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RELAP5/MOD3.3 Model Assessment of Maanshan Nuclear Power Plant with SNAP Interface

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ABSTRACT

RELAP5 is a very important analysis tool for Taiwan Power Company and is still used for the transient analysis of the Taiwan NPPs. The version of RELAP5 for Taiwan Power Company is RELAP5/MOD3.3 and the input deck of RELAP5 is established by the ASCII files. Symbolic Nuclear Analysis Package (SNAP) is an interface of NPP analysis codes which developed by U.S. NRC and Applied Programming Technology, Inc. Different from the traditional input deck in ASCII files, the graphical control blocks and thermal hydraulic connections make researches comprehend the whole power plant and control system more easily. Additionally, for the last few years, the TRACE/SNAP model of Maanshan NPP was developed and several kinds of transient events were performed. Based on the past research experience and SNAP advantages, the RELAP5/MOD3.3 model of Maanshan NPP was developed with SNAP interface in this research. Maanshan NPP is located on the southern coast of Taiwan. Its nuclear steam supply system is a type of PWR designed and built by Westinghouse for Taiwan Power Company. A startup test data and two transient results were used to compare with the results of RELAP5/MOD3.3 model for the new-developed analysis model assessment. The predictions of RELAP5/MOD3.3 were consistent to the startup test and historical transient data results. It indicates that there is a respectable accuracy for the Maanshan NPP RELAP5/MOD3.3 model.

FOREWORD

U.S. NRC (United States Nuclear Regulatory Commission) is developing an advanced thermal hydraulic code named TRACE for nuclear power plant safety analysis. The development of TRACE is based on TRAC, integrating RELAP5 and other programs. U. S. NRC has determined that in the future, TRACE will be the main code used in thermal hydraulic safety analysis, and no further development of other thermal hydraulic codes such as RELAP5 and TRAC will be continued. A graphic user interface program, SNAP (Symbolic Nuclear Analysis Program) which processes inputs and outputs for TRACE is also under development. One of the features of TRACE is its capacity to model the reactor vessel with 3-D geometry. It can support a more accurate and detailed safety analysis of nuclear power plants. TRACE has a greater simulation capability than the other old codes, especially for events like LOCA.

Taiwan and the United States have signed an agreement on CAMP (Code Applications and Maintenance Program) which includes the development and maintenance of TRACE. However, the above CAMP also includes the application of RELAP5. Additionally, in Taiwan, RELAP5 is a very important analysis tool for Taiwan Power Company and is still used for the transient analysis of the Taiwan NPPs. Therefore, the RELAP5/MOD3.3 model of Maanshan nuclear power plant has been developed in this research. A startup test data and two transient event historical results were used to compare with the results of RELAP5/MOD3.3 model.

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EXECUTIVE SUMMARY

RELAP5/MOD3.3 Patch04 code, which was developed for light water reactor (LWR) transient analysis at Idaho National Engineering Laboratory (INEL) for U.S. NRC, is applied in this research. This code is often performed to support rulemaking, licensing audit calculations, evaluation of accident, mitigation strategies, evaluation of operator guidelines, and experiment planning analysis. Same as other thermal hydraulic analysis codes, RELAP5/MOD3.3 is based on nonhomogeneous and non-equilibrium model for the two-phase system. However, calculations in this code will be solved by a fast, partially implicit numerical scheme to permit economical calculation of system transients. It can produce accurate transient analysis results in relatively short time.

Symbolic Nuclear Analysis Package (SNAP) is an interface of NPP analysis codes which developed by U.S. NRC and Applied Programming Technology, Inc. Different from the traditional input deck in ASCII files, the graphical control blocks and thermal hydraulic connections make researches comprehend the whole power plant and control system more easily. Due to these advantages, the RELAP5/MOD3.3 model of Maanshan NPP was developed with SNAP interface.

Maanshan NPP is located on the southern coast of Taiwan. Its nuclear steam supply system is a type of PWR designed and built by Westinghouse for Taiwan Power Company (Taipower, TPC). In this research, a RELAP5/MOD3.3 model of Maanshan NPP is developed. Further, the model in this research is developed with the SNAP interface. A startup test data and two transient results were used to compare with the results of RELAP5/MOD3.3 model for the new-developed analysis model assessment. The predictions of RELAP5/MOD3.3 were consistent to the startup test and historical transient data results. It indicates that there is a respectable accuracy for the Maanshan NPP RELAP5/MOD3.3 model.

ABBREVIATIONS

BPV	Bypass valve
CAMP	Code Applications and Maintenance Program
FWPT	Feedwater pump(s) Trip
kg	kilogram(s)
kW	kilowatt(s)
MPa	Megapascal(s)
MSIVC	Main Steam Isolation Valves Closure
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NRWL	Narrow Range Water Level
Pa	Pascal(s)
PORVs	Power-Operated Relief Valves
psi	pounds per square inch
PWR	Pressurized Light Water Reactor
RCS	Reactor Coolant System
S/G	Steam Generator
SNAP	Symbolic Nuclear Analysis Program
SRV	Safety/Relief Valves
TCVC	Turbine Control Valves Closure
TRACE	TRAC/RELAP Advanced Computational Engine
TSVC	Turbine Stop Valve Closure
W	Watt(s)
US	United States

1 INTRODUCTION

Maanshan Nuclear Power Plant (NPP) is the third NPP in Taiwan. Also, it is the first Pressurized Water Reactor (PWR) located at the south of Taiwan. There are two units in the Maanshan NPP. The total power of the Nuclear Steam Supply System (NSSS) is 2785 MWt, which consist of 2775 MWt for reactor power and 10 MWt for cooling pumps [1]. For the last few years, our group has developed the models of Taiwan NPPs with TRACE code in SNAP interface [2, 3]. Further, it is necessary to perform the NPP transients with several analysis codes so that the data results could be compared with each other to ensure the consistency. Therefore, the RELAP5/MOD 3.3 code was chosen to develop a new Maanshan NPP model. Different from the traditional ASCI input deck, the RELAP5/MOD 3.3 model was developed with SNAP interface.

RELAP5/MOD3.3 Patch04 code, which was developed for light water reactor (LWR) transient analysis at Idaho National Engineering Laboratory (INEL) for U.S. NRC, is applied in this research. This code is often performed to support rulemaking, licensing audit calculations, evaluation of accident, mitigation strategies, evaluation of operator guidelines, and experiment planning analysis [4]. Same as other thermal hydraulic analysis codes, RELAP5/MOD3.3 is based on nonhomogeneous and non-equilibrium model for the two-phase system. However, calculations in this code will be solved by a fast, partially implicit numerical scheme to permit economical calculation of system transients. It can produce accurate transient analysis results in relatively short time, which means large amounts of sensitivity or uncertainty analysis might be possible.

Symbolic Nuclear Analysis Package (SNAP) is an interface of NPP analysis codes which developed by US NRC and Applied Programming Technology, Inc. Different from the traditional input deck in ASCII files, the graphical control blocks and thermal hydraulic connections make researches comprehend the whole power plant and control system more easily [5]. Due to these advantages, the RELAP5/MOD3.3 model of Maanshan NPP was developed with SNAP interface. Moreover, due to the SNAP interface, the analysis results could be transferred into animations which were more attractive and more understandable. With the animation, interactions of different components and parameters could be easily observed.

To ensure the applicability of this model, three startup tests including feedwater pumps trip (FWPT), turbine trip (PAT50) and main isolation valves closure (MSIVC) would be analyzed first. With the comparison of RELAP5 results and startup tests data, it shows that the RELAP5/MOD 3.3 model of Maanshan NPP is consistent with the startup tests data.

Moreover, three hypothetical accidents which referred to the TRACE model [5,6] would be performed. The comparison of RELAP data results and TRACE data results show great consistency.

2 MODEL ESTABLISHMENT

As the Maanshan NPP operated in normal conditions, coolant water in primary system will carry the heat generated by the fuel rods to the steam generator. Feedwater in the secondary system then obtain the heat, evaporate and drive turbines to generate electricity. According to the energy conservation principle, internal energy of steam, which had driven turbines, will decrease. This lower internal energy steam will then go through the condenser and be transferred into feedwater and re-injected into the steam generator. However, it is difficult to develop the entire recirculation system with the analysis code. The computational time will be impractically long and the mass balance will be hard to reach. Hence, as developing the RELAP5 model, it is practical to define the feedwater pumps and the turbines as the boundary conditions because the main purpose of this model is to obtain the NSSS reactions during the transient. For the NSSS system of Maanshan NPP, the feedwater pumps, auxiliary feedwater pumps, turbines, safety/relief valves, steam dump valves and Power Operated Relief Vavles (PORVs) were defined as the boundary conditions and developed by the Time Dependent Volume component in the RELAP5 program [6].

2.1 Hydraulic Components

As mentioned in section 2, there are 3 recirculation loops in the Maanshan NPP. In each loop, there is a Reactor Coolant Pump (RCP) and Steam Generator (S/G). On the hot leg of second loop, a Pressurizer which can adjust the pressure of RCS with the spray valves and electronic heater was developed. In this analysis model, there are several Branch components developed to simulate the reactor vessel. According to the core arrangement, Branch components from number 140 to 156 were connected together as the average fuel channel. Branch components from number 120 to 136 were connected together as the hottest fuel channel. Branch components from number 100 to 116 were connected together as the bypass flow channel, as shown in Figure 1. Also, these channels will be connected to the heat structure components to obtain the heat and do the reactor kinetic analysis.

For those 3 recirculation loops in primary side, they were developed by Pipe, Valve, Branch, Jump and Single Volume, as shown in Figure 2. For these three-digit components, the first digit stands for the loop number (2 for first loop, 3 for second loop and 4 for third loop). Further, the other digits of these components represents to the component types. For instance, component 280 is the recirculation pumps in first loop and component 380 is the recirculation pumps in second loop. Though the Pump component in RELAP5 code has been developed with the pump parameters from Westinghouse, pump characters of the RCPs in this model was input according to Taiwan Power Company NPP training materials and past research models which were calculated by RELAP5-3D and TRACE codes, as shown in Figure 3.

In addition to RCPs, another important thermal hydraulic component in the primary side is heat exchanger. Pipe 250, which was developed for heat exchanger in first loop, was divided into 8 nodes. According to the geometry of heat exchanger, junction between fourth cell and fifth cell was 180 degree as shown in Figure 4. Further, the heat structure component can be view as structural component once the both side of the heat structure were connected to thermal hydraulic components. Hence, the Pipe 250 was connected to the left boundary of heat structure 2500, as shown in Figure 5. Likewise, the Pipe 350 of secondary loop was connected to the left boundary of heat structure 3500 and Pipe 450 of third loop was connected to the left boundary of heat structure 4500.

Similar to primary loops, the secondary loops of Maanshan NPP were developed with Pipe, Valve, Branch, Pump and Single Volume. Specially, to simulate the feedwater, auxiliary feedwater and steam dump systems, which flow rate was determined by system feedback, the Time Dependent Junction was used. With the same rules of primary loops, the components' number in secondary loops was numbered in three digits. The first digit stands for the loop's number and the other two digits stand for the component types. For instance, the component "520" were heat exchanger in first loop because the first digit "5" represents the first loop and the latter two digits "20" represents the heat exchanger. Component 520 was connected to the right boundary of heat structure 2500, which allows the heat transfer from component 250 in primary side to component 520 in secondary side. Due the heat from primary side, the water in component 520 will evaporate and go through the next component 522. Component 522 was a separator which can increase the quality up to 99.7%. This dried steam will then leave the separator and go through the Main Steam Line Isolation Valve (component 543), Turbine Control Valve (component 774), Turbine Stop Valve (component 775) and drive turbine, as shown in Figure 6.

As mentioned in section 2, the steam dump system was composed by 10 steam dump valves, 6 turbine bypass valves and several controlling equipment. To save the computational time, this RELAP5 model merged 10 steam dump valves into 4 groups. Each group was developed by a Time Dependent Junction component which the total steam flow rate was consistent to the operating conditions. Likewise, 6 turbine bypass valves were developed by 2 Time Dependent Junction components.

To simplify the feedwater control system, the feewater pumps and valves were developed by Time Dependent Volume and Time Dependent Junction respectively. For the Time Dependent Volume components, the fluid boundary conditions were referred to the thermal hydraulic properties of feedwater during operation. Therefore, the control system need only concern the effect of Narrow Range Water Level (NRWL), steam flow rate and feedwater flow rate to determine the feedwater flow rate. Once the flow rate was determined, Time Dependent Junctions which were connected with the control blocks in the feedwater control system will inject the adequate feedwater into recirculation loops. Details of the feedwater control system will be discussed in the following section.

2.2 Control Systems

In operation, the purpose of water level/feedwater control system is to ensure that water in the steam generator can cover the heat exchanger. For Maanshan NPP, the feedwater flow rate was determined by three units including NRWL in steam generator, steam flow rate and feedwater flow rate. As the water level deviate the setting values, the control system will adjust the injection of the feedwater flow rate to maintain the water level of the steam generator. Further, two water level measuring systems, including the NRWL and Wide Range Water Level (WRWL), calculated water level with pressure difference. Different from the TRACE model of Maanshan NPP which our group had developed before, there is no water level sensor signal component in the RELAP5 code. As a result, the measurements of water level were developed and composed with density, pressure and volume signal components which was shown in Figure 7.

In addition to the feedwater control, the steam dump system was also an important response mechanism. As mentioned above, the steam dump system of Maanshan NPP can be divided into two types including the pressure control mode and the T_{ave} mode. The pressure mode was initiated as core power was in range from 0% to 10%, which will not be discussed and applied in this research. Hence, the setting of the steam dump system was only referred to the response of T_{ave} mode. As shown in Figure 8, there are 3 control blocks with “sum” function calculated the average core temperature values of loop 1 to loop 3 respectively. Then, the control block 308 with “max” function will compare the maximum of average core temperature (T_{ave}) in loop 1 to loop 3 with No Load Temperature ($T_{no\ load}$, 564K in Maanshan NPP). Referring this comparison, the control blocks 318 can convert the difference of T_{ave} and $T_{no\ load}$ into steam dump flow rate with Table 15. As the temperature difference exceeded 0% (0℉), the first group of dump valves was opened. As the temperature difference exceeded 16% (15.8℉), the first groups of dump valves was fully opened and the second group of dump valves started to open and so on.

The pressure and water level control system of pressurizer includes the heater and the spray valve. There two types of heater including control heater and backup heater. The control heater and spray valves were applied for adjusting the pressure inside the pressurizer.

Frome Figure 9, the pressure of pressurizer will be compared to rated pressure in control block 120. With the comparison of these two pressure values, the difference can be transferred into the open of the spray valve (control blocks 123) and the power of heater (control block 121 and heat structure 1212 and 1222). However, the control heater is also related to the lower water level of pressurizer (control block 121). As the water level has been lower than 14%, the power of control heater will be zero (control block 124), which means the heater trip. In addition, if the trip setting of control block 121 was assigned to other trip signals, then the control heater can be tripped manually.

The backup heater is related to the charging control system of pressurizer. As shown in Figure 10, the maximum core temperature will be transferred into program water level through control block 130. Then, the water level will be subtracted from the actual water level. If the difference of these two water levels is larger than 5%, the backup heater will be initiated (control block 132). Further, the water level will be transferred into charging flow rate (control block 136) to adjust the water level inside the pressurizer. However, as the safety injection signal is initiated, the charging flow rate will be forced to zero.

2.3 Reactor Kinetics

In this RELAP5 model of Maanshan NPP, there are two sets of heat structures, which component numbers are 1201 and 1601, developed to simulate the hot fuel channel and average fuel channel. These heat structures were divided into 16 nodes (shown in Figure 11) in axial and 7 nodes in radial (shown in Figure 12). For the axial nodes, they were connected to the Brach components of the reactor core respectively. For radial nodes, the first 4 nodes stand for fuel pellets; the fifth node is filled helium inside the fuel rod and the sixth and seventh nodes are fuel cladding. The materials of each node can be defined manually. In this model, thermal properties (thermal conductivity and thermal capacity) of material 1 for the first 4 nodes were referred to that of Uranium dioxide. The material 2 for node 5 was referred to the helium thermal properties and the material 3 for node 6 and 7 was referred to that of Zircalloy.

Heat source of the heat structure can be set with the total reactor power or power table. In this model, at the beginning of the model assessment the heat source was set with power table which was referred to the startup test data results of Maanshan NPP to ensure the applicability of the thermal hydraulic components. After that, heat source of the heat structure would be set with total reactor power to ensure the point kinetic feedback calculations. For those heat structure components which were developed as the fuel bundles, the left boundaries were set as “symmetry” and the right boundaries were connected to the Branch components. For these connections of Brach components and heat structures, the power ratio should be defined (as shown in Figure 13) respectively to calculate the correct heat transfer. The power ratio setting of this RELAP5 model was referred to the TRACE and RETRAN model which were fully developed and assessed before.

For the point kinetic model, in addition to defining the power ratio of each node of heat structures, the ratio and position of reactivity feedback should also be defined. The reactivity feedback is dominated by Doppler Effect and Moderator density effect. The previous one is related to the temperature of fuel rods; hence, the check list of fuel temperature and reactivity should be added into the Power components. With this table (as shown in Figure 14), the RELAP5 code can calculate the corresponding reactivity feedback due to fuel temperature. Further, the fuel temperature feedback ratio should also be defined manually in the “Heat Weighting” settlement of Power component (shown in Figure 15). Similarity, to calculate the Moderator feedback, the checklist of coolant density and reactivity should be defined (shown in Figure 16). Then, with the volume weighting list (shown in Figure 17), the Branch components which were developed as the reactor core would be connected to the point kinetic calculation. With these settings, the RELAP5 code could calculate the power variation due to temperature and density changes inside the reactor core during transient events.

As mentioned above, the startup assessment transient events were calculated with power table first to ensure the applicability of thermal hydraulic components. Then, the point kinetic model would be applied to do the whole assessment of Maanshan RELAP5 model. As performed with power table, the RELAP5 model needs no control system to simulate reactor scram. However, when performed with point kinetic model, the reactor scram control is required. For instance, Table 100 is the scram reactivity feedback table which will start to dominate the power variation once the trip logic/variable gate is initiated as shown in Figure

18. From this figure, it is obvious that the table could cause a large negative reactivity feedback in few seconds to simulate the control rods insertion.

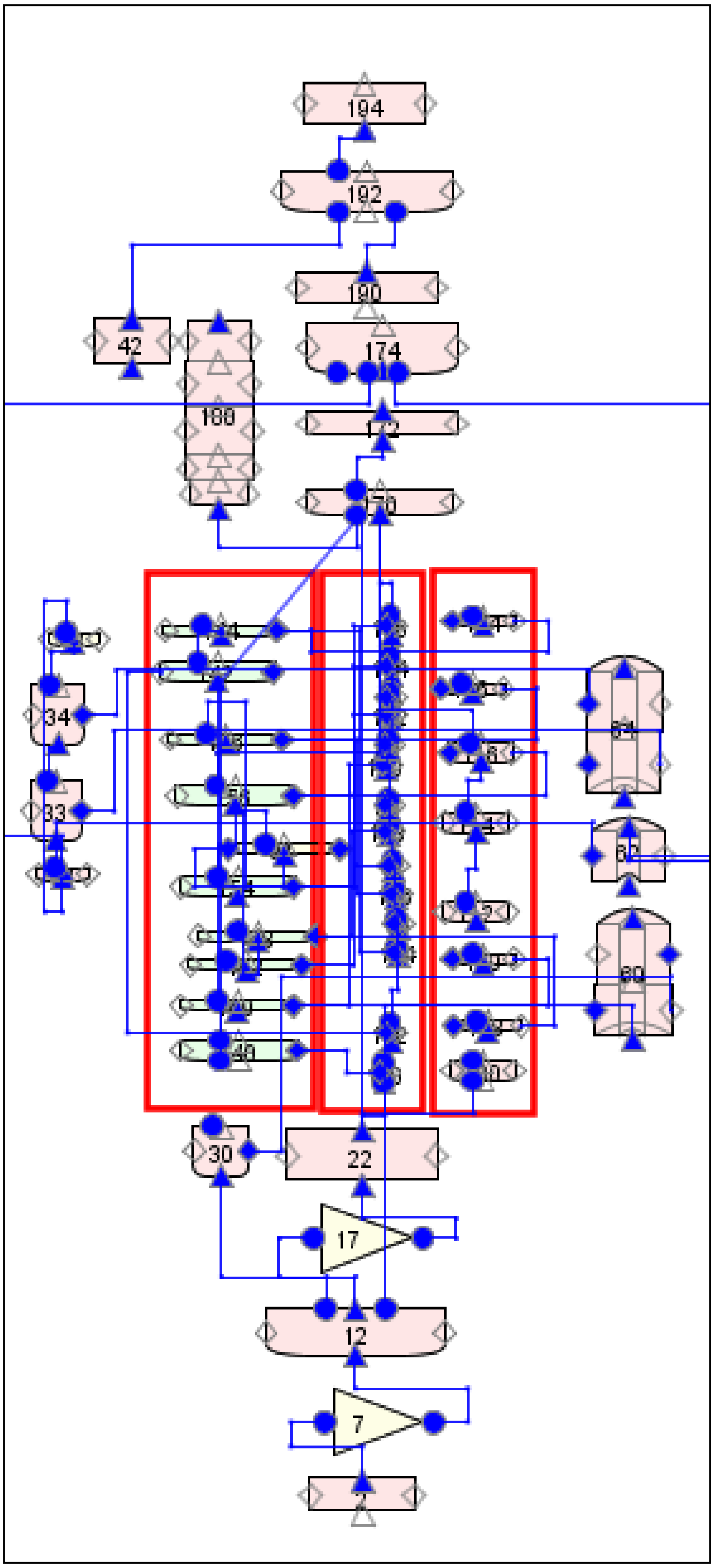


Figure 1　Reactor Core Components of Maanshan NPP in SNAP Interface

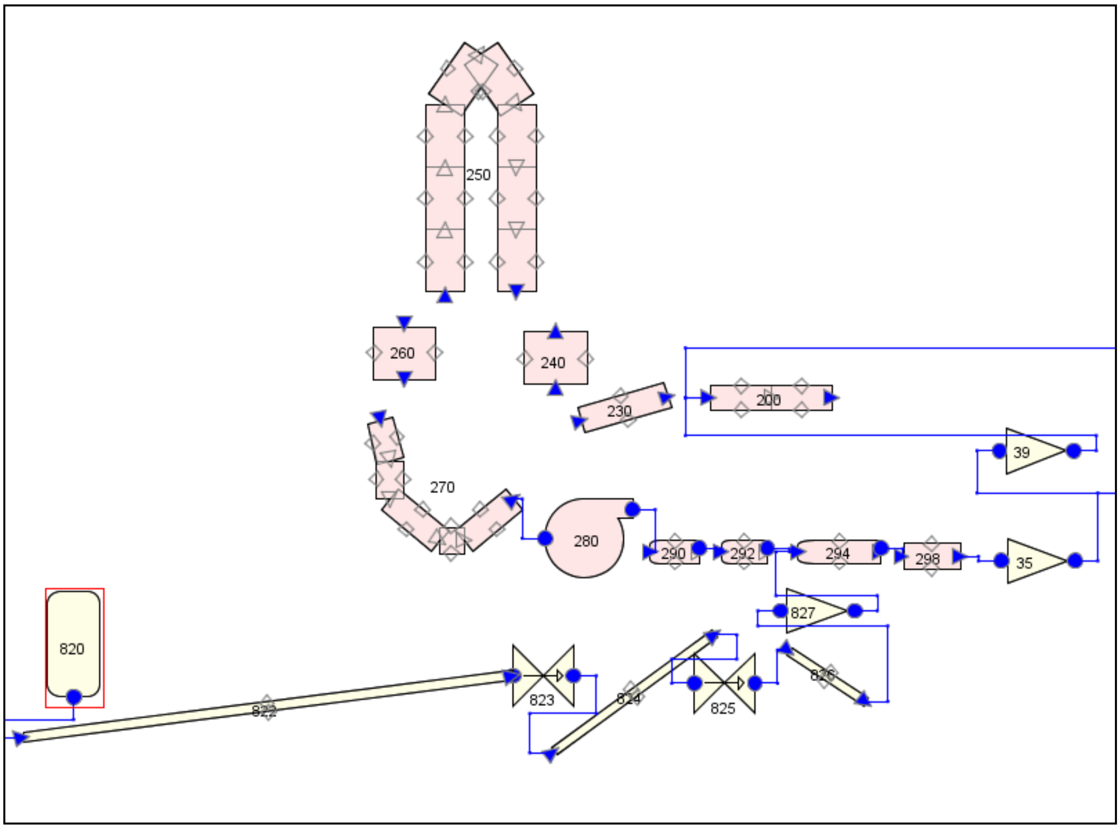


Figure 2 Components of First Loop of Maanshan NPP in SNAP Interface

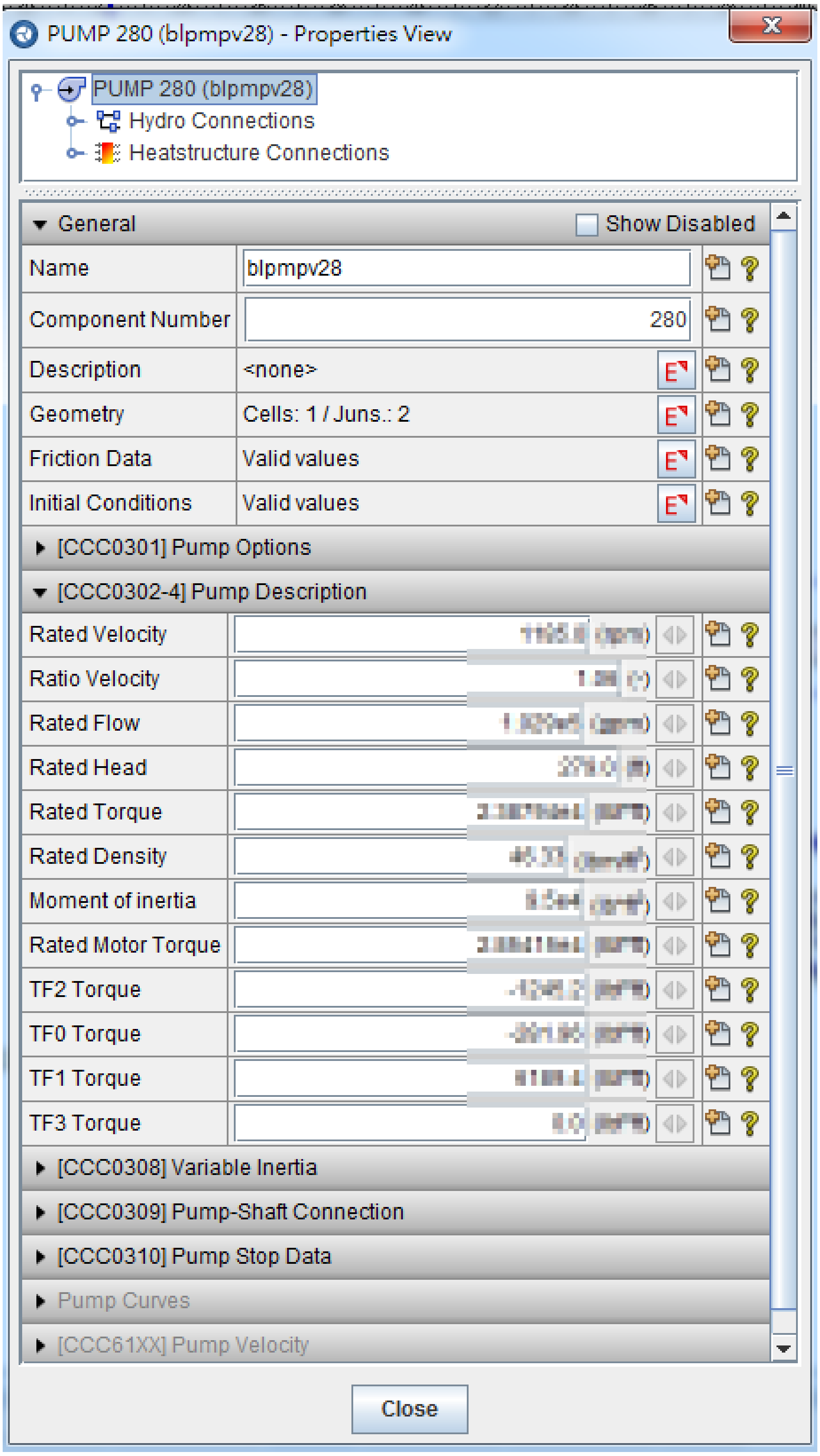
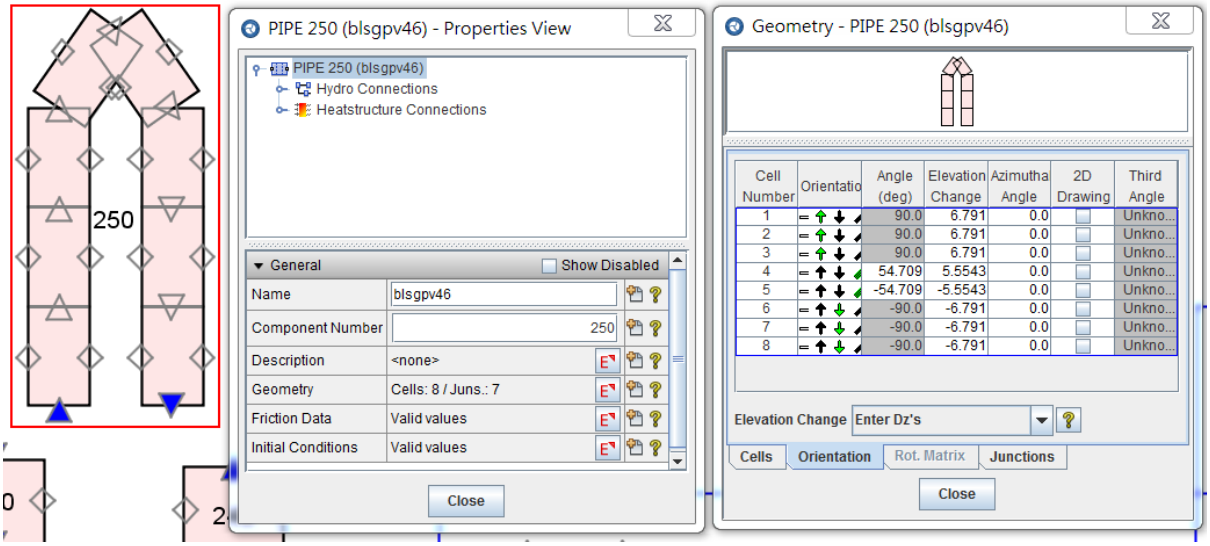


Figure 3 Recirculation Pump Properties of Maanshan NPP in SNAP Interface



Cell Number	Angle (deg)	Elevation Change	Azimuthal Angle	Third Angle
1	90.0	6.791	0.0	Unkno...
2	90.0	6.791	0.0	Unkno...
3	90.0	6.791	0.0	Unkno...
4	54.709	5.5543	0.0	Unkno...
5	-54.709	-5.5543	0.0	Unkno...
6	-90.0	-6.791	0.0	Unkno...
7	-90.0	-6.791	0.0	Unkno...
8	-90.0	-6.791	0.0	Unkno...

Figure 4 Heat Exchanger Component 250 of Maanshan NPP in SNAP Interface

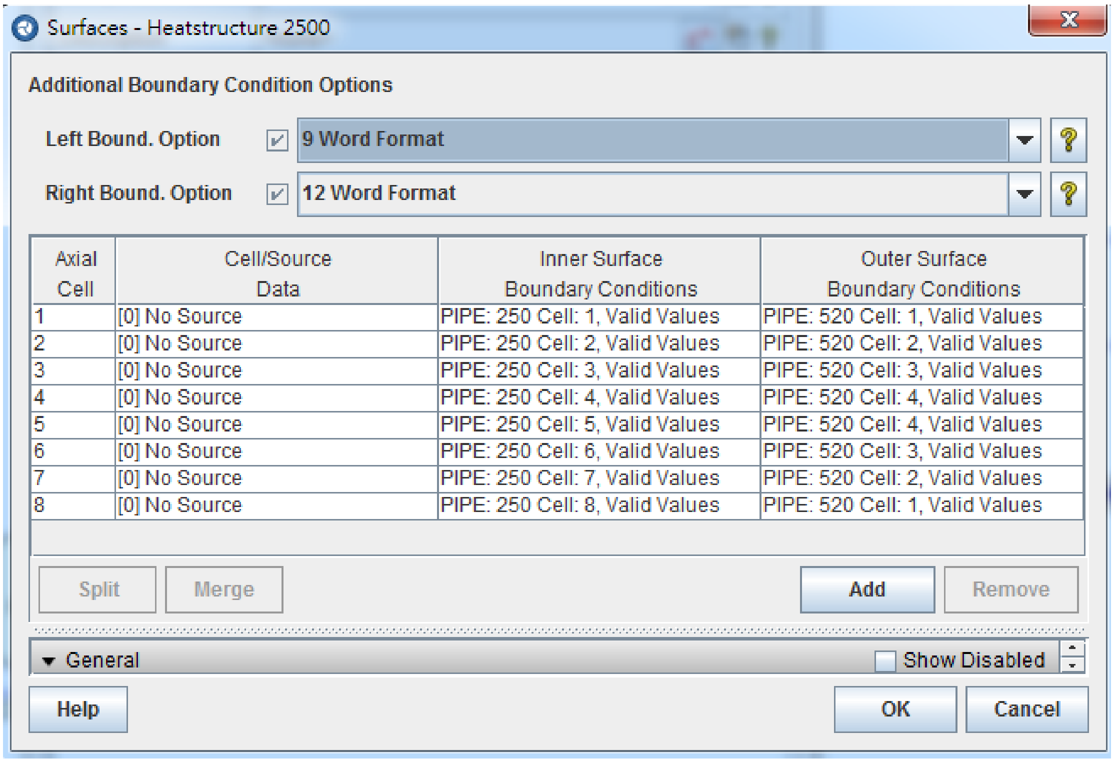


Figure 5 Heat Structure 2500 Properties in SNAP Interface

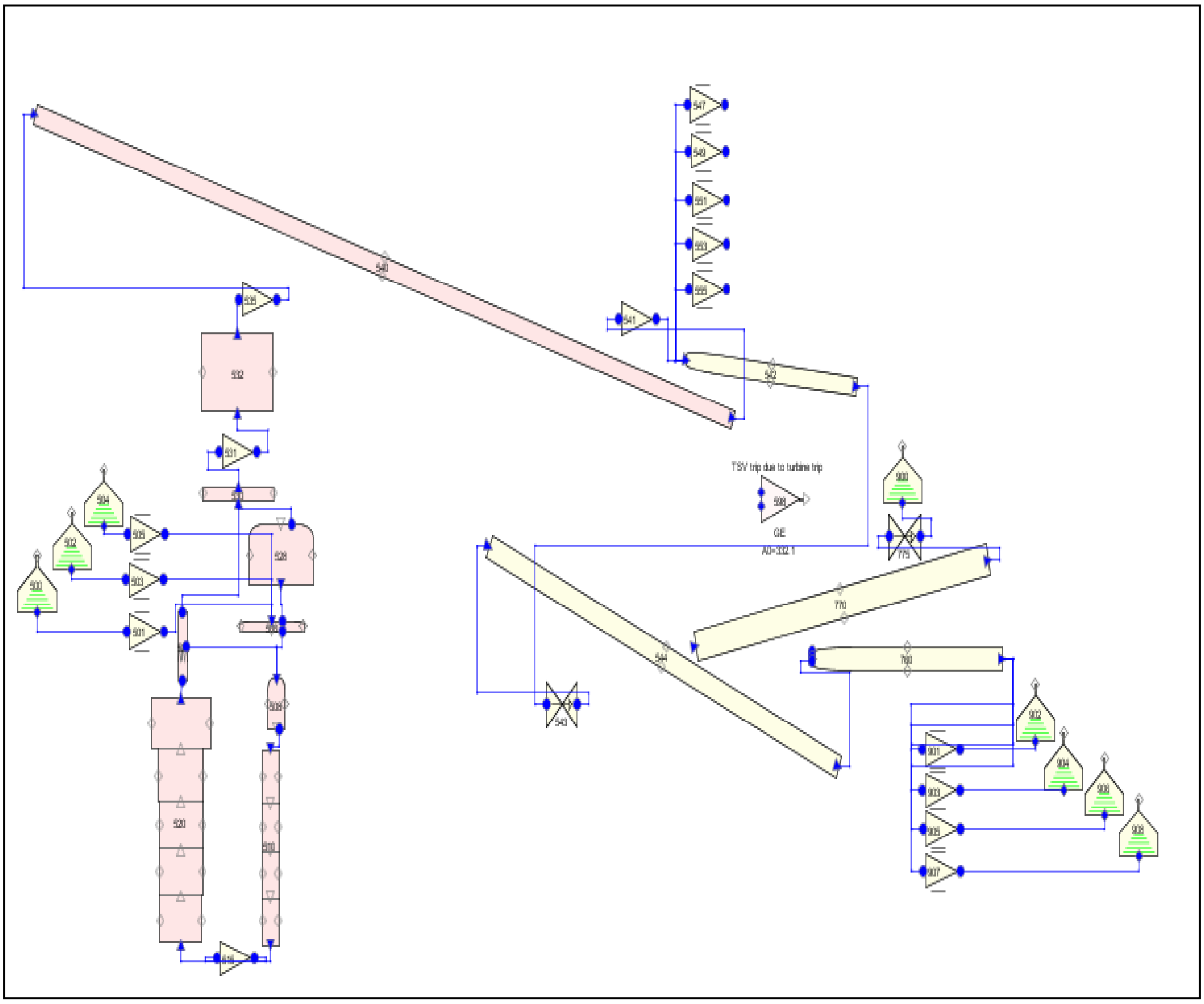


Figure 6 Components in Secondary Side of Maanshan NPP in SNAP Interface

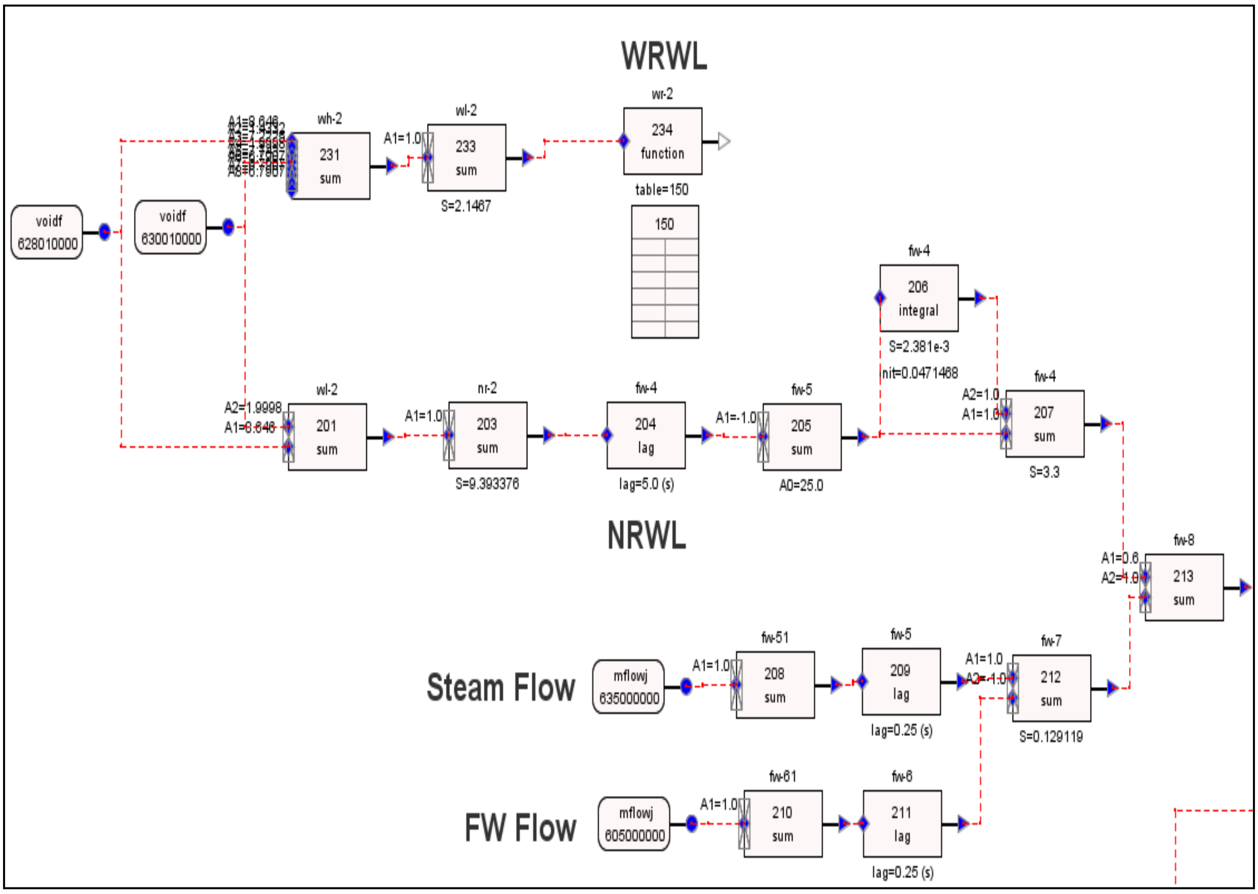


Figure 7 Feedwater Control System in SNAP Interface

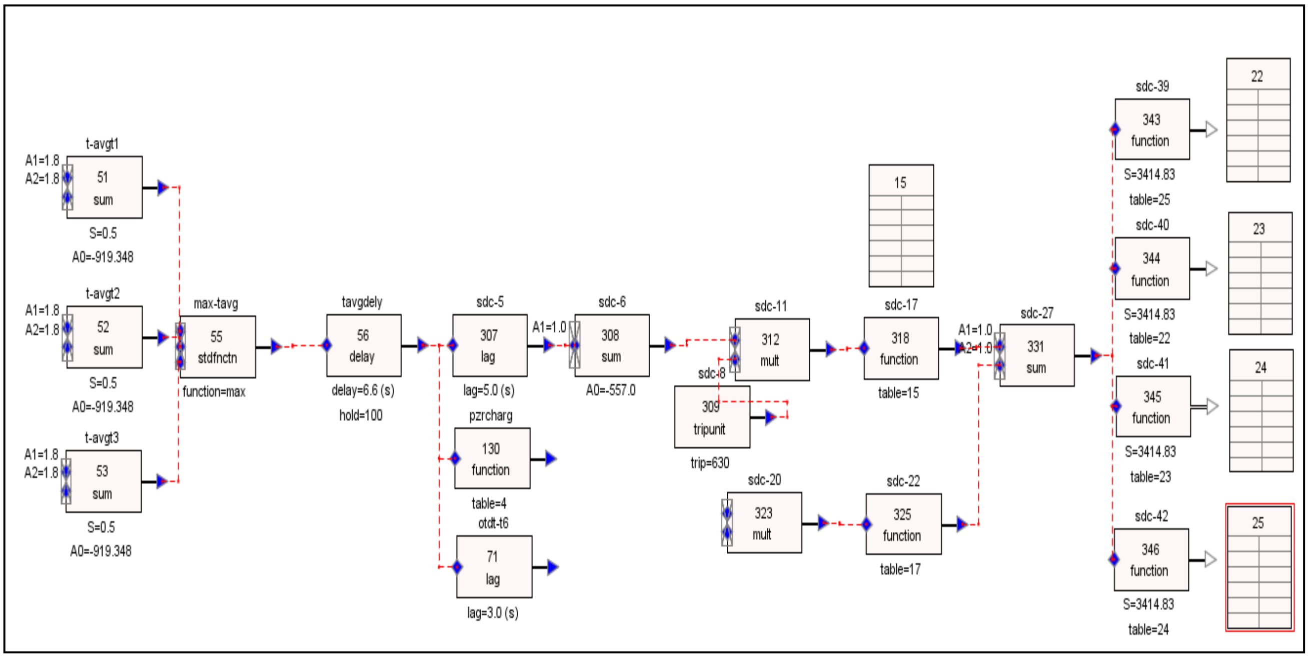


Figure 8 Steam Dump Control System in SNAP Interface

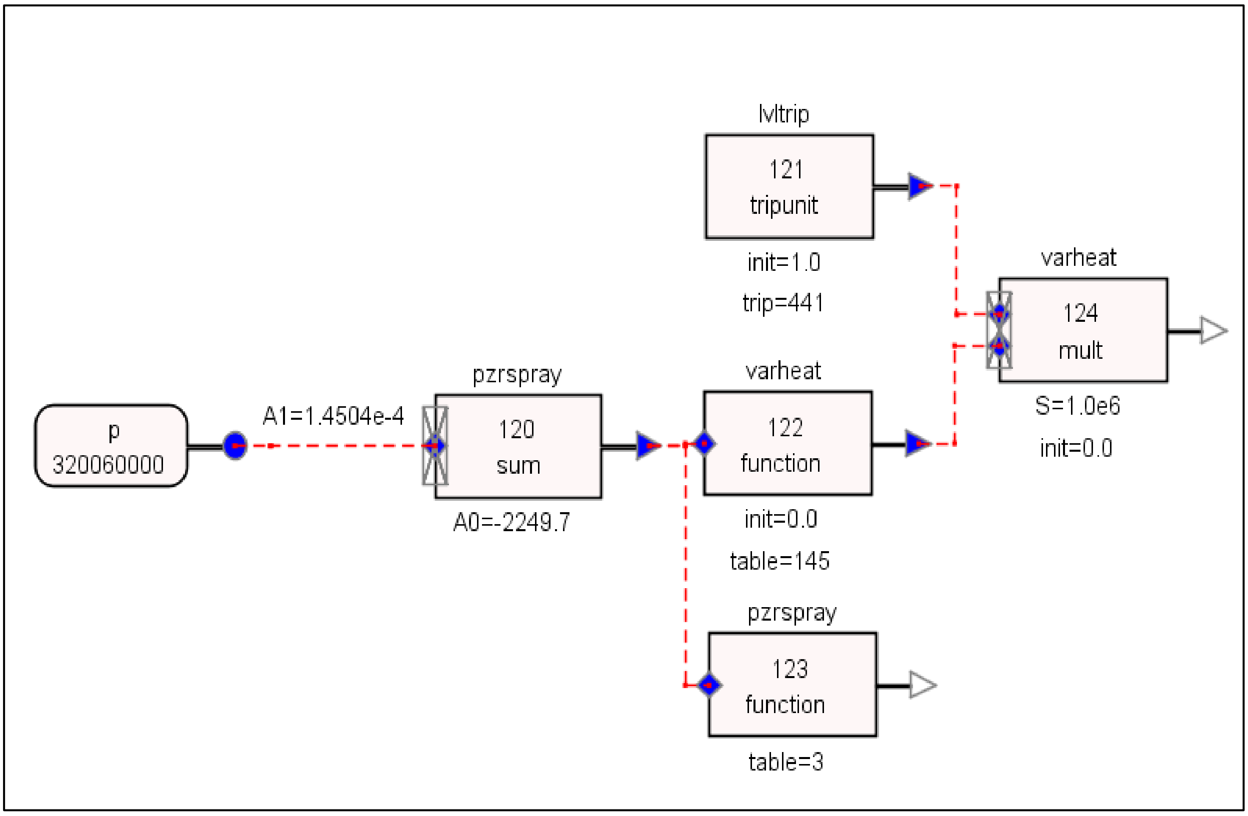


Figure 9 Heater of the Pressurizer Control System in SNAP Interface

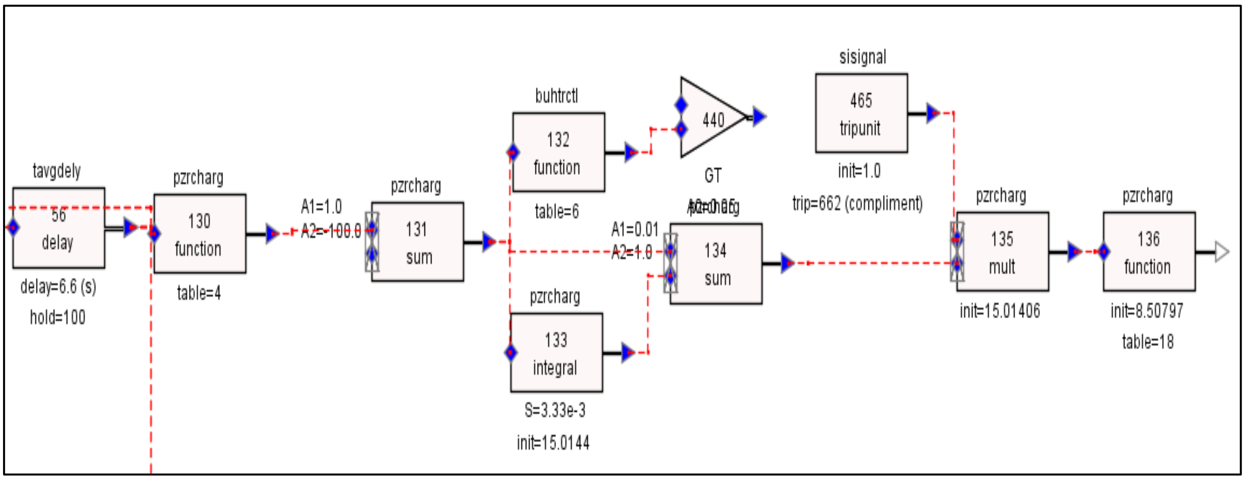


Figure 10 Pressurizer Injection Control System in SNAP Interface

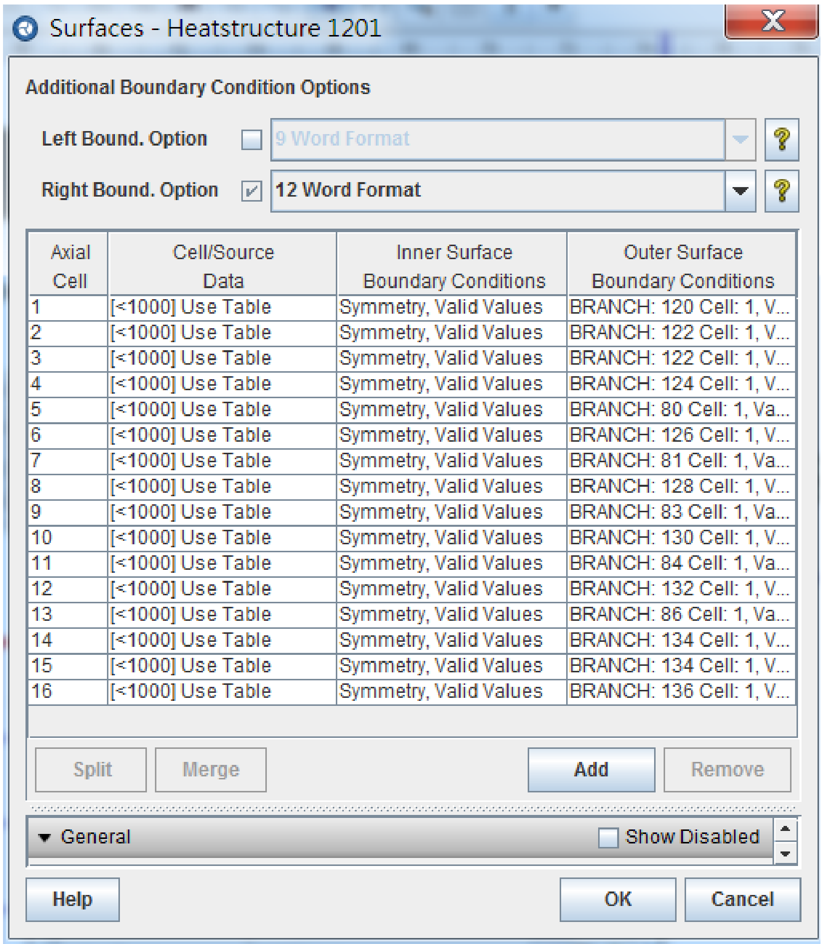


Figure 11 Properties in Axial Direction of Heat Structure 1201 in SNAP

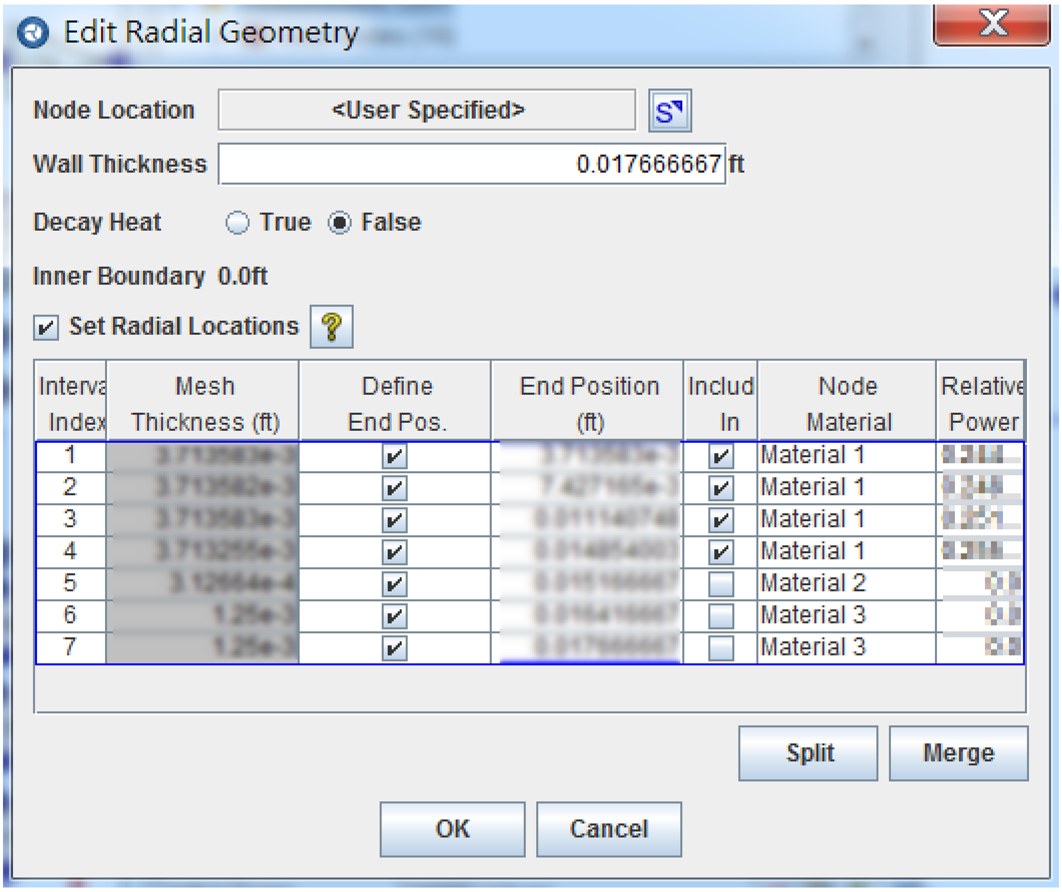


Figure 12 Properties in Radial Direction of Heat Structure 1201 in SNAP

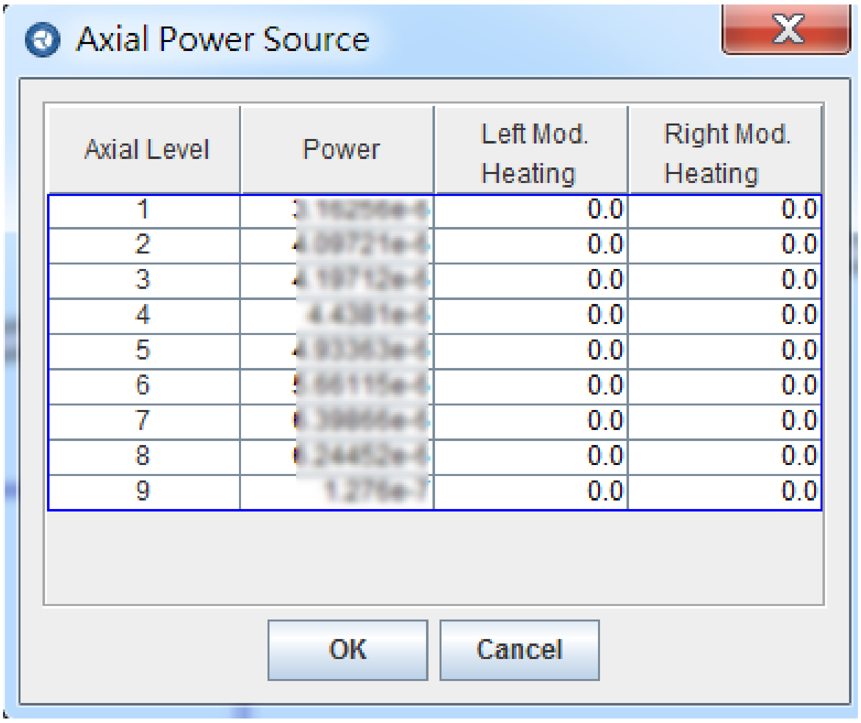


Figure 13 Power Ratio for Branch 100 to 116 of Heat Structure 1201

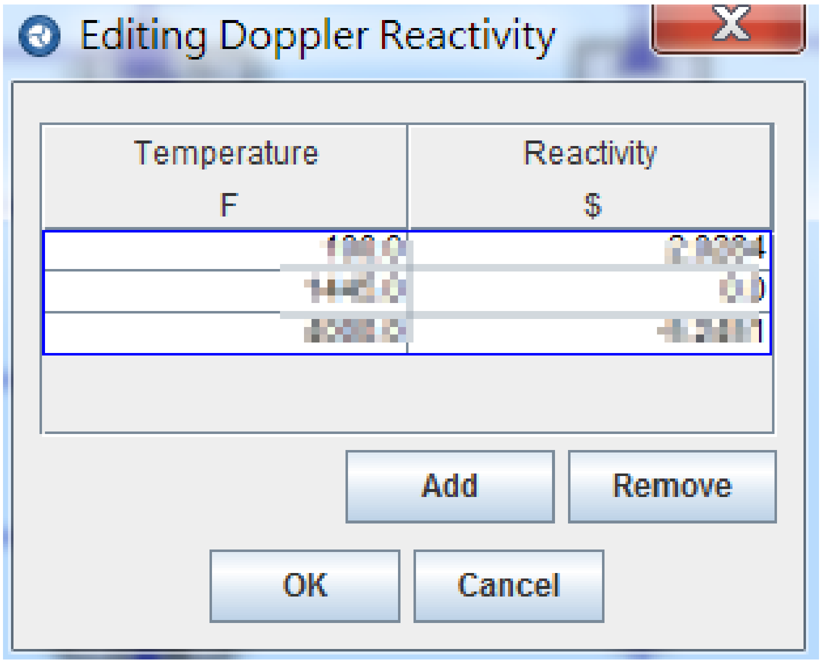


Figure 14 Doppler Effect Reactivity Feedback Table

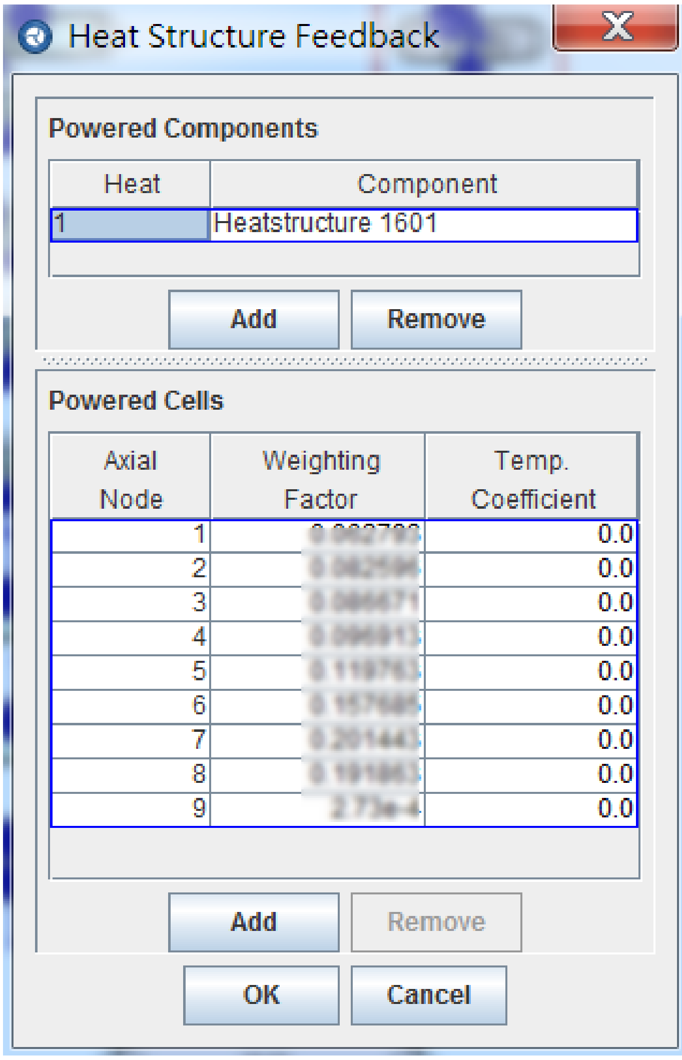


Figure 15 Doppler Effect Heat Structure Weighting Factor

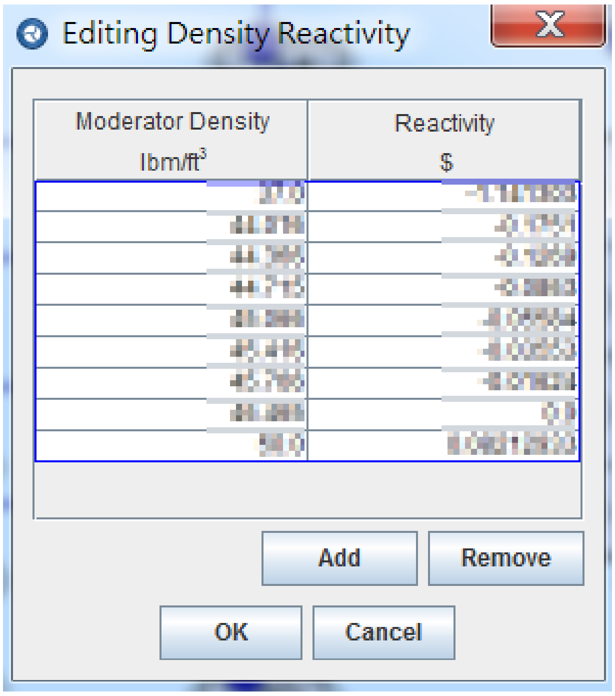


Figure 16 Density Effect Reactivity Feedback Table

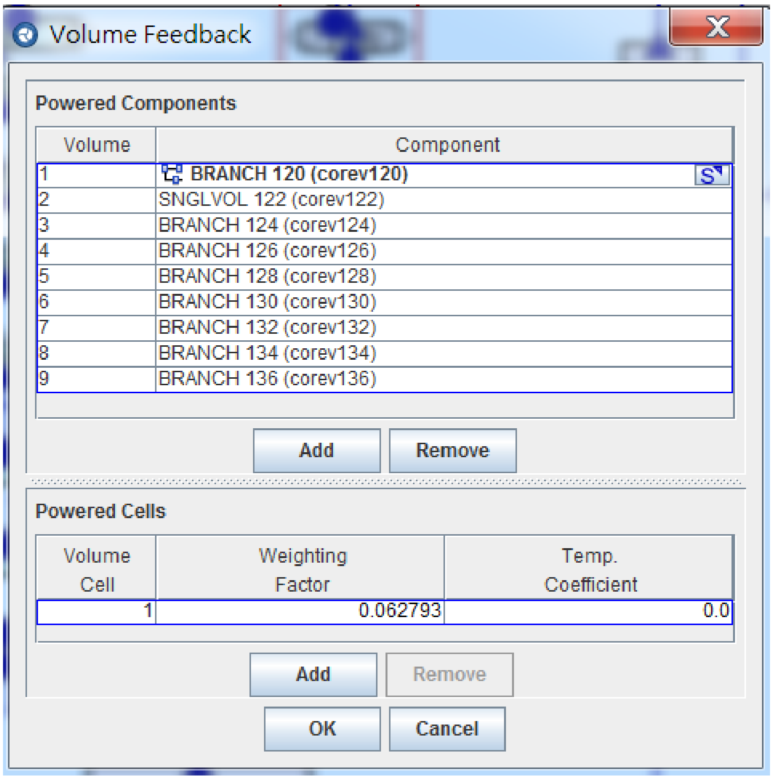


Figure 17 Moderator Density Effect Volume Weighting Factor

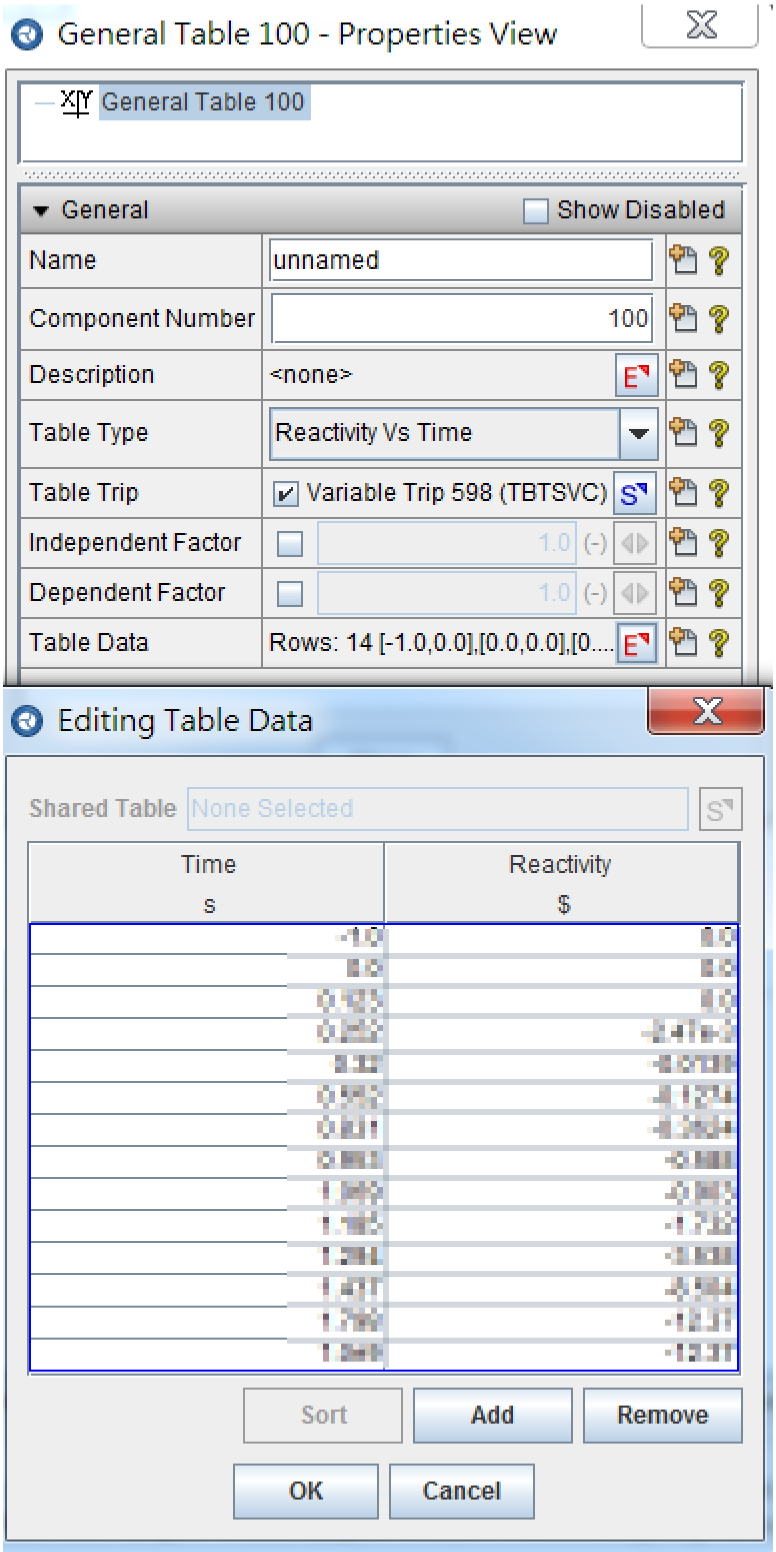


Figure 18　Reactor Scram Reactivity Feedback Properties

3 ANIMATION MODEL IN SNAP INTERFACE

The RELAP5 animation model of Maanshan NPP is developed referring to the TRACE animation model which was developed by our group before [3]. Figure 19 is the RELAP5 animation model of Maanshan NPP in SNAP interface. In this figure, three steam generators were surrounded the reactor vessel. Three pairs of “pipe segment” components with three “pump” components were developed to connect the steam generators and reactor vessel. Pumps turn green if they are on work and turn red if they are out of work. The recirculation pumps are connected to the trip signal 595 to determine whether it works or not. The pipe segment components are connected to the corresponding “Pipe” components of the RELAP5 model respectively. Further, these components in the animation model will change color with different fluid condition. The fluid condition and temperature color maps are developed at the left hand side.

Steam lines with several valves are developed from the outlet of the steam generators to the turbine inlet. On the steam lines, there are MSIVs, PORVs and SRVs which will turn green if they are opened and turn red if they are closed. The MSIVs are connected to the trip signal 670 which is the MSIVC signal in the FWPT transient. The connection might be changed for different transient analysis if the control signals are different. Likewise, the PORVs and SRVs are connected to the corresponding trip signal to show the status of components during the transient. The turbine was assumed as the boundary condition of the Maanshan NPP model; hence, it was not developed in the animation model. At the inlets of the steam generators, there are several feedwater pumps animation model developed including feedwater pumps, auxiliary feedwater pumps and so on. In the FWPT transient, for instance, the feedwater pumps tripped as the trip-signal 744 was initiated. As a result, the feedwater pumps were all connected to this signal to determine when they should turned red.

In the reactor vessel, the upper plenum and lower plenum are connected to “Branch” components 192 and 12 respectively. A “Volume Stack” animation component is developed inside the reactor vessel to illustrate the fluid condition of the reactor cores during the transient. This component is developed with 12 by 4 rectangles, which can be connected to different hydraulic components to show the fluid condition respectively. Covering the stack components, 4 “Control Rod” animation components are developed to illustrate the reactor scram timing. As a result, for instance, the scram trip signal 622 (for FWPT transient) was connected to these Control Rod components to control the control rods insertion.

In addition to the hydraulic component animation model developing, several indicators of important parameters are also developed in this animation model. “Data Value“ components of the animation model are developed on the top of every feedwater pumps and steam lines to show exact values of the flow rate. At the lower left corner of the animation model, there is a list to show some important parameters such as core power, steam generator pressure, pressurizer pressure, core flow rate, cladding temperature and hot leg temperature. Right from the list, there are three “Linear Dial” animation components, which was connected to control variable 153, 203 and 253 respectively to show the NRWL of each steam generator. At the bottom of the reactor vessel, there is a list composed with several “Annunciator” animation components to show the event sequence of the transient. As the specific event or operation is initiated, this component will turn yellow.

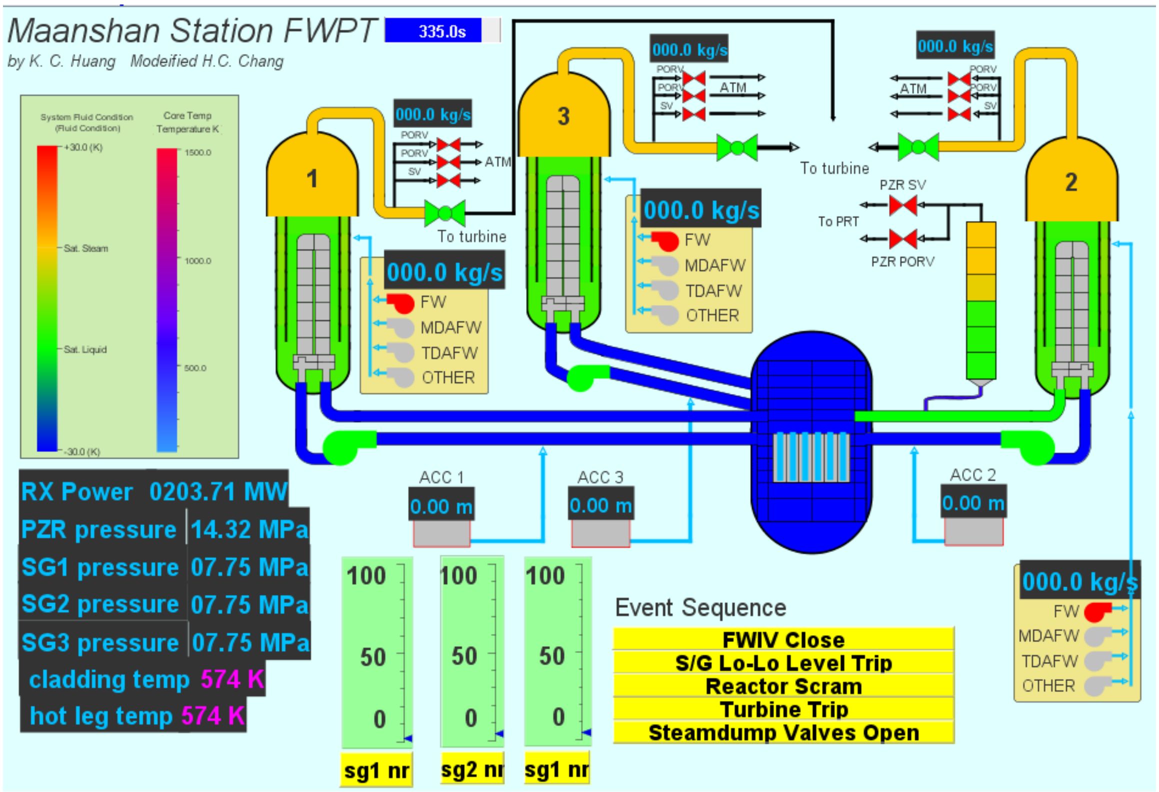


Figure 19 RELAP5 Animation Model of Maanshan NPP in SNAP Interface

4 ANALYSIS RESULTS

Based on the researches before, the turbine trip (PAT50), main steam line isolation valves closure (MSIVC) and feedwater pumps trip (FWPT) were applied to assess the applicability of Maanshan NPP RELAP5/MOD3.3 model.

In the RELAP5 analyses, the reactor power could be defined manually with table or calculated with point kinetic model. In this research, the power table mode would be performed first to ensure all the hydraulic components were suitable for the transient. After confirming all the thermal properties of hydraulic components were correct, the point kinetic model would be applied to assess the entire RELAP5 model. In the following sections, test conditions and event sequences of three selected transients were described [7].

4.1 Main Steam Line Isolation Valves Closure (MSIVC)

Event Sequence

The Main Steam Line Isolation Valves Closure (MSIVC) event happened at 15:00, 09/17/1987. Before the transient, the MSIV control signal failed. After checking, the staffs thought that this error might come from the open circuit of coil B. However, before repairing coil B, the coil A was accidently touched, which caused coil A loss power. As a result, the MSIV 108B closed rapidly and steam line of second loop was in low pressure, which initiated the safety injection. The reactor scram and turbine tripped.

The initial conditions of MSIVC were described in Table 1. There are three data sets including the plant data, RELAP5 (PT) which reactor power were defined with power table and RELAP5 (PK) which reactor power were calculated with Point Kinetics. In the beginning of the analysis, the power was 2752 MWt (99.17%) for all these three cases. From this table, it is obvious that both calculations of RELAP5/MOD3.3 model were consistent with the plant data.

As the MSIV closed rapidly, the steam flow rate decreased rapidly. The pressure in the steam generator hence increased, which would then initiate the PORV open. The PORV opened at 5.35 second after the transient start. The reactor scrammed at 5.8 second due to steam line low pressure in second loop. The turbine tripped and turbine stop valves closed together with the reactor scram; however, the turbine stop valves would be initiated 0.1 second after the reactor scram due to the electronic signal delay. After 6.1 seconds, pressure of steam generator reached to the limit of the steam dump system. The steam dump valves opened. Details of the event sequence were described in Table 2.

Table 1 Initial Conditions of MSIVC Transient

Parameters	Plant data	RELAP5 (PT)	RELAP5 (PK)
Power (MW)	2752	2752	2752
Core Temperature (K)	581.37	582.12	582.45
Feedwater Flow Rate (kg/sec)	521.63	533.92	539.22
Steam Flow Rate (kg/sec)	522.99	533.52	538.70
PZR Pressure (MPa)	15.41	15.27	15.34
PZR Water Level (%)	54.3	56.0	56.5
S/G Pressure (MPa)	6.84	6.84	6.85
S/G Water Level (%)	50.0	49.8	49.8

Table 2 The Sequence of MSIVC Transient

Event (sec)	Plant data	RELAP5 (PT)	RELAP5 (PK)
Loop 1 MSIV Close	1.0	1.0	1.0
S/G PORV Open	5.35	6	5.11
Reactor Trip	5.8	5.8	6.0
Turbine Trip	5.9	6.0	6.01
Steamdump Valves Open	6.1	6.1	6.1

Analysis Results

This section describes the analysis data results of MSIVC transient event. Plant data, RELAP5 (PT) and RELAP5 (PK) analysis results were plotted together to ensure the consistency. The important parameters during the transient were shown in Figure 20 to Figure 25.

Before the transient start, the reactor was operated at 2752 MWt (98.18%). At one second, the MSIVs in first loop closed rapidly due to the coil A being touched. The steam flow rate decreased immediately and the pressure of steam generator increased gradually. At about 5.35 second, the pressure of first loop reached to the set-point of PORV; the PORV opened as a result. At 5.7 second, the pressure drop signal, after the lead lag processing by the electronic circuit, was lower than the steam line low pressure set-point 4.13 MPa so that the safety injection was initiated. Because of the safety injection, the reactor scrammed and the turbine tripped. The core power dropped immediately. The turbine first step pressure and the reference temperature, which were related to the core power, decreased as a result. Due to the turbine trip, the turbine stop valves closed and the steam flow rate decreased. After the steam dump valves open, steam flow rate of the second and third loop decreased with a vibration.

Pressure of the header decreased at the beginning because of the turbine control valves open. After the reactor scram, the pressure of the header increased due to the closure of the TSVs but in the end, the pressure decreased again because the steam dump valves opened. The pressure of pressurizer and water level only increase slightly and after reactor scram, these two parameters decreased slowly. The core temperature decreased after the reactor scram. It should be noted that the core temperature of first loop was slightly higher than that of second loop and third loop because of the MSIV closure.

Figure 20 was the variation of core power during the transient. The power dropped down after the reactor scram due to the steam line pressure lower than 4.13 MPa at 5.3 second. Figure 21 was the variation of core temperature. Different from the next two transient, the analysis results of core temperature in MSIVC transient was not so consistent to the plant data. The difference might come from the system balance of the calculation. Further, to save the computational time, the analysis model was developed with less component nodes. However, the variation trend was similar.

Figure 22 and 23 were water level and pressure of pressurizer respectively. From these figures, it was obvious that the pressure and water level decreased due to the reactor scram. Further, the pressure decreasing in the analysis results of both RELAP5 models were much more than that in plant data because of less nodes in pressurizer component.

Figure 24 shows the variation of water level of steam generator in first loop. Because of the closure of MSIV, the pressure increased substantially and as a result the water level of steam generator decreased. Figure 25 was the steam flow rate variation in first loop. After the MSIV closure, the steam flow rate decreased. Later, because of the steam dump valves open, the steam flow rate increased slightly to exhaust the decay heat inside the steam generator.

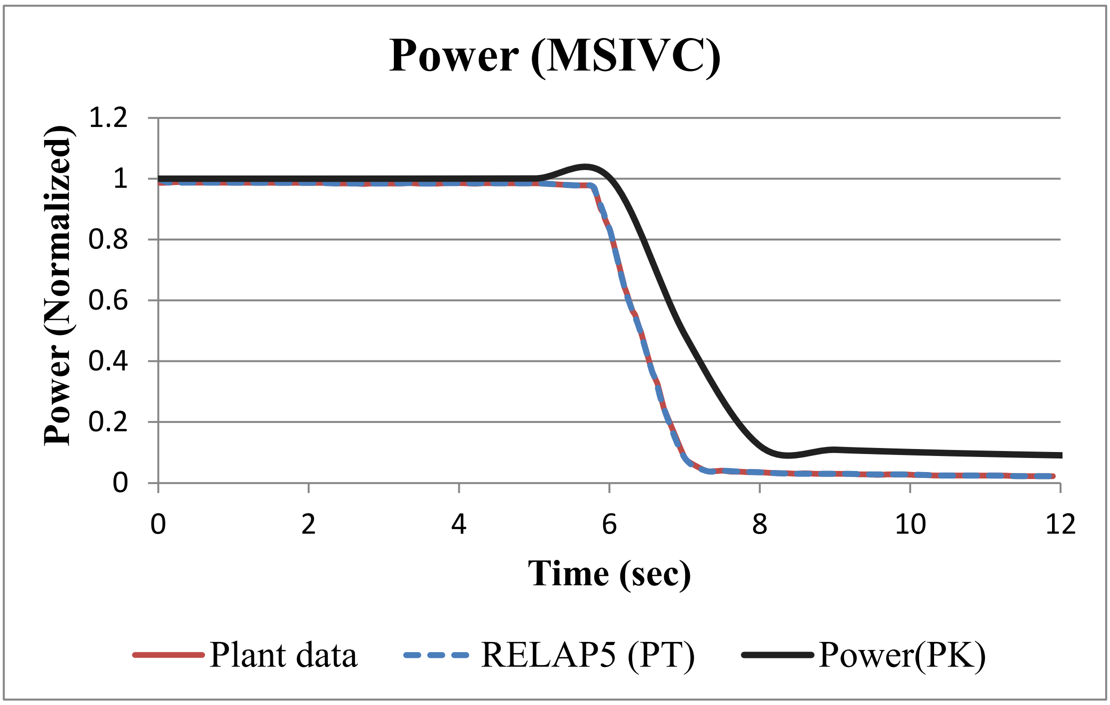


Figure 20 Power History during MSIVC Transient

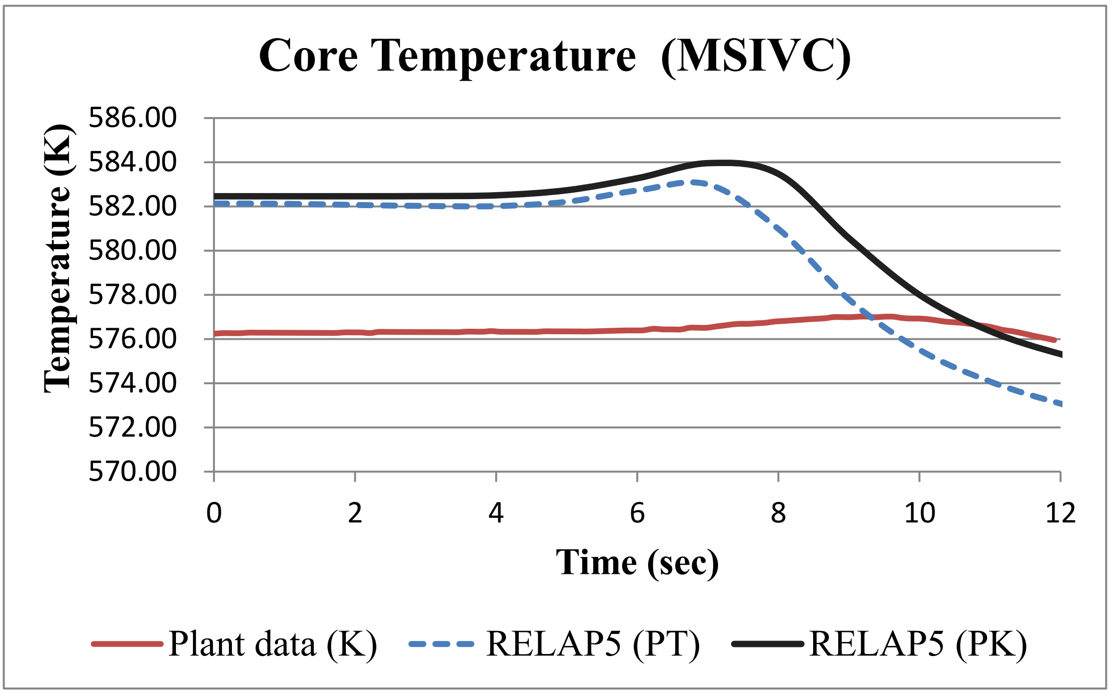


Figure 21 Core Temperature Variation during MSIVC Transient

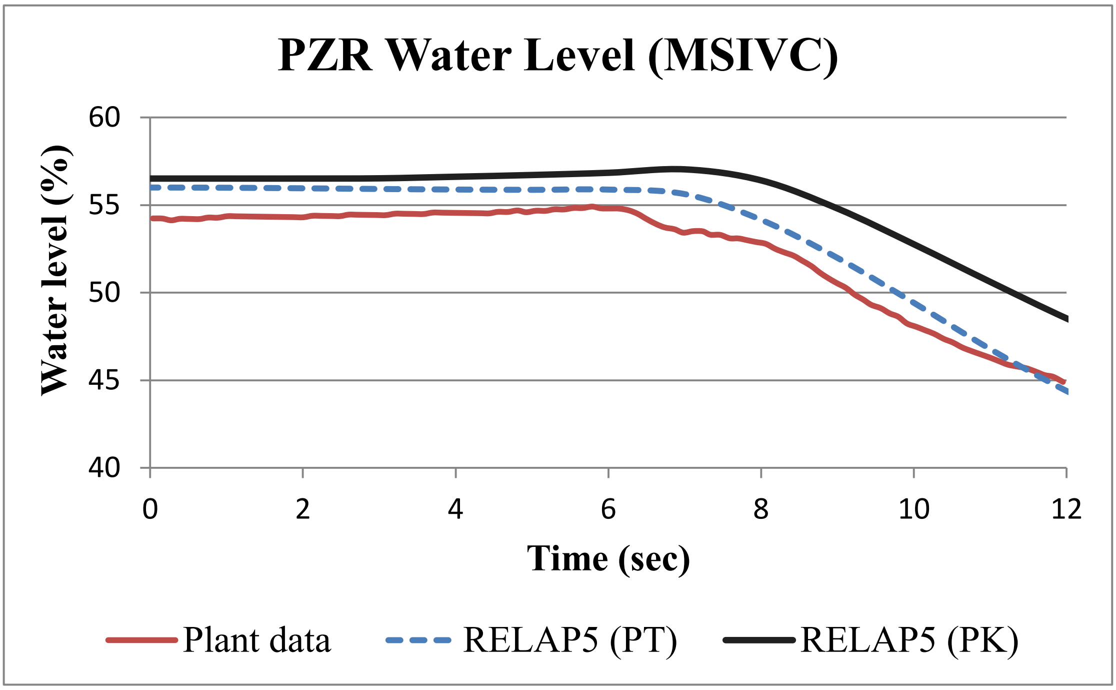


Figure 22 Water Level Variation of the Pressurizer during MSIVC Transient

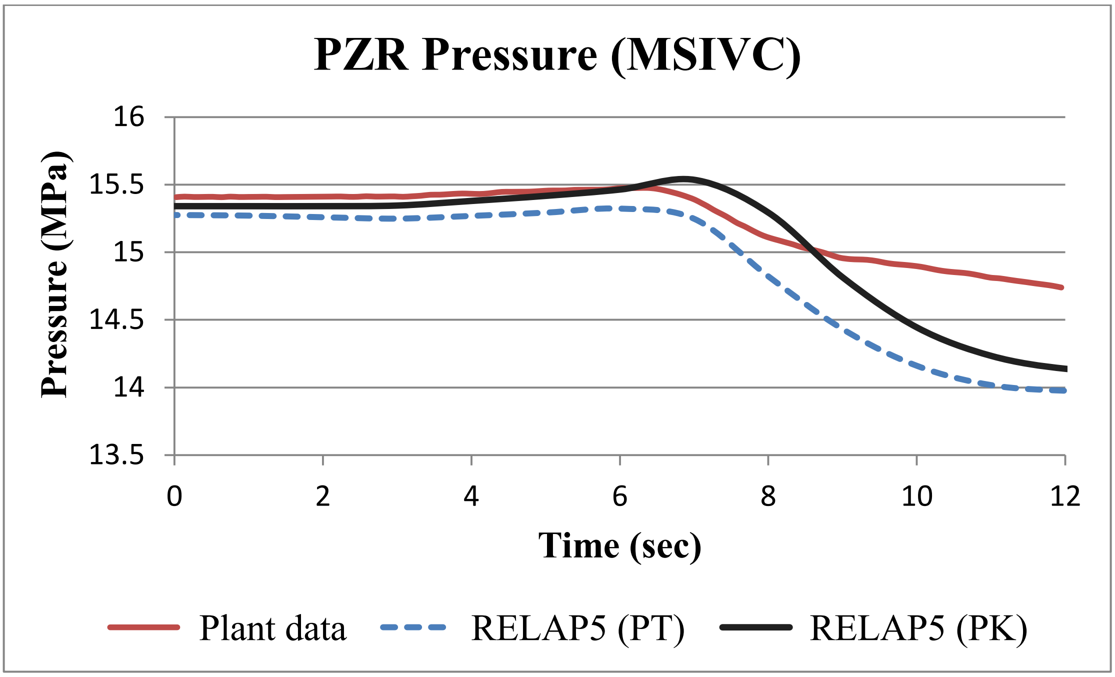


Figure 23 Pressure Variation of the Pressurizer during MSIVC Transient

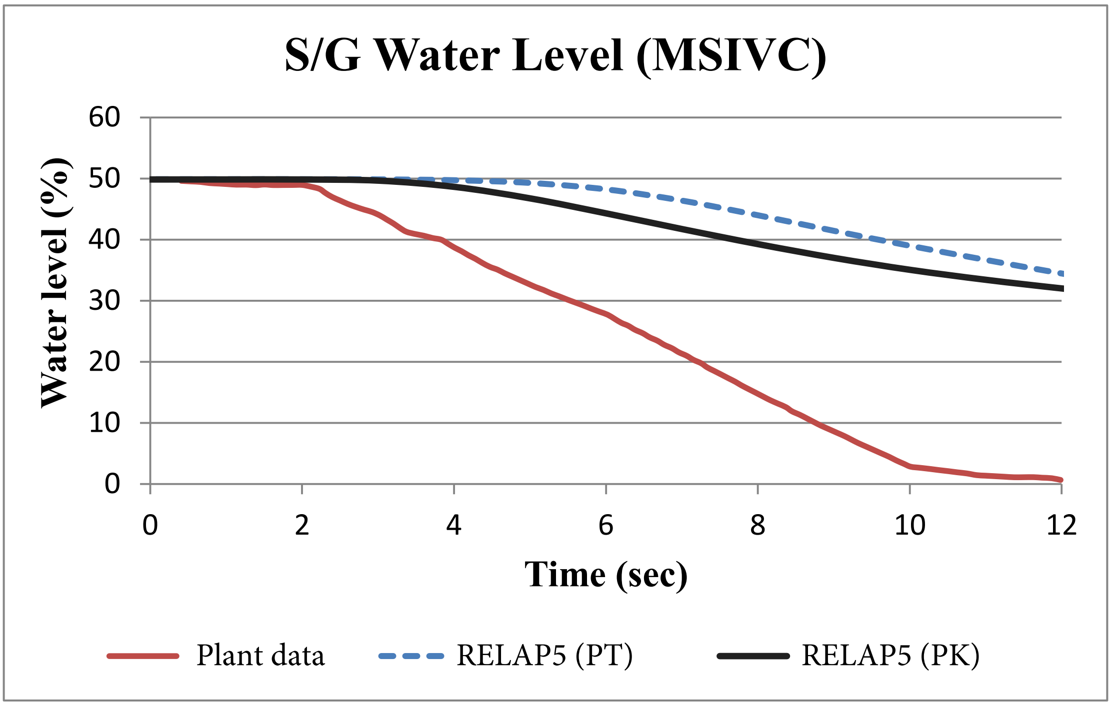


Figure 24 Water Level Variation of Steam Generator during MSIVC Transient

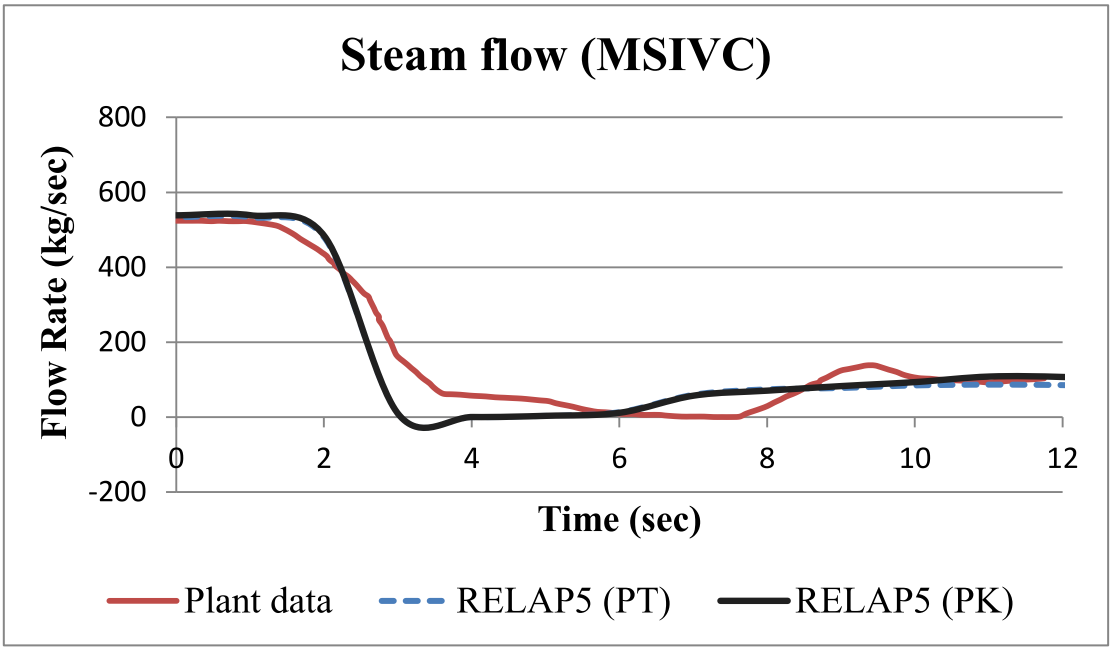


Figure 25 Steam Flow Rate Variation during MSIVC Transient

4.2 Feedwater Pumps Trip (FWPT)

Event Sequence

The Feedwater Pumps Trip (FWPT) event happened in 08/25/1987. The fuse of feedwater isolation valves power control failed and as a result, the feedwater isolation valves closed rapidly. The low water level of steam generator reached and initiated reactor scram. The turbine tripped along with the reactor scram.

The Maanshan NPP RELAP5 model was modified with this historical transient event sequence to assess whether the control system of this model could bring the NSSS back to safe operation condition as the feedwater tripped in full power operating. The model was initially developed in full power steady state. At 10 seconds, the feedwater trip signal was initiated and the feedwater flow rate decreased, which caused the water level decrease in the steam generator. As the water level was lower than 17% (normal NRWL is typically 50%), the reactor protection system initiated the reactor scram and the power decreased. The turbines were also tripped in few seconds after the reactor scram. The whole analysis period was 80 seconds.

The initial conditions of FWPT were described in Table 3. There are three data sets including the plant data, RELAP5 (PT) which reactor power was defined with power table and RELAP5 (PK) which reactor power were calculated with Point Kinetics. In the beginning of the analysis, the power was 2775 MWt (100%) for all these three cases. From this table, it is obvious that both calculations of RELAP5/MOD3.3 model were consistent with the plant data.

Table 4 is the event sequence comparison of analytical data results and the plant data. In the beginning of the test, there was a ten-second steady state. At 10 second, the feedwater tripped which caused the feedwater flow rate decrease. The plant data shows that the feedwater flow rate reached to 0 kg/sec in 6 seconds after reactor scram (16 seconds from the transient start). On the other hand, the analytical data shows that the feedwater flow rate reached to 0 kg/sec in 7 seconds after the reactor scram (17 seconds from the transient start). Details of the transient variation and mechanism are described below.

Table 3 Initial Conditions of FWPT Transient

Parameters	Plant data	RELAP5 (PT)	RELAP5 (PK)
Power (MW)	2775	2775	2775
Core Temperature (K)	580.9	582.4	582.4
Feedwater Flow Rate (kg/sec)	519.36	538.14	562.07
Steam Flow Rate (kg/sec)	517.09	538.14	562.07
PZR Pressure (MPa)	15.45	15.33	15.33
PZR Water Level (%)	54.4	56.45	56.52
S/G Pressure (MPa)	6.87	6.82	6.82
S/G Water Level (%)	50	49.96	49.96

Table 4 The Sequence of FWPT Transient

Event (sec)	Plant data	RELAP5 (PT)	RELAP5 (PK)
FWIV Close	10	10	10
S/G Lo-Lo Level Trip	32.1	35.2	35.1
Reactor Scram	32.2	32.2	35.1
Turbine Trip	32.3	33.0	35.1
Steam dump Valves Open	32.5	33.01	33.01
Steam Flow Rate Near 0	80.0	80.0	100.0

Analysis Results

This section describes the analysis data results of FWPT transient event. Plant data, RELAP5 (PT) and RELAP5 (PK) analysis results were plotted together to ensure the consistency. The important parameters during the transient were shown in Figure 26 to Figure 33.

Figure 26 is the feedwater flow rate variation during the transient. From curves on this figure, it shows good consistency of data results from plant data and REALP5 analysis. The feedwater flow rate of plant data reached to 0 kg/sec at 13 second while the RELAP5 reached to 0 kg/sec at 11 second. The feedwater flow rate was the boundary condition of this transient. As a result, the trip table feedwater control system must be developed to fit the plant data as precise as possible. However, once the trip table fixed, the feedwater flow rate in other cases would be not consistent. Hence, to have a general model for all transient analyses, the difference of trip curve between plant data and RELAP5 model was kept.

Due to the closure of feedwater isolation valves at 10 second, the feedwater flow rate decreased as shown in Figure 26. Further, the reactor was still in full power as the feedwater tripped. The water level of steam generator decreased as shown in Figure 27. Once the water level reached to 17%, the Lo-Lo Level Trip signal was initiated and hence the reactor scrammed; the turbine was tripped as a result. For the plant data, the Lo-Lo Level trip was at 32.5 second. However, the trip signal from analysis result of RELAP5 calculation was at 35 second because the total water level of the model decreased slower than that of the plant data.

Figure 28 is the power variation of FWPT transient. With 0.5 second delay time, the reactor scrammed at 33 second after the Lo-Lo Level trip signal initiated at 32.5 second. For the RELAP5 (PT) curve, the core power was referred to the power history of plant data so the power variation (scrammed at 32.5 second) was same as plant data in spite that the Lo-Lo Level trip signal was initiated at 35.1 second. However, for the RELAP5 (PK) model, the core power was calculated with point kinetic model. The reactor scrammed due to the trip signal at 35.1 second. Nevertheless, the negative reactivity came from the scram table was stronger enough to dominate the core power. The power variation of RELAP5 (PK) model was soon consistent to the plant data after scram. The final power of RELAP5 (PK) model was about 2% of the design power, which was different from the plant data. This difference might come from the decay heat model and the scram table. However, with the same reason of feedwater trip table, to have a general model for all transient analyses, the difference of scram curve and decay heat model between plant data and RELAP5 (PK) model was kept.

Figure 29 was the core temperature variation of FWPT transient. This parameter was the maximum of three loops in Maanshan NPP. From this figure, it is obvious that the core temperature of both the plant data and RELAP5 analysis results increased gradually after 25 second. It is because the water level of the steam generator decreased (Figure 27) and as a result the heat could not be carried out immediately. After 33 second (35.1 second for the point kinetic model), the reactor scrammed and the core power dropped rapidly. In addition, the steam dump valves opened to exhaust the steam. Hence, the core temperature decreased.

Figure 30 is the variation of pressure inside the pressurizer during FWPT transient. At the previous 19-second interval, the pressure from both plant data and RELAP5 results was steady. At 19 second, the pressure from RELAP5 results increased while the pressure from plant data increased at 21 second. Because the water level in the steam generator decreased, the heat was harder to be carried out through the evaporation. Further, the closure of TSVs (due to turbine trip) increased the pressure of the steam generator. From both these two reasons, the water temperature in primary loops would increase (as shown in Figure 29). The pressure of the pressurizer increased as a result. Because the increment of pressure, the water level of pressurizer decreased (as shown in Figure 31).

Figure 32 is the variation of steam line pressure. In the beginning, the pressure of all cases was about 6.82 MPa. After the reactor scram and the TSVs closure, the steam pressure increased as a result. For the plant data, the peak value of steam line pressure was 7.86 MPa at 39 second while for the RELAP5 results, the peak value of steam line pressure was 7.9 MPa also at 39 second. After the steam dump system initiated, the steam line pressure decreased. Because the pressure variations of these three cases were different, the peak value would not be consistent. Further, form the past research our group has finished, the prediction of pressure in RELAP5 model would mostly be higher.

Figure 33 is the steam flow rate variation during the FWPT transient. At the beginning, flow rate variations of all of these three cases were similar. The flow rate increased slightly before the turbine trip. After the closure of TSVs, the steam flow rate dropped. Later, with the steam dump valves open, the steam flow rate increased rapidly and even caused a peak value. The peak value for the plant data was 312 kg/sec and the peak values for the RELAP5 analyses were about 242 kg/sec. Then, due to the reactor scram, the core temperature and hence the steam flow rate decreased gradually.

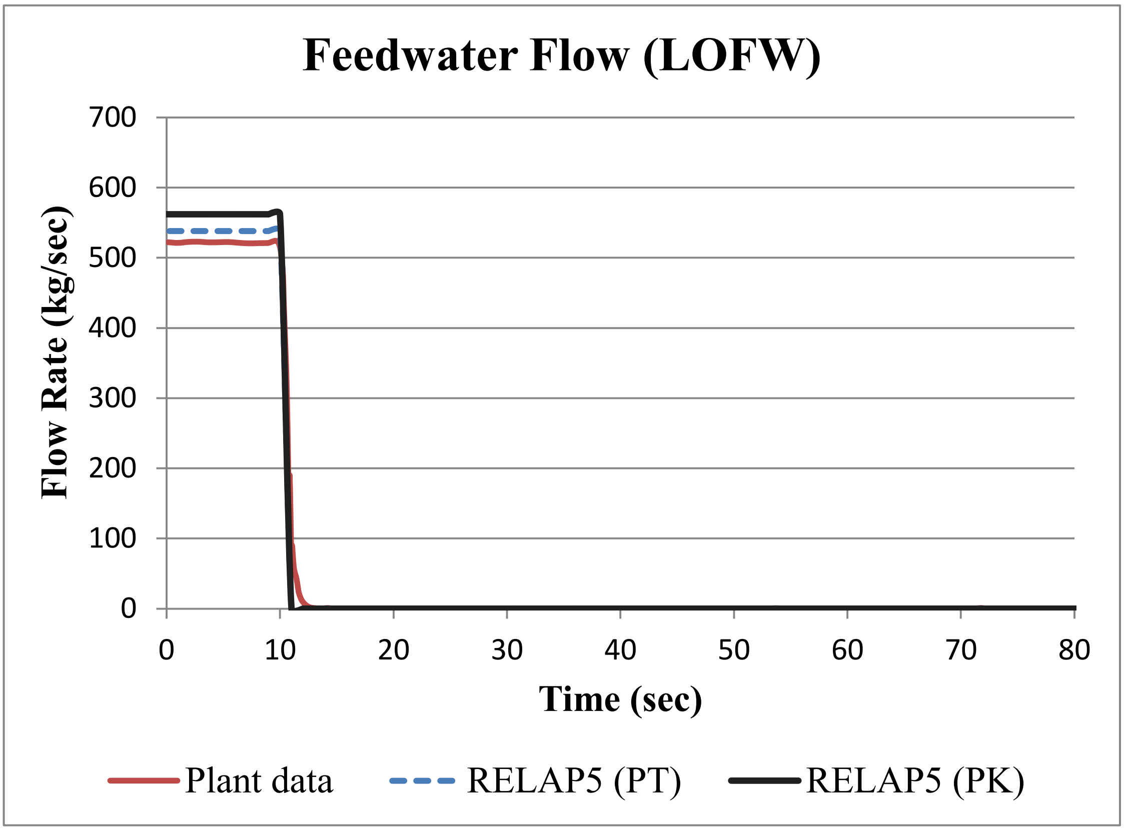


Figure 26 Feedwater Flow Rate Variation during FWPT Transient

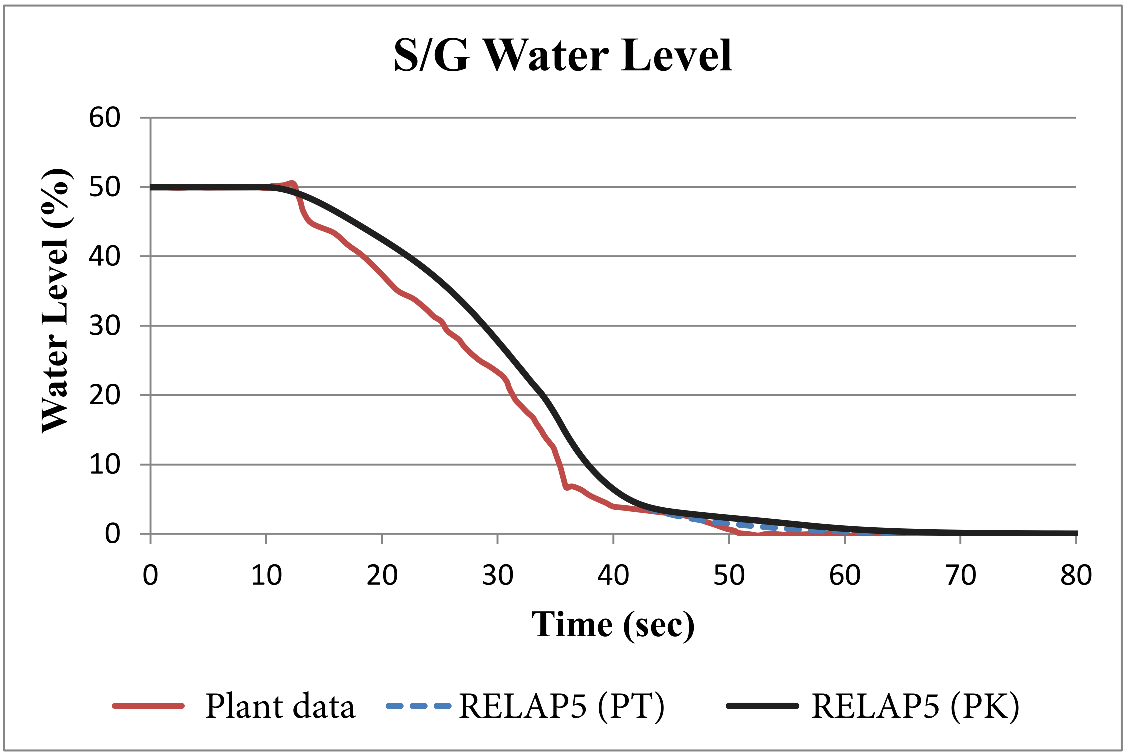


Figure 27 Water Level of the Steam Generator during FWPT Transient

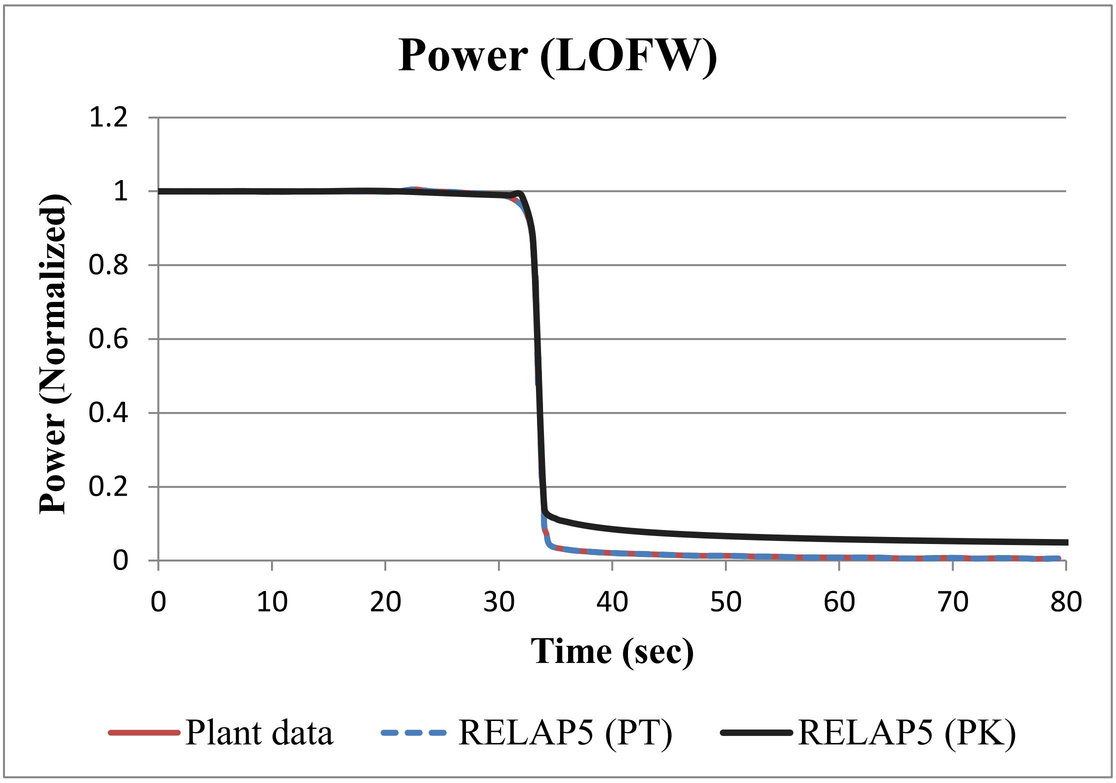


Figure 28 Power History of FWPT Transient

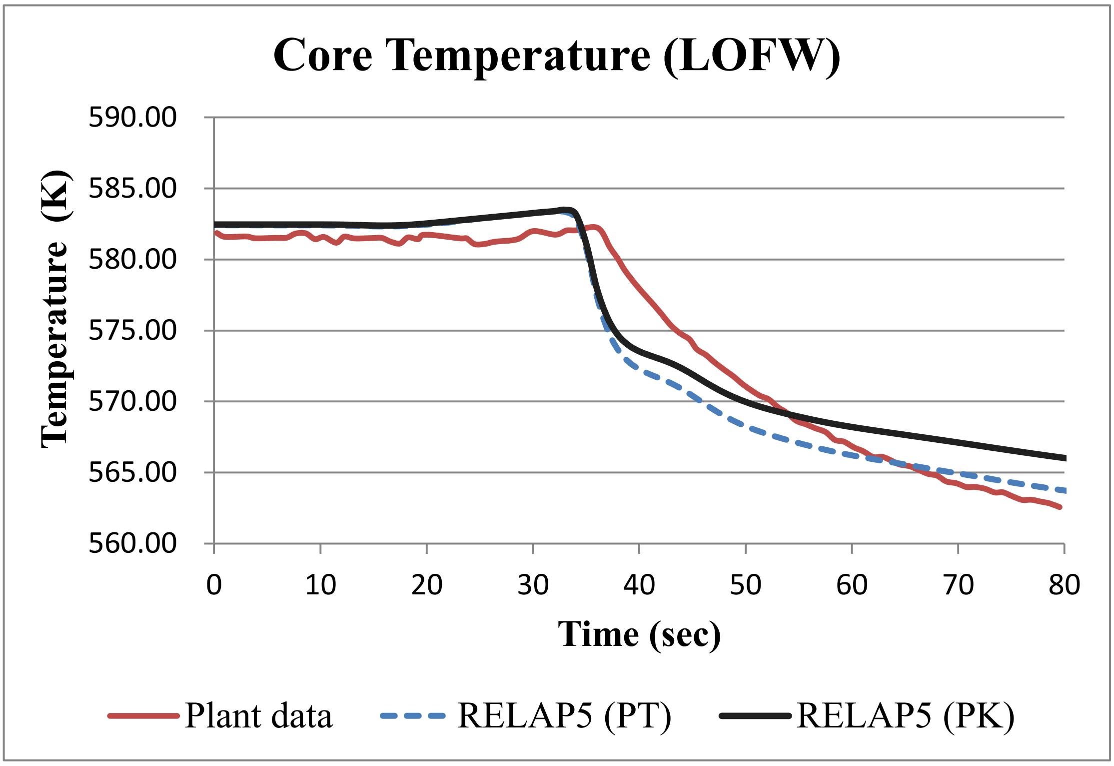


Figure 29 Core Temperature Variation of FWPT Transient

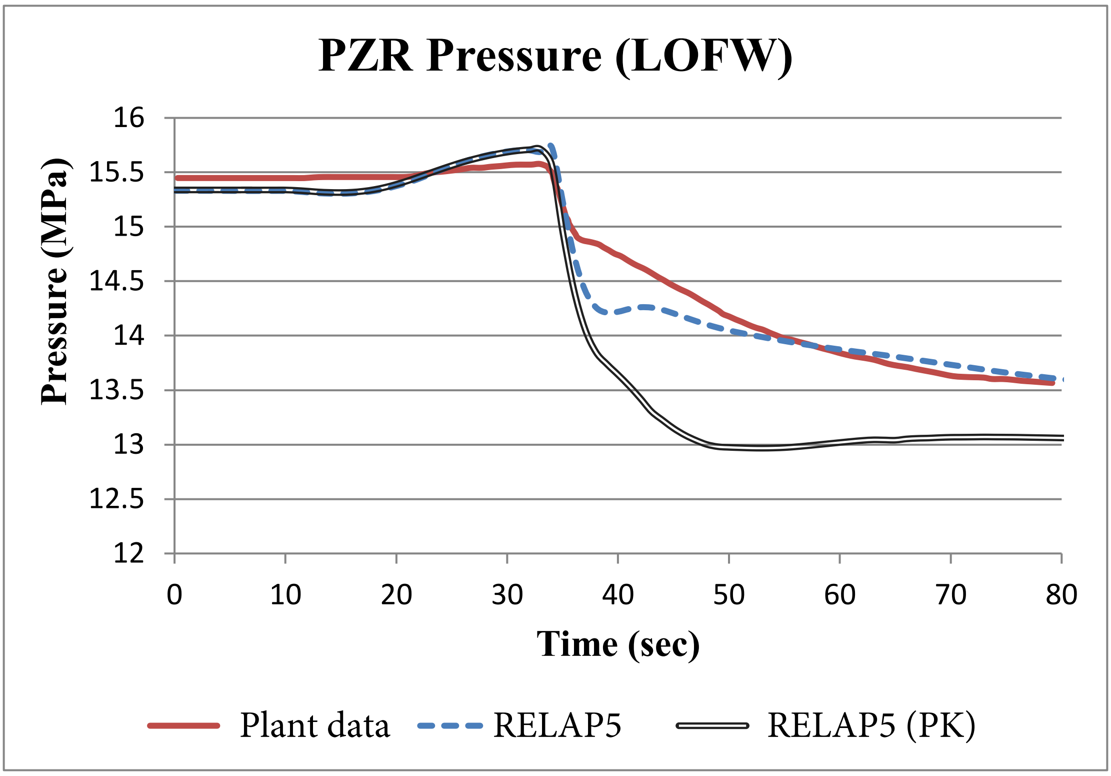


Figure 30 Pressure Variation of Pressurizer during FWPT Transient

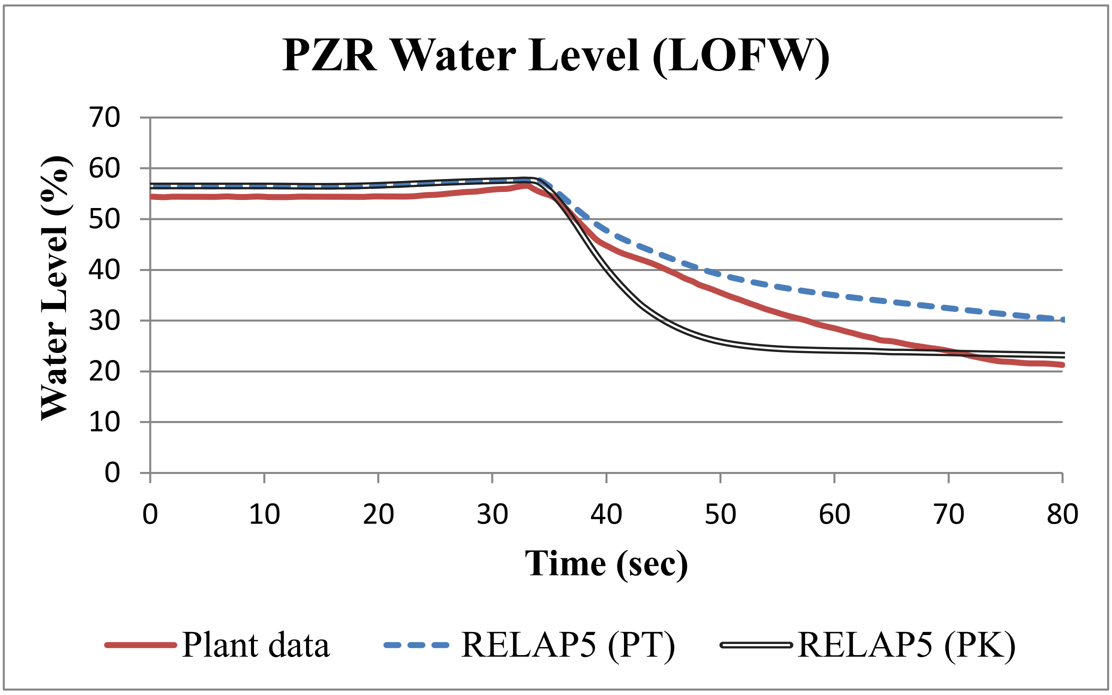


Figure 31 Water Level of Pressurizer during FWPT Transient

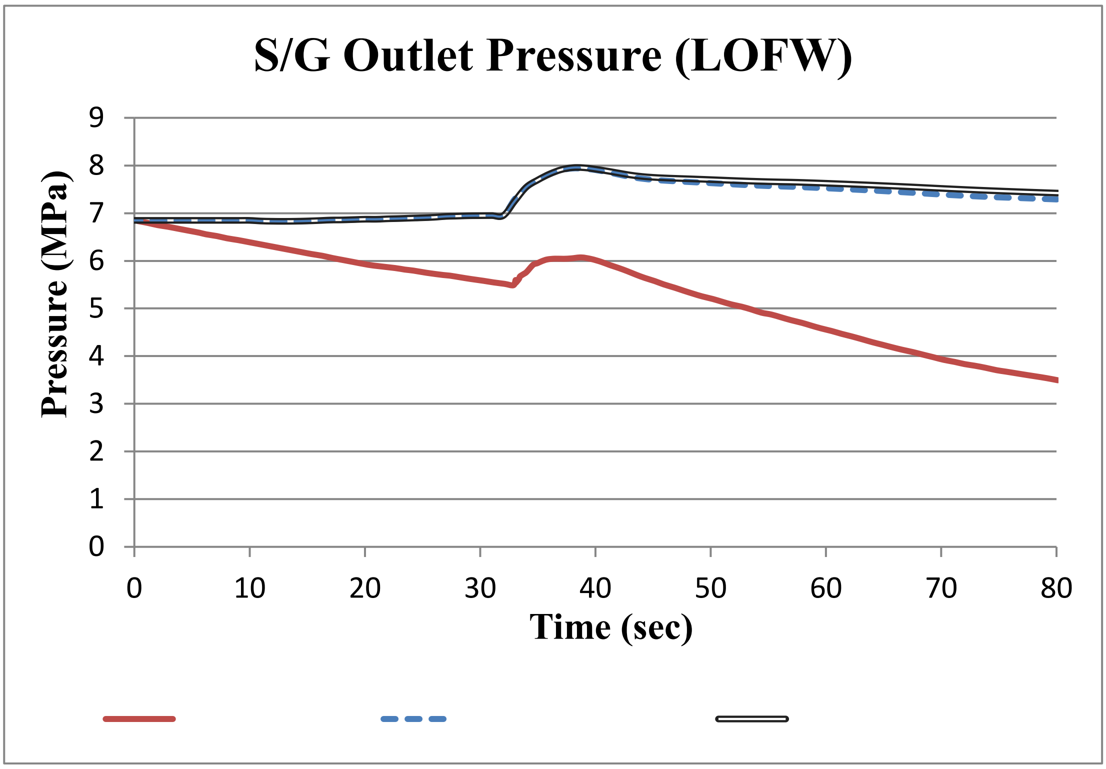


Figure 32 Steam Pressure of Steam Generator during FWPT Transient

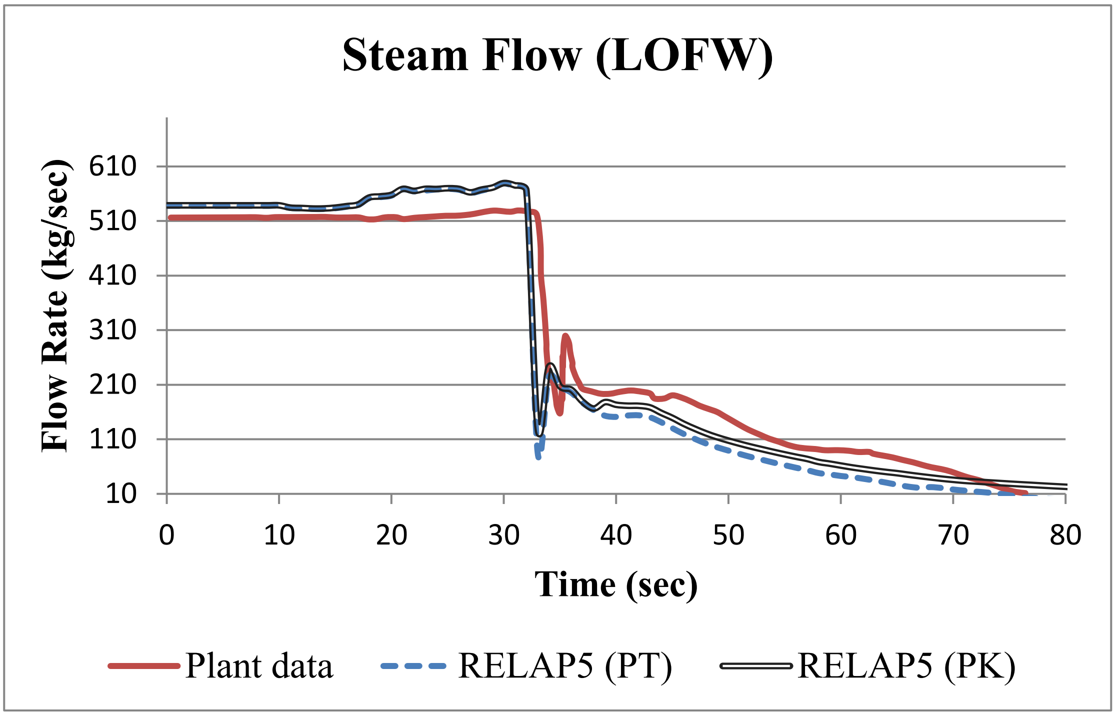


Figure 33 Steam Flow Rate Variation of FWPT Transient

4.3 Turbine Trip (PAT50)

Event Sequence

The full power turbine trip test, which was performed in 04/26/1985 and numbered 2-PAT-50, was one of the NPP startup tests. The purpose of this test is to ensure whether the NPP control system could bring the NSSS back to safe conditions during the turbine trip. First, the NPP was operated in full power, all the control system was in automatic mode. At the given time, the operator tripped the turbines manually. After 0.1 second, the turbine stop valves (TSVs) started to close and at the same moment, the reactor protection system initiated the reactor scram. The power decreased rapidly. Further, due to the closure of TSVs, the steam in the main steam line could only be exhausted through the steam dump system, which opened in 0.1 second after the turbine trip. The steam dump system stabilized the NSSS till the test finished in 80 seconds.

The initial conditions of PAT50 were described in Table 5. There are three data sets including the plant data, RELAP5 (PT) which reactor power was defined with power table and RELAP5 (PK) which reactor power were calculated with Point Kinetics. In the beginning of the analysis, the power was 2752.5 MWt (99.2%) for all these three cases. From this table, it is obvious that both calculations of RELAP5/MOD3.3 model were consistent with the plant data.

Table 6 is the event sequence comparison of analytical data results and the plant data. In the beginning of the test, there was a ten-second steady state. At 10 second, the turbine tripped manually which caused the reactor scram after 0.1 second. The reactor power decreased to 2% (after 2 seconds from the reactor scram) due to the control rod insert. 0.4 second after the turbine trip, the steam dump valves opened to exhaust the 31.9% rated steam flow rate in secondary loops into the condenser. Water temperature in the primary loops was lower that $T_{no\ load}$ (564K) because the reactor power was only about 5% of full power and the steam was exhausted in the secondary loops. Due to the temperature decreasing, the volume of water shrank and the water level of Pressurizer decreased as a result. After 50 seconds, the water level of Pressurizer reached to minimum 20.56%. The pressure of pressurizer also decreased to minimum 13.14 MPa after 30 seconds from the reactor scram. Until the end of steam exhausting, the core temperature increased slowly due to the decay heat. Likewise, the pressure in the pressurizer increased from the minimum because of the increasing core temperature and the initiating of the pressurizer heater.

Table 5 Initial Conditions of PAT50

Parameters	Plant data	RELAP5 (PT)	RELAP5 (PK)
Power (MW)	2752	2752	2752
Core Temperature (K)	582.03	581.99	581.99
Feedwater Flow Rate (kg/sec)	541.13	540.50	534.02
Steam Flow Rate (kg/sec)	544.31	540.50	534.02
PZR Pressure (MPa)	15.26	15.3	15.3
PZR Water Level (%)	56.5	56.07	56.05
S/G Pressure (MPa)	6.82	6.85	6.84
S/G Water Level (%)	50	49.9	49.9

Table 6 The Sequence of PAT50

Event (sec)	Plant data	RELAP5 (PT)	RELAP5 (PK)
Turbine trip	10.0	10.0	10.0
Turbine stop valve fully close	10.1	10.1	10.1
Reactor scram	10.1	10.1	10.1
Bypass valve fully open	12.0	12.0	11.9
S/G low level	16.2	21.2	22.5
Feedwater pump trip	28.5	32.9	31.7
Turbine Bypass valve fully close	44.4	44.5	44.5

Analysis Results

This section describes the analysis data results of PAT50 startup test. Plant data, RELAP5 (PT) and RELAP5 (PK) analysis results were plotted together to ensure the consistency. The important parameters during the transient were shown in Figure 34 to Figure 41.

Figure 34 is the power variations of three cases during the PAT50 startup test transient. In both the plant data and analyses, the turbine was tripped manually at 10 second. The reactor scrammed in 0.1 to 0.2 second so that the reactor power dropped rapidly. Data result of RELAP5 (PT) was consistent to the plant data because the power was input as boundary conditions in RELAP5 (PT) model by the power history of plant data. However, for the RELAP5 (PK) analysis result, the power dropped which was also consistent to the plant data until the 12 second. The difference of RELAP5 (PK) and plant data after 12 second might come from the decay heat model choice. For the point kinetic model, the default ANS-73 decay heat was chosen. Nevertheless, this model may be not suitable for the Maanshan NPP. Further, the PAT50 was a startup test which means the fuel cycle and arrangement were not same as the normal operating conditions.

The core temperature was shown in Figure 35. The core temperature decreased after 10.2 second because the reactor scrammed and the steam dump system was initiated. From this figure, it can be noticed that the decreasing trends of analyses and plant data were similar but the plant data decreased later. The possible reason for this difference might be that the temperature variation was more sensitive in the computational model while the variation might not be recorded immediately for the plant data. After 30 to 40 second, core temperature of all of cases reached near to the $T_{no\ load}$ (564K). However, after 40 second, the core temperature of both analysis model was higher than that of plant data because more decay heat was generated in analysis model as described in previous paragraph.

Figure 36 and 37 were comparisons of pressure and water level of the pressurizer. Both the pressure and water level dropped in one second after the reactor scram, which were consistent to the plant data. Before 20 seconds, the analysis results predicted properly but later the analysis data results were higher because of the decay heat model difference as mentioned above.

Figure 38 was the steam flow rate variation during the transient. To maintain the steady state, flow rate of the analysis models was lower than that of plant data. As the turbine tripped and TSVs closed, the steam flow was blocked and decreased in one second. The analysis models were more sensitive for the TSVs closure because of few computational nodes. After the closure, the steam valves opened to exhaust steam so that the flow rate of the plant data increased suddenly which caused two peak values. As mentioned above, the computational was few and as a result both the computational steam flow rate curves vibrated as the steam valves opened. After 20 seconds, flow rates and decreasing trend of all cases were similar. After 40 seconds, the steam flow rate reached near to 0 because the lower heat generation.

Steam line pressure variations were shown in Figure 39. The initial values of all cases were about 6.9 MPa. After the turbine trip, the TSVs closure so that the steam line pressure increased rapidly. For the plant data, there was a peak value during the pressure increasing interval because of the initiation of steam dump system. However, for the computational models, the peak value was not as clear as that of plant data. The peak pressure of plant data was 7.59 MPa at 15 second and that of analysis results were about 7.54 MPa at 13.7 second, which was similar. Further, during the time interval 10 to 16 seconds, all of the curves were almost overlapping, which means that the simulation of the pressure increasing is quite proper. Figure 40 shows the water level of steam generator. Because of the low power and high steam line pressure, the water level decreased as the reactor scrammed. Compared to the steam line pressure, the water level curves of all cases were almost overlapping before 16 second. However, due to the feedwater difference as shown in Figure 41, the water level of computational models decreased much slower than that of plant data. The water level of plant data was lower than the lower limit of NRWL at 35 second which could not be recorded. Because of the higher feedwater flow rate, the water level of computational results was lower than the lower limit at 70 second.

Figure 41 shows a huge difference of feedwater flow rates between plant data and computational data. This difference might come from the operator intervention which was not documented. The feedwater of PAT50 startup test might be limited as a constant during 10 to 30 second of the transient while the feedwater of analysis model was automatically controlled by the feedwater control system. The details of this difference should be confirmed by collecting more startup test documents and comparing with data results from other thermal hydraulic analysis codes.

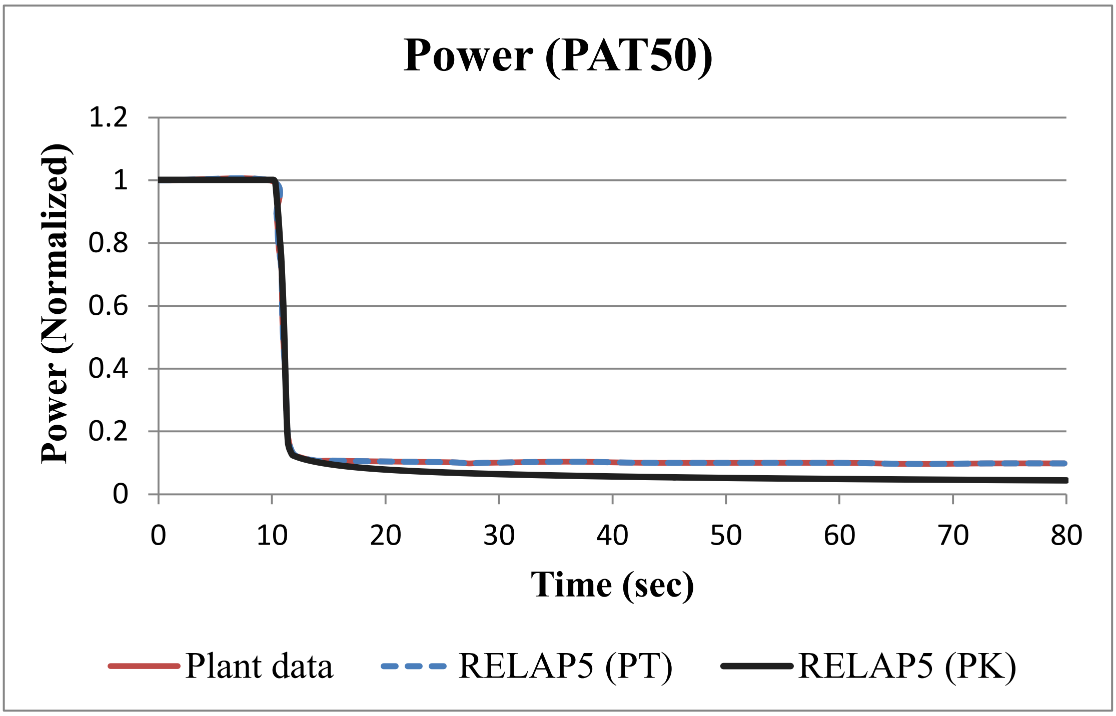


Figure 34 Power History of PAT50 Transient

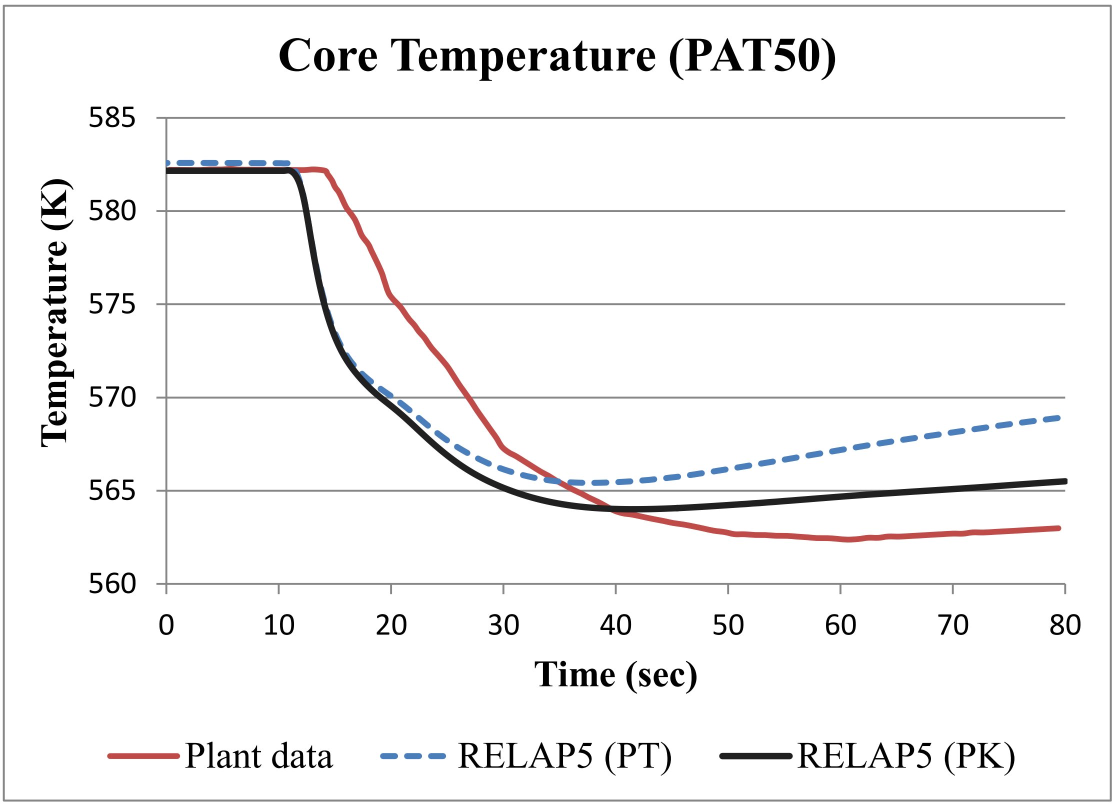


Figure 35 Core Temperature Variation of PAT50 Transient

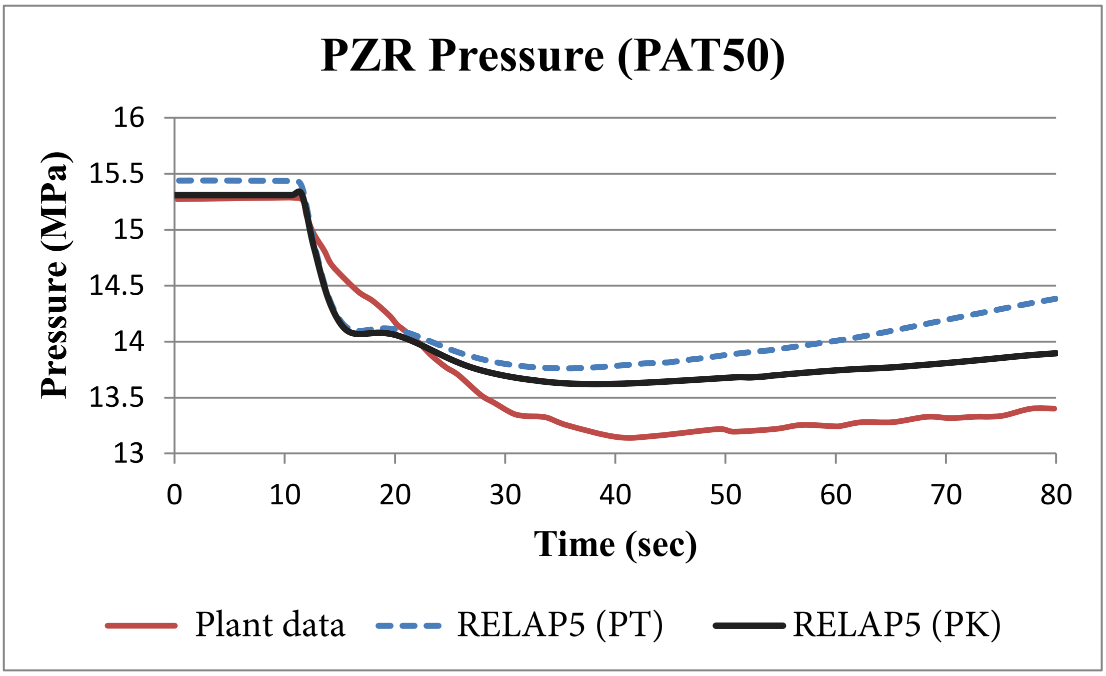


Figure 36 Pressure Variation of Pressurizer during PAT50 Transient

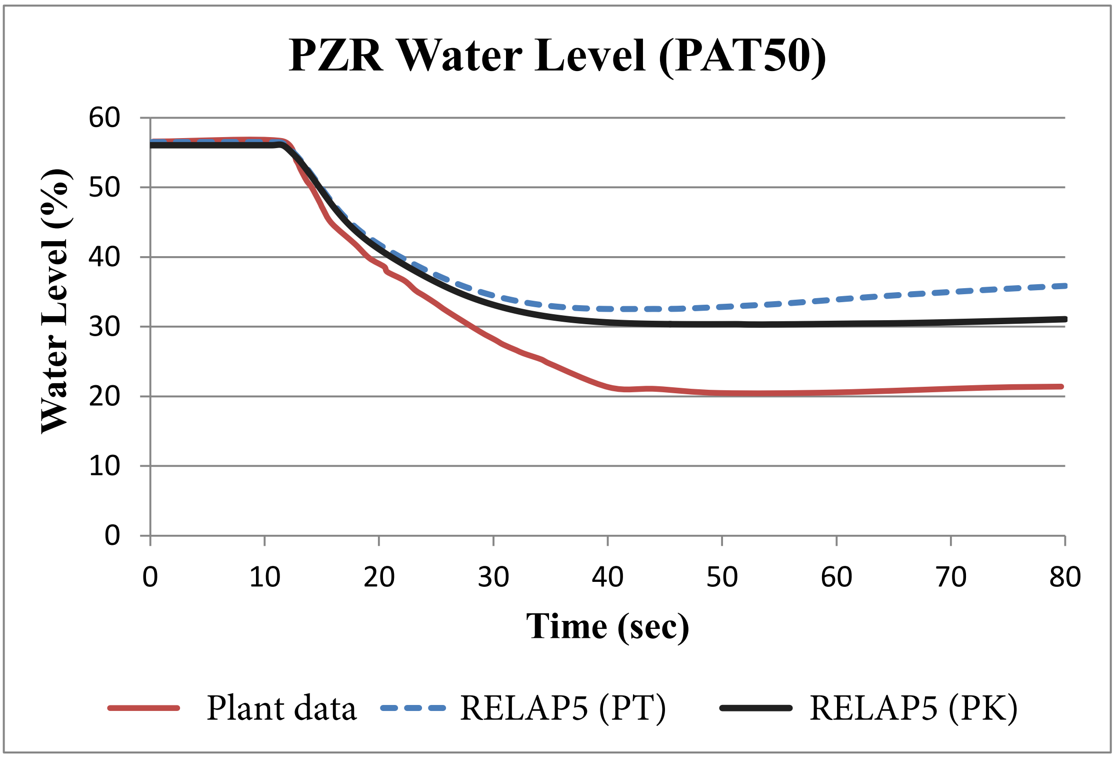


Figure 37 Water Level Variation of Pressurizer during PAT50 Transient

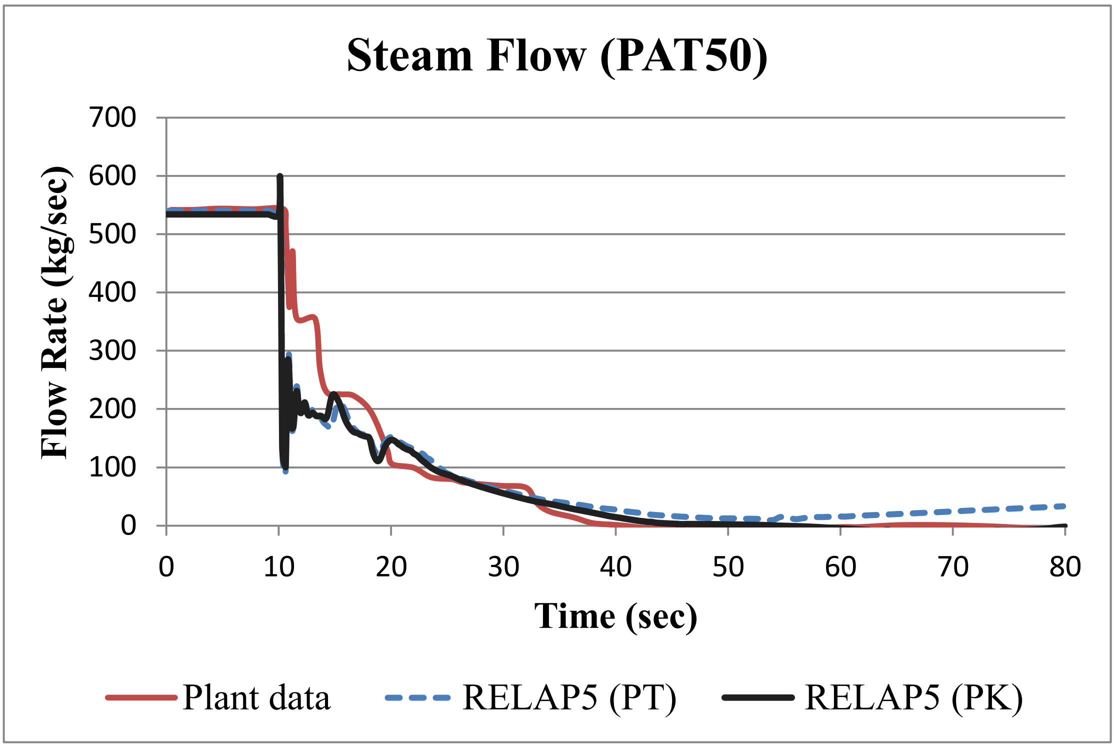


Figure 38 Steam Flow Rate Variation of PAT50 Transient

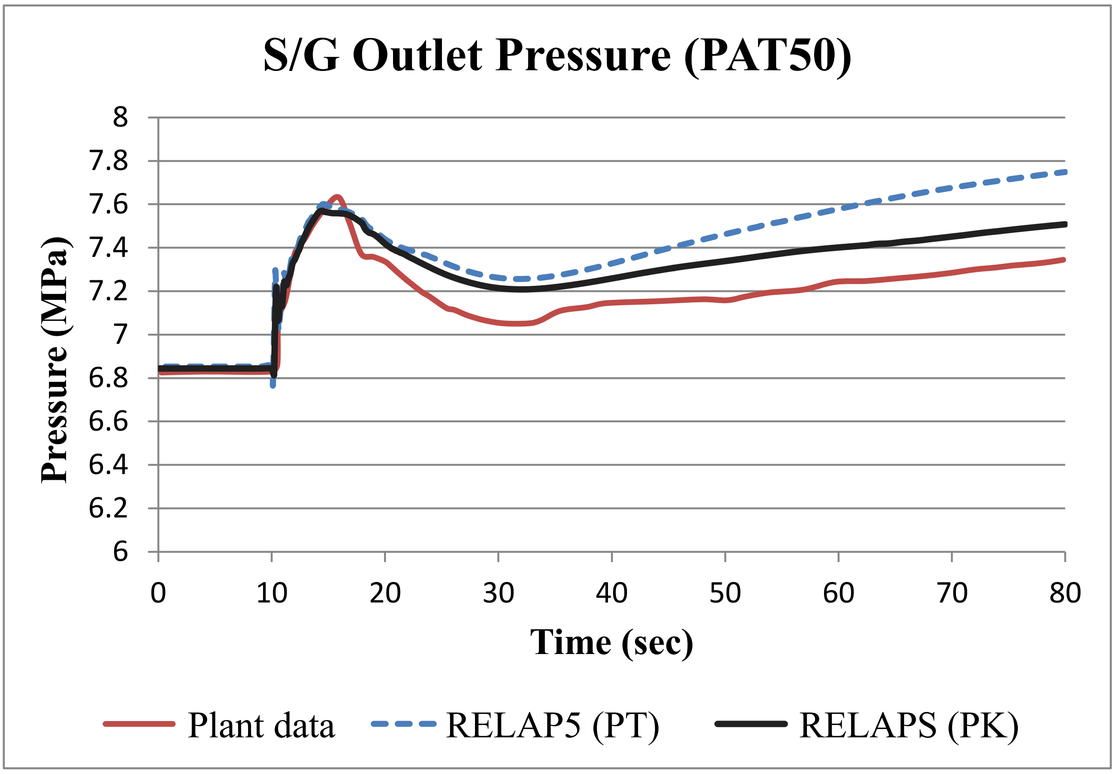


Figure 39 Steam Pressure of Steam Generator during PAT50 Transient

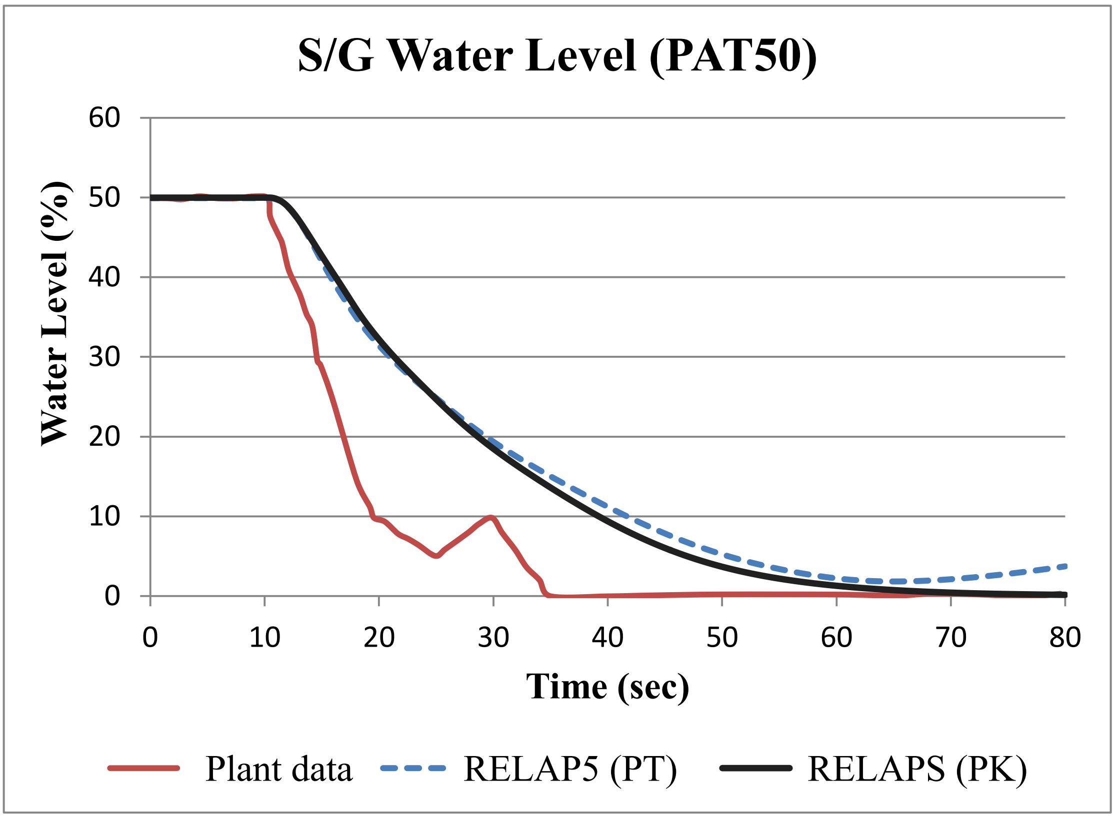


Figure 40 Water Level of Steam Generator during PAT50 Transient

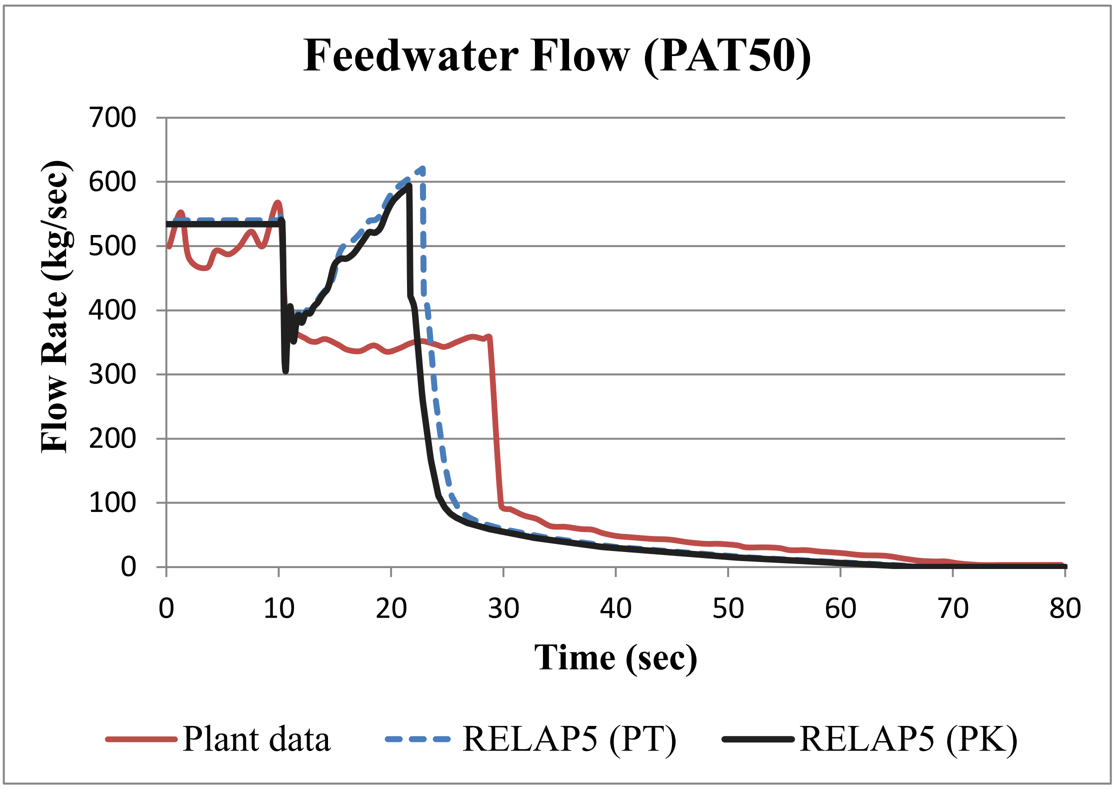


Figure 41 Steam Flow Rate Variation of PAT50 Transient

5 CONCLUSIONS

From the analysis data results above, the RELAP5/MOD 3.3 model of Maanshan NPP can accurately predict the results for the startup test transients. It can correctly simulate the important parameters variation and trend during the transients. For the thermal hydraulic components, the feedwater flow rate, steam flow rate, water level of steam generators and the water level pressurizer were mostly predicted by this model. For the control systems such as turbine trip, reactor scram, steam dump valves open were consistent to the operating of NPP. Further, this analysis model will be applied for other hypothetical accidents and complicated transient events so that the plant personnel might have some expectations during the accident.

The SNAP interface was successfully applied in this research. With the SNAP interface, the Maanshan NPP model was developed quickly and efficiently. In the SNAP interface, the visualized thermal hydraulic components allow users adjust the geometric parameters easily. Further, the component connections were also more understandable. Additionally, the control signals, trips and functions were developed with the component nodes, which mean that the users need not draw the nodding diagram manually. Moreover, the RELAP5 animation model of Maanshan NPP was successfully developed. In the future, the data results can be animated immediately which allows researchers observe different parameters and their interactions at the same time.

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10. SUPPLEMENTARY NOTES

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11. ABSTRACT (200 words or less)

RELAP5 is a very important analysis tool for Taiwan Power Company and is still used for the transient analysis of the Taiwan NPPs. The version of RELAP5 for Taiwan Power Company is RELAP5/MOD3.3 and the input deck of RELAP5 is established by the ASCII files. Symbolic Nuclear Analysis Package (SNAP) is an interface of NPP analysis codes which developed by U.S. NRC and Applied Programming Technology, Inc. Different from the traditional input deck in ASCII files, the graphical control blocks and thermal hydraulic connections make researches comprehend the whole power plant and control system more easily. Additionally, for the last few years, the TRACE/SNAP model of Maanshan NPP was developed and several kinds of transient events were performed. Based on the past research experience and SNAP advantages, the RELAP5/MOD3.3 model of Maanshan NPP was developed with SNAP interface in this research. Maanshan NPP is located on the southern coast of Taiwan. Its nuclear steam supply system is a type of PWR designed and built by Westinghouse for Taiwan Power Company. A startup test data and two transient results were used to compare with the results of RELAP5/MOD3.3 model for the new-developed analysis model assessment. The predictions of RELAP5/MOD3.3 were consistent to the startup test and historical transient data results. It indicates that there is a respectable accuracy for the Maanshan NPP RELAP5/MOD3.3 model.

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