**NUREG/IA-0483**

# International Agreement Report

## Loss of Flow Analysis of Maanshan Nuclear Power Plant With RELAP5/SNAP

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**Washington, DC 20555-0001**

**Manuscript Completed:** May 2017
**Date Published**: October 2018

Prepared as part of
The Agreement on Research Participation and Technical Exchange
Under the Thermal-Hydraulic Code Applications and Maintenance Program (CAMP)

**Published by**
**U.S. Nuclear Regulatory Commission**

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# ABSTRACT

The objective of this study is to assess the applicability of the RELAP5/MOD3.3 model of Maanshan NPP on Loss of Flow transient. Maanshan NPP was the first three-loop PWR in Taiwan constructed by Westinghouse. For the last few years, the TRACE model of Maanshan NPP was developed and several kinds of transients were performed. Recently, the RELAP5/MOD3.3 code is another important development priority for our group. In 2015, the RELAP5/MOD3.3 model of Maanshan NPP was developed with SNAP interface. To expand the applicability of Maanshan RELAP5/MOD3.3 model, the loss of flow transient was analyzed in this research. Hence, two analyses- Partial Loss of Flow (PLOF) and Complete Loss of Flow (CLOF) were performed in this research. From the analysis results, including power, core flow, pressurizer pressure and cladding average temperature, were similar to the results of TRACE model. It indicates that Maanshan RELAP5/MOD3.3 model can predict this transient well.

# FOREWORD

The U.S. NRC (United States Nuclear Regulatory Commission) has developed a thermal hydraulic analysis code, RELAP5, has been designed to perform best-estimate analysis of loss-of-coolant accidents (LOCAs), operational transients, and other accident scenarios in reactor systems. Models used include multidimensional two-phase flow, non-equilibrium thermo-dynamics, generalized heat transfer, reflood, level tracking, and reactor kinetics. Traditionally, the RELAP5 code analysis model was developed by ASCII file, which was not intelligible for the beginners of computer analysis. Fortunately, and graphic input interface, SNAP (Symbolic Nuclear Analysis Program) is developed by Applied Technology Incorporation Inc. and conducted by the U.S. NRC, the model development process becomes more conveniently.

To obtain the authorization of these codes, Taiwan and the United States have signed an agreement on CAMP (Code Applications and Maintenance Program) which includes the development and maintenance of RELAP5 code. NTHU (National Tsing Hua University) is the organization in Taiwan responsible for the application RELAP5 and SNAP in thermal hydraulic safety analysis. The NTHU should record user's experiences of these two programs and provide suggestions for development of RELAP5 and SNAP. To meet this responsibility, the RELAP5/MOD3.3 model of Maanshan nuclear power plant has been developed. This model was used to perform the loss of flow transient analysis.

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# EXECUTIVE SUMMARY

RELAP5/MOD3.3 Patch04 code, which was developed for light water reactor transient analysis at Idaho National Engineering Laboratory 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. 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. 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. In addition, the loss of flow transients which includes Partial Loss of Flow (PLOF) and Complete Loss of Flow (CLOF) were analyzed in this research using this RELAP5/MOD3.3 model. The analysis results such as power, core flow, pressurizer pressure and cladding average temperature were similar to the TRACE results. It is shown that Maanshan NPP RELAP5/MOD3.3 model can predict such transient well.

# ABBREVIATIONS

| | |
|---|---|
| BPV | Bypass valve |
| CAMP | Code Applications and Maintenance Program |
| FWPT | Feedwater pump(s) Trip |
| FSAR | Final Safety Analysis Report |
| 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 |
| PLOF | Partial Loss of Flow |
| CLOF | Completed Loss of Flow |
| RCS | Reactor Coolant System |
| RCP | Reactor Coolant Pump |
| 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) |
| UV | Under Voltage |
| UF | Under Frequency |
| 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 ASCI 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.

In our previous study, three startup tests including feedwater pumps trip (FWPT), turbine trip (PAT50) and main isolation valves closure (MSIVC) had been analyzed in 2015 [5]. 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. In addition, the Loss of Coolant events were performed [6, 7]. From the comparison, it shows that most of the thermal hydraulic properties are consistent to FSAR and TRACE data results [6, 7]. Hence, in this research, to expand the applicability of Maanshan RELAP5/MOD3.3 model, the loss of flow transient was analyzed. Two analyses- Partial Loss of Flow (PLOF) and Complete Loss of Flow (CLOF) were performed in this research.

# 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, as developing the RELAP5 model, it is practical to define the feedwater pumps and the turbines as the boundary conditions. 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 feedwater 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 <u>Control Systems</u>

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 Tave 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 Tave 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 (Tave) in loop 1 to loop 3 with No Load Temperature (Tno load, 564K in Maanshan NPP). Referring this comparison, the control blocks 318 can convert the difference of Tave and Tno 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. From 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.

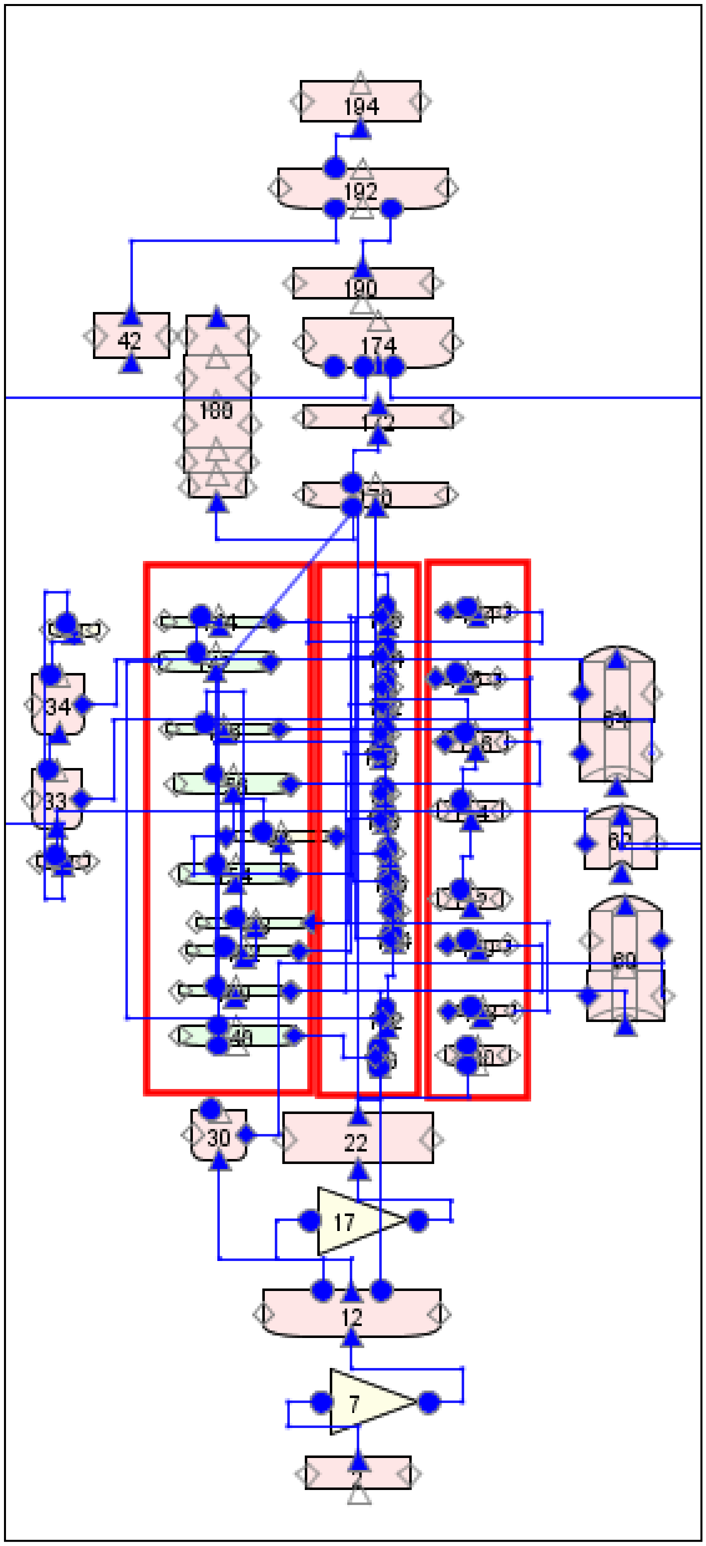


**Figure 1  Reactor Core Components of Maanshan NPP in SNAP Interface**

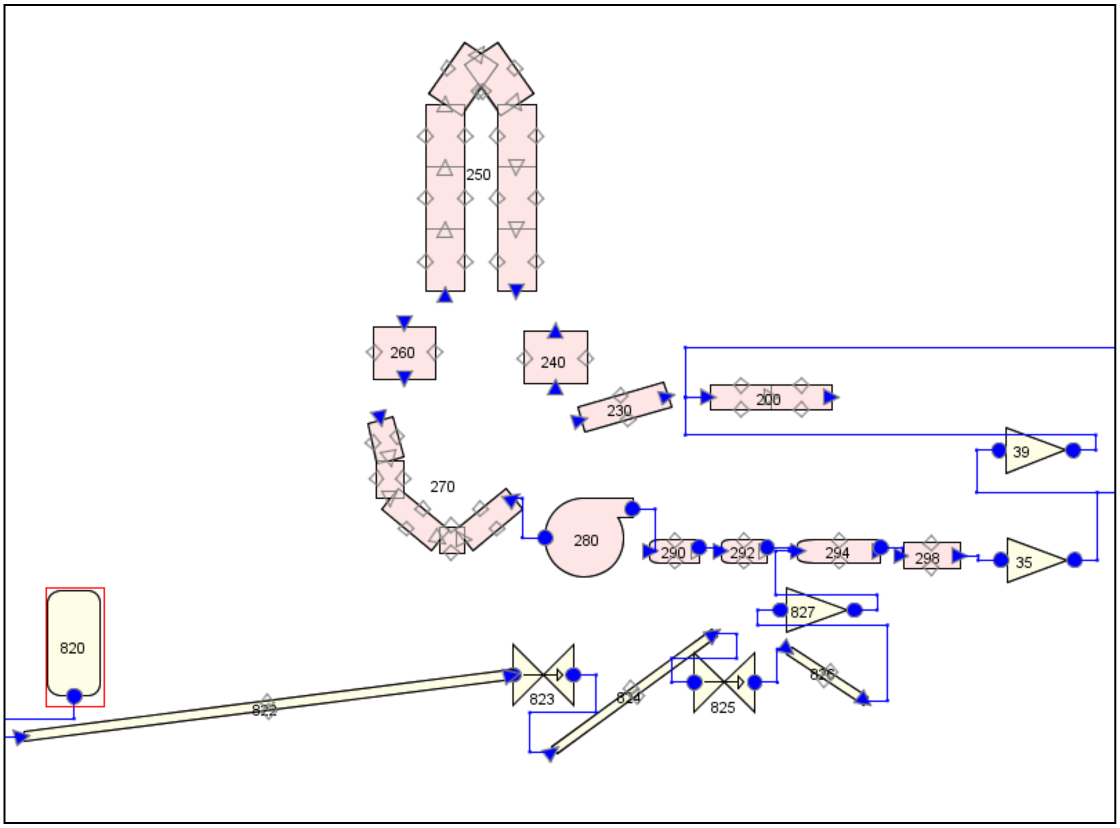


**Figure 2  Components of First Loop of Maanshan NPP in SNAP Interface**

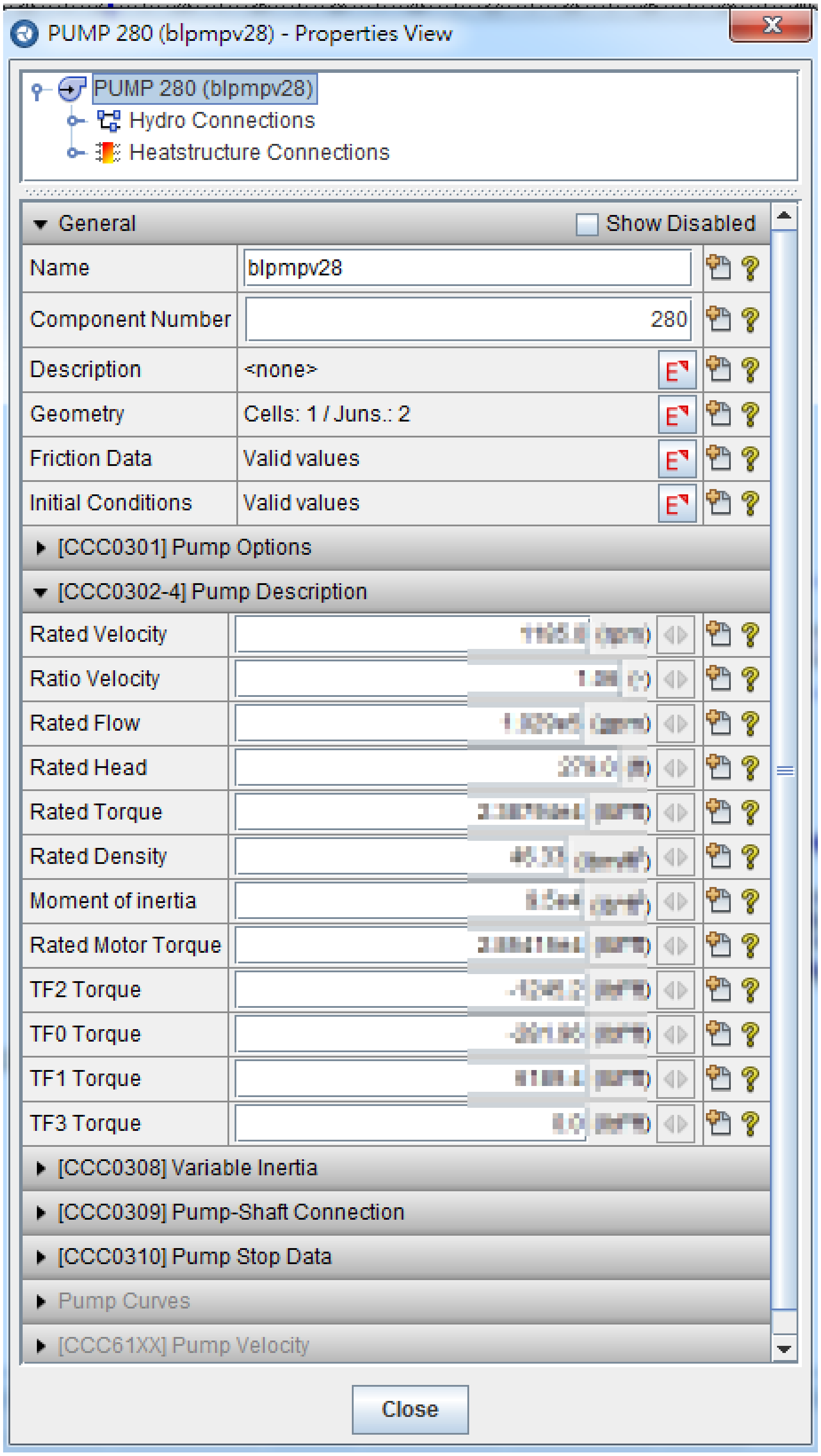


**Figure 3 Recirculation Pump Properties of Maanshan NPP in SNAP Interface**

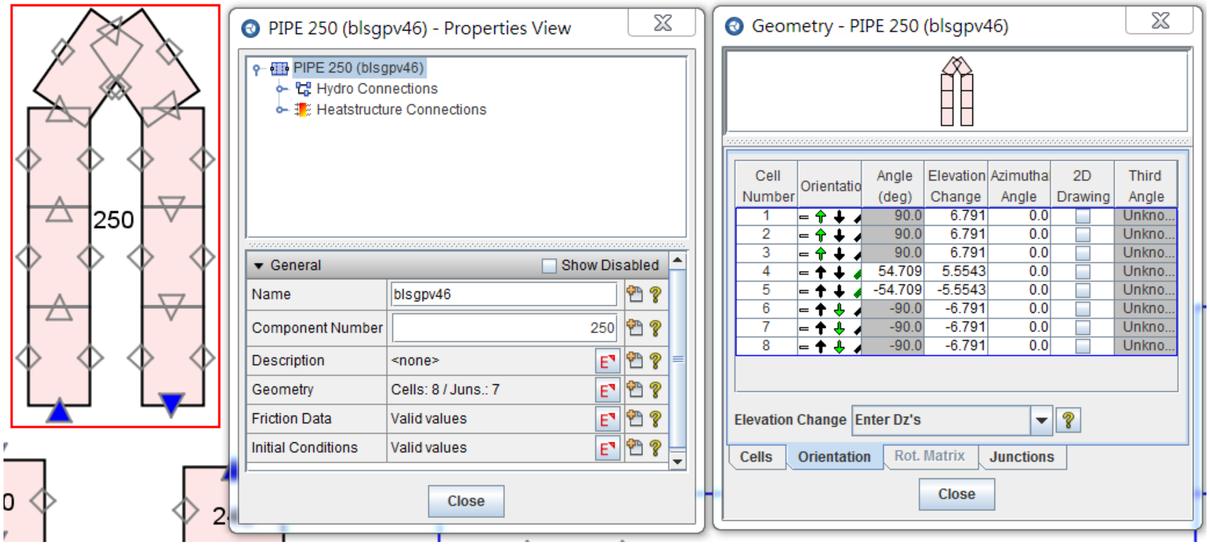


Figure 4 Heat Exchanger Component 250 of Maanshan NPP in SNAP Interface

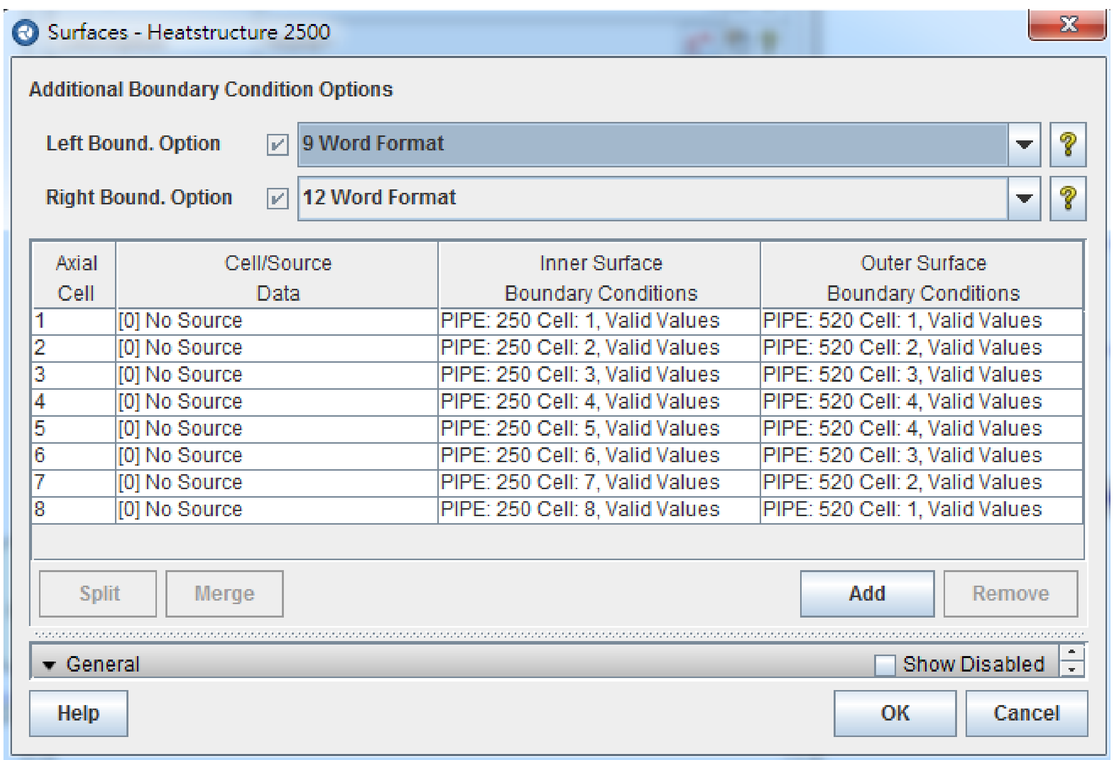


Figure 5 Heat Structure 2500 Properties in SNAP Interface

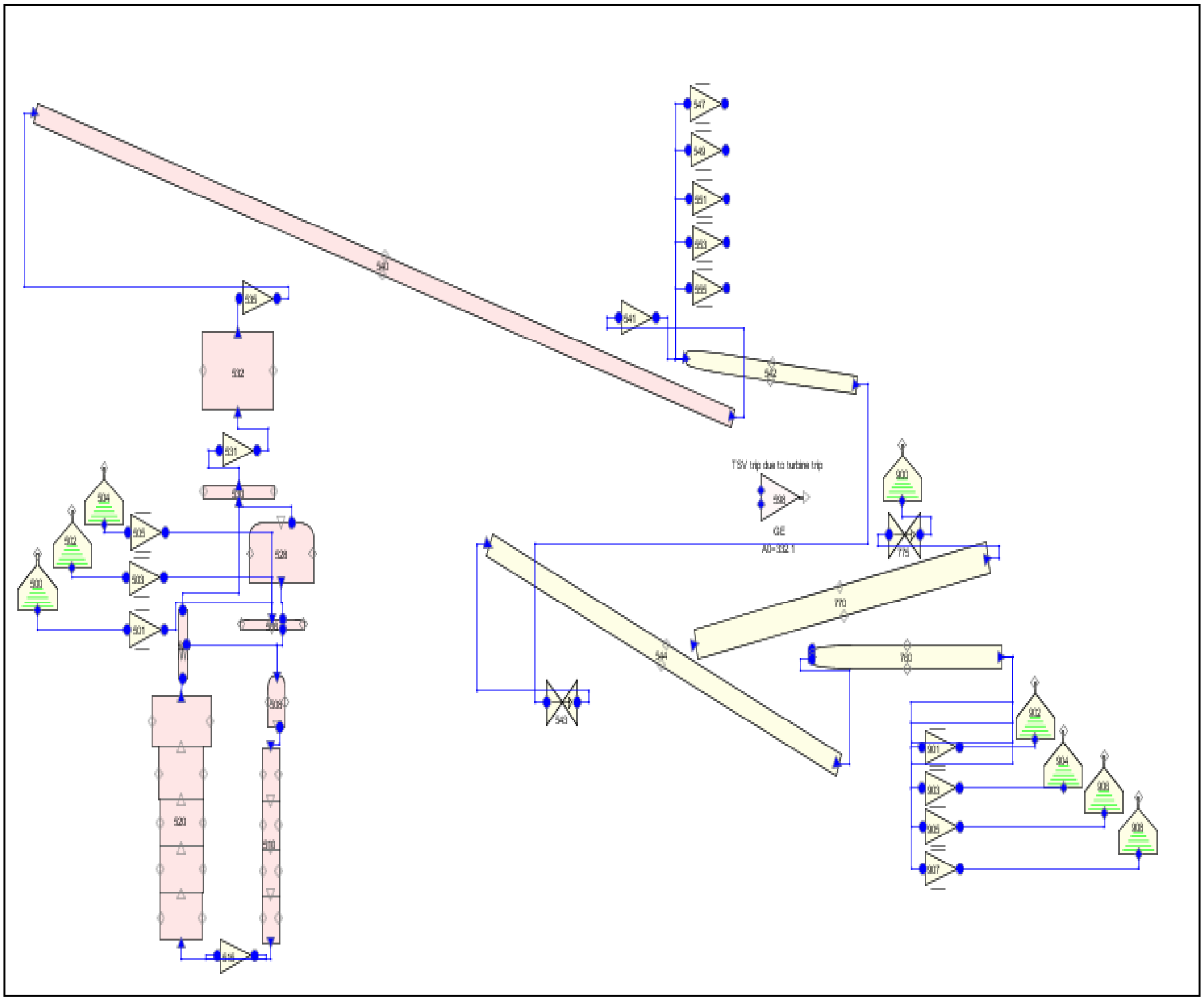


**Figure 6  Components in Secondary Side of Maanshan NPP in SNAP Interface**

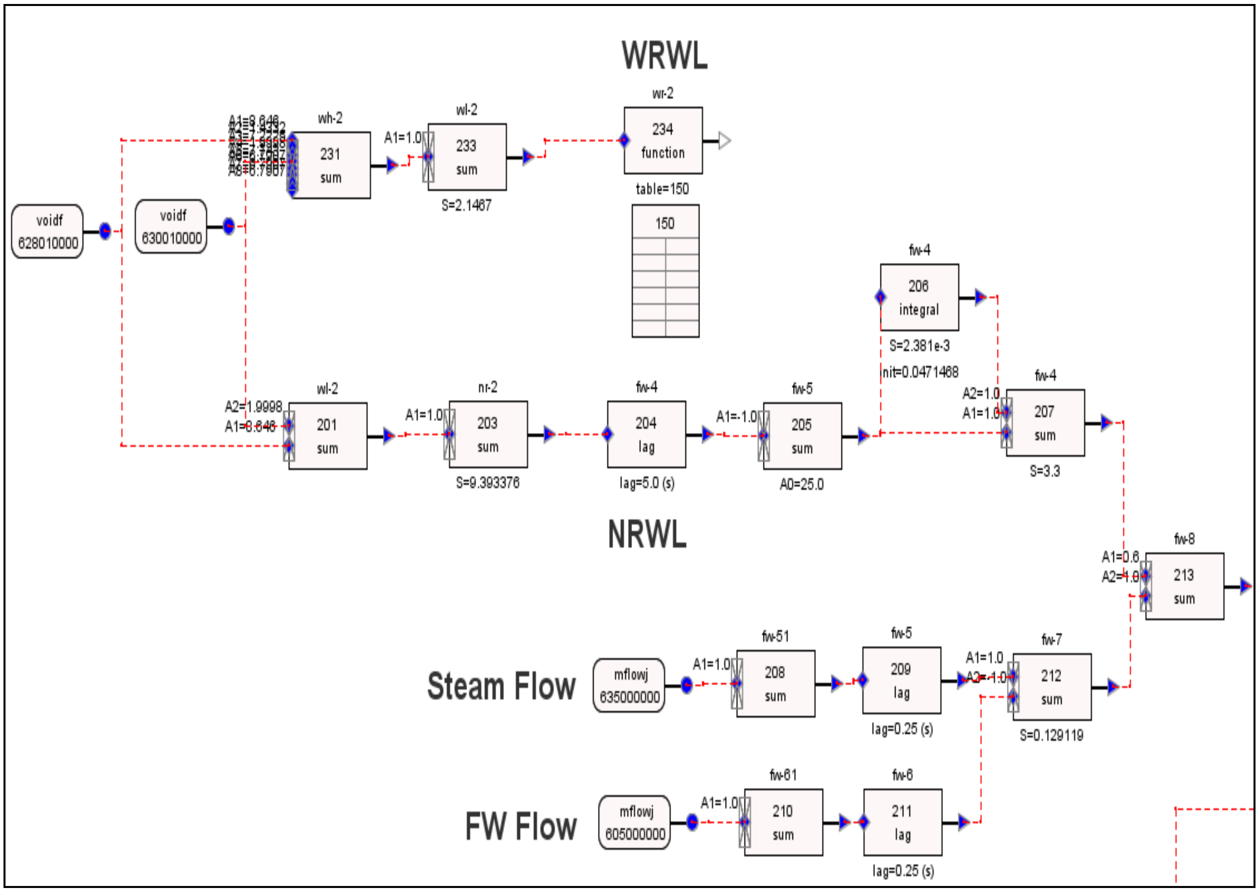


**Figure 7 Feedwater Control System in SNAP Interface**

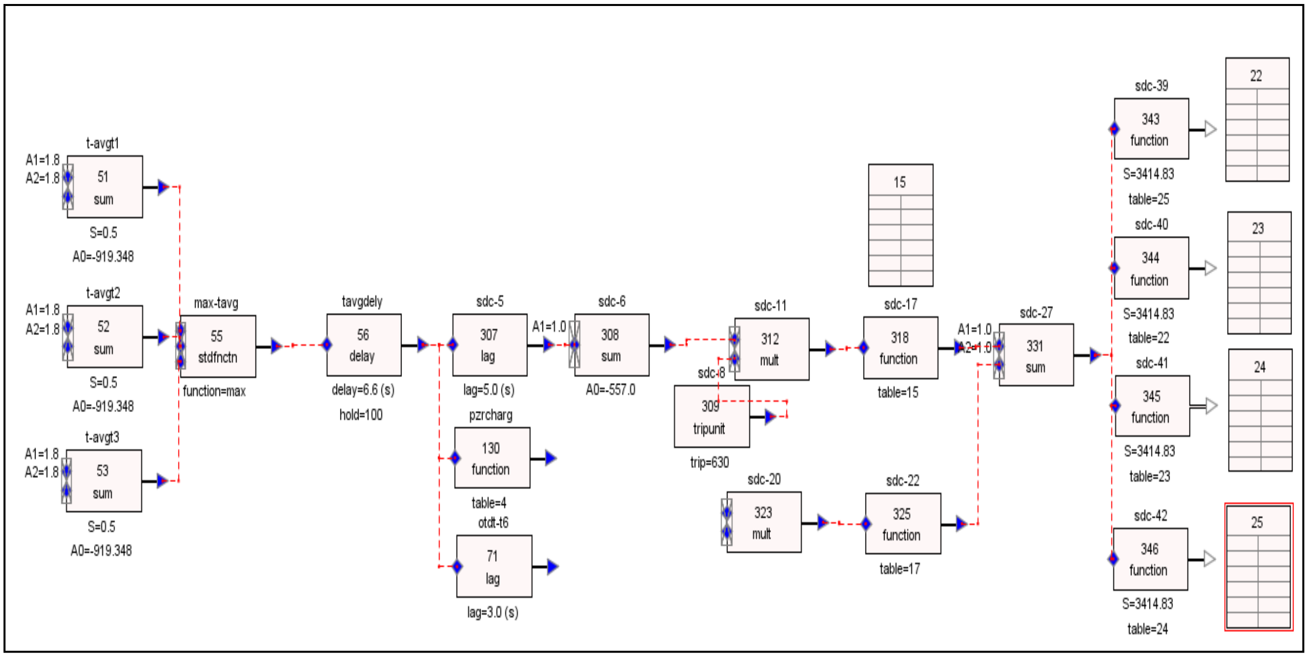


**Figure 8 Steam Dump Control System in SNAP Interface**

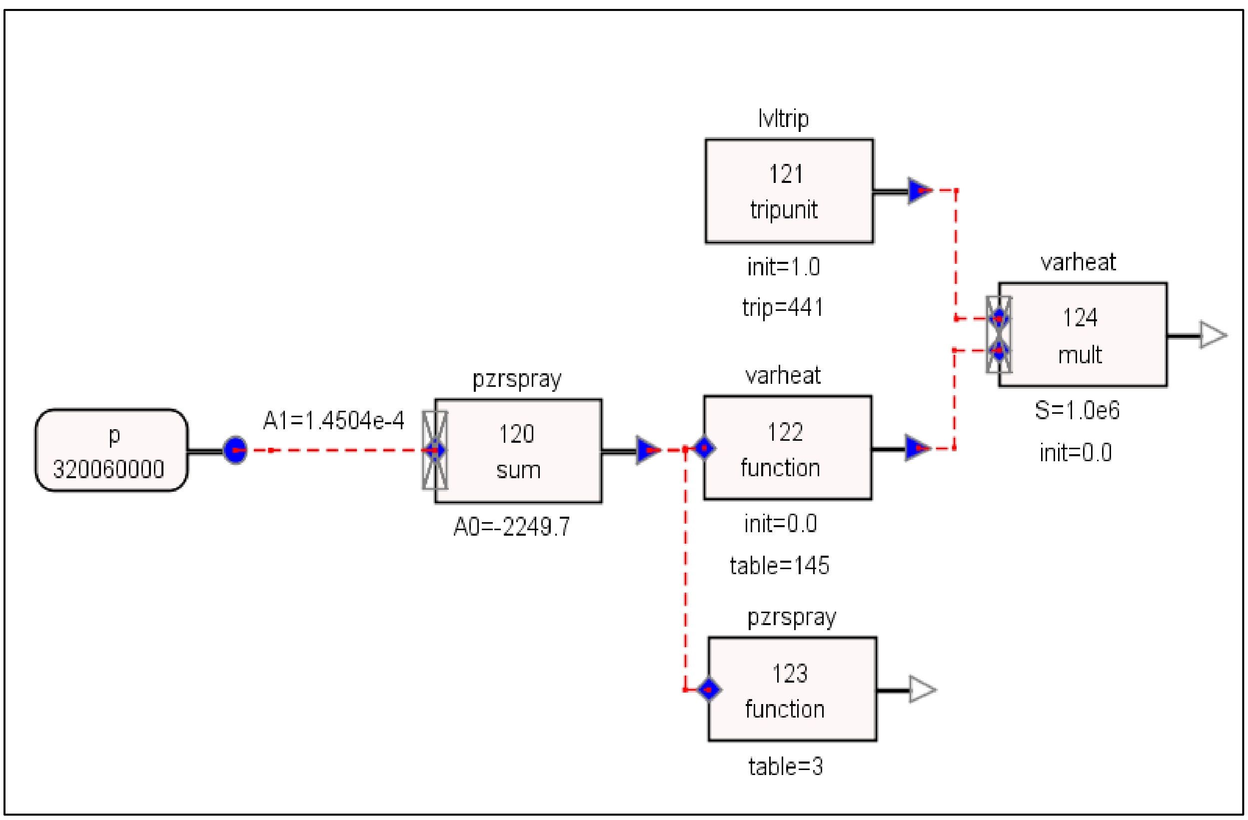


**Figure 9 Heater of the Pressurizer Control System in SNAP Interface**

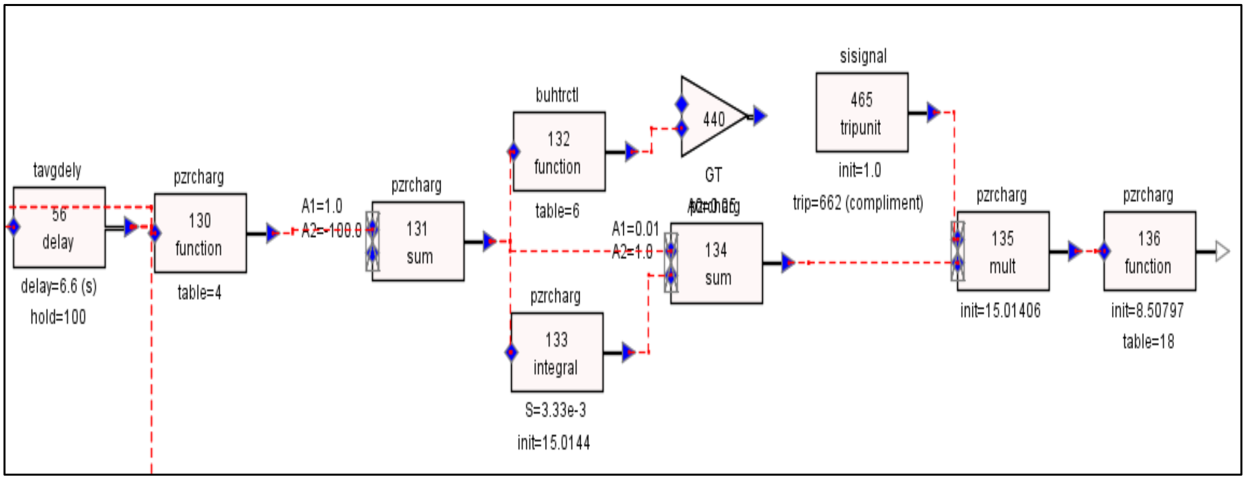


**Figure 10 Pressurizer Injection Control System in SNAP Interface**

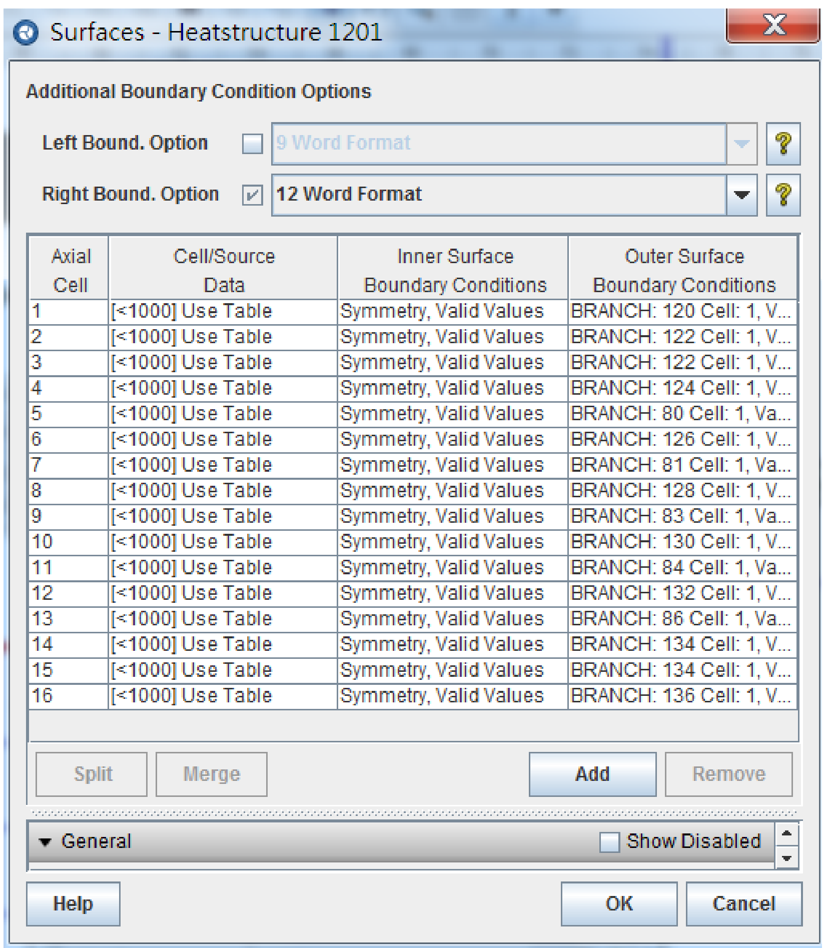


**Figure 11 Properties in Axial Direction of Heat Structure 1201 in SNAP**

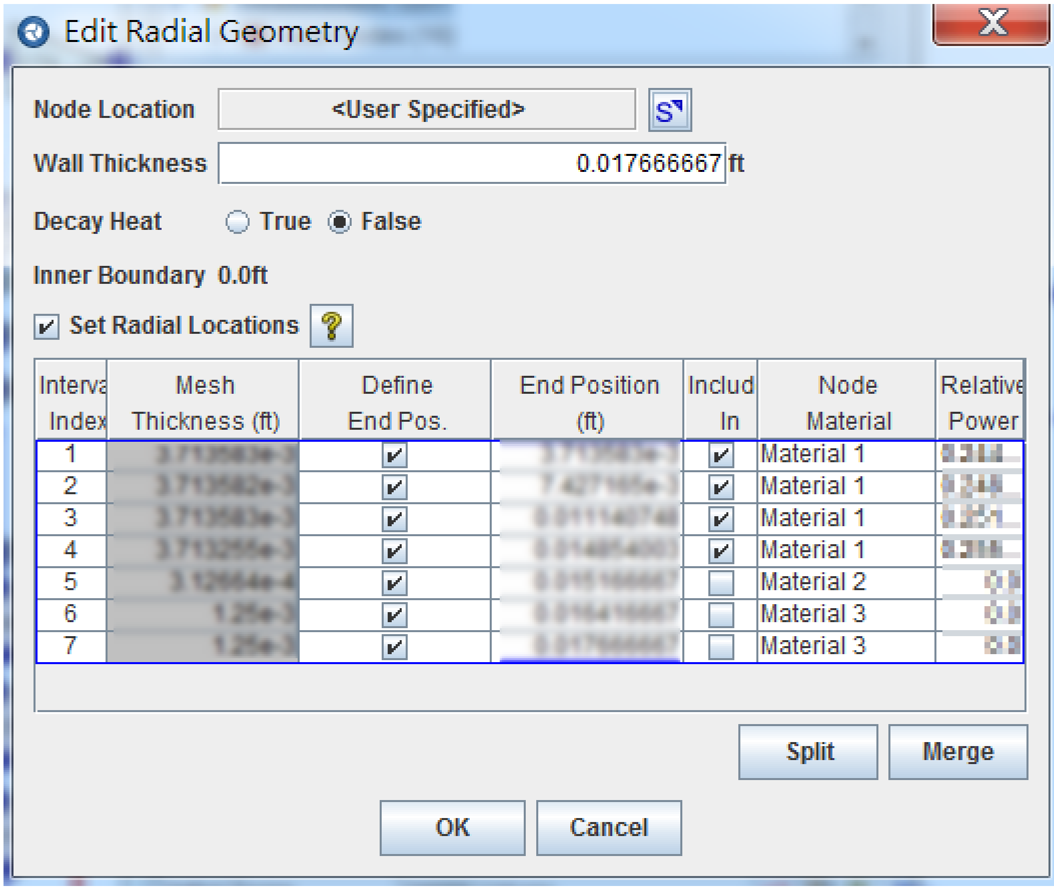


**Figure 12 Properties in Radial Direction of Heat Structure 1201 in SNAP**

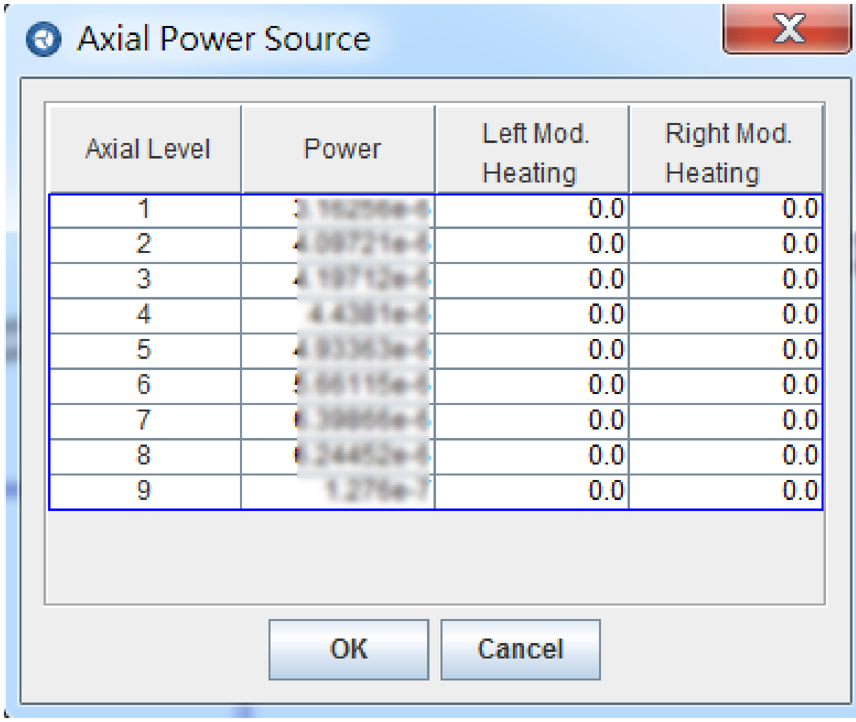


**Figure 13 Power Ratio for Branch 100 to 116 of Heat Structure 1201**

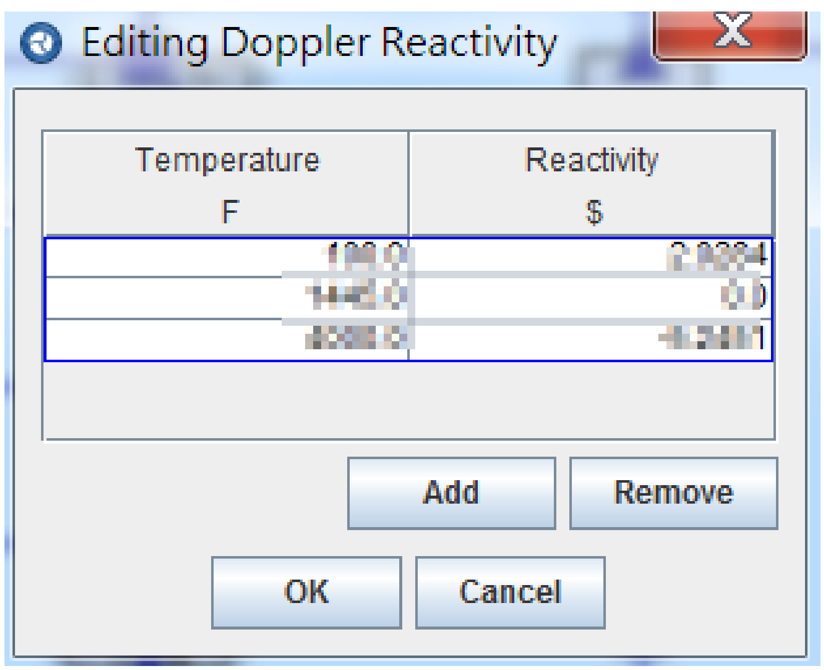


**Figure 14 Doppler Effect Reactivity Feedback Table**

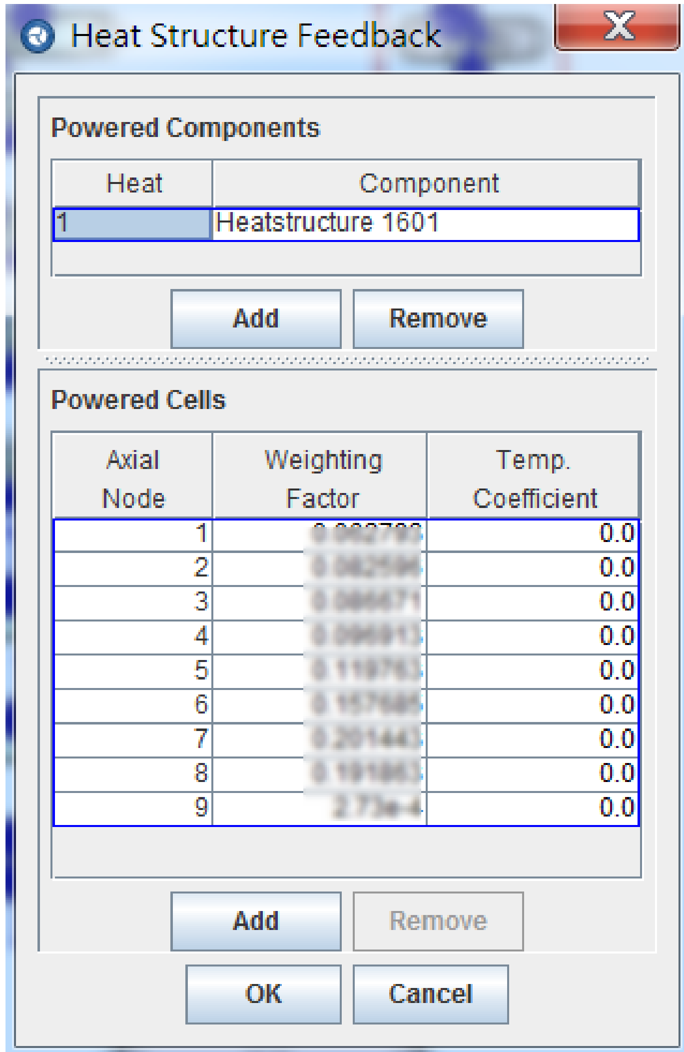


**Figure 15 Doppler Effect Heat Structure Weighting Factor**

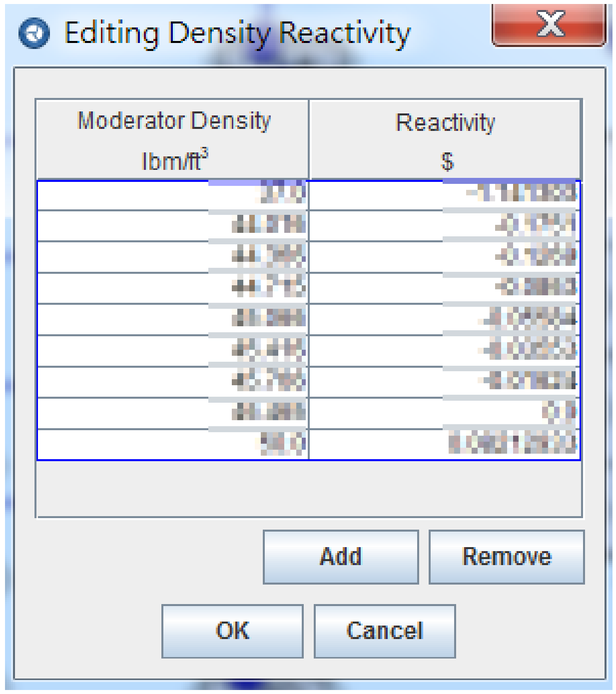


**Figure 16 Density Effect Reactivity Feedback Table**

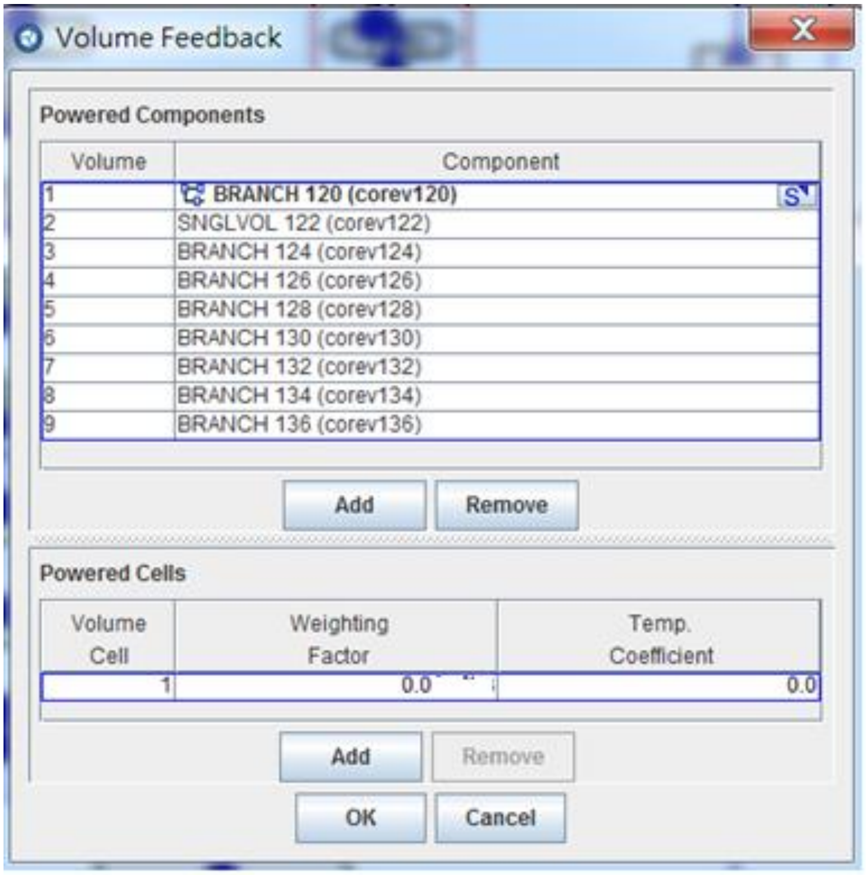


**Figure 17 Moderator Density Effect Volume Weighting Factor**

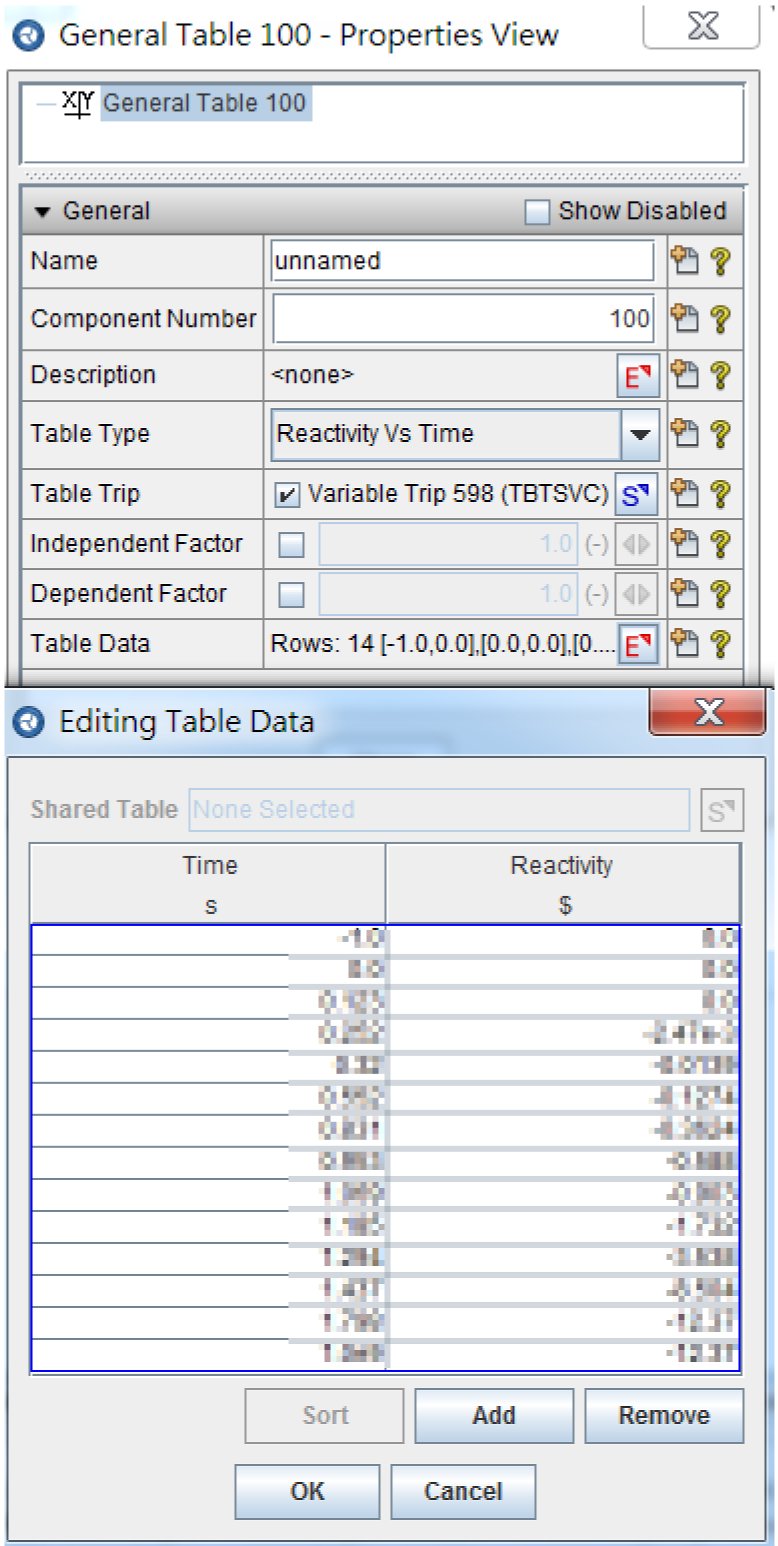


**Figure 18 Reactor Scram Reactivity Feedback Properties**

# 3 LOSS OF FLOW ANALYSIS ESTABLISHMENT

## 3.1 Partial Loss of Flow (PLOF)

Assumptions

The Maanshan NPP is a three loop PWR, and each loop has one RCP with different power buses. All of the systems were assumed to be well functioned in the beginning. The nominal power was 2775MWt (100%) and the core flow was in 100% nominal flow rate. According to the ANS classification of plant conditions, one RCP failure belongs to ANS Condition II event, which is called Partial Loss of Flow event. Moreover, Maanshan Final Safety Analysis Report (FSAR) in chapter 15-3 had mentioned that the main reason of Loss of Flow event was due to RCP failure. Therefore, in this case, it was assumed that the RCP motor of the second loop is tripped because of losing power. Then the RCP started to slow down and also made the flow rate decreased. The protection for a partial loss of flow accident is provided by the low coolant flow reactor trip. Any loop reaches the low flow set point (90% of nominal flow rate) will activate the reactor trip signal. In real situation, there is an electronic signal delay time that results in the extra time for control rod starting to drop in real situation. Hence, as setting the reactor trip control, the delay time should be concerned. Details of the electronic signal delay time is shown in Table 1.

Model Establishment

The Maanshan NPP RELAP5 model would be modified to simulate the Partial Loss of Flow event. To ensure the expected nominal conditions, the model would be run 30-second steady-state to ensure all the thermal hydraulic properties reach to the nominal value. Then, the RCP is tripped at 30 second. Hence, there is a Variable Trip 431 (Figure 23) connected to Pump 380 (Figure 22) to make the RCP tripped at 30 second. The decreasing curve of pump velocity is referred to the torque-inertia equation.

Afterward, a constant Control Block 801 is made as a reference value according to the nominal flow rate of the second loop. There was a Junction Signal (mflowj65000000) that recorded the real-time flow rate in the second loop. Further, the Control Block 802 (Figure 24) will record the number of “mflowj65000000” divided by “Control Block 801”, which represents the current time flow ratio. This ratio will be sent to Variable Trip 597 (Figure 25) which will determine the ratio is less than 0.87 or not. Once the flow ratio is less than 87%, Variable Trip 597 turns on. However, as mentioned above, this trip could not directly activate the reactor scram. Variable Trip 598 (Figure 26) was made to simulate the electronic signal delay by comparing “time of 597” with “time 0”. As this trip turns true, the signal would activate reactor scram control table (General Tables 100). The reactor scram table had been verified in RELAP5/MOD3.3 Model Assessment of Maanshan Nuclear Power Plant with SNAP Interface published in 2016. Therefore, if the flow rate is below the set point, the reactor scram table will make the power decreased rapidly. The whole case would be stopped at 12 second.

## 3.2 <u>Complete Loss of Flow (CLOF)</u>

Assumptions

All the RCPs tripped at the same time belongs to ANS Condition III event, which is called Complete Loss of Flow (CLOF). Like the previous case, all the systems were assumed to be well functioned at the beginning. The nominal power was 2775MWt (100%) and the core flow was in 100% nominal flow rate. Besides, it was assumed that all the RCPs were failed at transient start. However, the analysis of CLOF would be divided into two cases. In CLOF (UV), three of the RCP motors tripped because of losing power and it was due to "low voltage signal" that resulted in the reactor scram (1.5 second delay). In CLOF (UF) case, there was a frequency interference which caused the RCP motors velocity changed. The frequency of the power supply was 60 Hz at the beginning. It was assumed that the interference made the frequency dropped in 5 Hz per second. According to the principle of the induction motor, the motor velocity change is related to the frequency change. Therefore, the velocity curve was made base on the frequency change which was referred to FSAR as shown in Figure 20. The "low frequency signal" would be triggered at 57 Hz which made the reactor scram (0.6 second delay). Details of the velocity curve were described in Figure 20. The electronic signal delay was shown in Table 1.

Model Establishment

CLOF (UV)

The model of these two cases would both run 30-second steady-state to ensure the expected nominal conditions. The model setting of CLOF (UV) was similar to the PLOF. Variable Trip 431 was connected to all of the RCPs in each loop (Pump 280, Pump 380, Pump 480), and would cut down the power supply of RCP motors at 30 seconds. The decelerating curve of velocity was according to the torque-inertia equation for CLOF UV case. Variable Trip 597 (Figure 28) was made to simulate the electronic signal delay by comparing "time 0" with "null 0", and would be connected to the reactor scram control table (General Tables 100). Therefore, reactor shut down at 31.5 second. The whole case would be stopped at 12 seconds.

CLOF (UF)

To simulate the pump decelerating, the pump velocity table mentioned above was settled in velocity menu of RCPs in each loop. Variable Trip 431 was connected to the velocity table trip. The Variable Trip 431 determined whether the time of low frequency set point was reached or not. All the RCP motors would follow the velocity table if the velocity table trip (Variable Trip 431) turns on. Variable Trip 597 (Figure 32) was used to simulate the signal delay by comparing "time 0" with "null 0" and was connected to the reactor scram control table (General Tables 100). In this case, the reactor scrammed at 31.25 second and the whole case would be stopped at 12 second.

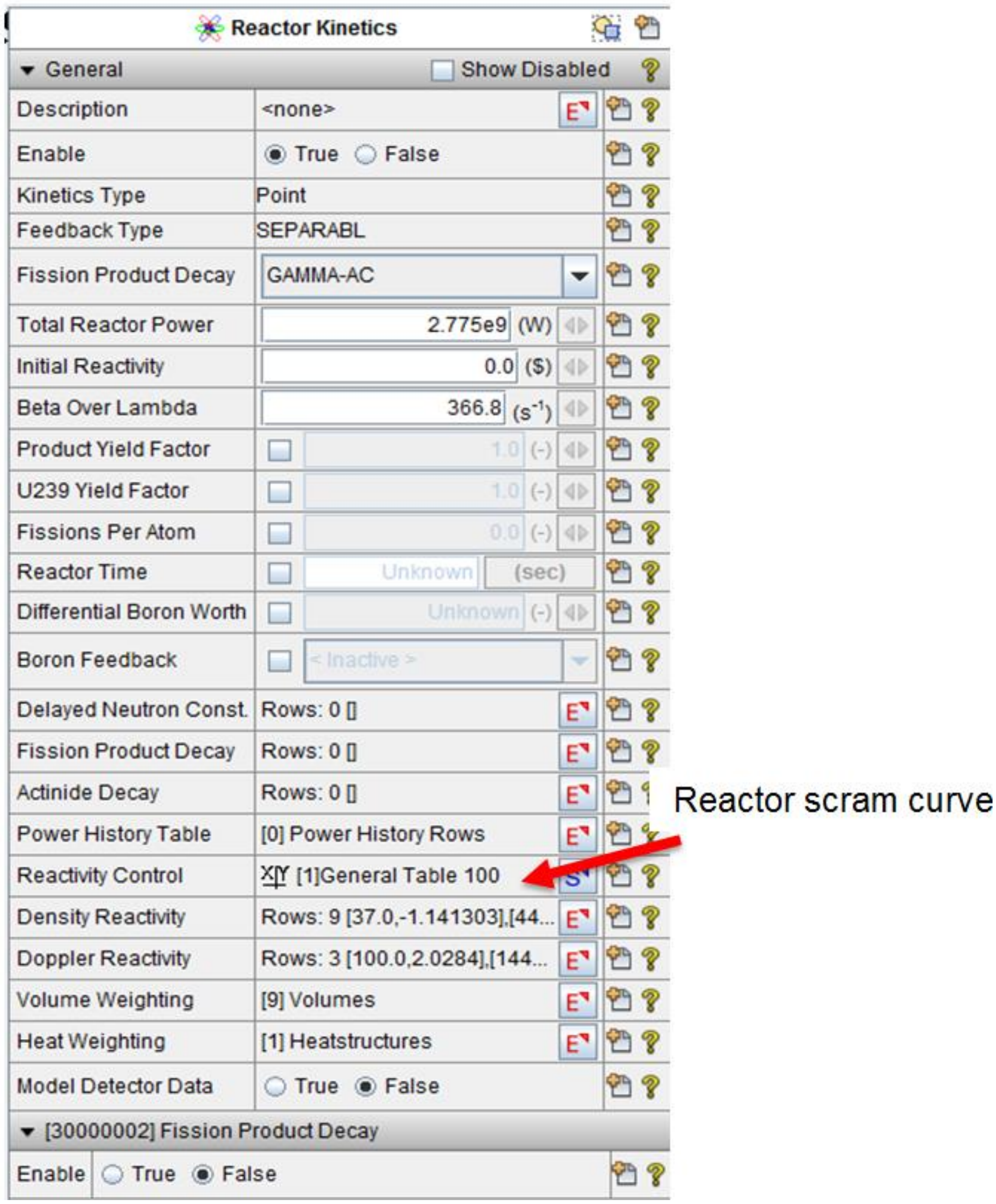


**Figure 19 Reactor Kinetics**

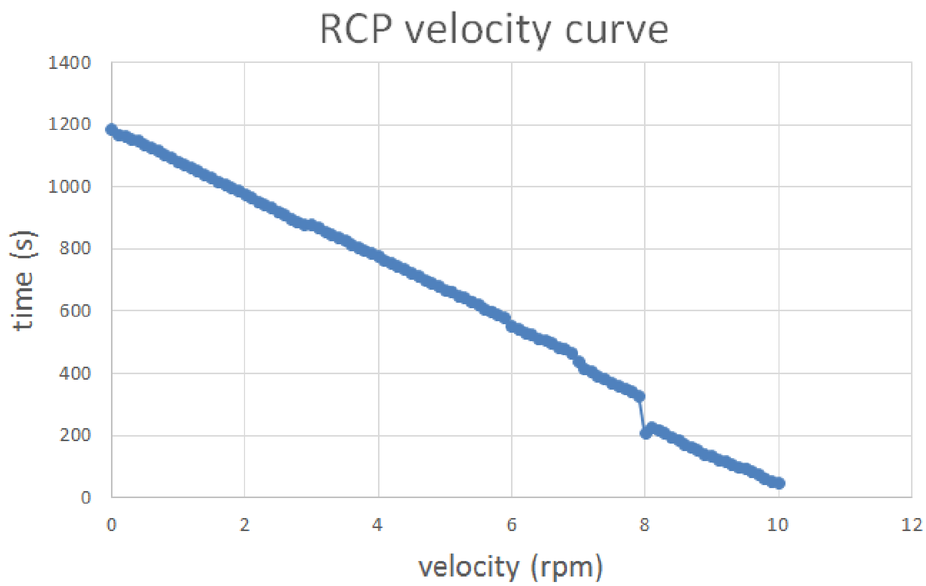


**Figure 20 RCP Velocity Curve in CLOF (UF)**

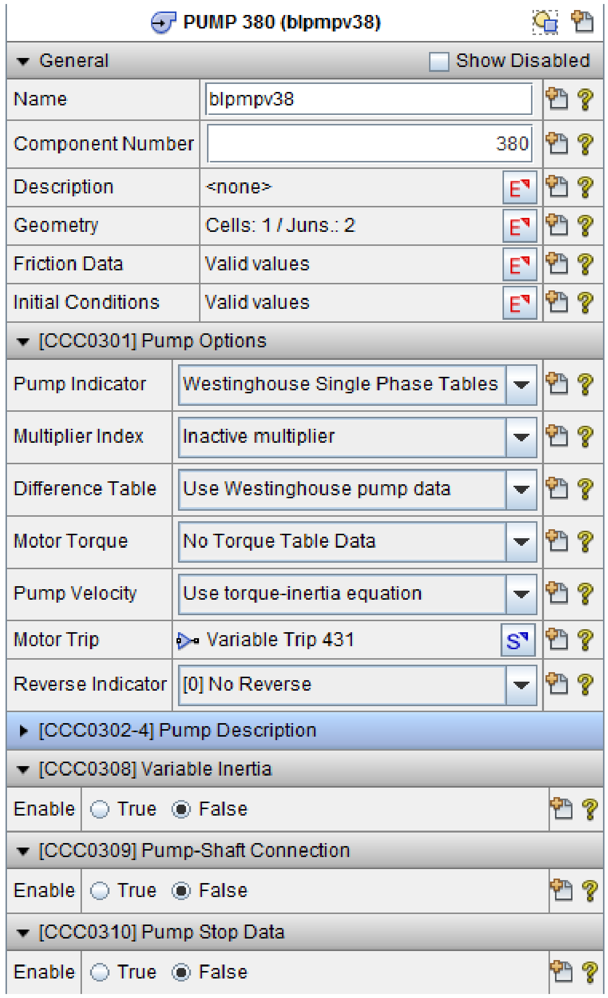


**Figure 21 Pump 380 (PLOF)**

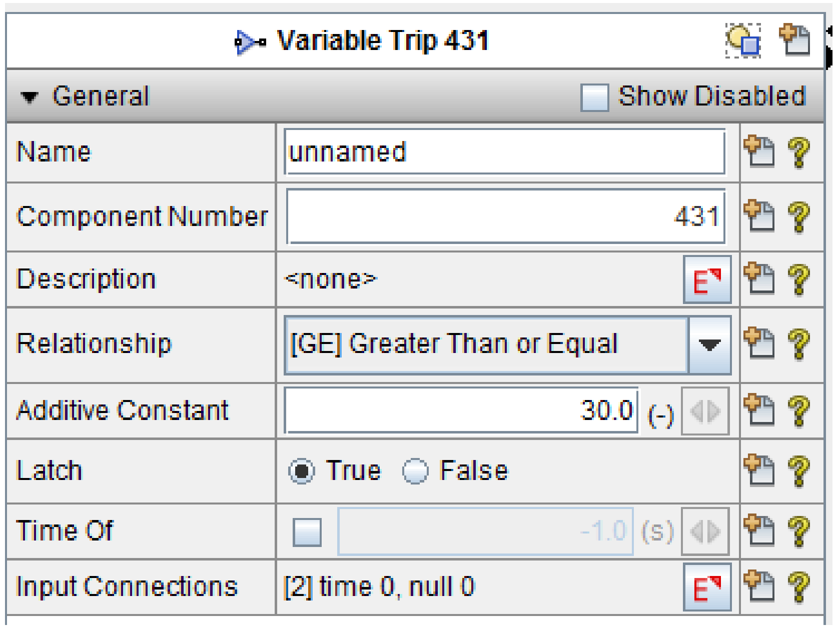


**Figure 22 Variable Trip 431**

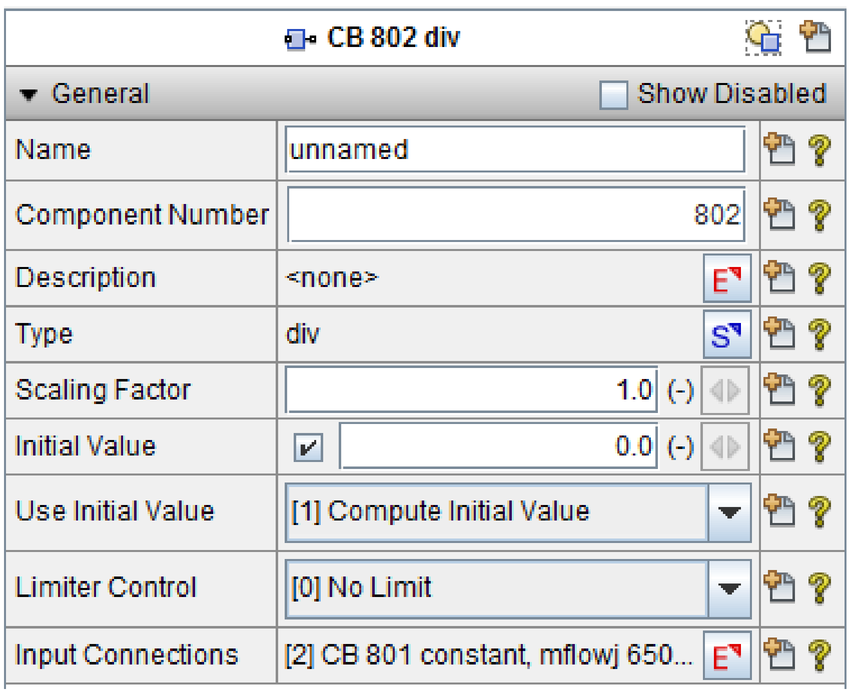


**Figure 23 Control Block 802 (PLOF)**

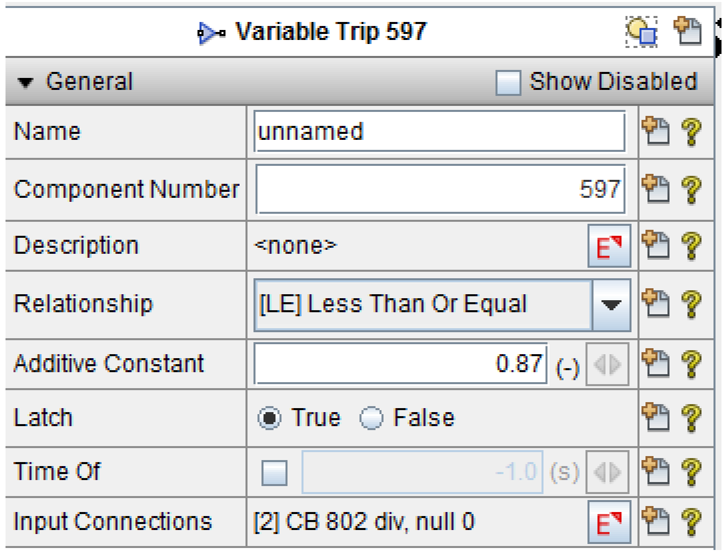


**Figure 24** Variable Trip 597 (PLOF)

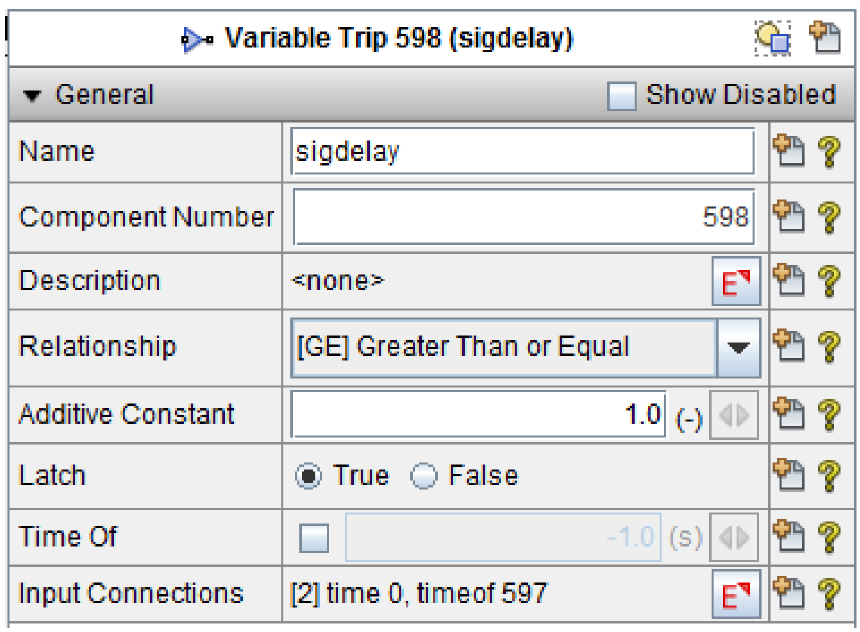


**Figure 25** Variable Trip 598 (PLOF)

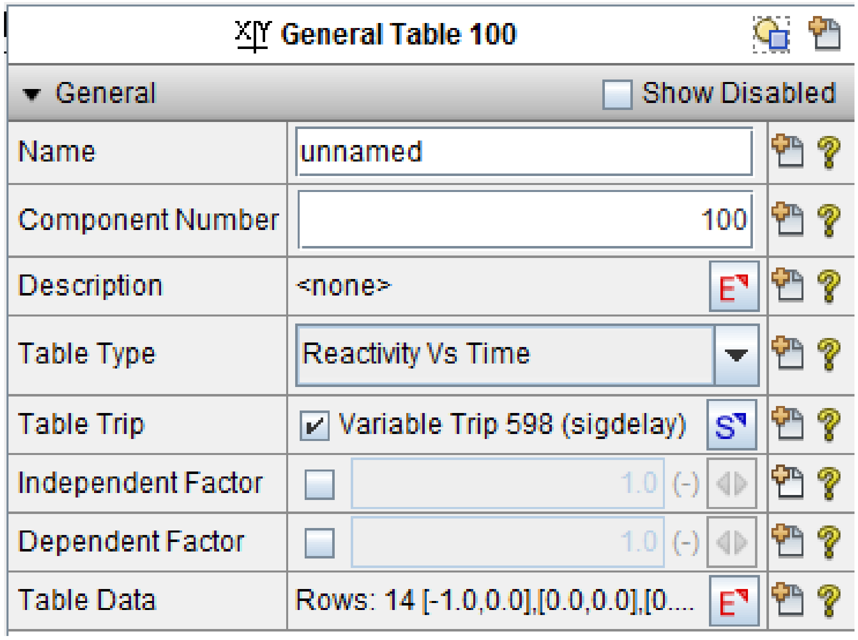


**Figure 26 Reactor Scram Control Table (PLOF)**

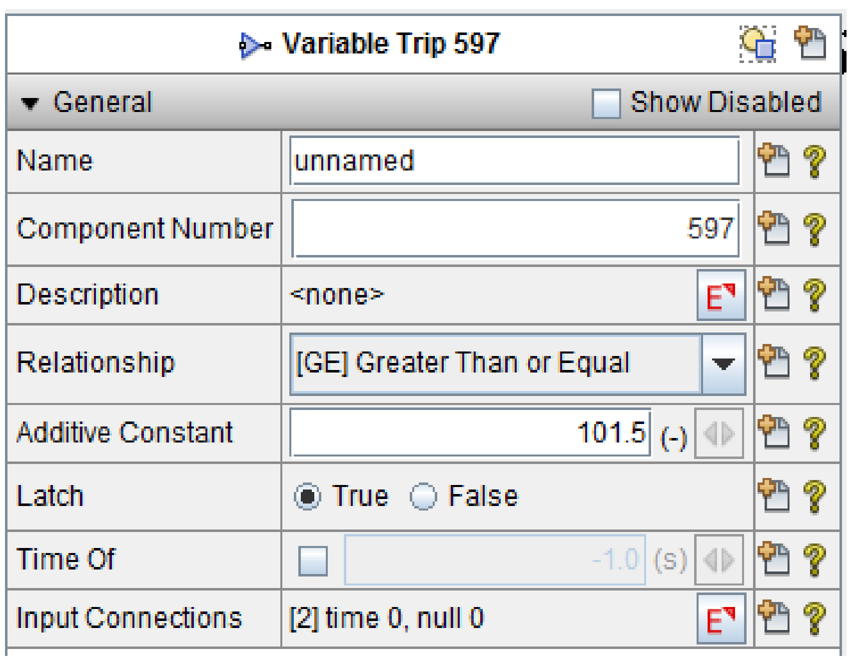


**Figure 27 Variable Trip 597 (CLOF UV)**

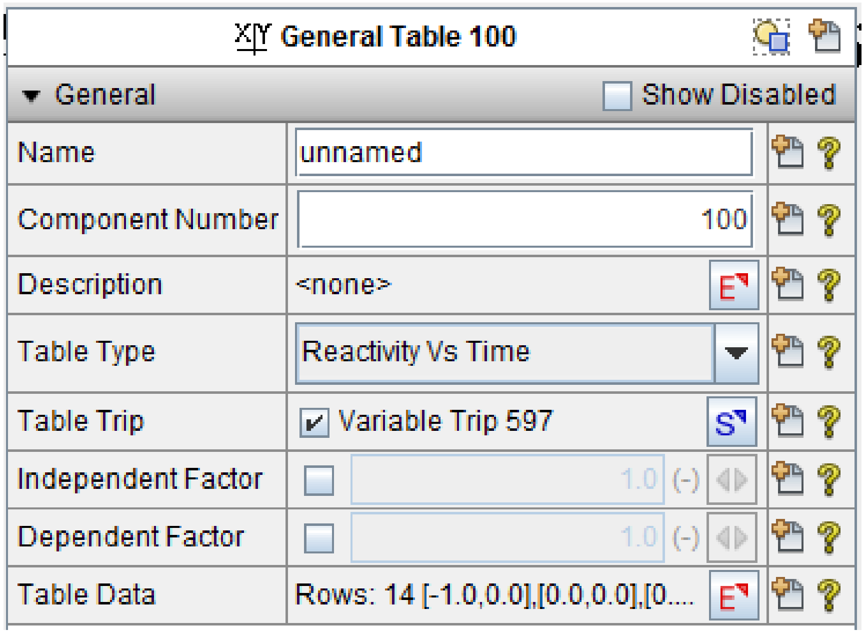


**Figure 28 Reactor Scram Control Table (CLOF UV)**

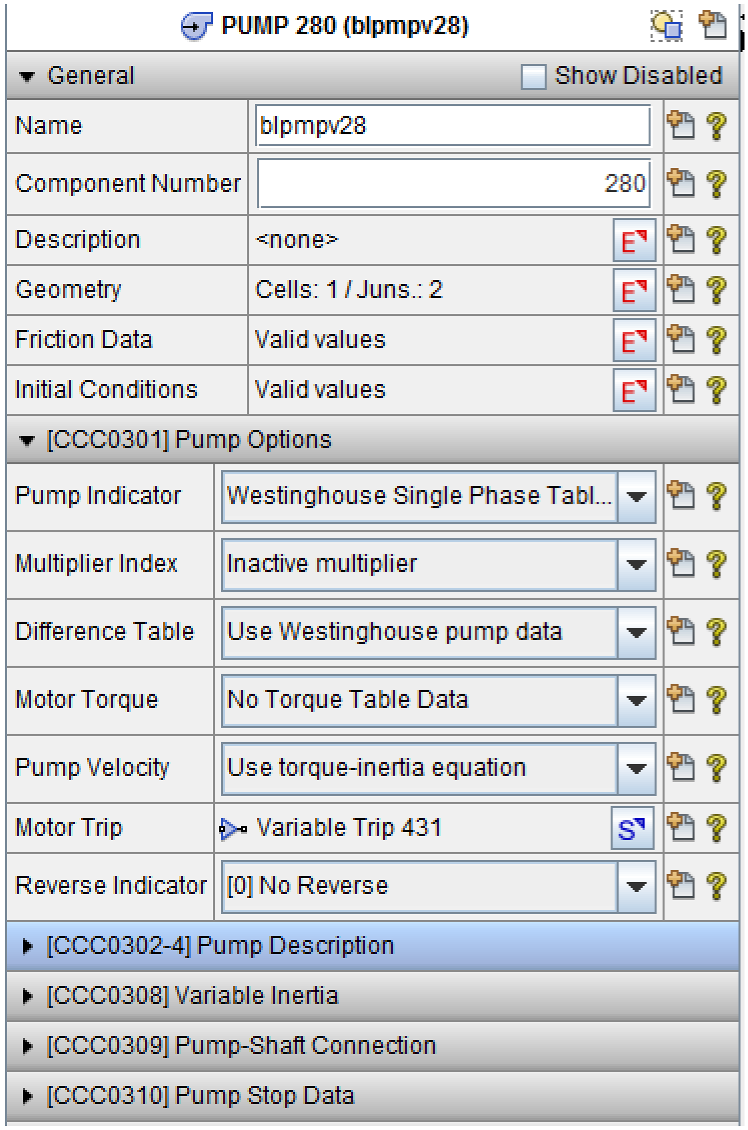


**Figure 29 Pump 280 (CLOF UV)**

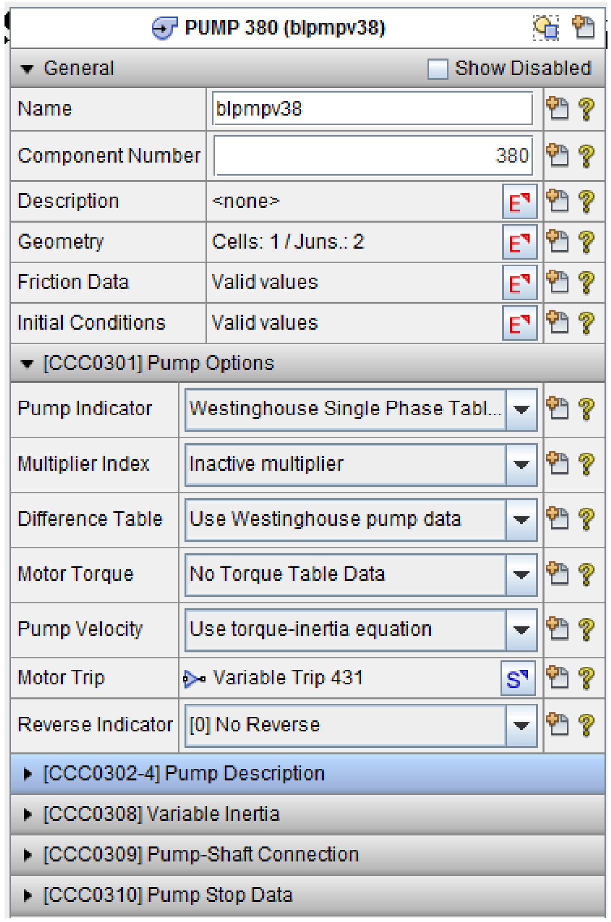


**Figure 30 Pump 380 (CLOF UV)**

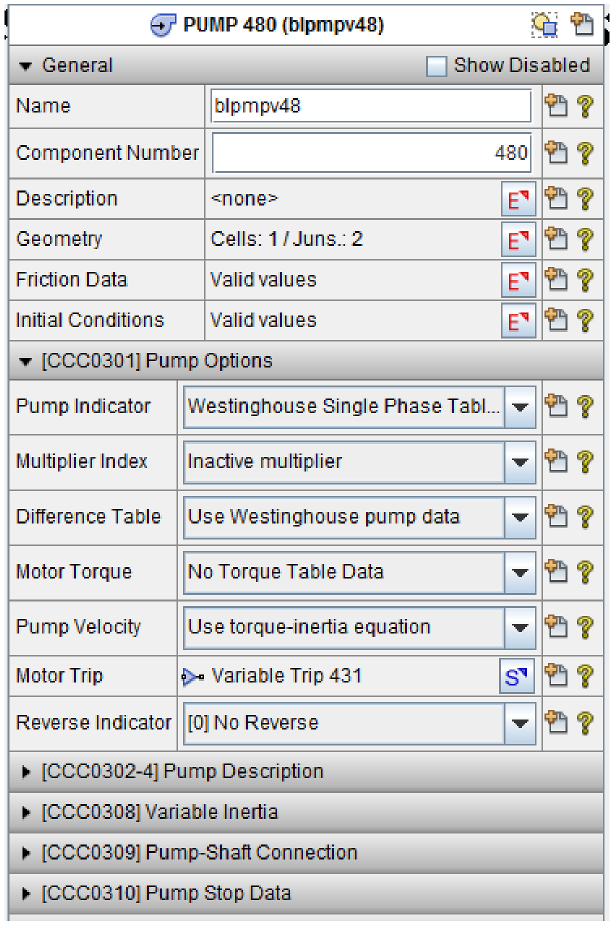


**Figure 31 Pump 480 (CLOF UV)**

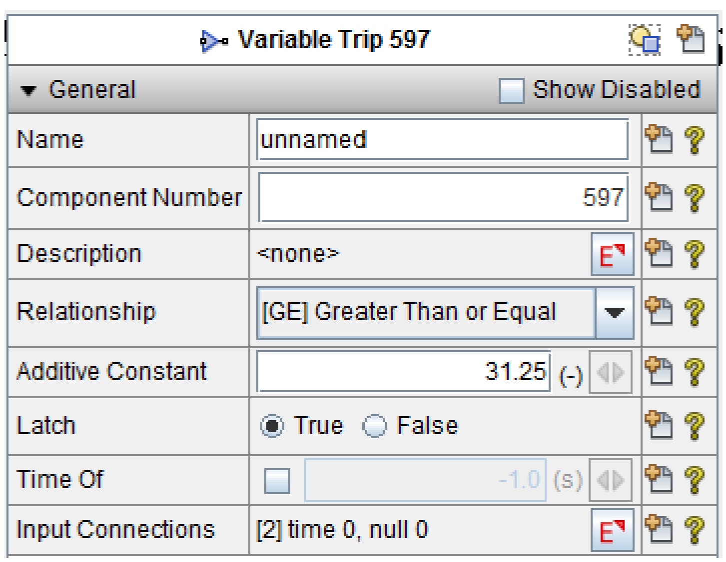


**Figure 32 Variable Trip 597 (CLOF UF)**

**Table 1 Maanshan NPP Reactor Scram Electronic Signal Delay**

| Reactor scram signal | Set point | Delay time (second) |
|---|---|---|
| **RCS Low Flow** | 87% Nominal | 1.0 |
| **RCP Undervoltage** | --- | 1.5 |
| **RCP Underfrequency** | 57 Hz | 0.6 |

# 4 ANALYSIS RESULTS

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 and it was according to the Maanshan NPP Final safety analysis report. 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 Loss of Coolant cases were described [7]. To compare the data analysis results, the 30-second steady-state data was eliminated. The 0-second data results in the following tables and plots represent to the beginning of the transient.

## 4.1 Partial Loss of Flow (PLOF)

Event Sequence

The initial conditions of the CLOF (UV) event were described in Table 2. 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. At the beginning of the analysis, the power was 2775MWt (100%). From this table, it is shown that both the calculations of RELAP5/MOD3.3 model were consistent with the plant data.

The PLOF transient started at 0 second. One of the RCPs motor was tripped and the flow rate started to decrease due to the RCP slowing down. The fuel average temperature increased and the pressure in the pressurizer also increased. At 1.35 second, the flow rate reached the low flow rate set point that triggered the reactor scram signal. Finally, the control rod started to drop at 2.35 second because of the electronic signal delay. Therefore, the fuel average temperature and the pressure in the pressurizer decreased with the power decreasing. At 6.75 second, the total reactor power was reduced to 10% of the nominal power. Details of the event sequence were described in Table 3.

**Table 2 Initial Conditions of PLOF Transient**

| Parameters | Plant data | RELAP5 (PK) | TRACE (PK) |
|---|---|---|---|
| Power (MW) | 2775 | 2775 | 2775 |
| Core Temperature (K) | 581.37 | 584.62 | 589.9 |
| Core Flow Rate (kg/sec) | 13775 | 14070 | 13380.23 |
| Steam Flow Rate (kg/sec) | 522.99 | 533.52 | --- |
| PZR Pressure (MPa) | 15.41 | 15.76 | 15.39 |
| Hot Leg Temperature (K) | 599.6 | 599.3 | --- |
| Cold Leg Temperature (K) | 565.2 | 565.2 | --- |

**Table 3 The Sequence of PLOF Transient**

| Event (sec) | RELAP5 (PK) | TRACE (PK) |
|---|---|---|
| Loop 2 RCP Trip | 0.0 | 0.0 |
| Low Flow Set point | 1.75 | 1.67 |
| Reactor Scram | 2.75 | 2.67 |

Analysis results

This section describes the analysis data results of PLOF transient event. The RELAP5/MOD3.3 model of Maanshan NPP was first performed in power table (PT) mode. The table of reactor power in table mode was according to the analysis of Final Safety Analysis Report of Maanshan NPP. The main purpose was to check the rationality of the model setting. Then, the model would be operated in point kinetic mode. In this research, the analysis data in PT mode and PK mode would be compared with each other. Finally, the results were also compared with the TRACE data which were made few years ago.

Figure 33 shows the reactor total power in power table (PT) mode and point kinetic (PK) mode during PLOF transient. As mentioned in previous paragraph, when the flow rate reached the set point, reactor scram would be activated. Theoretically, the total power of two modes should be similar. However, it was obvious that the power curve were different. The power in PK mode decreased more quickly than that in PT mode. The reactor scram of PK mode leads about 1 second. After checking the reactor scram table, it was because of different negative reactivity feedback that made the power drop of RELAP5 PK model more quickly. In the RELAP5 PK model, the negative scram feedback table was based on all control- rod-in condition which would cause more efficient negative feedback. However, there is no related information to re-evaluate the scram table for this case, this assessed scram table was kept in this research.

Figure 34 shows the core flow rate of PK and PT modes. In this case, the decreasing of flow rate was mainly resulted from second loop RCP slowing down. Both modes were operated with the same RCP properties. Therefore, the tendencies of these two curves were consistent. It was also reasonable that the flow rate went down to about 65% of the nominal core flow rate with one RCP tripped.

Figure 35 was the core temperature. The two temperature curves had similar tendency. They both had a little rise in the beginning of transient due to the decreasing of flow rate and both reached the same peak. Then the temperature started to decrease because of the reactor scram and power decrease. In Figure 33, total reactor power went down to about 10% of rated power at 4 second in PK mode and it took about 6 second in PT mode to reach about 15% of rated power. Similarly, the moment when core temperature started to decrease in PK mode also leads about 1.5 second.

The pressurizer pressure is shown in Figure 36. The reducing of flow rate made the whole system core temperature increased. Therefore, the pressurizer pressure increased during 0-4 second. After the reactor scram occurred, the decreasing of core temperature made the pressurizer pressure dropped. As the temperature which was shown in Figure 35, the pressurizer pressure also corresponded to the power variation.

Although it had different power curve that caused different results between PT mode and PK mode, the thermal hydraulic properties of two modes still matched their power curve individually. Not only flow temperature but also system pressure made correct response to the power variation.  It indicates that the hydraulic setting and boundary conditions was reliable.

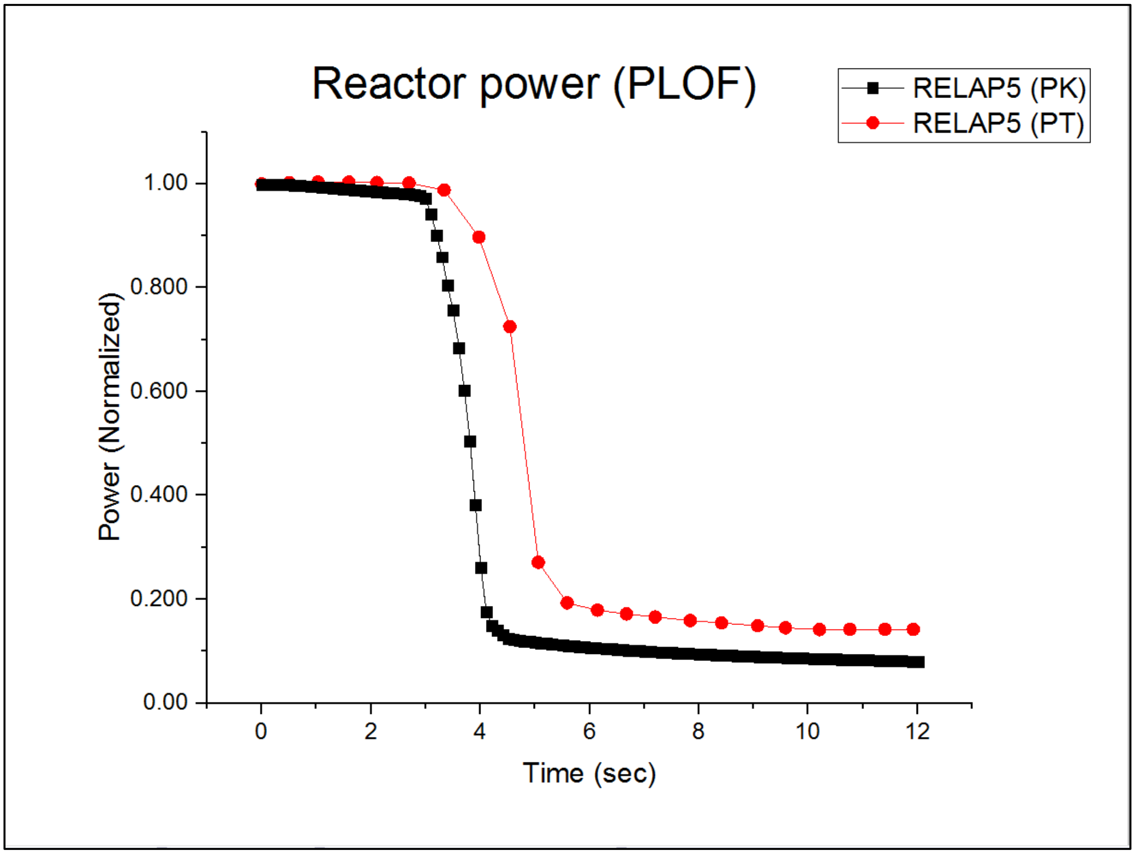


**Figure 33  Power (Normalized) Variation During PLOF Transient**

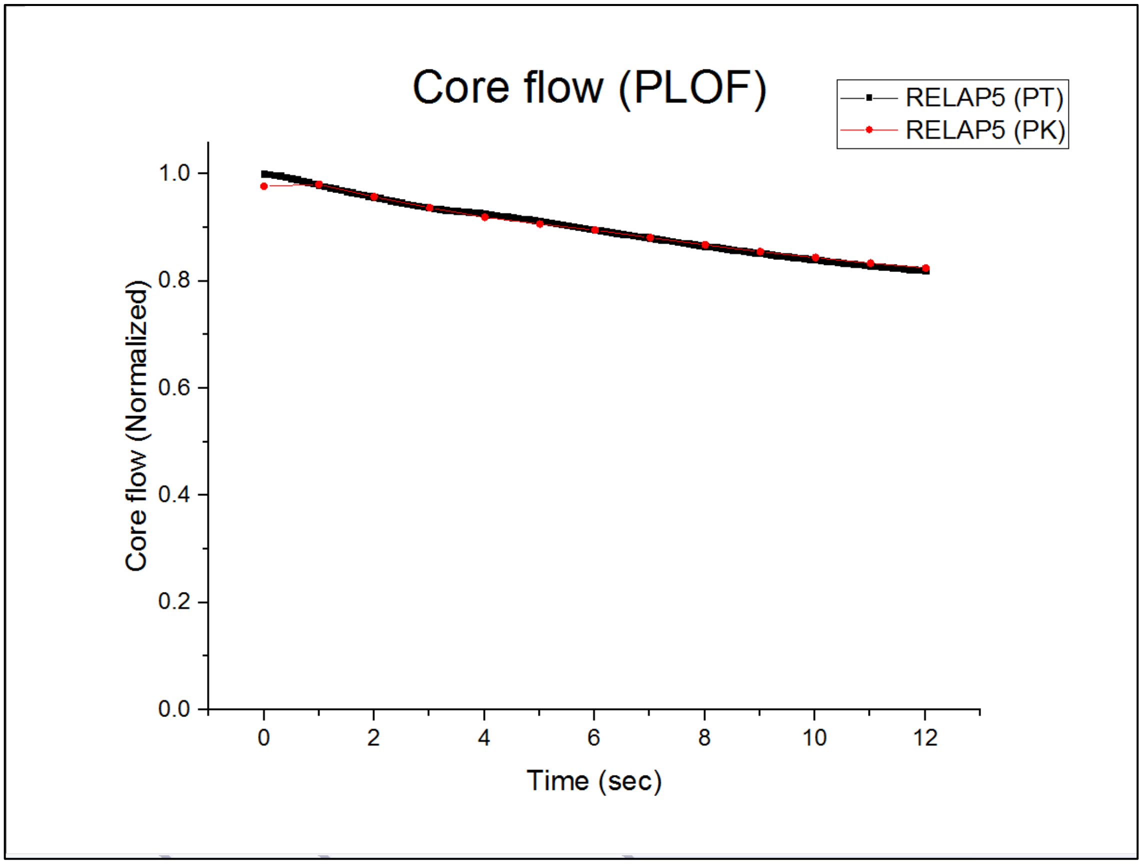


**Figure 34 Core Flow During PLOF Transient**

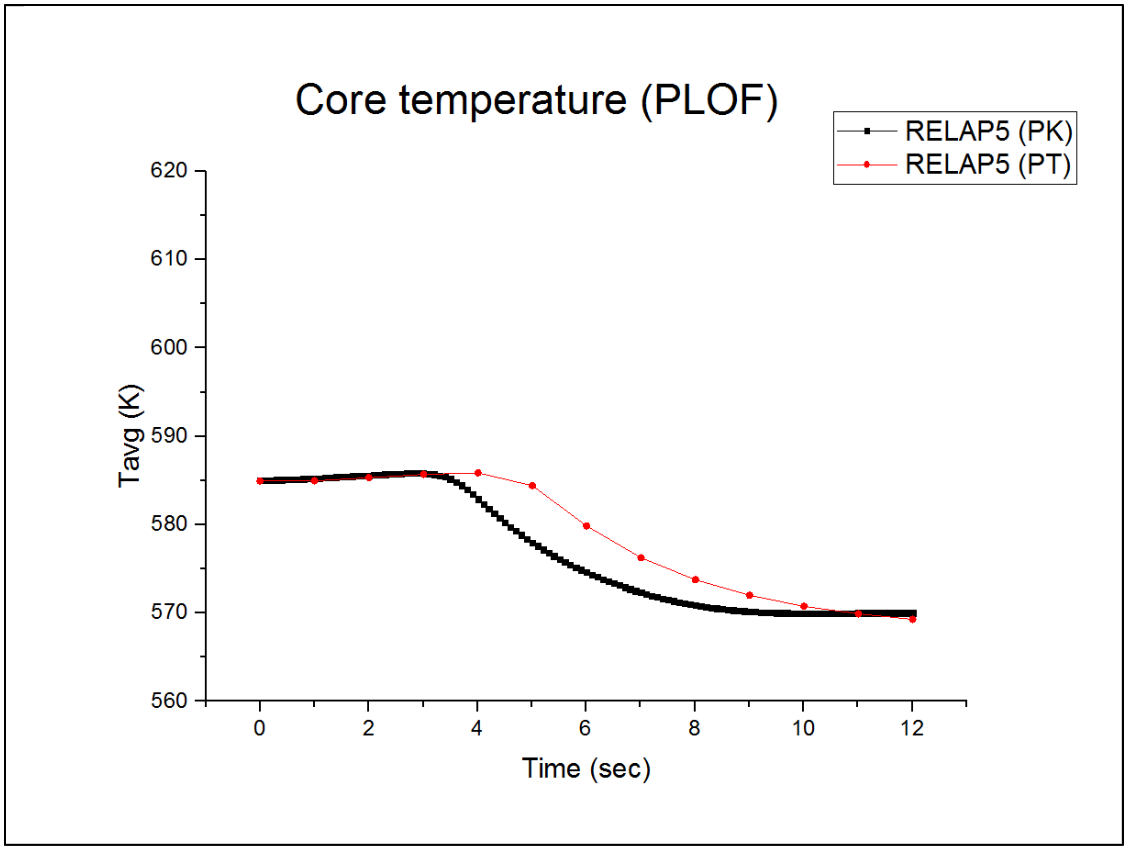


**Figure 35 Core Average Temperature During PLOF Transient**

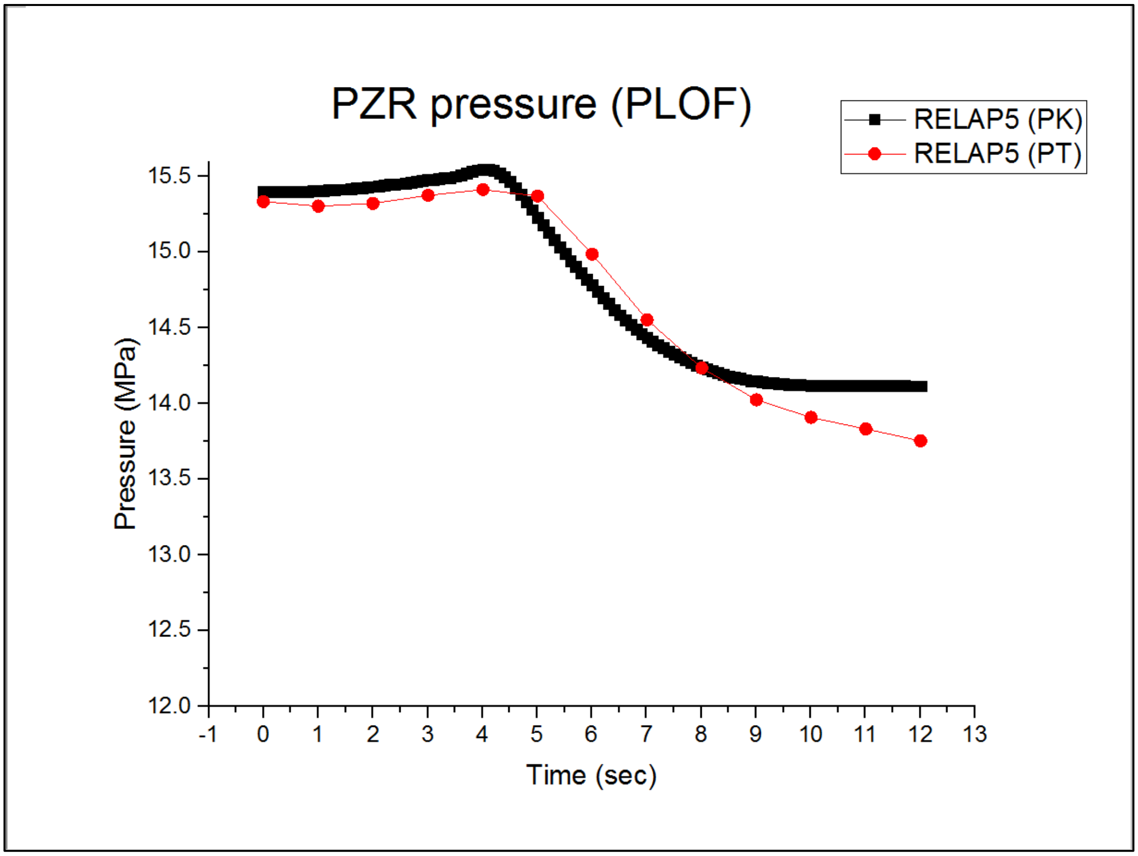


**Figure 36 Pressurizer Pressure During PLOF Transient**

Then, the results of RELAP5 (PK) would be discussed further by comparing with the results of TRACE model which was made few years ago [7]. The same case was simulated in Maanshan TRACE model by Jung-Hua Yang, and the results were mentioned in ref. 7. The assumption and model setting of TRACE model was also based on FSAR of Maanshan NPP.

According to the PLOF event results in TRACE model, it was shown that low flow set point reached at 1.67 second (Table 3). Because of the electronic signal delay, reactor scram occurred at 2.67 second. On the other hand, the low flow set point reached at 1.75 second in RELAP5 model and the power started to drop at 2.75 second. Figure 37 includes the core flow and shows the difference. In spite of the same RCP properties input, the core flow in TRACE model decreased a little faster than in RELAP5 model. Therefore, the low flow set point reached earlier in TRACE analysis as it was shown in Table 3.

Figure 38 shows the power variation of these three models. It was obvious that the power curve in RELAP5 (PK) simulation was different from which in TRACE and RELAP5 (PT) simulation. The reactor scram in Figure 38 seemed to advance about 1 second. However, according to Table 3 and Figure 37, it is shown that the time difference of reaching low flow set point was only 0.08 second. It indicates that reactor scram almost occur at the same time. Hence, the difference comes from different negative reactivity feedback that made the power drop in different shape.

Figure 39 illustrates the core temperature. The curves were both at about 585 K at the beginning. At about 3 second, the temperature of RELAP5 started to decrease which is corresponding to the power drop time. After about 1 second, the temperature curve of TRACE then started to decrease. Moreover, it was easily to find that the different temperature decreasing shape between two models. Temperature in RELAP5 model dropped more rapidly than that in TRACE model. In Figure 38, it was shown that the total power reduced to about 8% of rated power in RELAP5 model. On the other hand, it merely went down to about 18% in TRACE model. As mentioned above, this difference comes from the different scram reactivity feedback. It could also fully explain why the core temperature in RELAP5 model was lower than in TRACE model in the end.

Pressurizer pressure was shown in Figure 40. In this case, the pressurizer pressure was mainly influenced by the core temperature. Therefore, the curve tendency was similar to the core temperature. The curves increased along with the increasing temperature. After the reactor scram, the temperature decreased and also made the pressurizer pressure went down. Because the reactor scram time of TRACE analysis was longer than that of RELAP analysis, the TRACE model may generate more vapors to increase the system pressure. The peak value of TRACE curve was about 15.7 MPa and 15.5 MPa of RELAP5 curve.

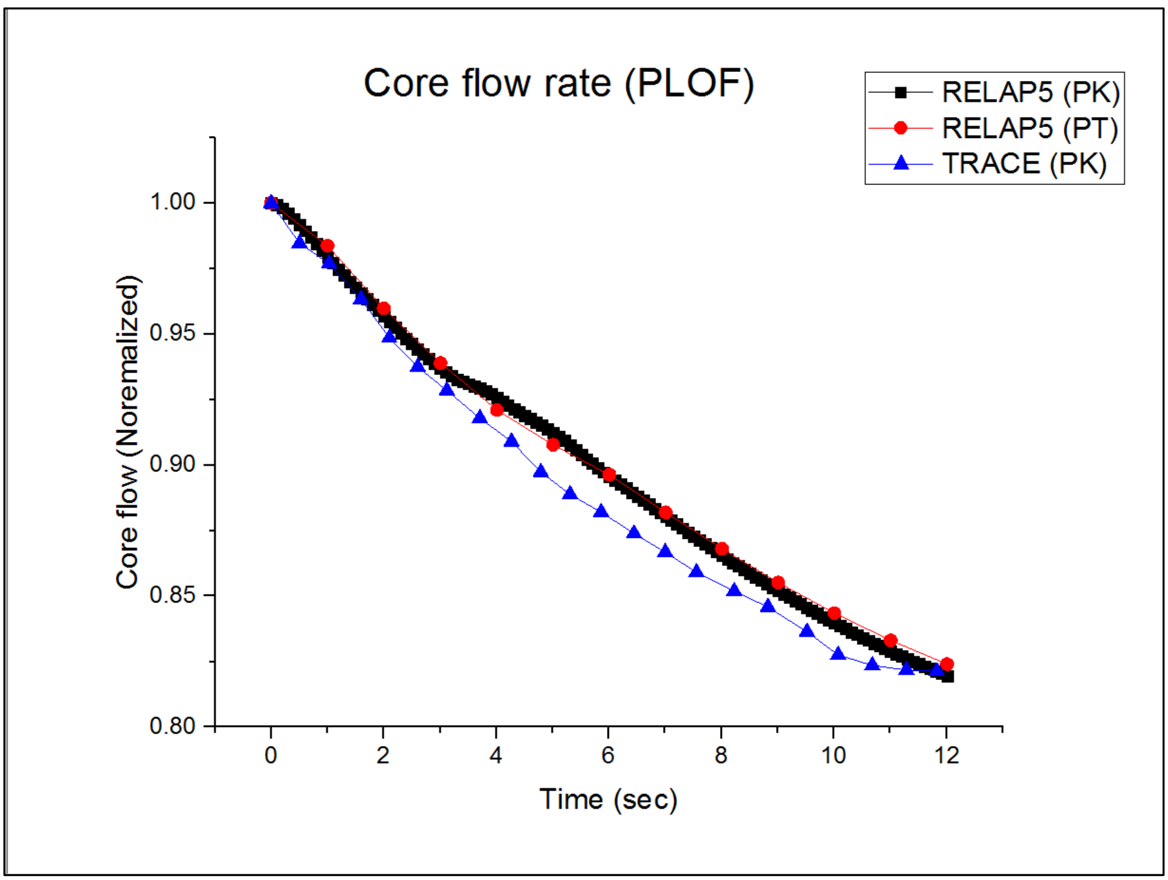


**Figure 37 Core Flow During PLOF Transient**

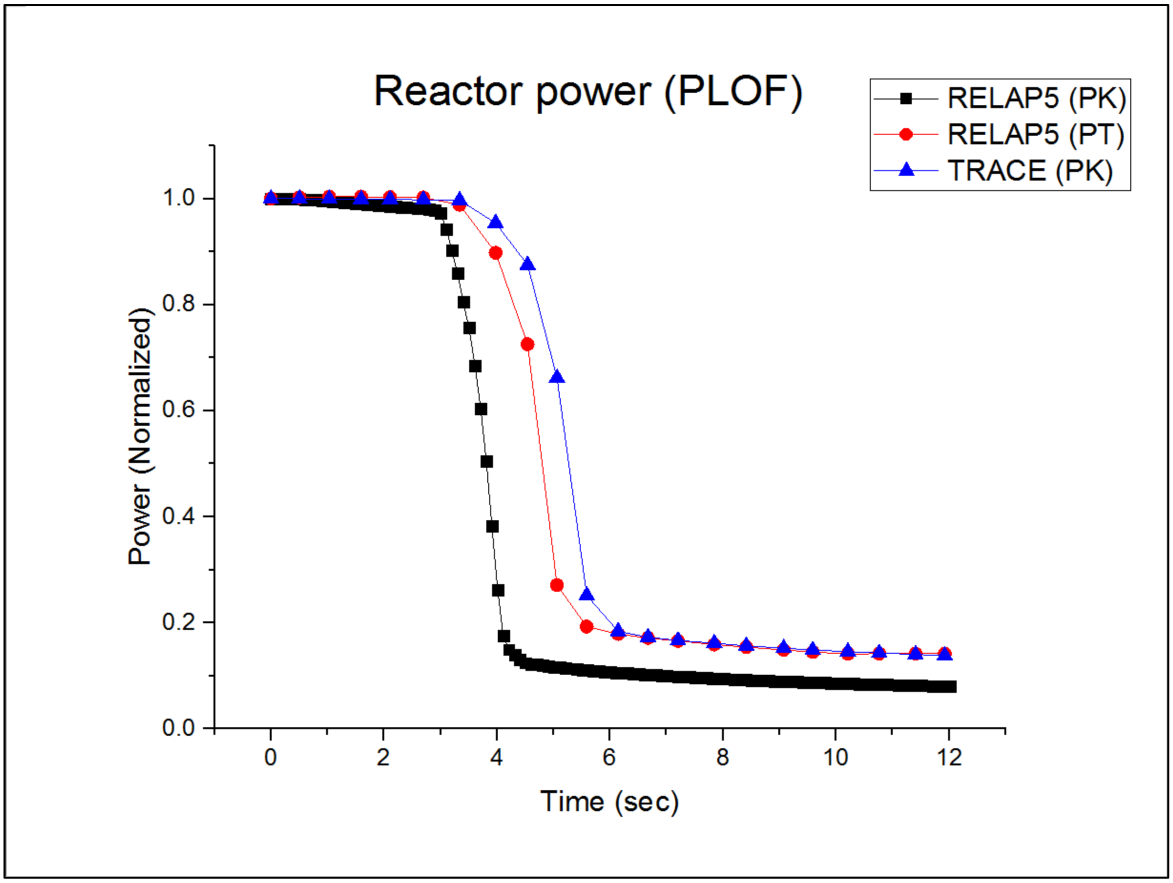


**Figure 38 Reactor Power During PLOF Transient**

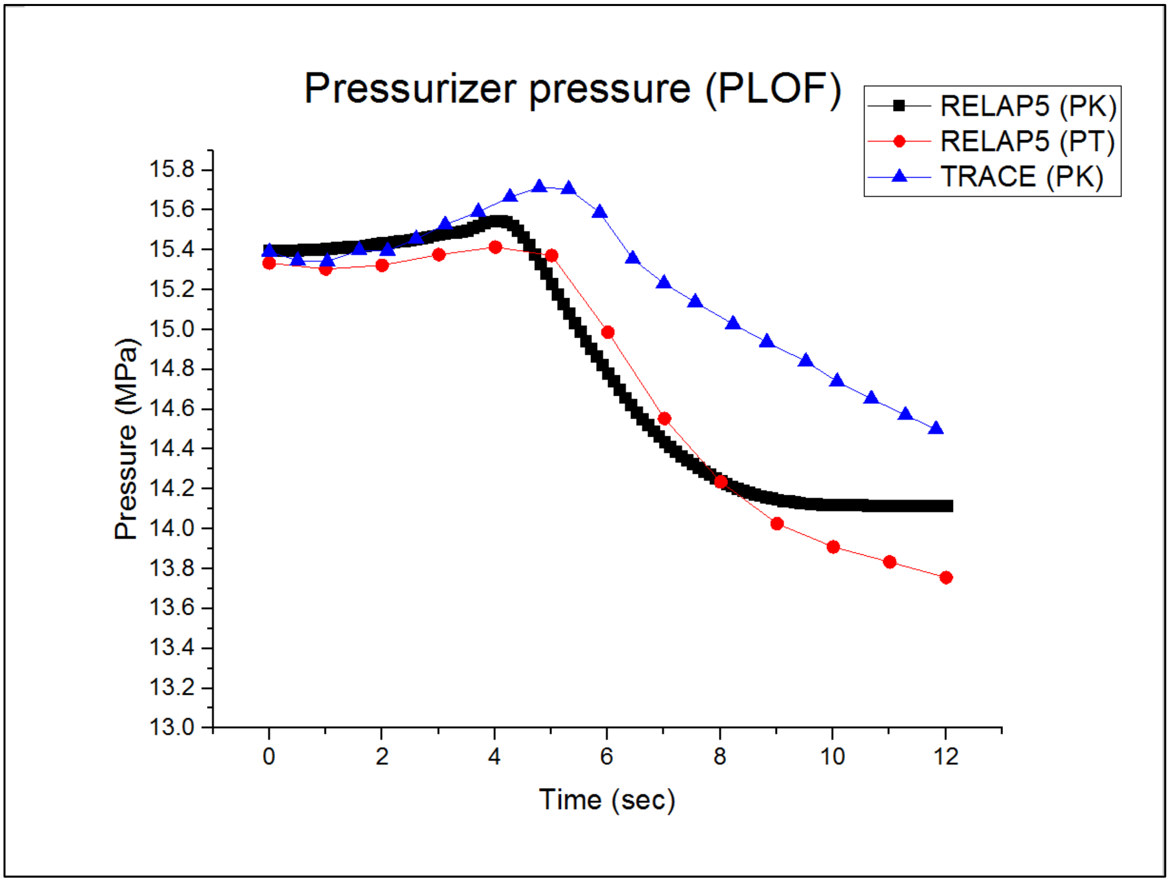


**Figure 39 Core Temperature During PLOF Transient**

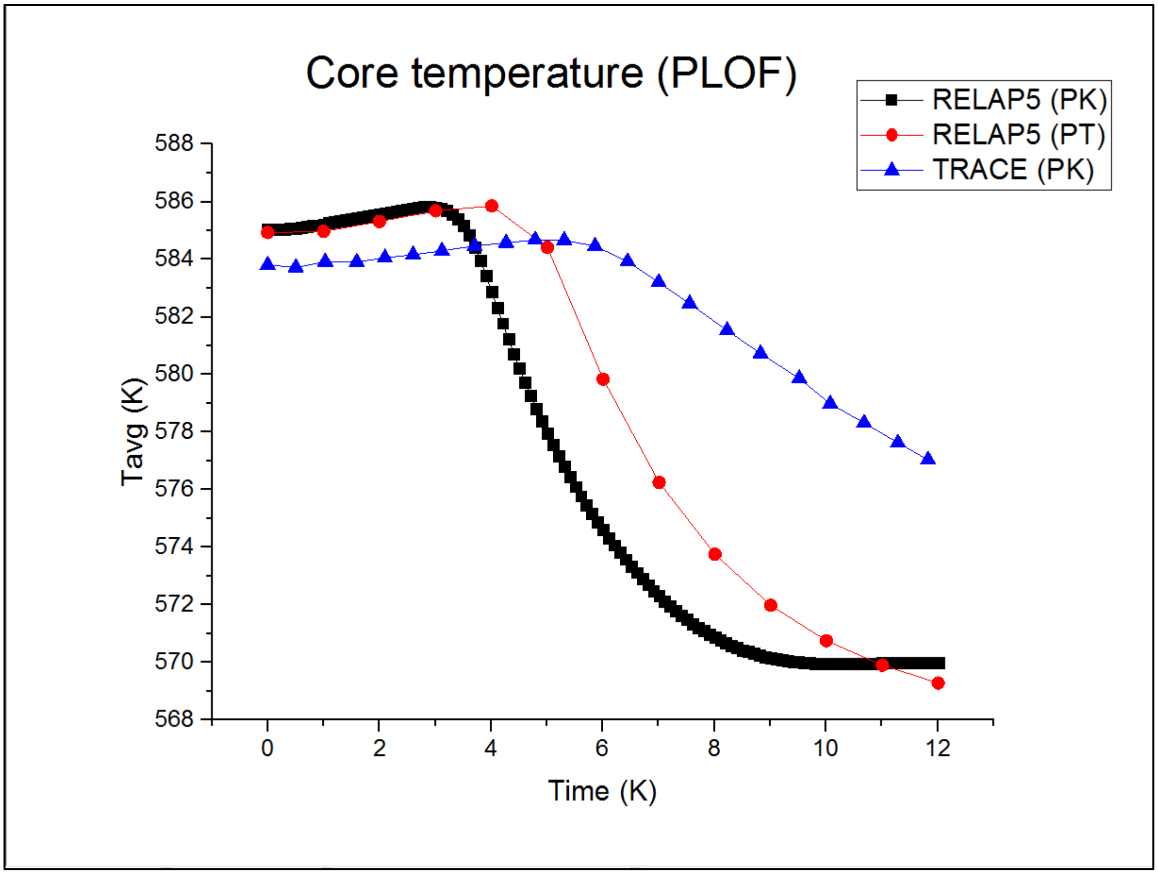


**Figure 40 Pressurizer Pressure During PLOF Transient**

## 4.2 Complete Loss of Flow (CLOF)

The initial conditions of CLOF (UV) and CLOF (UF) events were described in Table 5 and Table 6. There are 3 data sets including the plant data (FSAR), RELAP5 (PK), which reactor power were calculated with Point Kinetics model, and the TRACE data. In the beginning of the analysis, the power was 2775MWt (100%). From this table, it is shown that both the calculations of RELAP5/MOD3.3 model were consistent with the plant data.

CLOF (UV)

At 0 second, the transient started. All the power supply of three RCP motors were cut off. The flow rate started to decrease because of the RCPs velocity slowing down. Further, this also caused the core temperature and the pressurizer pressure increased. Right after the power being cut off, the “low voltage signal” was triggered and the reactor security system initiated the reactor scram. The control rods would drop at 1.5 second because of the signal delay. The power decreased to 10% of rated power at 4.35 second. The core temperature and the pressurizer pressure went down again. The whole analysis period was 12 seconds. Details of the event sequence were described in Table 5.

CLOF (UF)

The transient started at 0 second, which is the time that frequency interference appeared. The frequency interference was supposed to make the power supply frequency decrease in 5 Hz per second. The RCPs decelerated due to the decreasing of the power supply frequency. Then, the core flow rate reduced which made the core temperature and the pressurizer pressure increased. At 0.65 second, the low frequency set point reached and the reactor scram signal was triggered. After 0.6 second (signal delay), control rods started to drop to make the power decrease rapidly. Therefore, the core temperature and the pressurizer pressure decreased. Details of the event sequence were described in Table 6.

**Table 4 Initial Conditions of CLOF Transient**

| Parameters | Plant data | RELAP5 (PK) | TRACE (PK) |
|---|---|---|---|
| Power (MW) | 2775 | 2775 | 2775 |
| Core Temperature (K) | 581.37 | 584.62 | 589.9 |
| Core Flow Rate (kg/sec) | 13775 | 14070 | 13380.23 |
| Steam Flow Rate (kg/sec) | 522.99 | 533.52 | --- |
| PZR Pressure (MPa) | 15.41 | 15.76 | 15.39 |
| Hot Leg Temperature (K) | 599.6 | 599.3 | --- |
| Cold Leg Temperature (K) | 565.2 | 565.2 | --- |

**Table 5 The Sequence of CLOF (UV) Transient**

| EVENT (SEC) | RELAP5 (PK) | TRACE (PK) |
|---|---|---|
| Loop 2 RCP Trip | 0.0 | 0.0 |
| Low Voltage Set point | 0.0 | 0.0 |
| Reactor Scram | 1.5 | 1.5 |

**Table 6 The Sequence of CLOF (UF) Transient**

| EVENT (SEC) | RELAP5 (PK) | TRACE (PK) |
|---|---|---|
| Loop 2 RCP Trip | 0.0 | 0.0 |
| Low Frequency Set point | 0.65 | 0.65 |
| Reactor Scram | 1.25 | 1.25 |

Analysis results

This section describes the analysis data results of the CLOF transient event. As mentioned in previous paragraph, the results of the transient event were also divided into two cases, CLOF (UV) and CLOF (UF). Different from the PLOF analysis, the analysis results of RELAP5 (PT), RELAP5 (PK) and TRACE (PK) would be included in the same figure and discussed together.

Starting from CLOF (UV), Figure 41 shows the core flow rate in CLOF (UV) transient. Comparing to the PLOF transient (Figure 34), the core flow rate in CLOF (UV) decreased more rapidly due to all RCPs tripped in this case. The curve started from the rated value and went down to about 55% of nominal flow rate in 10 second. Like the analysis results of PLOF event, the core flow in TRACE model decreased a little faster than that in RELAP5 model.

Figure 42 shows the power variation of CLOF (UV). In this case, reactor scram was activated by low voltage signal. It means that the reactor scram signal was triggered at the moment of RCP motors losing power. Therefore, the reactor scram all occurred at 1.5 second (as shown in Table 5). The RELAP5 (PK) curve started from nominal power, then went down to about 10% rated power at 3 second. However, it took extra 1 second for RELAP5 (PT) and TRACE (PK) curve and just decreased to about 20% rated power due to different reactivity feedback applied in reactor scram.

Figure 43 is the core temperature during the CLOF (UV) transient. For the RELAP5 (PK) curve, the core temperature started from about 585 K then increased to 587 K, was resulted from the decreasing of flow rate. After reactor scram occurred, the core temperature dropped to about 570 K at 10 second. Because the reactor power of RELAP5 (PT) dropped slower than RELAP5 (PK), there was a higher peak value in RELAP5 (PT) core temperature curve. The temperature curve of RELAP5 (PT) also decreased later. Besides, the core flow in CLOF (UV) reduced more rapidly than in PLOF which caused the temperature climbed up even higher. The peak value of RELAP5 (PT) and RELAP5 (PK) shown in Figure 43 was 590 K and 587 K respectively. In Figure 35, however, was both 585 K. The TRACE curve also had similar tendency that increasing in the beginning, then decreasing after the reactor scram occurred. However, the exact shape was quite different from other two curves. In spite that the core flow rate reduced more quickly in CLOF (UV), the TRACE temperature curve just rose little amount.

Figure 44 shows the pressurizer pressure during the CLOF (UV) transient. The RELAP5 (PK) and the RELAP5 (PT) pressure curves had a similar shape and identical tendency. The curves both started at 15.4 MPa, then climbed up because of the flow temperature increased. Finally, the low voltage signal made the reactor scram occur which caused the temperature and pressure dropped. According to the temperature curve in Figure 43, the higher temperature value in RELAP5 (PT) also resulted in higher pressure curve. Figure 44 shows the peak value of REALP5 (PT) and RELAP5 (PK) were about 16.2 MPa and 15.5 MPa respectively. In the same figure, the TRACE pressure curve, however, was different from its temperature behavior. It climbed up rapidly than other two curves and reached its peak value about 1 second earlier than RELAP5 (PT) curve.

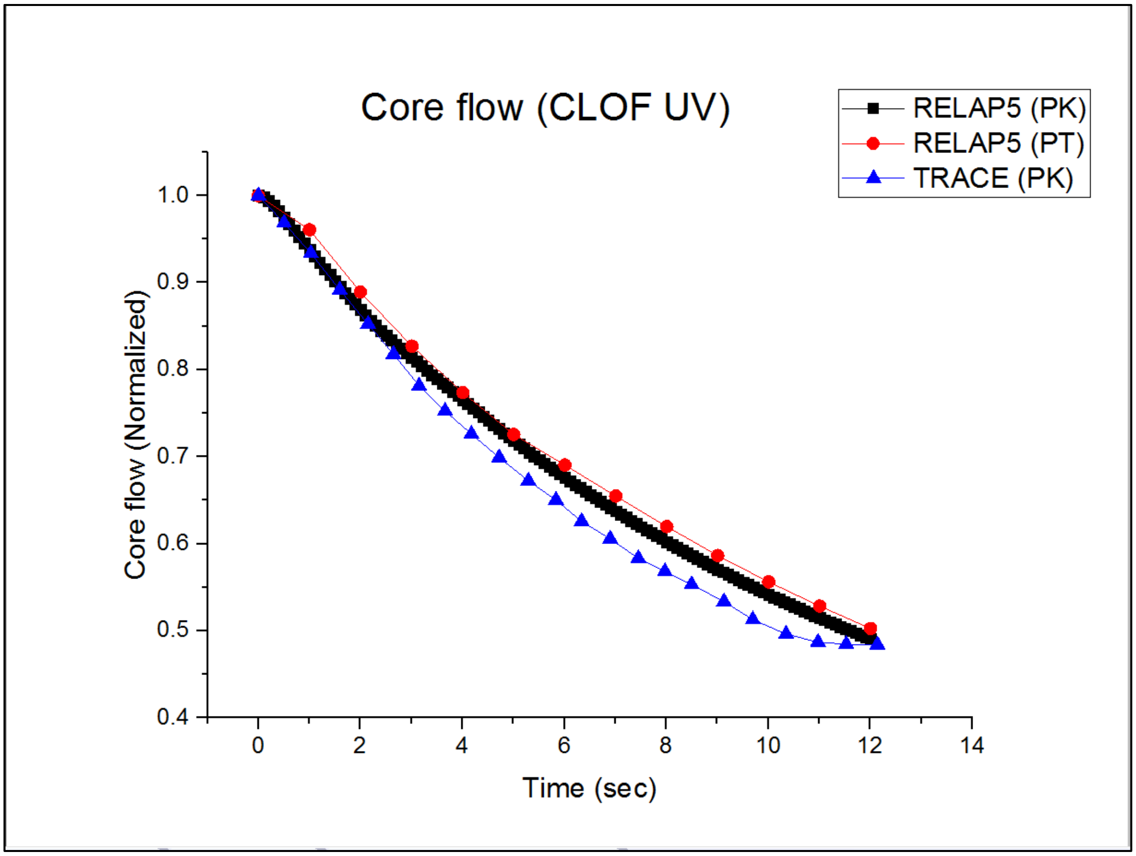


**Figure 41 Core Flow During CLOF (UV) Transient**

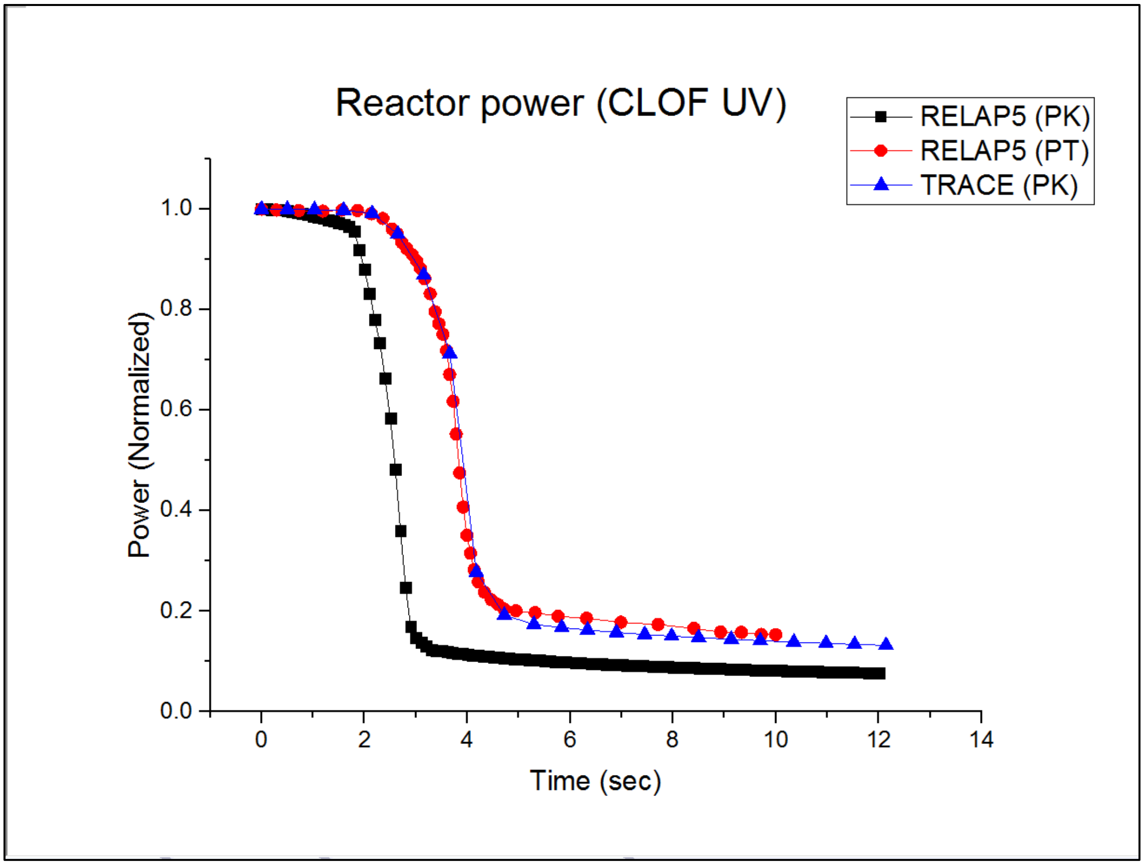


**Figure 42 Reactor Total Power Variation During CLOF (UV) Transient**

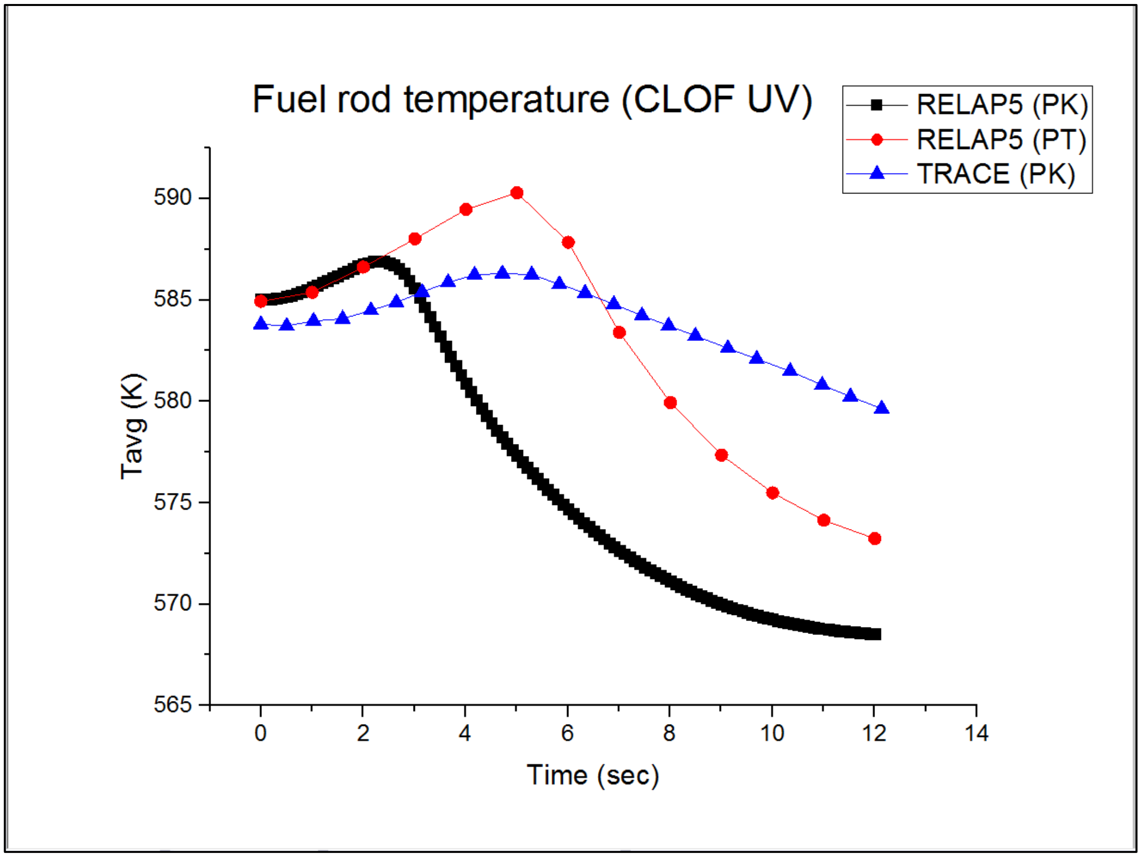


**Figure 43** **Core Temperature Variation of CLOF (UV) Transient**

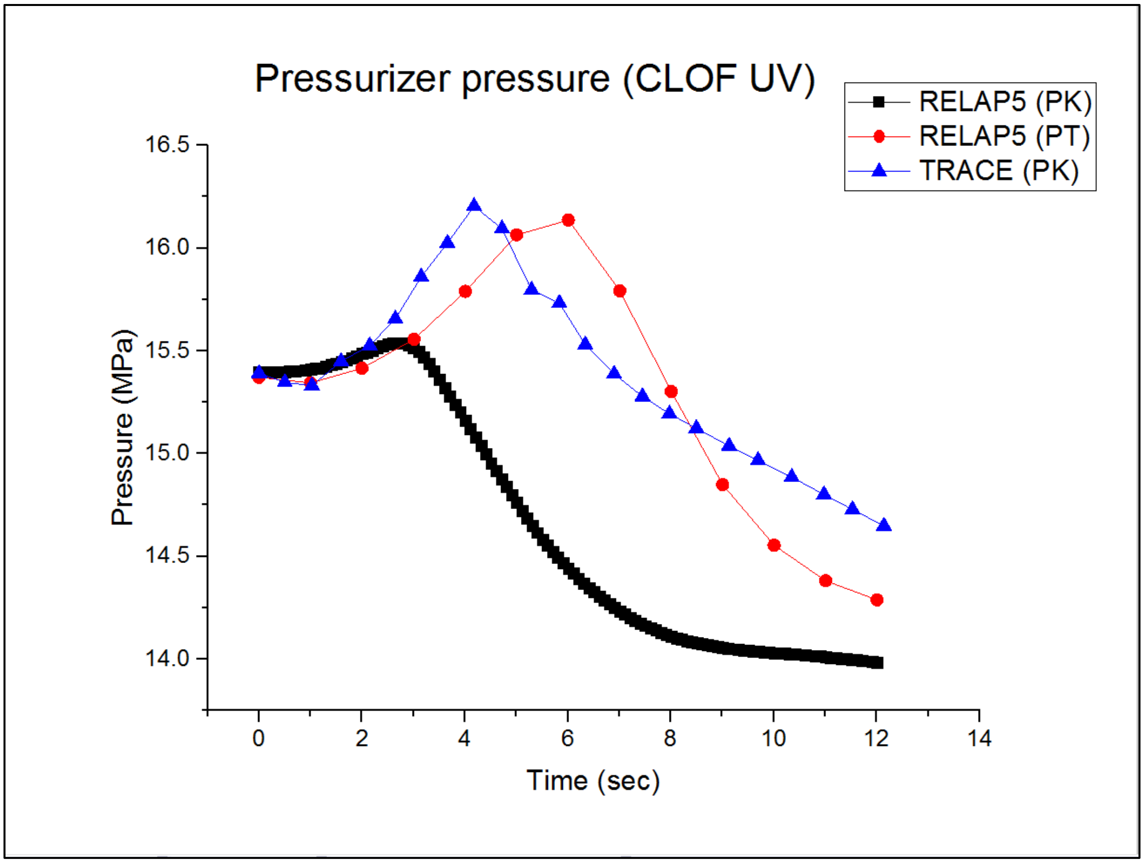


**Figure 44 Pressurizer Pressure of CLOF (UV) Transient**

In the last part, the CLOF under frequency analysis results would be discussed. Figure 45 shows the core flow during CLOF (UF) transient. In this case, the power supply frequency of the RCP motor changed due to the interference. The frequency change caused the RCP motors decelerating more quickly than usual loss of power event. Therefore, the core flow decreased more severely, which could be easily found in Figure 45. The curves of RELAP5 (PT) and RELAP5 (PK) were almost the same. It merely took about 6 seconds to reached 50% nominal flow rate and went down to 10% in about 10 seconds. Comparing to CLOF (UV), the core flow reduced to about 50% at 10 second (Figure 41). Different from the previous cases, the core flow in TRACE decreased slower in CLOF (UF) transient event. The RCP slowing down curves for both RELAP5 and TRACE model in this case were according to the velocity table not the torque-inertia equation. This calculation method with velocity table is the main reason that makes difference that caused the variation.

Figure 46 shows the reactor total power during the CLOF (UF) transient. According to the Table 6, the low frequency signal was triggered at 0.6 and the reactor scram occurred at 1.25 second. In the Figure, it could be found that the power declined slightly before 1.25 second, which also could be found in previous cases. It was because of the effect of core flow reducing that resulted in this phenomenon. The core flow decreasing amplitude of CLOF (UF) was greater than other cases, which made the power decline before the reactor scram more distinct. After the reactor scram occurred, the power curves were all similar to previous cases.

The core temperature was illustrated in Figure 47. The curves of RELAP5 (PT) and RELAP5 (PK) both began at 585 K. Temperature of RELAP5 (PT) went up to about 592 K whereas RELAP5 (PK) was about 586 K. Similarly, the increasing amplitude was influenced by the core flow. The comparison could be illustrated in Figure 49, in which the curve of CLOF (UF) was fully higher than CLOF (UV). As for TRACE temperature curve, the core flow difference also made little influence, which it had been found in CLOF (UV) in Figure 43. The highest temperature during the CLOF (UV) and CLOF (UF) transient were both about 586 K.

Figure 48 explained the pressurizer pressure variation during CLOF (UF) transient. The initial conditions of all the pressure curves were about 15.4 MPa. Different from the core temperature, the pressure between CLOF (UV) and CLOF (UF) did not change so much. The highest value in RELAP5 (PK) was 15.7 MPa, and PELAP5 (PT) and TRACE (PK) were both about 16.2 MPa. The pressure variation between CLOF (UV) and CLOF (UF) was shown in Figure 50. Moreover, pressurizer pressure of TRACE (PK) also increased a little earlier than RELAP5 (PT), which it had already been found in CLOF (UV).

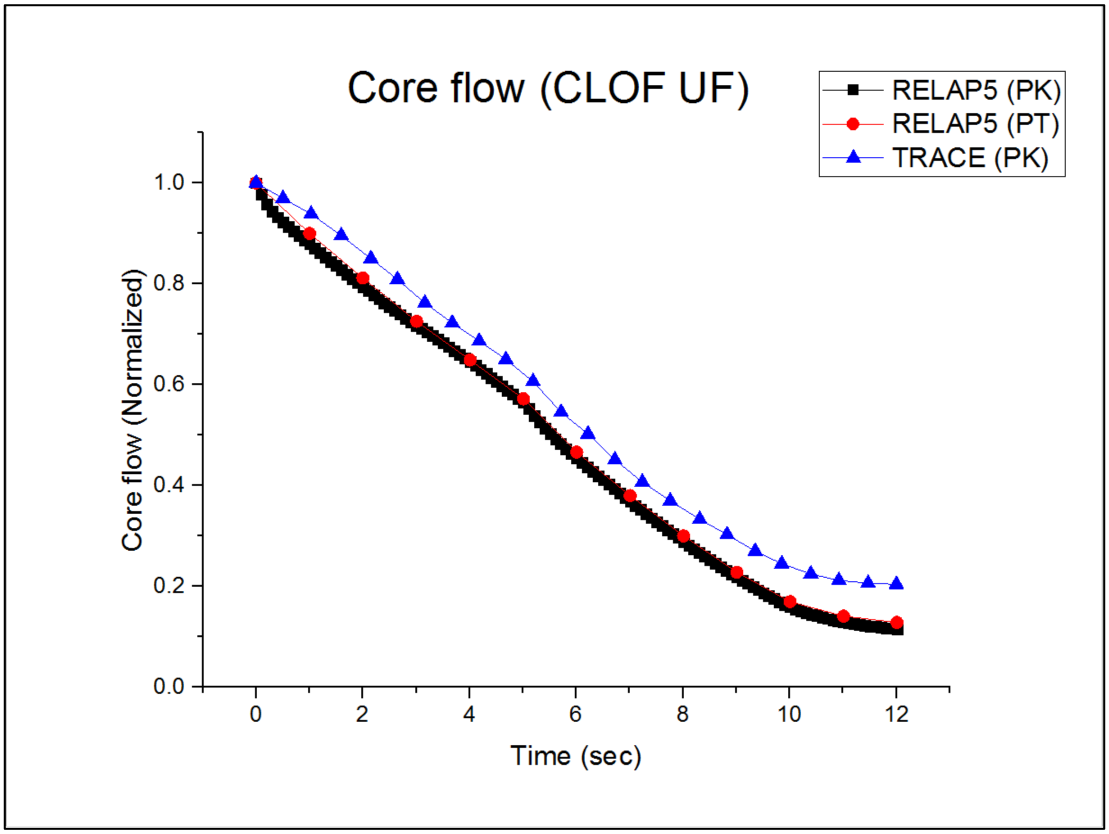


**Figure 45 Core Flow During CLOF (UF) Transient**

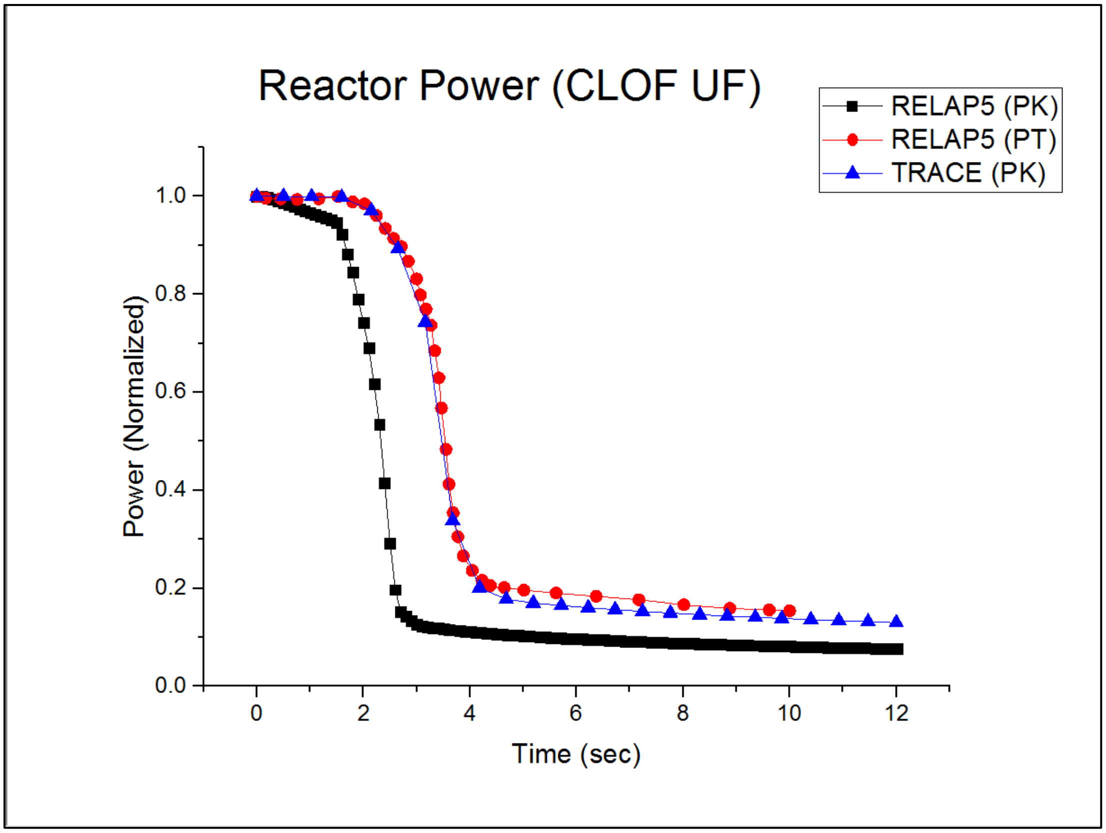


**Figure 46 Reactor Power During CLOF (UF) Transient**

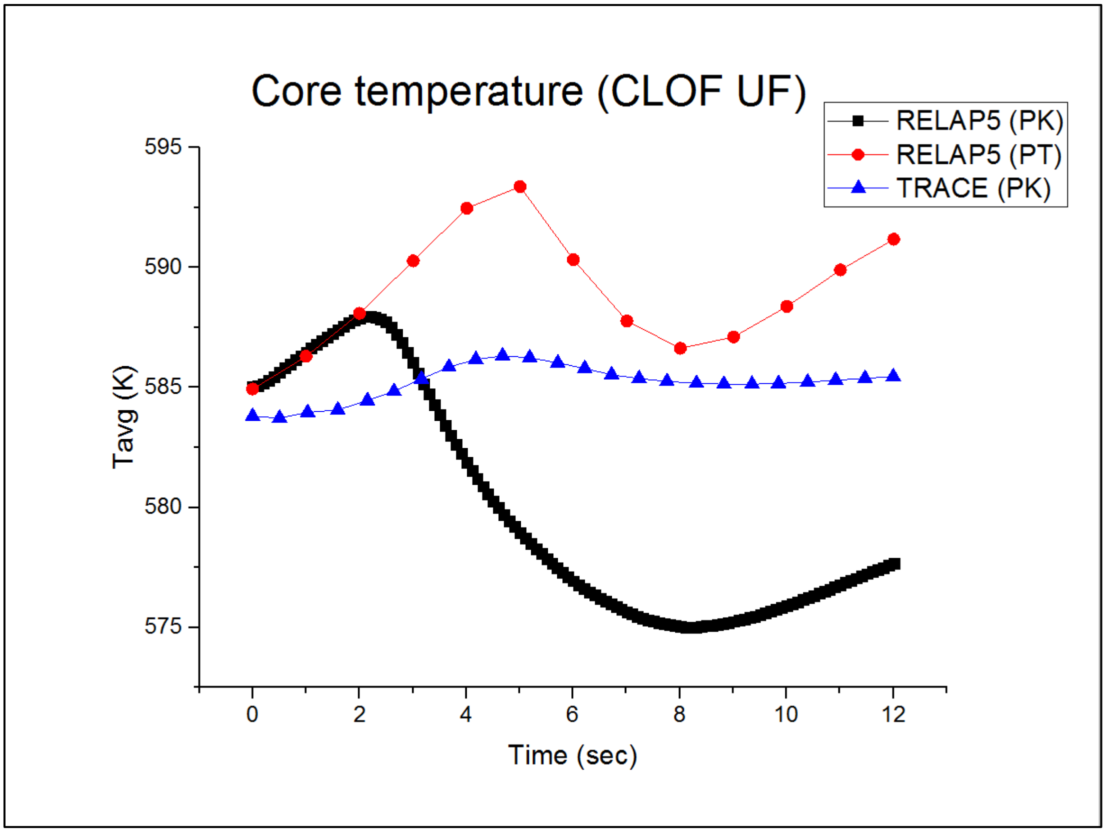


**Figure 47 Core Temperature During CLOF (UF) Transient**

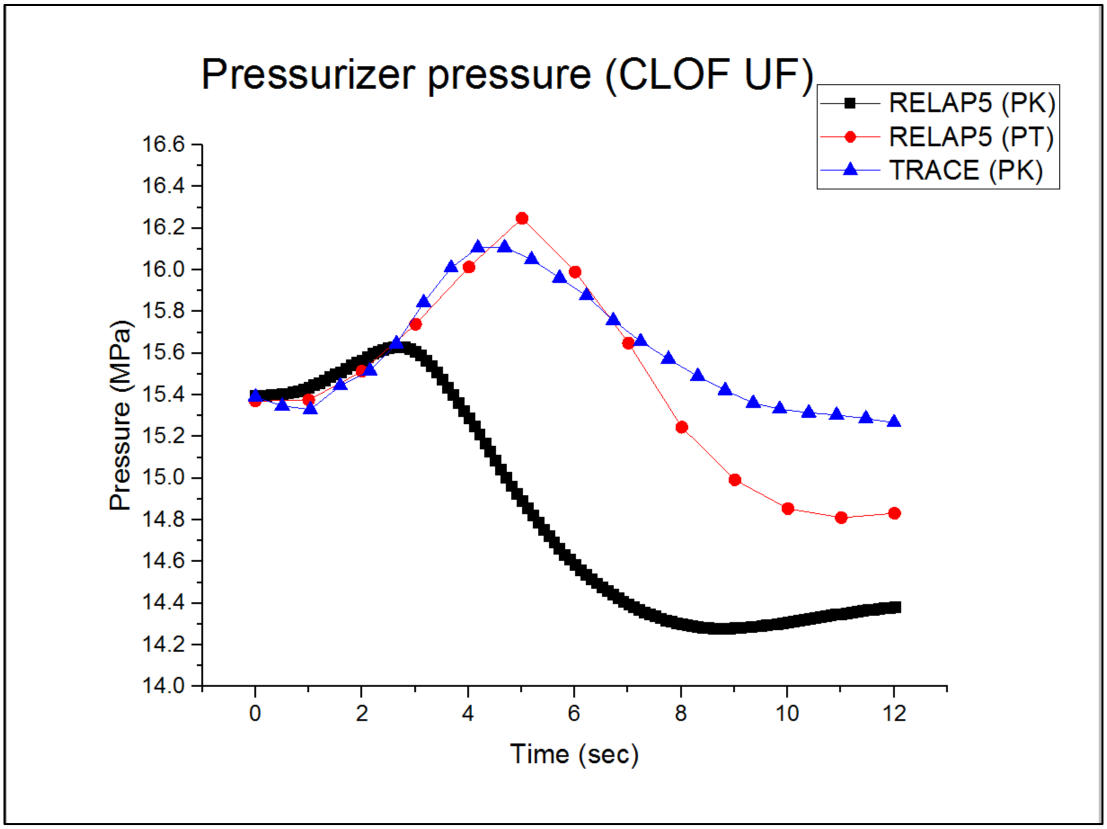


**Figure 48 Pressurizer Pressure of CLOF (UF) Transient**

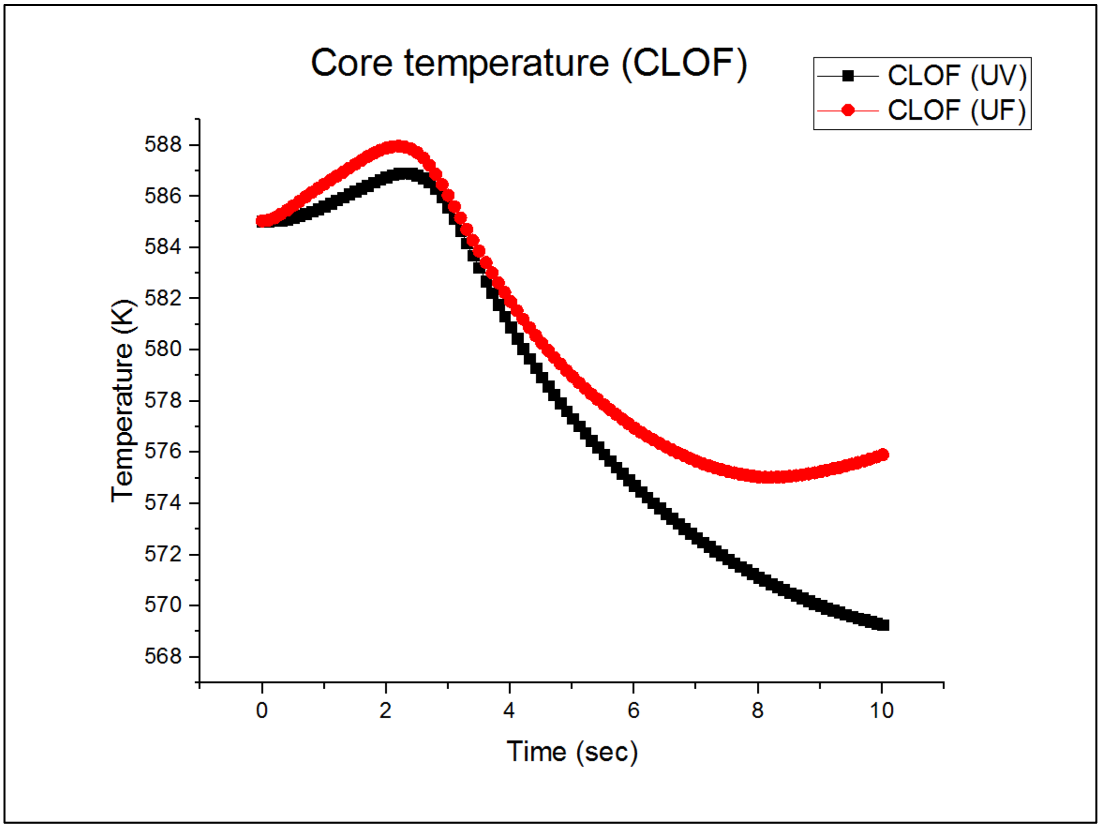


**Figure 49 Core Temperature (CLOF)**

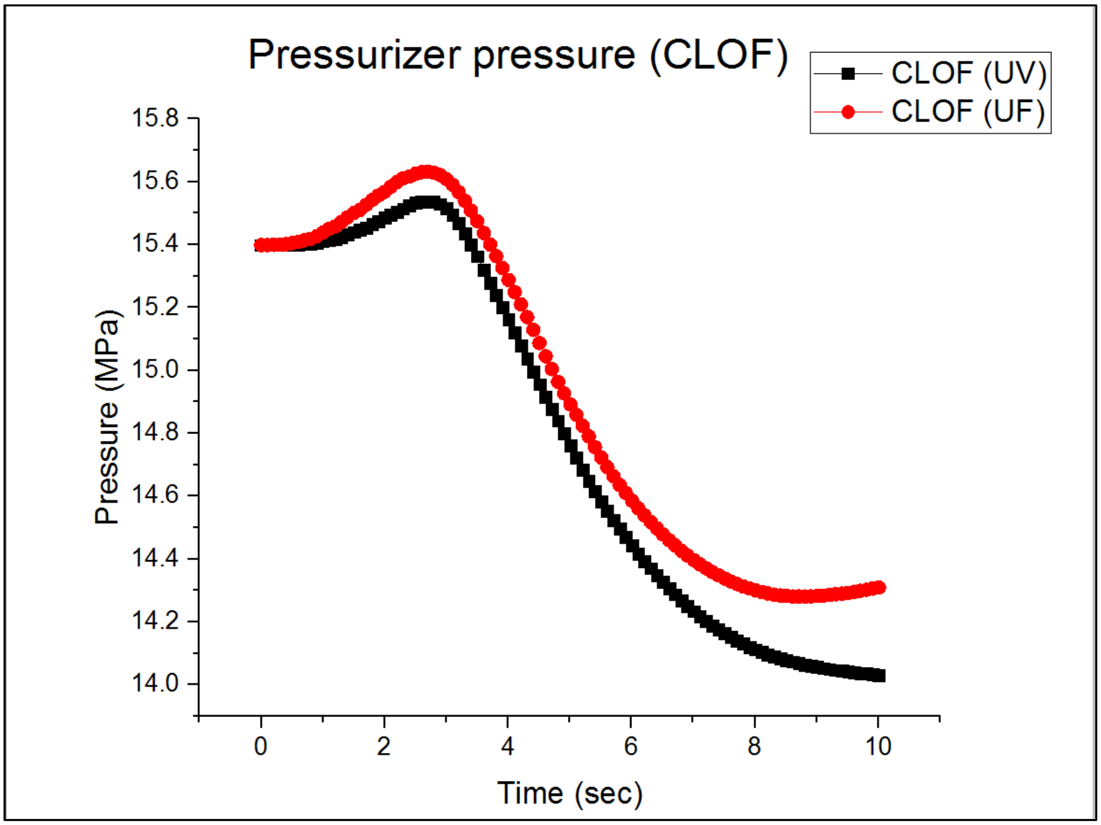


**Figure 50 Pressurizer Pressure (CLOF)**

# 5 CONCLUSIONS

The purpose of this study is to analysis the loss of flow transient (Partial Loss of Flow (PLOF) and Complete Loss of Flow (CLOF)) using Maanshan NPP RELAP5/MOD 3.3 model. By comparing the RELAP5 results with the TRACE predictions, it indicates that the temperature (Tavg) variation in RELAP5 model was more significant than in TRACE model, but the trend is similar and reasonable. Except for the temperature, all of the results were close. It shows that the RELAP5/MOD 3.3 model of Maanshan NPP can be applied correctly to predict important parameter variation and tendency in this transient.

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| NRC FORM 335 (12-2010) NRCMD 3.7 | U.S. NUCLEAR REGULATORY COMMISSION | 1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) |
|---|---|---|
| **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)* | | **NUREG/IA-0483** |
| 2. TITLE AND SUBTITLE **Loss of Flow Analysis of Maanshan Nuclear Power Plant With RELAP5/SNAP** | | 3. DATE REPORT PUBLISHED MONTH: **October** YEAR: **2018** |
| | | 4. FIN OR GRANT NUMBER |
| 5. AUTHOR(S) Chunkuan Shih, Jong-Rong Wang, Shao-Wen Chen, Jian-Ting Chen, Yuh-Ming Ferng, Ting-Yi Wang*, Hsien-Lang Chiu* | | 6. TYPE OF REPORT Technical |
| | | 7. PERIOD COVERED (Inclusive Dates) |

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)


Nuclear and new Energy Education and Research Foundation
Institute of Nuclear Engineering and Science,
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*Department of Nuclear Safety, Taiwan Power Company
242, Section 3, Roosevelt Rd., Zhongzheng District
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9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address.)

Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

K.Tien, NRC Project Manager

11. ABSTRACT (200 words or less)


The objective of this study is to assess the applicability of the RELAP5/MOD3.3 model of Maanshan NPP on Loss of Flow transient. Maanshan NPP was the first three-loop PWR in Taiwan constructed by Westinghouse. For the last few years, the TRACE model of Maanshan NPP was developed and several kinds of transients were performed. Recently, the RELAP5/MOD3.3 code is another important development priority for our group. In 2015, the RELAP5/MOD3.3 model of Maanshan NPP was developed with SNAP interface. To expand the applicability of Maanshan RELAP5/MOD3.3 model, the loss of flow transient was analyzed in this research. Hence, two analyses- Partial Loss of Flow (PLOF) and Complete Loss of Flow (CLOF) were performed in this research. From the analysis results, including power, core flow, pressurizer pressure and cladding average temperature, were similar to the results of TRACE model. It indicates that Maanshan RELAP5/MOD3.3 model can predict this transient well.


| 12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.) | 13. AVAILABILITY STATEMENT |
|---|---|
| Feedwater Pump Trip (FWPT) | unlimited |
| Partial Loss of Flow (PLOF) | 14. SECURITY CLASSIFICATION |
| RELAP5/SNAP/ | *(This Page)* unclassified |
| Complete Loss of Flow (CLOF) | *(This Report)* unclassified |
| Turbine Control Valve Closure (TCVC) | 15. NUMBER OF PAGES |
| Maanshan Nuclear Power Plant (NPP) | 16. PRICE |

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WASHINGTON, DC 20555-0001

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NUREG/IA-0483 Loss of Flow Analysis of Maanshan Nuclear Power Plant With RELAP5/SNAP October 2018