Application of RELAP5-MV for some accident scenarios analysis of AP1000

YOUSIF Eltayeb[1,*], ZHANG Zhijian[1], TIAN Zhaofei[1], JU Haoran[1], and WANG Yafeng[1]

1. *Harbin Engineering University, 145 Nantong Street, Harbin, Heilongjiang, China, 150001 (eltayeb2003@gmail.com)*

Abstract: RELAP5 code developed in U.S.A. to predict the accident behaviour of light water reactor is widely used around the world. On the other hand the RELAP5-MV had been developed by the authors of this paper to improve the input preparation functions of the RELAP5 to help users of RELAP5 for fast modeling the plant system to conduct simulation practice by advanced graphics displays. However, it is needed to validate the help functions of the RELAP-MV for users by testing different accident scenarios in actual plant conditions. Accuracy and reliability of RELAP5-MV have been confirmed by simulating small break loss of coolant accident (SBLOCA) and steam generator tube rupture (SGTR) accident in AP1000. The results obtained by RELAP5 and RELAP5-MV are in good agreement as they are based on the same models of light water reactor accident analysis. But in comparison with RELAP5, RELAP5-MV resulted in making simulation of nuclear power systems easier and more convenient than RELAP5 for users especially the beginners. With RELAP5-MV the simulation process is simplified and simulation efficiency is increased. By both component and modular modeling methods supported in RELAP5-MV. In this paper, the predicted plant behaviour of the AP1000 by using RELAP5-MV is presented for SBLOCA and SGTR scenarios. The results obtained by RELAP5-MV have demonstrated the effectiveness of RELAP5-MV for such transients and accident scenarios simulations.

Keywords: RELAP5; loss of coolant accident; SGTR; AP1000; graphical modeling

1 Introduction

Since the time RELAP5/MOD2 was introduced, many changes have been made to make the code better and more user friendly, with large range of analytical capability [1] since it was difficult to simulate complex phenomena with the old version of code. Several challenges are associated with usage of RELAP5, some of which are highlighted below. First, too many components make it hard and complex for new users to understand and make the use of RELAP5 difficult due to over ten categories of input cards with dozens of components. Then for editing input cards by notepad, too many data with complex logic relationship makes it quite hard to seek and modify specific parameters in the input file. Searching appropriate input file for information of equipment is a complicated task.

A RELAP5 problem input deck consists of at least one title card, optional comment cards, data cards, and a terminator card. A list of these input cards is printed at the beginning of each RELAP5 problem. For parameter studies and for temporary changes, a new title card with the inserted, modified, and deleted data cards and identifying comment cards should be placed just ahead of the terminating card [2]. Because of the complex process of modeling and debugging, it takes long time costs with no efficiency.

Oriented to solve these disadvantages, the graphical software RELAP5-MV was originally developed by several authors of this paper with the use Extensive Markup Language (XML) based on inner kernel model and solver of RELAP5 code [3]. The RELAP5-MV software has been made to be more interactive and user friendly through some developed graphic user interfaces (GUIs). As expected these GUIs have their models based on RELAP5 code. Those user aids and interfaces have been categorized into three elements: input model builders, transient aids and post-processors. Input aids helps to make an input deck for RELAP5 easy. RELAP5-MV Visualized Modularization software is recognized as one of the best estimate transient simulation program of light water reactor, in combination with new options for improved modeling methods, computational simulation techniques and integrated graphics displays.

RELAP5-MV supplies a friendly human-computer interface through which users can build a system model more conveniently, rapidly and visually.

The effectiveness of RELAP5-MV has been demonstrated in this paper by applying it to the accident analysis of SBLOCA and SGTR for AP1000. With the software, the accidents analyses can be conducted easier and faster in time than by RELAP5.

In this paper, the detail of RELAP5-MV is explained in 2 with respect to the design of the interface software, component modeling and modular modeling. In 3, the two accident analysis, SBLOCA and SGTR, of AP1000 by RELAP5-MV are described together with the overview of AP1000 and its modeling by RELAP5-MV. In addition, a brief explanation of the input setup for the two accident scenarios was given. Lastly, the results of the two accidents simulations by RELAP5-MV are presented as the proof that the developed RELAP5-MV can accomplish the successful accident simulations by comparing with RELAP5/MOD3.

2 RELAP5-MV

RELAP5-MV software is based on RELAP5/MOD3 thermal hydraulic best estimate code. It has an interactive graphic user interface which is based on two modeling methods: one is the actual component-based modeling method to describe how the whole plant is comprised by various components like reactor, steam generator, pressurizer, main pump, steam turbine, and other real components. Component modeling method deal with the plant configuration and operating characteristics of the modeling objects. The other is to build a simplified block diagram representation of the system components by the form of "control volume” as the basic unit of division and modeling, and for some special components, the direct application of the "special model" of the system such as main pump, valves and so on. Table 1 shows classification table of RELAP5 basic input cards. Hydrodynamic components include single volume, single junction, pipe, branch, valve, separator, time-dependent volume, time-dependent junction, pump and accumulator and so on [4].

Table 1 Classification table of RELAP5 basic input cards [5]

Type of component	Function
Hydrodynamic component	Simulate flow path
Thermal property data	Show thermal property of heat structure
Control component	Edit control method of nuclear system and control variable
Time-step control component.	Control the time step for calculation and total calculation time.
General table component	Input variable table
Heat structure	Represent metal structures in a facility
Reactor kinetics component	Edit parameters in “point” reactor kinetics model
Problem control options	Set up parameters about problem and calculation
Trip component	Set trip function
Output component	Control output process

All of these components have their corresponding parts in RELAP5-MV by component modeling method.

In view of the perspective of the novice users, the RELAP5-MV is effective for the novice user to master plant components and configurations for the purpose of modeling the whole plant. In addition, the expertise users can easily conduct on accident analysis to see the nuclear plant performance from many aspects by using various functions of the RELAP5-MV software.

2.1 Software interface design

RELAP5-MV supplies a friendly human-computer interface through which users can build a system model more conveniently, rapidly and visually. Figures 1 to 4 show the main interface of RELAP5-MV, consisting of menu bar, toolbar, design sketch, navigation tree, component library, outline, *etc*. The software interface has the following components: hydraulic model, thermal component model, control system model, start or close logic signal, menu bar and toolbar as shown in Fig 1. The available function tools are given by Menu bar and tool bar for the following operations:
(i)file operation (new, open, save files generated by

RELAP5-MV, icons adjustment in the interface, RELAP5 input file generation, calling RELAP5 to start simulating calculation, *etc.*).

(ii) Design icons for sketching of components.

(iii)Navigation tree for easy navigation between different possible modules and operations

(iv)Library: collecting icons of components and modules, including component library and module library.

2. 2 Components modeling

The input rules for a single control volume diagram is to start from the left side of the interface, select a single control volume map source, drag and drop to the software work window, right-click the mouse button and select appropriate pop-up as shown in Fig. 2.

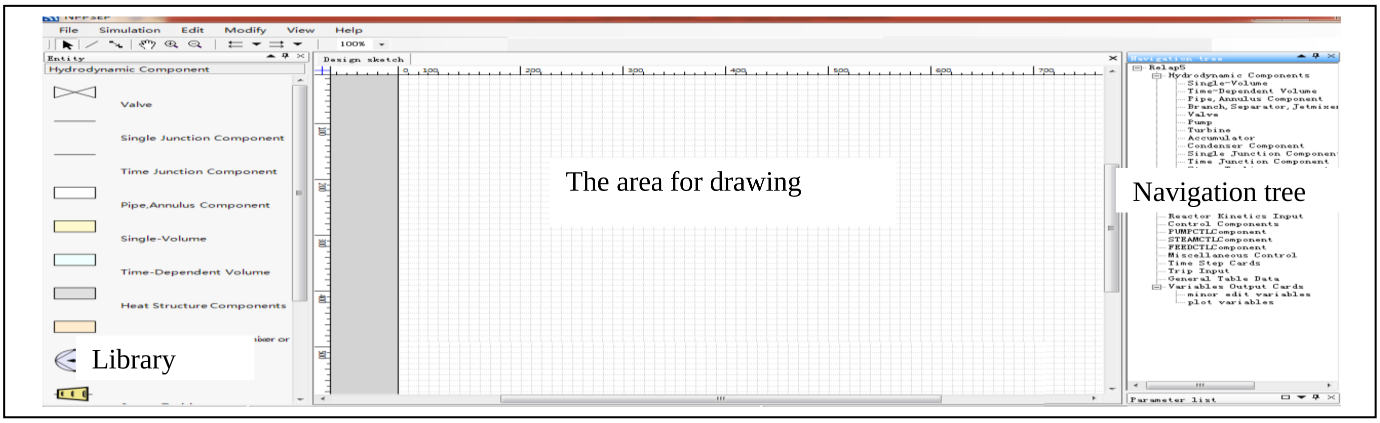

Fig. 1 software interface.

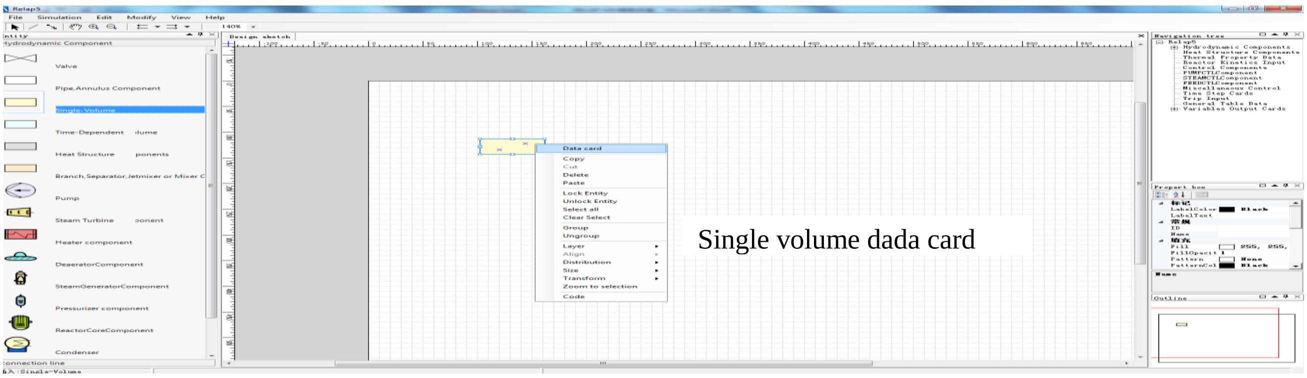

Fig. 2 Single volume input display.

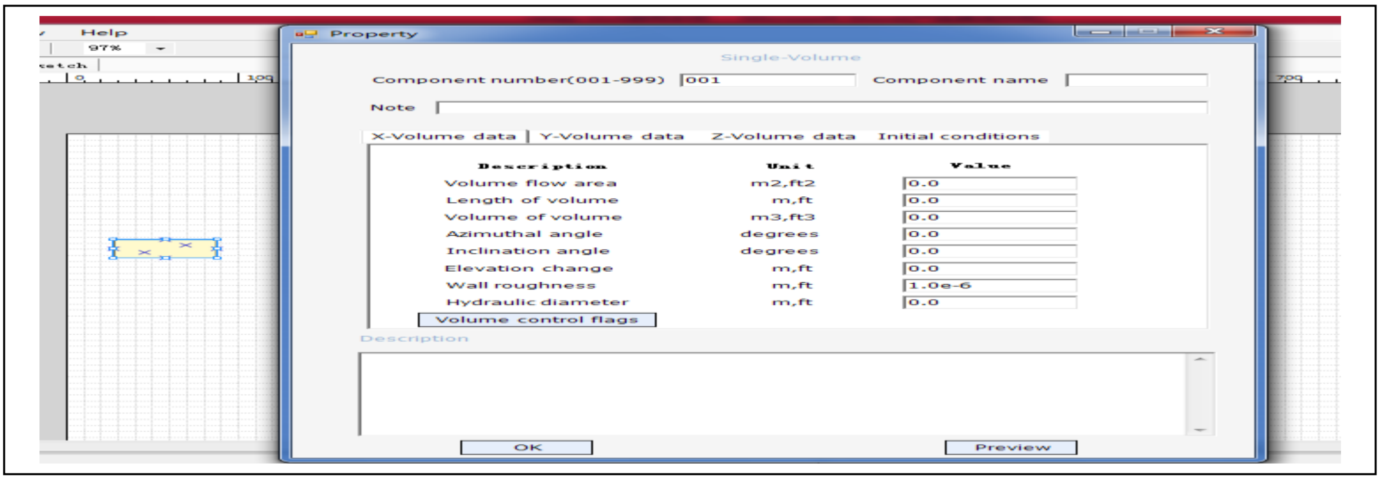

Fig.3 Single volume parameter input interface.

The first option on the pop-up shown above is data card. The data card icon can be clicked upon to input the required parameter. In addition, clicking on the control volume parameters icon displays the control volume parameters of the configuration, as shown in Fig. 3.

Figure 4 displays the way to input the parameters through the software window / interface.

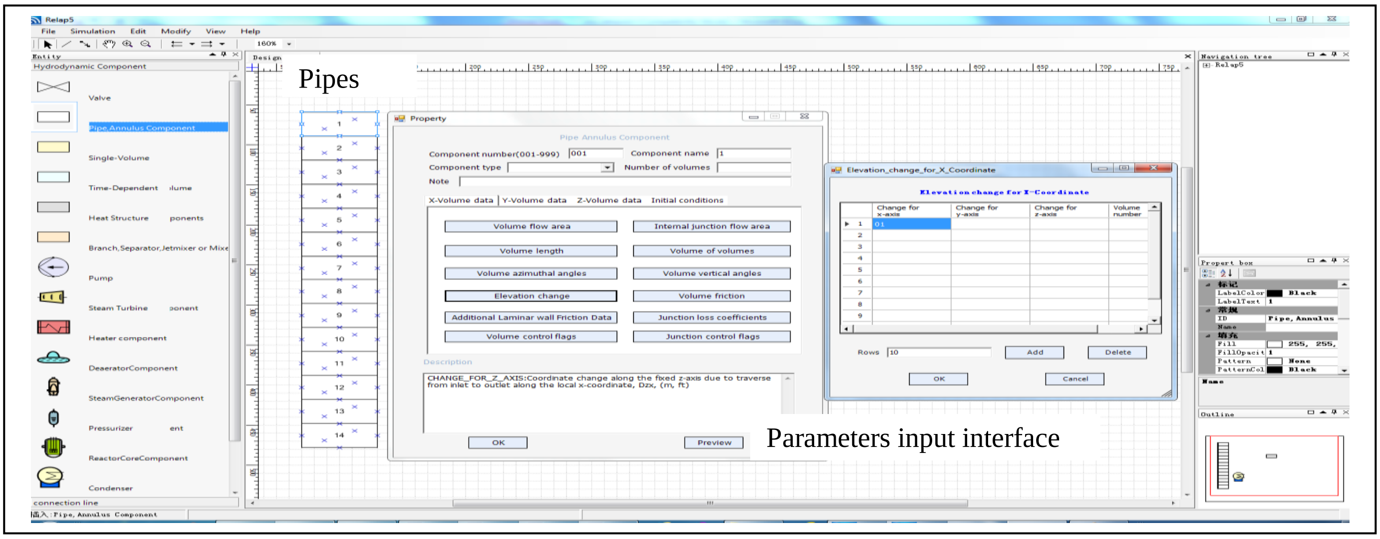

Fig .4 Main interface of RELAP5-MV.

2.3 Modular modeling

RELAP5-MV provides an easy and fast modeling platform, which is suggested to be used by beginner of RELAP5/MOD3 simulation. Modular modeling method is a modeling method using device modules in the module library to build simulation system directly. Already prepared nuclear power device modules using standard structure, function and operating characteristics of devices, modules are collected in a module library embedded in RELAP5-MV. Filling in parameters step by step in each device and then connecting with junctions, simulation system could be established more rapidly. Thus, modular modeling method acts as one kind of technological means for easy and fast nuclear reactor system design. During component modeling process, kinds of node division strategies will be tested to find the optimized one, not only to cover every part and achieve main functions and performances of the equipment, but also to show the static and dynamic properties accurately by calculation results as well. In another word, component modeling method is fit for skilled researchers.

The main interface of the reactor core can be seen in Fig. 5 below. Input data file will be generated automatically for RELAP5-MV. Simulation calculations are carried out in background based on the incorporated models and format once the necessary modeling data are filled on the interface according to the prompt dialog box. These components are used in similar manner with other components.

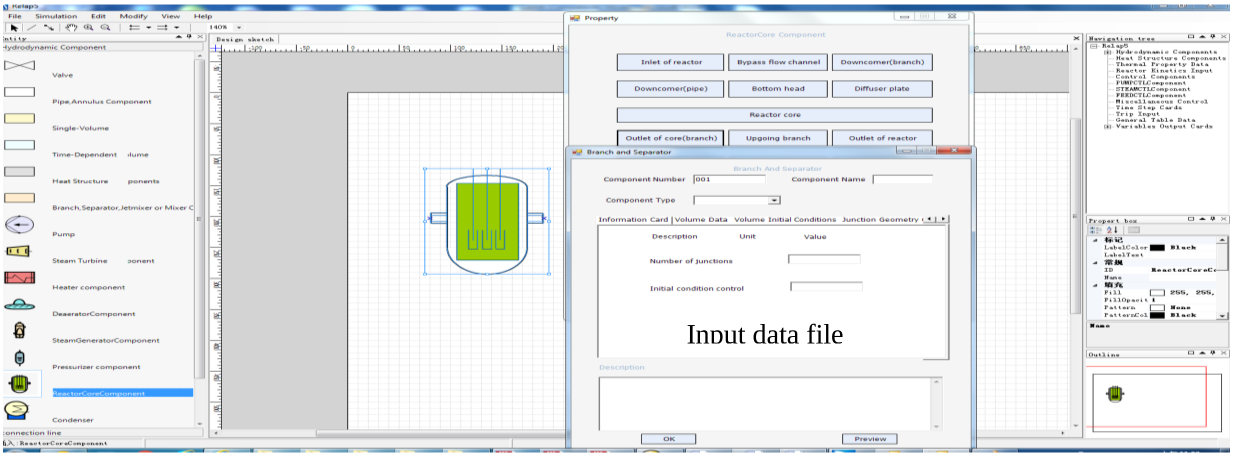

Fig. 5 Filling parameters interface of reactor core.

Other reactor equipment can be modeled by following the same procedure. Shown above is an illustration of the software for modular modeling.

Figure 6 shows simulation system of primary loop of NPP with modular modeling method. It shows a simple simulation of a typical pressurized water reactor with its main coolant system and the specific modeling results. The software forms the module library of the device. Dragging the source module which represents the corresponding equipment from the module library to carry on the system simulation modeling, and then the modules are connected by lines between the modules. Meanwhile, the corresponding control can be described by the program described in the right side of Fig. 6. Help information, editing and drawing the required parameters such as input, follows exactly the same modeling rules as RELAP5/MOD3.

Beside the block diagrammatic representation of components in RELAP5-MV, the full component modeling can also take this form as shown below.

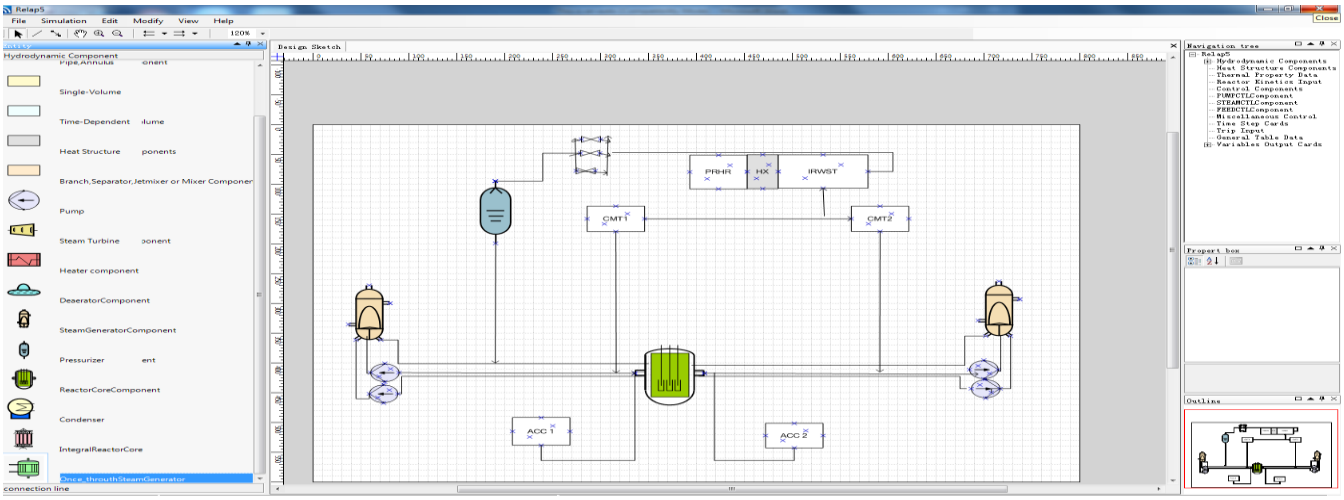

Fig. 6 AP1000 model diagram.

Refer to the description of each module above. After inputting and saving the parameters, click Generate input File (I file) in the menu bar of simulation. One can also generate the executable file of current simulation system, which can be called RELAP5 calculation. RELAP5-MV interface is quite user friendly. Other reactor parameter and their respective values can be selected in the same way as demonstrated above.

3 SBLOCA and SGTR simulations for AP1000

3.1 Overview of AP1000

The AP1000 is an advanced nuclear power plant with a rated thermal power of 3400MWt and the net electrical power of at least 1117 MWe. The core of AP1000 has two loops in the RCS. Instead of depending on active systems like diesel generators and pumps, the AP1000 reactor uses natural forces such as compressed gases, gravity and natural circulation to prevent the reactor core and the containment from excessive heating. The AP1000 reactor's passive systems include: Passive Core Cooling System (PXS), In-containment Refueling Water Storage Tank (IRWST), Passive Containment Cooling System (PCS), Main Control Room Emergency Habitability System, High Pressure Safety Injection with Core Makeup Tanks (CMTs), Medium Pressure Safety Injection with Accumulators, Low Pressure Reactor Coolant Make from the IRWST, Passive Residual Heat Removal (PRHR-HX), and Automatic Depressurization System (ADS) as shown in Fig.7 [6].

In the AP1000 reactor, the three main sources of water are provided for cooling the reactor core and decay (residual) heat removal to avoid core uncover. They are In-containment Refueling Water Storage Tank (IRWST), Core Makeup Tank (CMT) and the Accumulators systems [7]. The two CMTs are slightly raised above the reactor core. The discharge injection line and an inlet pressure balance line connect the CMTs to the RCS.

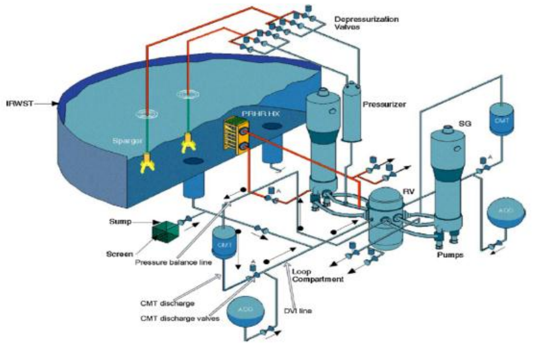

Fig.7 passive core cooling system of AP1000 [20]

3.2 Modeling of AP1000 by RELAP5-MV

A RELAP5-MV model representing the transient analysis of AP1000 has been developed, with the primary loop, partial secondary loop and the passive safety system components of AP1000. Figure 8 shows nodal scheme used in the RELAP5-MV simulation.

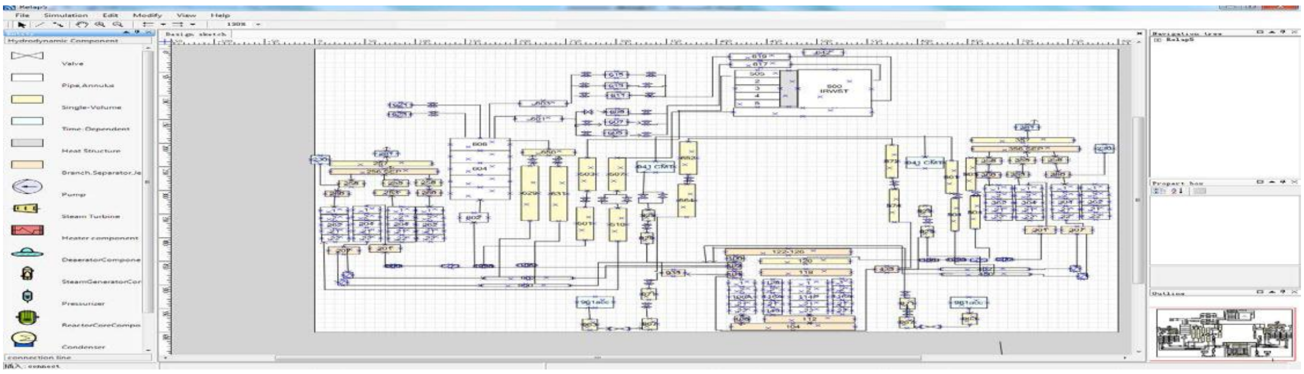

Fig. 8 Nodal scheme of AP1000 by RELAP5-MV.

The diagram shows the corresponding parts of the source for system simulation modeling. The parameters on the diagram can be controlled by the navigation tree on the right side of the window above. By pointing the cursor to the navigation tree, the user can enter the required parameters like thermal properties, editing and graphics. Building the diagram with RELAP5 follows exactly the same rules. A small break LOCA is simulated by adding a trip valve and a time dependent control volume (TMDPVOL) to the broken cold leg in the non-pressurizer side [8].

3.2 SBLOCA in AP1000

In this study the SBLOCA is simulated by considering a break in one of the cold legs in non-pressurized side of the primary loop. The break size considered is 10 inch which is maximum size of SBLOCA. The SBLOCA transient in AP1000 can be characterized into four different phases: blow-down phase, ADS blow-down phase, natural circulation phase and IRWST injection phase. At the start of SBLOCA, the leakage through the break results in drop of primary system pressure due to loss of energy and water inventory from the break [8].

During the accident, as the primary system pressure drops to 12.41 MPa, the control system automatically actuates the reactor trip. The actuation of reactor trip reduces the core power to decay heat level. The further primary system pressure drop generates safety actuation, a safeguard signal known as “S” signal at 11.72 MPa. The actuation of the CMT and PRHR-HE starts immediately after the “S” signal. The main reactor coolant pumps trip follow the “S” signal after a short time delay of 6 sec [9,10]. The natural circulation loop is established on actuation of CMTs. The CMTs inject cold borated water in the reactor core by gravity injection and receive hot water through the upper part of CMTs. The PRHR heat exchanger is immersed in the IRWST. The actuation of PRHR heat exchanger established another natural circulation loop. The hot water from hot leg enters into PRHR heat exchanger, and transfers residual heat to the IRWST and the cold water then injected into the primary system through cold leg. The further drop in the primary system pressure results in the actuation of two ACCs at 4.82MPa. The ACCs are filled with cold borated water which is injected in the primary system through direct vessel injection (DVI) line. The high accumulator ejection flow rate temporarily impedes the CMT injection because accumulators and CMTs are connected on the same DVI line. The AP1000 reactor is equipped with automatic depressurization system (ADS). The main purpose of ADS system is to sequentially depressurize the primary system close to containment pressure and start the IRWST injection. IRWST injects cold water into the primary system by gravity for long-term cooling. The actuation of ADS depends on the water level in CMTs. The AP1000 automatic depressurization is divided into four stages. The ADS stage 1 actuates when water level in either of the CMTs drops below 67.5%. The stages 2 and 3 are designed to actuate at the specified time delay following ADS stage 1[11]. The protection and safety monitoring system (PMS) set-points and time delay assumed in the SBLOCA analysis for AP1000 are listed in Table 2.

Table 2. AP1000 safety system actuation set-point during SBLOCA [8].

System/ Function	Actuation set-points	Time delays (sec)
Reactor trip	12.41 MPa	2.0
“S” signal generation	11.72 MPa	2.0
SG feed water control valves starts to closing	After reactor trip signal	3.2
Main steam isolation valves start to closing	After “S” signal	4.8
Reactor coolant pumps trip	After “S” signal	6.0
PRHRS isolation valve starts to opens	After “S” signal	0.0
CMT actuation	After “S” signal	0.0
Accumulator actuation	4.83 MPa	0.0
ADS-1 actuation	after CMT water volume reduces to 67.5%	20.0
ADS-2 actuation	70 s after ADS-1 actuation	30.0
ADS-3 actuation	120 s after ADS-2 actuation	30.0
ADS-4A actuation	20.0% liquid volume fraction in CMT	2.0
ADS-4B actuation	60 s after ADS-4A actuation	2.0
IRWST injection	pressure < 89.6 KPa + the containment p	2.0

3.4 Steam generator tube rupture in AP1000

The SGTR event causes a direct flow of primary coolant from the high-pressure reactor coolant system (RCS) to the secondary system of the steam generator (SG). This leads to contamination of the secondary system and possible release of radiological products to the environment. The loss of primary coolant can significantly exceed the make-up capacity of the charging pumps. [12]

For these reasons, SGTR constitutes an important safety concern and has been classified as a design-basis event (DBE) for PWRs [13].

Induced SGTRs are a consequence of other events, such as incorrect installation of anti-vibration devices, abnormal secondary water chemistry conditions or loose objects left inside the SGs during earlier maintenance works [12]. Several studies and experiments have been performed to investigate SGTR events in order to verify the features and capacity of the reactor safety systems, as well as to optimize the effectiveness of the emergency operating procedures.

Traditional modeling method and modular modeling method are supported with RELAP5-MV to achieve aims of device and system simulation. For traditional modeling method, all kinds of components are developed such as single volume, single junction, pipe, branch, time dependent volume, *etc*. For modular modeling method, the module library is established in the software. The library packages include the main system equipment of primary and secondary loops such as reactor core, U-tube steam generator, once-through steam generator, pump, pressurizer, steam turbine, condenser, heat exchanger, de-aerator, *etc*. From the library, it is easy to select icons interface from the library packages. The nodal scheme of steam generator is as shown in Fig. 9.

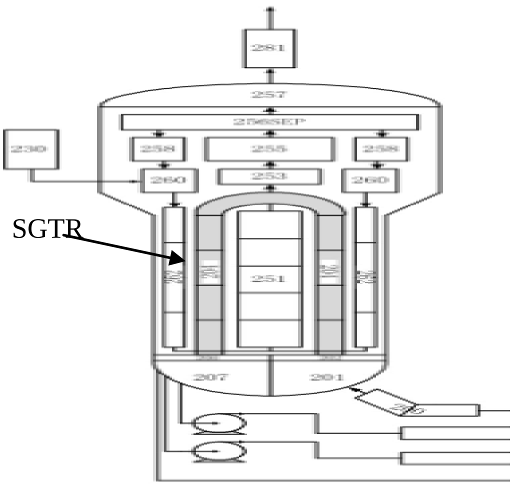

Fig. 9 Nodal scheme of steam generator.

4 Results and discussion

4.1 SBLOCA analysis

The steady state analysis of AP1000 reactor is performed in RELAP5-MV by using AP1000 model as described in the previous section. The comparison between the steady-state results obtained by the RELAP5-MV code and the RELAP5/MOD3 simulation results of the AP1000 reactor is given in Table 3. It can be seen from Table 3 that the calculated values are in close agreement with the RELAP5/MOD3 simulation results of AP1000.

Table 3 Steady state calculated results of AP1000 obtained by the RELAP5-MV.

Parameters	RELAP5-MV Results	RELAP5 simulation results[16]
Core thermal power (MW)	3415.00	3400.00
RCS pressure (MPa)	15.60	15.52
Vessel inlet temperature (K)	553.90	553.82
Vessel outlet temperature (K)	595.00	594.26
Vessel average temperature (K)	585.00	574.04
SG secondary pressure (MPa)	5.61	5.61
SG feed water temperature (K)	499.83	499.82

In this study, 10 inch small break LOCA is simulated by using the RELAP5-MV in a cold leg of AP1000 in non-pressurizer loop. Table 2 presents the actuation set points of safety system during the small break LOCA. RELAP5-MV, SBLOCA accident scenarios has been simulated with the event sequence as shown in Table 4.

Table 4 Sequence of events of 10-in. SBLOCA in AP1000 by RELAP5-MV

Event	RELAP5-MV(s))
Break open	00.0
Reactor trip signal	5.5
“S” signal	8.0
Main feed isolation valves begin to close	8.2
Steam turbine stop valves clos	9.8
Reactor coolant pumps start to coast down	12
Accumulator injection starts	94
Accumulator 1 empties	440
Accumulator 2 empties	442
ADS stage 1	732
ADS stage 2	802
ADS stage 3	922
ADS stage 4	1181

The results obtained from SBLOCA analysis using RELAP5-MV code can be presented in graphical form.

As the LOCA starts, the primary system pressure drops due to break flow. The break flow causes mass and energy loss from primary loop, and the primary system pressure decrease causes various actuation of safety systems; This causes actuation of all passive safety systems and ADS stages, and the primary system finally reaches the containment pressure and results in actuation of IRWST. The pressure drop during small break LOCA calculated by RELAP5-MV is shown in Fig. 10.

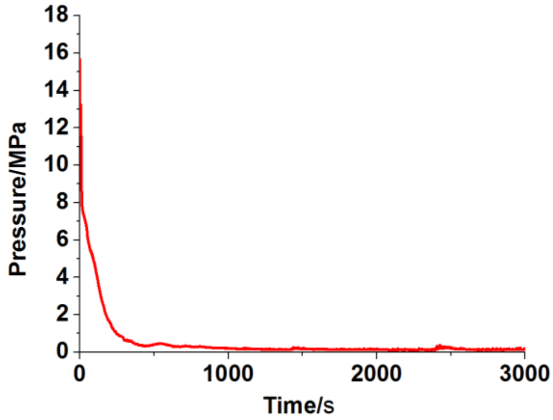

Fig.10 RCS pressure transient by RELAP5-MV.

During primary system pressure drop, the reactor scram is initiated as the primary system pressure drop to 12.41 MPa, and thereafter the reactor power reaches to decay heat level. Figure 11 shows the nuclear power during small break LOCA. The “S” signal is generated when primary system pressure reduction reaches to 11.72 MPa. The generation of S signal actuates the water injection through CMT and PRHR HX. The reactor coolant pumps stops with a short time delay after “S” signal generation which result in reduction of coolant flow through the break.

The break flow rate during small break LOCA as obtained through RELAP5-MV is shown in Fig 12. The break flow is usually of two different phases: the sub-cooled blow-down phase and the two-phase blow-down phase. The former is at the onset of the accident and it is characterized by decreasing pressure to the saturation pressure with a high break flow rate and remain only for a short time while the latter decreases the reactor coolant pressure and last for the remaining time of the accident [14,15]. The results are nearly similar. It is obvious from the results that the actuation of CMTs and accumulators causes rapid increase in break flow.

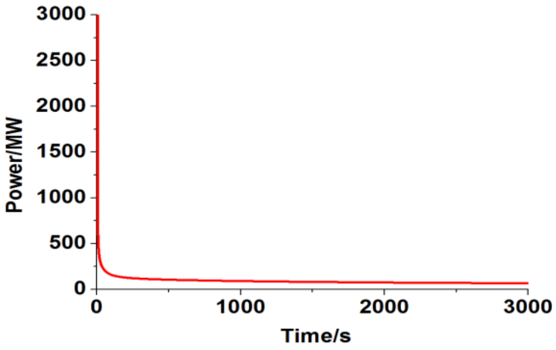

Fig.11 Nuclear power.

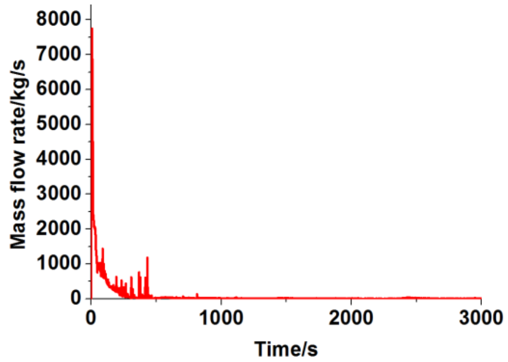

Fig.12 Liquid break discharge flow rate.

The actuation of CMT results in injection of relatively cold borated water into the primary system. The upper part of CMT is connected with one of the cold leg and the relatively hot water is collected into the CMT from cold leg and a loop is established due to natural circulation flow. This natural circulation flow through CMT is called recirculation mode [16]. Similarly, the IRWST actuation also causes a natural circulation flow. The heat of hot coolant is transferred to the IRWST and the relatively cold and heavier water is injected into the primary system. The CMT draining mode will start when the the hot coolant collection stops at the top of the CMT [17].

As described earlier, the CMT recirculation mode is converted to CMT draining mode when the water level in the line connecting the cold leg and upper part of CMT start draining. The CMT draining mode causes the reduction in CMT level. The actuation set point of ADS stage 1 is related to the CMT level and reduction in CMT level actuated the automatic depressurization system. The CMT flow rate obtained by REPLAP5-MV code is shown in Fig 13.

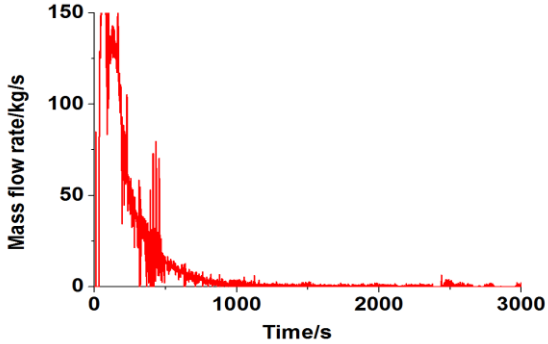

Fig. 13 CMT injection rate.

The accumulator injection starts when the primary system pressure reduces to 4.83 MPa. The accumulators also inject relatively cold borated water into the reactor core through direct vessel injection line. As accumulators and CMT are connected to the direct vessel injection (DVI) line, the relatively high accumulator flow temporarily reduces the CMT flow due to back pressure [16,18,19]. The calculated results by RELAP5-MV show that the accumulator injection starts at 94 sec and the accumulator become empty at around 440 sec as shown in Fig. 14 [19]. The first increasing and then decreasing trend of flow through accumulator as shown in Fig. 14 is because of the pressure difference between the accumulator compressed gas and the primary system pressure [8].

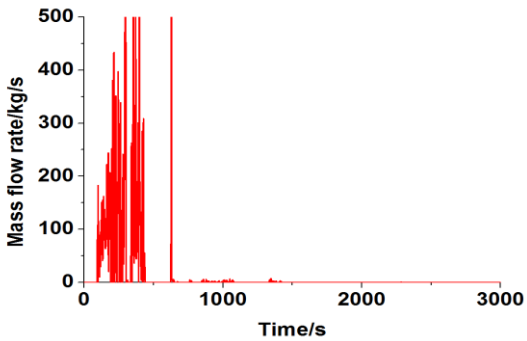

Fig 14 Accumulator injection rate.

The actuation of ADS stages starts when the CMT water level reaches to 67.5%. The ADS actuation causes the further rapid reduction in primary system pressure. The flow through ADS stages is given in Figs. 15, 16, and 17. The discharge flow is higher in ADS stages 2 and 3 as compared with the ADS stage 1 due to relatively large diameter of ADS stages 2 and 3 [11]. The calculated results show that the ADS stage 1 actuate at 732 sec and the ADS stage 2 and 3 actuates at 802 sec and 922 sec, respectively.

The further reduction in CMT level causes the actuation of ADS stage 4 at 20% CMT level. The simulated results show that the ADS stage 4 is actuated at around 1181 sec. The flow rate through ADS stage 4A and 4B obtained through REPA5-MV is given in Figs. 18 and 19, respectively.

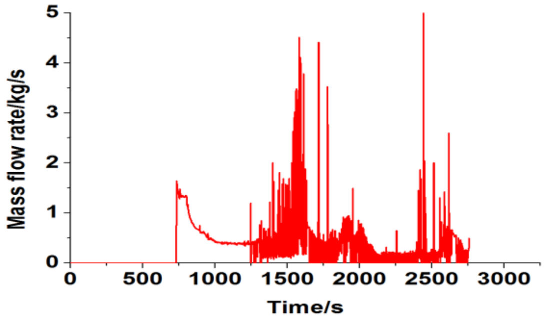

Fig.15 ADS-1 discharge flow rate.

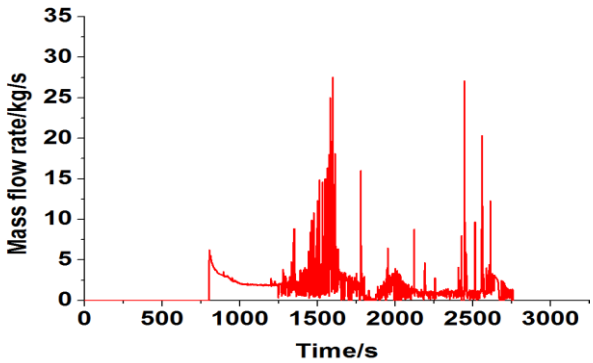

Fig. 16 ADS-2 discharge flow rate.

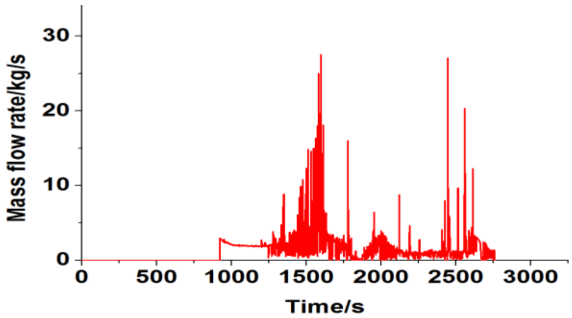

Fig. 17 ADS-3 discharge flow rate.

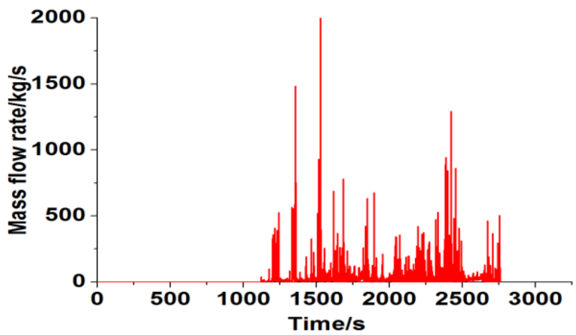

Fig. 18 ADS-4A discharge flow rate.

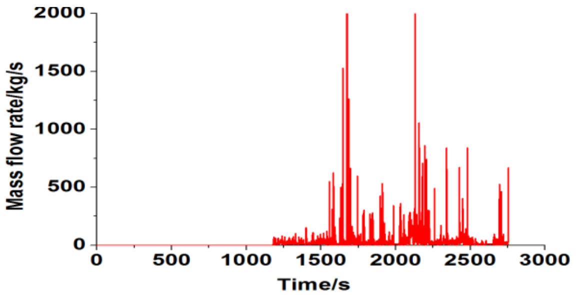

Fig.19 ADS-4B discharge flow rate.

The actuation of ADS stages causes effective and more rapid primary system depressurization and reduces the primary system pressure to the containment pressure and the gravity driven IRWST injection starts. The large water storage in IRWST results in core cooling for longer period of time.

4.2 Description and analysis of SGTR.

Transient scenario is modeled by using the RELAP5-MV code. The sequence of events for this transient is presented in Table 5.

With RELAP5-MV, SGTR accident scenarios have been simulated and the results obtained are in agreement with that of RELAP5/MOD3. The transient scenario for the AP1000 NPP is modeled using the RELAP5-MV. The calculation was performed up to 3,000.0 sec. Before investigating the transient, RELAP5 model was run in real plant equilibrium conditions up to 1500 s to establish steady state condition at nominal reactor power.

The initiating event of this analysis is the steam generator tube rupture in SG#1. The tube rupture causes a decrease in reactor coolant system pressure. For all simulated events, the reactor trip occurs due to the low pressurizer pressure. The turbine trip and the subsequent feed water pump trip cause an increase in steam generator pressure and a rapid drop in steam generator water level. Since the primary system pressure is initially much greater than the steam generator pressure, reactor coolant flows from the primary into the secondary side of the affected steam generator [21].

Table 5 SGTR sequence of events. [20]

Events	Time(s)
Double-ended steam generator tube rupture	0
Loss of offsite power	0
Reactor trip	0
Reactor coolant pumps and main feedwater pumps assumed to trip and begin to coastdown	0
Two chemical and volume control pumps actuated and pressurizer heaters turned on	0
Low-2 pressurizer level signal generated	2,498
Ruptured steam generator power-operated relief valve fails open	2,498
Core makeup tank injection and PRHR operation begins (following maximum delay)	2,515
Ruptured steam generator power-operated relief valve block valve closes on low steam line pressure signal	2,979

Figure 20 demonstrates considerable leakage at the beginning of the accident at 1500 s because of the high pressure difference between the primary and secondary side in the break position.

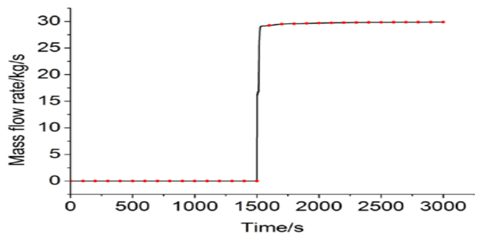

Fig. 20 Primary to secondary leak.

The main feed water pumps are assumed to coast down following reactor trip. The startup feed water pumps are conservatively assumed not to start. However, the pressure decreases in the primary loop from 15.63 MPa to approximately 15.4 MPa as shown in Fig. 21.

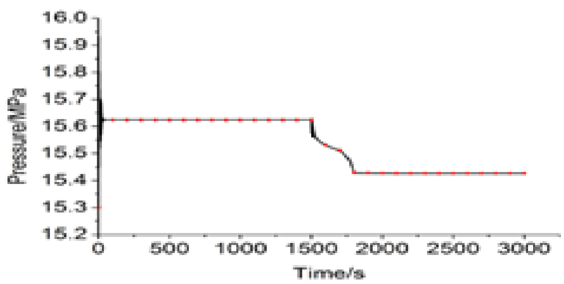

Fig. 21 Pressure at SG#1 in primary loop.

Following the tube rupture, reactor coolant flows from the primary into the secondary side. In response to this loss of reactor coolant, pressurizer level and reactor coolant system pressure decreases. Fast depressurization of the primary circuit and a rapid increase of the water level in the affected SG#1 characterize the initial phase of the transient. Water level goes back to rated value after a small fluctuation due to the SG control strategy as shown in Fig. 22.

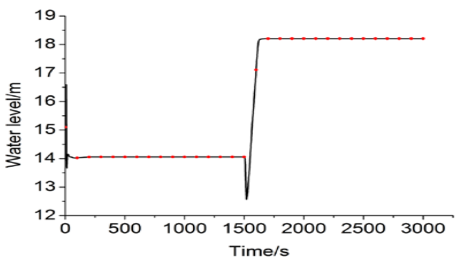

Fig. 22 Water level of SG#1.

After reactor trip, core power rapidly decreases to decay heat levels shown in Fig. 23.

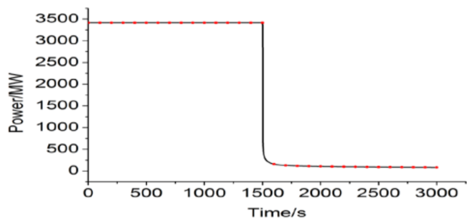

Fig. 23 Nuclear power

Maximum heat addition to the pressurizer from the pressurizer heaters increases the primary pressure. As the leak flow continues to decrease primary system inventory, low pressurizer level "S" and core makeup tank and PRHR actuation signals are reached. This results in an initial increase in primary-to-secondary leakage and a decrease in the reactor coolant system temperatures. When mass flow rate of steam decreases in the core, the pressure of steam rises. Negative reactivity must be inserted to meet the reduction in heat absorption capacity of the secondary loop. Thus, the inlet and outlet temperatures of the core decrease.

The decrease in the reactor coolant system temperature results in a decrease in the pressurizer level and reactor coolant system pressure, also the core inlet to outlet temperature differential decreases as shown in Figs. 24 and 25.Coolant discharge leads to further rapid decrease in pressure of the core outlet shown in Fig.25.

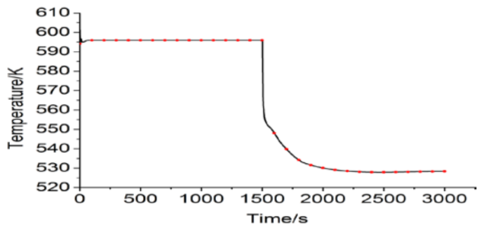

Fig. 24 Temperature at the core outlet.

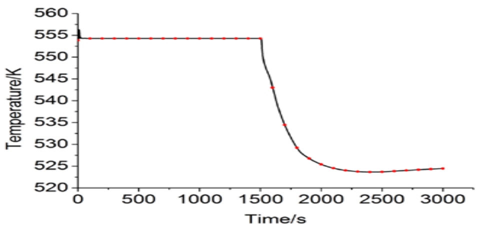

Fig. 25 Temperature at the core Inlet.

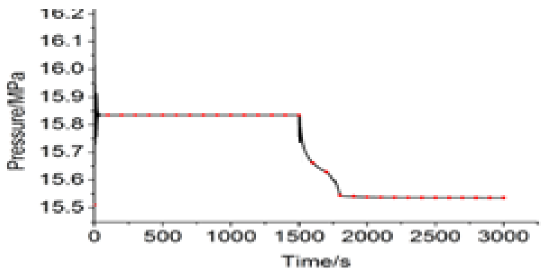

Fig. 26 Pressure at the core outlet.

The analysis and comparison of simulated results obtained by RELAP5-MV show that the code is capable of simulating the plant behavior appropriately during small break LOCA and SGTR similar to original RELAP5 code.

5 Conclusion

RELAP5-MV Visualized Modularization software has been proved to be one of the best estimate transient simulation program of LWRs as expected as it is based on RELAP 5 models.

RELAP5-MV provides a user-friendly human-computer interface (integrated graphics displays) through which users can build a system model more conveniently, rapidly and visually, and also makes simulation of nuclear power systems easier and convenient for the novice users.

The SBLOCA is simulated in one of the cold legs of the non–pressurized side of the primary loop, with 10 inch break size in addition to SGTR. In both cases, the results and accident sequence are very similar to that obtained with original RELAP5 which demonstrated the effectiveness of RELAP5-MV Visualized Modularization Software (GUI) in the same manner as RELAP5.

The accident simulation can be conducted by RELAP5-MV more in short time in setting up the input data with getting the same level of accuracy to be compared with those by using the RELAP5/MOD4.

References

[1] TALUKDER, M.H., *et al*.: "Modeling of the Pactelsbl－50 Transient Using RELAP5 Computer Code Department of Nuclear Engineering," Vol. 262, pp. 1653－4662, 2012.

[2] ROCKVILLE, M.: "RELAP5/MOD3.3 CODE MANUAL VOLUME II:APPENDIX A INPUT REQUIREMENTS," Information Systems Laboratories, Inc. Idaho Falls, Idaho, U. S. Nuclear Regulatory Commission Washington, DC 20555, 2002.

[3] JU H.R., TIAN Z.F., WANG Y.F., and SHANG G.P.: Development of graphical software for RELAP5, STSS/ISSNP 2015, JAPAN.

[4] The RELAP5 Code Development Team: RELAP5/MOD3CodeManual: UREG/CR-5535, Washington: U. S. Nuclear Regulatory Commission, 1995.

[5] FLETCHER, C. D., SCHULTZ, R. R.: RELAP5/MOD3.3CODE MANUAL, VOLUME IIUSER'S GUIDELINES. America: EG&G Idaho. Inc., 1995

[6] CORLETTI, M.M.: "AP1000 Plant Description and Analysis Report", Westinghouse Electric Company, LLC, 2000.

[7] XIE, H., HE, S.J.: "The SCDAP/RELAP5 3.2 model of AP1000 on SBLOCA", Progress in Nuclear Energy, vol. 61, no. 102-107 2012.

[8] YANG, J., WANG, W.W., QIU, S.Z., TIAN, W.X., SU, G.H., and WU., Y.W.: "Simulation and analysis on 10-in. cold leg small break LOCA for AP1000".Annals of Nuclear Energy, vol. 46, no. 81 – 89, 2012.

[9] FRIEND, M.T., WRIGHT, R.F., HOCHREITER, L.E., and OGRINS, M.: Simulated AP600 response to small-break loss-of-coolant-accident and non-loss-of-coolant accident events: analysis of SPES-2 integral test results. Nucl. Technol. Vol. 122, no. 19–42, 1998..

[10] MUFTUOGLU, A.K.: "Improved Methodology for Integral Analysis of Advanced Reactors Employing Passive Safety", Ph.D. thesis. The Pennsylvania State University, USA, 2004.

[11] Westinghouse AP1000 Design Control Document, Westinghouse Electric Company, 2004.

[12] KOSZELA, Z., and SOKOLOWSKI, L.: "Thermal-Hydraulic Analysis of Single and Multiple Steam Generator Tube Ruptures in a Typical 3-Loop PWR", Journal of Power Technologies 95 (3) (2015).

[13] ADAMS, J. P., and SATTISON, M. B.: "Frequency and consequences associated with a steam generator tube rupture event," Nuclear Technology, 90 (2) 168–185, (1990).

[14] ZHOU, R.B., LI, F.C., XU, G.H., YAO, R.Q., and WEI, G.F.: "Experimental investigation on 1% small break loss-of-coolant accident of pressurized water reactor cold leg", At. Energy. Sci. Technol. Vol. 32, no 9–15 , 1998.

[15] NO, H.C.: "Decay heat removal and operator's intervention during a very small LOCA", J. Korean Nucl. Soc. 16 (1), 11–17, 1983.

[16] WRIGHT, R.F.: "Simulated AP1000 response to design basis small-break LOCA events in APEX-1000 test facility", Nuclear Engineering and Technology , Vol. 39, no. 287–298, 2007.

[17] BURCHILL, W.E.: "Physical phenomena of small-break loss-of-coolant accident in PWR", Nucl. Saf. 23 (5), 525–536, 1982.

[18] YONONMOTO, T., KONDO, M., KUKITA, Y., GHAN, L.S., and SCHULTZ, R.R.: "Core makeup tank behavior observed during the ROSA-AP600 experiments", Nucl. Technol. Vol. 119, no. 112–122, 1997..

[19] ELSHAHAT, A, ABRAM, T, HOHORST, J, and CHRIS, A: "Simulation of the Westinghouse AP1000 Response to SBLOCA Using RELAP/SCDAPSIM", International Journal of Nuclear Energy Volume 2014, Article ID 410715.

[20] WHITEMAN, J. L.: "AP1000 European Design Control Document", PA 15230, DCP/DCA/EDCR: EPS-GW-GL-001, 002, 003, 004, and 006 (2009).

[21] GROUDEV, P. P., GENCHEVA, R. V., STEFANOVA, A. E., and PAVLOVA, M. P.: "RELAP5/ MOD3.2 investigation of primary-to-secondary reactor coolant leakage in VVER-440," Journal of Nuclear Engineering Design, 31, 961–74, (2004).