 bar. At this point, the steam is discharged through MSSVs and the SG water level decreases until the SG completely dries up (Figure 95) and loses heat removal capabilities (Figure 96).

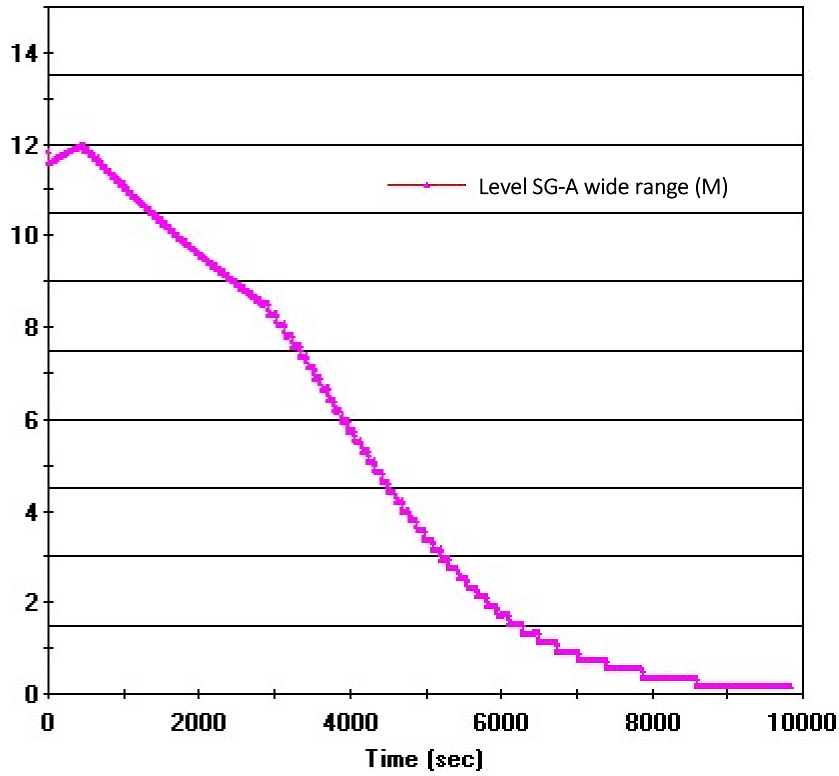

FIG. 94. SG-A water level (wide range) (m).

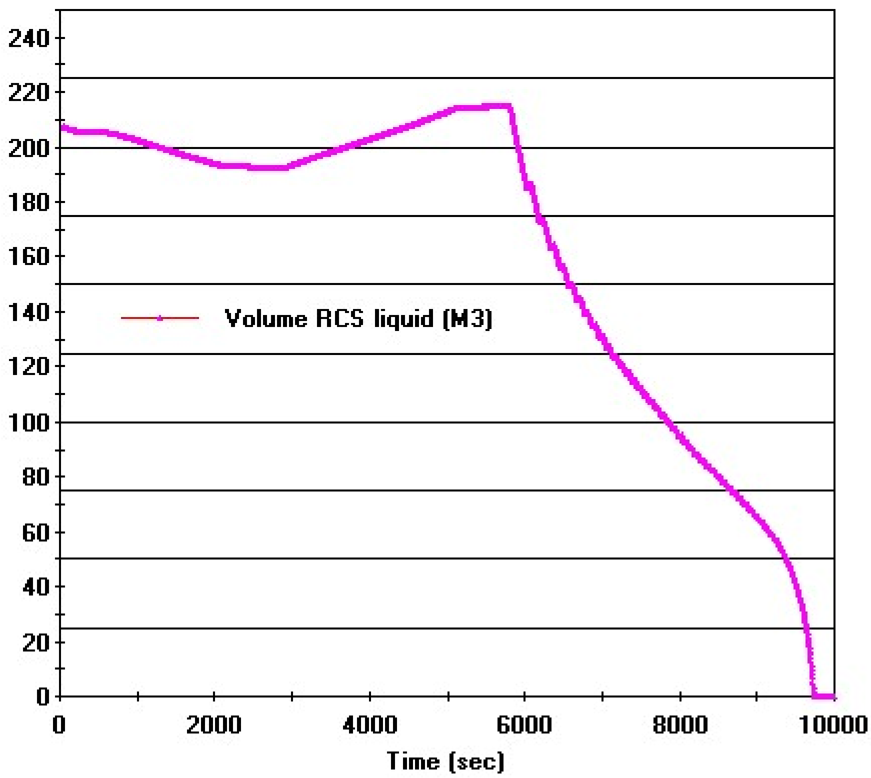

FIG. 95. RCS coolant volume (m^3).

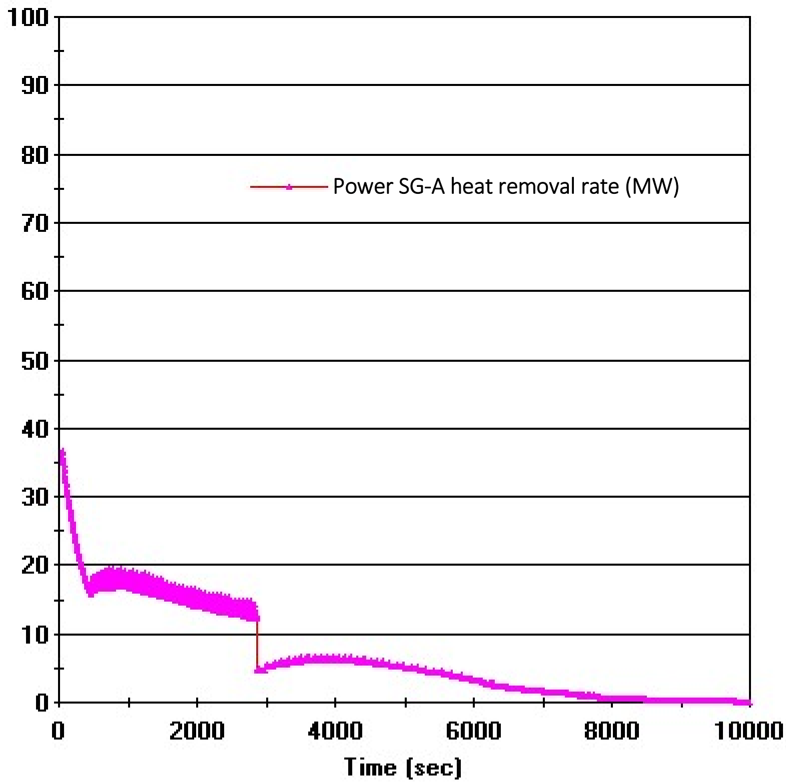

FIG. 96. SG-A heat removal rate (MW).

APPENDIX

A.1. IAEA PCTRAN SETPOINTS FOR PLANT COMPONENTS

Component	Description	Value
Pressurizer	PORV opening setpoint (bar)	165.2
	PORV reseat setpoint (bar)	163.2
	Safety valve open setpoint (bar)	176
	Safety valve reseat setpoint (bar)	175
	PRZ spray initiation error (bar)	0.6
	Proportional heater initiation error (bar)	0
RPS	High pressure reactor scram setpoint (bar)	165.7
	Low pressure reactor scram setpoint (bar)	132.2
	Low SG narrow range scram setpoint (%)	17
	High neutron flux reactor scram setpoint (fraction of full power)	1.18
	Low core flow reactor scram setpoint (fraction of full flow)	0.87
	High-high SG level turbine trip setpoint (%)	82
SG	SG relief valves opening setpoint (bar)	81.8
	Low T_{avg} SG isolation setpoint (°C)	281
ECCS	HPI automatic start setpoint (bar)	129.69
	Accumulator initiation pressure setpoint (bar)	43
	LPI system initiation pressure (bar)	11.36
	High RB press for SI initiation (bar)	1.3
	Simultaneous low PZR level with RCS pressure for SI initiation (fraction of full)	0.15
	Simultaneous low RCS pressure with PZR level for SI initiation (bar)	128
	Low SG press for SI initiation (bar)	38

A.2. IAEA PCTRAN BASIC DATA VARIABLES

Variable	Value	Description
POWER	1800	Rated Thermal Power (MW)
P0	155	RCS initial pressure (Bar)
T0	301	RCS initial average temperature (°C)
WRC0	30530	Total core flow rate (t/hr)
TCST	15	RCP coastdown time (sec)
RCP	10	Total RCP heat input (MWt)
APORV	11.8	Area of PORV (cm^2) per valve
ASAFT	23	Area of pressurizer safeties (cm^2) valves combined
PCSP	1.3	Containment spray initiation pressure (bar)
TEFW0	60	EFW initiation delay time after initiation signal (sec)
AL0	0.565	Initial pressurizer level (fraction of full)
TEMP	25	HPI and EFW temperature (°C)
ADV	261.09	SG relief valves total capacity (t/hr)
WLD0	13.7	Letdown flow for CVCS during normal operation(t/hr)
CHG	30	Charging flow (t/hr)
TRIN	5	Rod insertion minimum time (sec)
TROT	2	Rod withdrawal minimum time (sec)
AKRD	1.333	Reactivity worth of one rod ($)
AKCHG	-10	Accumulator/RWST boron activity (pcm/ppm)
GFW	0.5	Feedwater controller constant, a dimensionless number
VSG	565	Total SG volume (m^3)
TBV	0.4	Total TBV flow capacity in fraction of full power steam
HMFW	950	Feedwater enthalpy at full power (kJ/kg)
TRXT	99999	Reactor trip delay time after turbine trip (sec)
TER1	9.18	Maximum T_{avg} error for steam dump control (°C)
EFW	120	EFW capacity for turbine driven pump (t/hr)
VPRZ	35	Pressurizer volume (m^3)
VRCS	180	Total RCS volume excluding pressurizer (m^3)
MSG0	80000	Total SG water inventory including steam mass (kg)
TSG0	280	No-load steam generator temperature or RC T_{avg} (°C)
SSMS	170	Reactor vessel stainless steel mass (ton)
UO2MS	56	Fuel UO_2 mass (ton)
WTR0	114.5	SG tube rupture flow per tube break (t/hr)
CFT	60	CFT tank (accumulator) total water capacity (m^3)
WCF0	35.3	Nominal CFT flow rate at initiation (t/hr)
ULSG	8.535	Height of SG U-tubes (m)
HRLV	0.17	Pressurizer low level setpoint for heater shutoff (fraction of full)
SPRY	20	PRZ spray flow capacity (t/hr)
VTAF	90	RCS volume to top of active fuel (m^3)
ATAF	7	RCS or core cross section area at top of fuel (m^2)
LSG0	11.83	SG wide range level at full power (m)

A.2. IAEA PCTRAN BASIC DATA VARIABLES (cont.)

Variable	Value	Description
PSG100	55	Steam generator pressure at 100% power (bar)
HTRB	662	Backup heater capacity (kW)
LPSG	2	Number of steam generator loops
GSTM	2	Turbine control valve and bypass valve gain constant
TER2	2.78	Deadband for T_{avg} steam dump control for load rejection (°C)
ASG1	2.254	SG lower section cross section area (m^2)
RLSG	5.1	Range for SG narrow range level instrument (m)
CFTN2	28.227	Total nitrogen volume in Core Flood Tanks or accumulators (m^3)
GCSP	640	Containment spray capacity (t/hr)
USTC	0.01	Core uncovery steam cooling effectiveness as fraction of water cooling
TF0	788.9	Average fuel temperature at full power (°C)
PDSN	3.089	Containment design pressure (bar)
WSV	2240	Main steam safety valve total capacity (t/hr)
PSV	85	Main steam safety valve opening press (bar)
TRB0	50	Initial containment temperature (°C)
PRB0	1.034	Initial containment pressure (bar)
LWRB0	2.8	Initial sump water level (m)
ARB	420	Sump or containment cross section area (m^2)
VRB	40000	Containment volume (m^3)
PFCL	9.99	High containment pressure set point to start fan cooler (bar)
QCSP0	10	Emergency SI heat exchanger rated capacity (MW)
QCL0	28	Fan cooler capacity (MW)
TDSN	128	Containment design temperature (°C)
CRTM	10	Containment heat sink concrete mass (ton)
STLM	5	Containment heat sink steel mass (ton)
RLK0	0.1	Containment leak rate at design pressure (%/day)
PCRT	6.895	RC pressure for break flow changed to non-critical (bar)
RDSP	0.001	Rod speed constant, dimensionless number
QRHR0	10	RHR heat exchanger rate (MW)
TANK0	2600	RWST initial water volume (m^3)
TKMIN	400	RWST water volume to switch to sump (m^3)
HCOR	3.67	Fuel length (m)
AFUT	4500	Total core heat transfer area (m^2)
MZRKT	3500	Total mass of fuel channel (Zr) (kg)
MZRST	18000	Total mass of fuel cladding (Zr) (kg)
MCRT	16214	Total mass of control rods (kg)
MVES	3800	Mass of vessel bottom (kg)
CNH2B	5	Concentration of H_2 in containment to start burn (%)
FH2O	0.0533	Fraction of H_2O in concrete
FCO2	0.1939	Fraction of CO_2 in concrete
FDEC	0.1	Fraction of decomposed concrete gas reacts with corium
WVNT0	10	Containment vent flow at 1 psid (kg/s)
PPMEC	2000	ACC & RWST boron concentration (ppm)

Variable	Value	Description
PORV1	165.2	PORV open setpoint (bar)
PORV2	163.2	PORV reseat setpoint (bar)
SAFT1	176	Safety valve open setpoint (bar)
SAFT2	175	Safety valve reseat setpoint (bar)
PHIGH	165.7	High pressure reactor scram setpoint (bar)
PHPI	129.69	HPI automatic start setpoint (bar)
PSCRAM	132.2	Low pressure reactor scram setpoint (bar)
PADV	81.8	SG relief valves (atmospheric dump valve) opening pressure
PCFT	43	Pressure set point for core flood tank initiation (bar)
PLPI	11.36	LPI system initiation pressure (bar)
TAVGL	281	Low T_{avg} set point for SG isolation (°C)
PSP1	0.6	PRZ spray initiation error (bar)
PSP2	5	PRZ spray maximum error (bar)
SGHH	82	High-high SG level for turbine trip (%)
HTR1	270	PRZ proportional heater capacity (kW)
PHTR1	0	proportional heater initiation error (bar)
PHTR2	-1.1	proportional heater error for full capacity (bar)
SGLL	17	Low SG narrow range scram set point (%)
PRBH	1.3	High RB press for SI initiation (bar)
LPZL	0.15	Simultaneous low PZR level with RC press for SI initiation
PHPL	128	Simultaneous low RCS pressure with PZR level for SI initiation (bar)
PSGL	38	Low SG press for SI initiation (bar)

A.3. IAEA PCTRAN PRE-LOADED INITIAL CONDITIONS

IC number	Power (%)	RC Press (bar)	T_{avg} (°C)	SG Press (bar)	Time in life	Description
1	100	155	306.9	70	EOC	100% Power EOC
2	100	155	306.9	70	MOC	100% Power MOC
3	100	155	306.9	70	BOC	100% Power BOC
4	75	155	304.7	70	EOC	75% Power EOC
5	75	155	305.2	70	MOC	75% Power MOC
6	75.09	154.42	280.37	70.12	BOC	75% Power BOC
7	0.02	155.41	280.56	64.2	EOC	HZP (Hot zero power)
8	0.01	155.57	126.09	64.24	EOC	HZP 20% RCCA out
9	4.1	2.43	134.68	2.39	EOC	2000 cm^2 Cold leg LOCA without ECCS and core melt
10	1.44	3.13	136.81	3.28	EOC	Core collapsed
11	1.11	3.34	282.33	3.32	EOC	Vessel failed and CCI
12	1.41	155.17	282.86	64.21	EOC	1.4% decay heat

A.4. IAEA PCTRAN MALFUNCTIONS

Malfunction number	Description
1	Loss of Coolant Accident (Hot Leg)
2	Loss of Coolant Accident (Cold Leg)
3	Steam Line Break Inside Containment
4	Steam Line Break Outside Containment
5	Spark Presence for Hydrogen Burn
6	Loss of AC Power
7	Loss of Flow (Locked Rotor)
8	Anticipated Transient without Scram
9	Turbine Trip
10	Steam Generator A Tube Rupture
11	Steam Generator B Tube Rupture
12	Rod withdrawal/Insertion
13	Reserved for future
14	Moderator Dilution
15	Load Rejection
16	Containment Failure or Spark for H_2 Detonation
17	Fuel Failure at Power
18	Fuel Handling Accident in Containment
19	Fuel Handling Accident in Auxiliary Building
20	Letdown Line Break in Auxiliary Building

A.5. IAEA PCTRAN PLOTTABLE VARIABLES

Variable	Description
TAVG	Temperature RCS average (°C)
THA	Temperature Hot leg A (°C)
THB	Temperature Hot leg B (°C)
TCA	Temperature Cold leg A (°C)
TCB	Temperature Cold leg B (°C)
WRCA	Flow Reactor Coolant loop A (t/hr)
WRCB	Flow Reactor Coolant loop B (t/hr)
PSGA	Pressure Steam Generator A (bar)
PSGB	Pressure Steam Generator B (bar)
WFWA	Flow SG A feedwater (t/hr)
WFWB	Flow SG B feedwater (t/hr)
WSTA	Flow SG A steam (t/hr)
WSTB	Flow SG B steam (t/hr)
VOL	Volume RCS liquid (m^3)
LVPZ	Level Pressurizer (%)
VOID	Void of RCS (%)
WLR	Flow RCS leak (t/hr)
WUP	Flow Przr PORV and safeties (t/hr)
HUP	Spec Enthalpy Przr top discharge (kJ/kg)
HLW	Spec Enthalpy RCS leak (kJ/kg)
WHPI	Flow HPI (t/hr)
WECS	Flow Total ECCS (t/hr)
QMWT	Power Total megawatt thermal (MW)
LSGA	Level SG A wide range (m)
LSGB	Level SG B wide range (m)
QMGA	Power SG A heat removal (MW)
QMGB	Power SG B heat removal (MW)
NSGA	Level SG A narrow range (%)
NSGB	Level SG B narrow range (%)
TBLD	Power Turbine load (%)
WTRA	Flow SG A tube leak (t/hr)
WTRB	Flow SG B tube leak (t/hr)
TSAT	Temperature Przr saturation (°C)
QRHR	Power RHR removal rate (MW)
LVCR	Level Core water (m)
SCMA	Temp Loop A subcooling margin (°C)
SCMB	Temp Loop B subcooling margin (°C)
FRCL	Clad failure (%)
PRB	Press Reactor building (bar)
PRBA	Press Partial RB air (bar)
TRB	Temp Reactor building (°C)
LWRB	Level RB sump water (m)
DNBR	Ratio Departure from nuclear boiling

A.5. IAEA PCTRAN PLOTTABLE VARIABLES (cont.)

Variable	**Description**
QFCL	Power Fan cooler heat removal (MW)
WBK	Flow Total break entering RB (t/hr)
WSPY	Flow Pressurizer spray (t/hr)
WCSP	Flow Containment spray (t/hr)
HTR	Power Pressurizer heater (kW)
MH2	Mass H_2 generated by Zr-H_2O (kg)
CNH2	Concentration RB hydrogen (%)
RHBR	Reactivity Soluble boron (%dk/k)
RHMT	Reactivity Mod temperature (%dk/k)
RHFL	Reactivity Fuel (Doppler) (%dk/k)
RHRD	Reactivity Rod (%dk/k)
RH	Reactivity Total (%dk/k)
PWNT	Power Nuclear Flux (%)
PWR	Power Core thermal (%)
TFSB	Temp Submerged fuel average (°C)
TFPK	Temp Peak fuel (°C)
TF	Temp Average fuel (°C)
TPCT	Temp Peak clad (°C)
WCFT	Flow Accumulator (t/hr)
WLPI	Flow Low pressure injection (RHR) (t/hr)
WCHG	Flow Charging (t/hr)
RM1	Rad Monitor RB air (CPM)
RM2	Rad Monitor Steam Line (CPM)
RM3	Rad Monitor Condenser Off-gas (CPM)
RM4	Rad Monitor Aux Building Air (CPM)
RC87	Activity RC Coolant (CPM)
RC131	Concentration RC I-131 Eq (GBq/cc)
STRB	Rad Rel Rate RB (GBq/s)
STSG	Rad Rel Rate SG Valves (GBq/s)
STTB	Rad Rel Rate Condenser Off-gas (GBq/s)
RBLK	Mass Total Leakage out of RB (kg)
SGLK	Mass Total Leakage out of SGs (kg)
DTHY	Dose Rate EAB Thyroid (mSv/hr)
DWB	Dose Rate EAB Whole Body (mSv/hr)
P	Press RCS (bar)
WRLA	Flow SG A MSV/ADV (t/hr)
WRLB	Flow SG B MSV/ADV (t/hr)
WLD	Flow Letdown (t/hr)
MBK	Integrated Break Flow (kg)
EBK	Integrated Break Energy (MJ)
TKLV	RWST Water Volume (m^3)
FRZR	Fraction Zr Oxidation

A.5. IAEA PCTRAN PLOTTABLE VARIABLES (cont.)

Variable	Description
MDBR	Mass of Corium in DW (kg)
MCRT	Mass of molten concrete (kg)
MGAS	Mass of CCI gases (kg)
TDBR	Temp of Debris in Cavity (°C)
TSLP	Temp of Debris in Lower Plenum (°C)
TCRT	Temp of Molten Concrete (°C)
PPM	Concentration RCS Boron (ppm)

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ABBREVIATIONS

AC	alternating current
ACC	accumulator
ADV	atmospheric dump valve
AFW	auxiliary feedwater
AOO	anticipated operational occurrence
BOC	beginning of cycle
BWR	boiling water reactor
CCI	corium–concrete interaction
CFT	core flood tank
CVCS	chemical and volume control system
DBA	design basis accident
DEC	design extension condition
DW	drywell (reactor cavity in PWR)
ECCS	emergency core cooling system
EFW	emergency feedwater
EOC	end of cycle
FWIV	feedwater isolation valve
FWP	feedwater pump
HPI	high pressure injection
HPIS	high pressure injection system
IC	initial condition
LBLOCA	large break loss of coolant accident
LOCA	loss of coolant accident
LPI	low pressure injection
LPIS	low pressure injection system
MCCI	molten core–concrete interaction

MOC	middle of cycle
MSSV	main steam safety valve
MSIV	main steam isolation valve
MTC	moderator temperature coefficient
MFWP	main feedwater pump
PORV	pilot-operated relief valve
PRZ	pressurizer
PWR	pressurized water reactor
RB	reactor building
RBS	reactor building spray
RC	reactor coolant
RCCA	rod control cluster assemblies
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal
RPS	reactor protection system
RWST	refuelling water storage tank
SBLOCA	small break loss of coolant accident
SBO	station blackout
SG	steam generator
SI	safety injection
TBV	turbine bypass valve
TMI	Three Mile Island

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