International Agreement Report

RELAP5/MOD3.3 Model Assessment and Hypothetical Accident Analysis of Kuosheng Nuclear Power Plant with SNAP Interface

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ABSTRACT

After the measurement uncertainty recapture power uprates, Kuosheng nuclear power plant (NPP) has been uprated the power from 2894 MWt to 2943 MWt. For these power upgrade, several analysis codes were applied to assess the safety of Kuosheng Nuclear Power Plant. In our group, there were a lot of effort on thermal hydraulic code, TRACE, had been done before. However, to enhance the reliability and confidence of these transient analyses, thermal hydraulic code, RELAP5/MOD3.3 will be applied in the future. The main work of this research is to establish a RELAP5/MOD3.3 model of Kuosheng NPP with SNAP interface. Model establishment of RELAP5 code is referred to the Final Safety Analysis Report (FSAR), training documents, and TRACE model which has been developed and verified before. After completing the model establishment, three startup test scenarios would be applied to the RELAP5 model. With comparing the startup test data and TRACE model analysis results, the applicability of RELAP5 model would be assessed.

Recently, Taiwan Power Company is concerned in stretch power uprated plan and uprates the power to 3030 MWt. Before the stretch power uprates, several transient analysis should be done for ensuring that the power plant could maintain stability in higher power operating conditions. In this research, three overpressurization transient scenario including main steam isolation valves closure, turbine trip with bypass failure and load rejection with bypass failure would be performed. These analysis results would be compared with the TRACE model results which had been performed before.

FOREWORD

Taiwan and the United States have signed an agreement on CAMP (Code Applications and Maintenance Program). INER (Institute of Nuclear Energy Research, Atomic Energy Council, R.O.C.) is the organization in Taiwan responsible for the application of TRACE and RELAP in thermal hydraulic safety analysis, for recording user's experiences of it, and providing suggestions for its development. To meet this responsibility, the RELAP5/MOD3.3 model of Kuosheng nuclear power plant has been built. The startup test data and TRACE model analysis results were used to compare with the results of RELAP5/MOD3.3 model.

CONTENTS

FIGURES

TABLES

EXECUTIVE SUMMARY

RELAP5/MOD3.3Patch04 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 nonequilibrium 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 Kuosheng NPP was developed with SNAP interface.

Kuosheng NPP is located on the northern coast of Taiwan. Its nuclear steam supply system is a type of BWR/6 designed and built by General Electric on a twin unit concept. Each unit includes two loops of recirculation piping and four main steam lines, with the thermal rated power of 2894MWt. After the project of MUR for Kuosheng NPP, the operating power is 2943 MWt. Unit 1 started SPU from Cycle 24 and Unit 2 started SPU from Cycle 23. The operating power will be 3030 MWt. In this research, a RELAP5/MOD3.3 model of Kuosheng NPP is developed. Further, the model in this research is built up with the SNAP interface. The startup test data and TRACE model analysis results were used to compare with the results of RELAP5/MOD3.3 model for overpressurization transients and feedwater pump trip transient. The predictions of RELAP5/MOD3.3 were consistent with the data of startup test and TRACE roughly. It indicates that there is a respectable accuracy in Kuosheng NPP RELAP5/MOD3.3 model

ABBREVIATIONS

BWR	Boiling-Water Reactor
BPV	Bypass valve
FWPT	Feedwater pump(s) Trip
INER	Institute of Nuclear Energy Research Atomic Energy Council, R.O.C.
kg	kilogram(s)
kW	kilowatt(s)
LRWB	Load Rejection with Bypass
LRNB	Load Rejection with Bypass Failure
MPa	Megapascal(s)
MSIVC	Main Steam Isolation Valves Closure
MUR	Measurement Uncertainty Recapture
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
Pa	Pascal(s)
psi	pounds per square inch
SNAP	Symbolic Nuclear Analysis Program
SPU	Stretch Power Uprated
SRV	Safety/Relief Valves
TBV	Turbine Bypass Valve
TCVC	Turbine Control Valves Closure
TRACE	TRAC/RELAP Advanced Computational Engine
TSVC	Turbine Stop Valves Closure
TTNB	Turbine Trip with Bypass Failure
W	Watt(s)
US	United States

1. INTRODUCTION

Kuosheng NPP is located on the northern coast of Taiwan. Its nuclear steam supply system is a type of BWR/6 designed and built by General Electric on a twin unit concept. Each unit includes two loops of recirculation piping and four main steam lines, with the thermal rated power of 2894MWt. After the project of MUR for Kuosheng NPP, the operating power is 2943 MWt [1, 2]. Unit 1 started SPU from Cycle 24 and Unit 2 started SPU from Cycle 23. The operating power will be 3030 MWt. To uprate the power, the assessments of power plant transient should be analyzed. In the past, a TRACE model of Kuosheng NPP has been developed. The analysis and simulation of the TRACE model for the startup tests and hypothetical transients has been done. In this research, a RELAP5/MOD3.3 model of Kuosheng NPP is developed referred to FSAR [1], training documents [2], RELAP5 3D and TRACE model which had been developed before [3]. Further, the model in this research is built up with the SNAP interface rather than the text files. The visible thermal hydraulic components and control blocks in the SNAP interface help users build the model more efficiently and concretely.

RELAP5/MOD3.3Patch04 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 nonequilibrium 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 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 Kuosheng NPP was developed with SNAP interface.

To ensure the applicability of this model, three startup tests including feedwater pumps trip (FWPT), load rejection with bypass (LRWB) 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 Kuosheng 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

Overview of the RELAP5/MOD 3.3 Model

Different from typical thermal hydraulic model establishment, the RELAP model in this research was developed in the SNAP interface. With SNAP interface, the components and control blocks are visible as shown in Figure 1. As a result, the users could set up component parameters and nodding diagram at same time. In this RELAP model, situation of NSSS in transient was mainly concerned. Hence, the turbines and feedwater pumps of Kuosheng NPP were assumed to be boundaries which were simulated by components Time dependent volume (TMDVOL). Four main steam pipe lines, which were consistent with the configuration of Kuosheng NPP, were developed. On these pipe lines, three important valves including main steam line isolation valves, turbine stop valves and turbine control valves are developed. Further, there are totally 16 safety/relief valves connected on the main steam pipe lines. All the opening and closing setpoints are also developed according to the arrangement of Kuosheng NPP.

The reactor vessel is developed by several kinds of components including Branch, Pipe, Single junction and single volume. Four pipes which are developed to simulate fuel assemblies are connected to heat structures inside the reactor vessel. Source data of these heat structures are referred to the total reactor power. To simplified the model and save the computational time, Point Kinetics and Separable feedback types are chosen for the reactor kinetics. Two recirculation loops and recirculation pumps are set up according to the configuration of the power plant. Further, there are two control valves developed on two recirculation loops respectively to adjust the recirculation flow rate. 20 jet pumps in the NPP are merged into two jet pump components to save the computational time. More details of the components properties and the setting of control systems are described below.
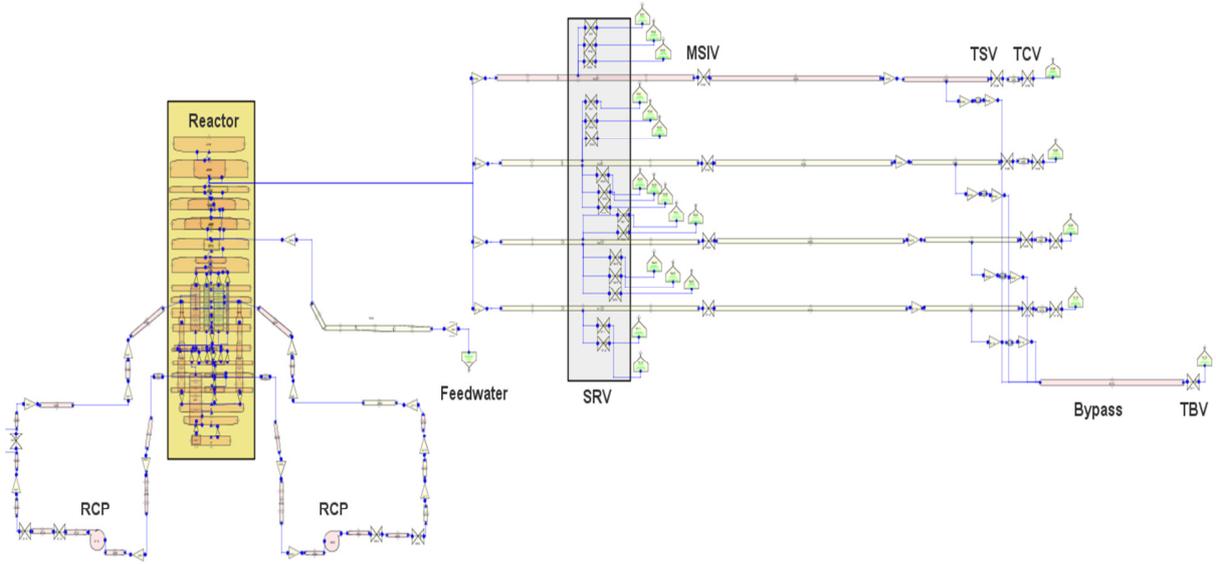
Figure 1   Overview of the Kuosheng NPP RELAP5/MOD3.3 Model

<u>Reactor Kinetics</u>

The Point kinetic and SEPARABL feedback types are developed in this transient. Further, GAMMA-AC fission product decay type is chosen to calculate the decay heat. The total reactor power was set according to different transient scenario respectively. Initial reactivity and value of Beta over Lambda were 0 and 125 respectively, which were referred to the manual of RELAP5. In addition, with some unit converting, density and Doppler reactivity feedback table were referred to the TRACE and RETRAN model which had been verified before. Details of the reactor kinetics are shown in Figure 2 to Figure 4.

To simplify the control model and reduce the computer time, the scram control systems were developed according to each case respectively. However, as shown in Figure 5, the scram reactivity feedback was written into Table 900, which was also referred to the TRACE model verified before, to control the power after the reactor scram. As the table trip set in Table 900 was initiated by the power control system in each case, the negative reactivity feedback, which was increasing with time, would be concerned in the power calculation.

Figure 2 Parameters of Reactor Kinetics in RELAP5 Model
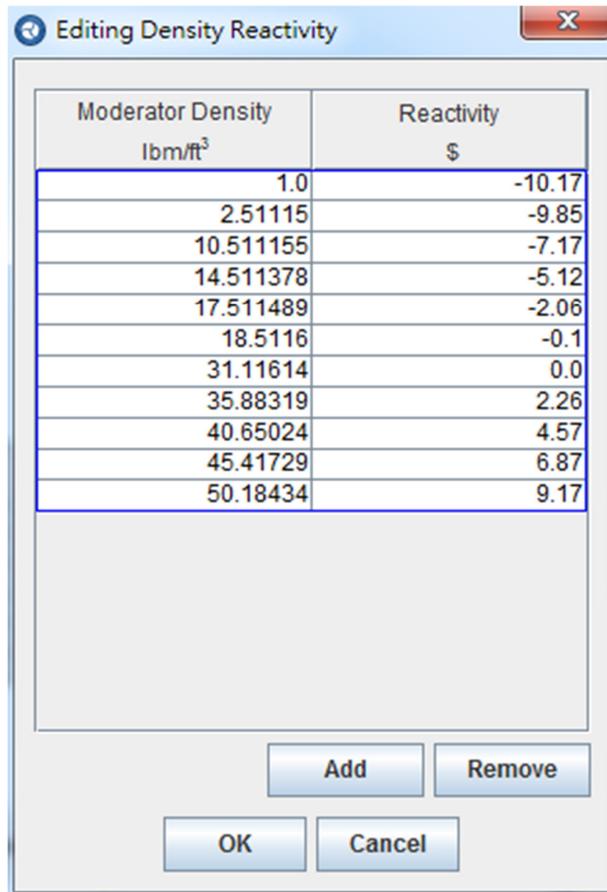
Figure 3　Density Reactivity Feedback Table of Reactor Kinetics
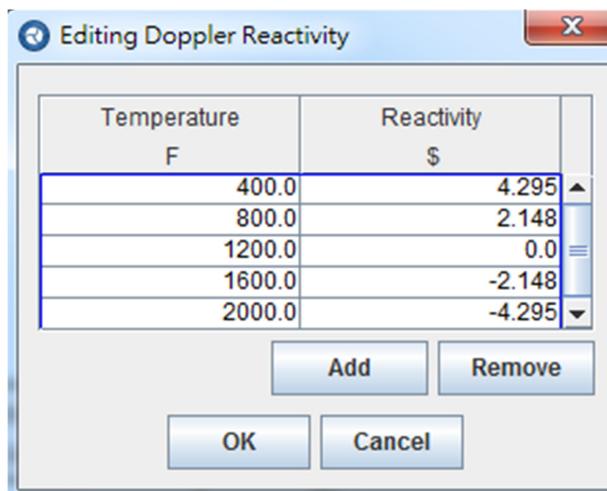
Figure 4　Doppler Reactivity Table of Reactor Kinetics
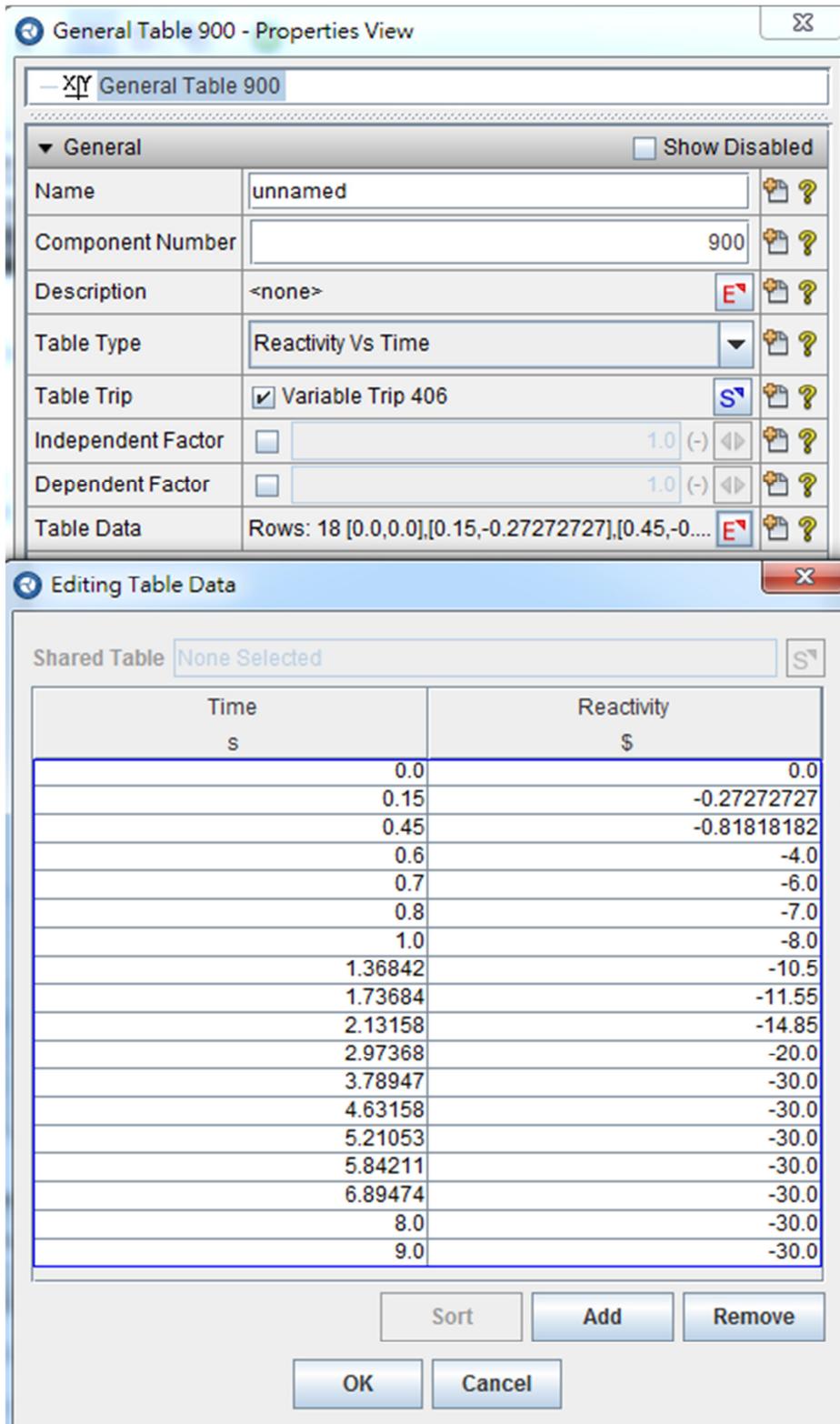
Figure 5   Scram Reactivity Feedback Table

Main Steam Pipe Lines

According to the configuration of Kuosheng NPP, there are four main steam pipe lines developed in this RELAP5 model. On the main steam pipe lines, there are one main steam line isolation valve, one turbine stop valve and one turbine control valve in each pipe lines respectively. Further, 16 safety/relief valves are also set up on the main steam pipe lines. At the ends of the pipe lines, Time Dependent Volume components are set to simulate the turbines, which are assumed to be the boundaries of this model. In addition, the turbine bypass pipe line and valve are also developed in this model as shown in Figure 1. All the geometry and thermal hydraulic conditions of these components mentioned above are referred to the FSAR, training materials and configuration of Kuosheng NPP. Details of the properties are shown in Figure 6 to Figure 10.

Figure 6   Properties of Main Steam Isolation Valves
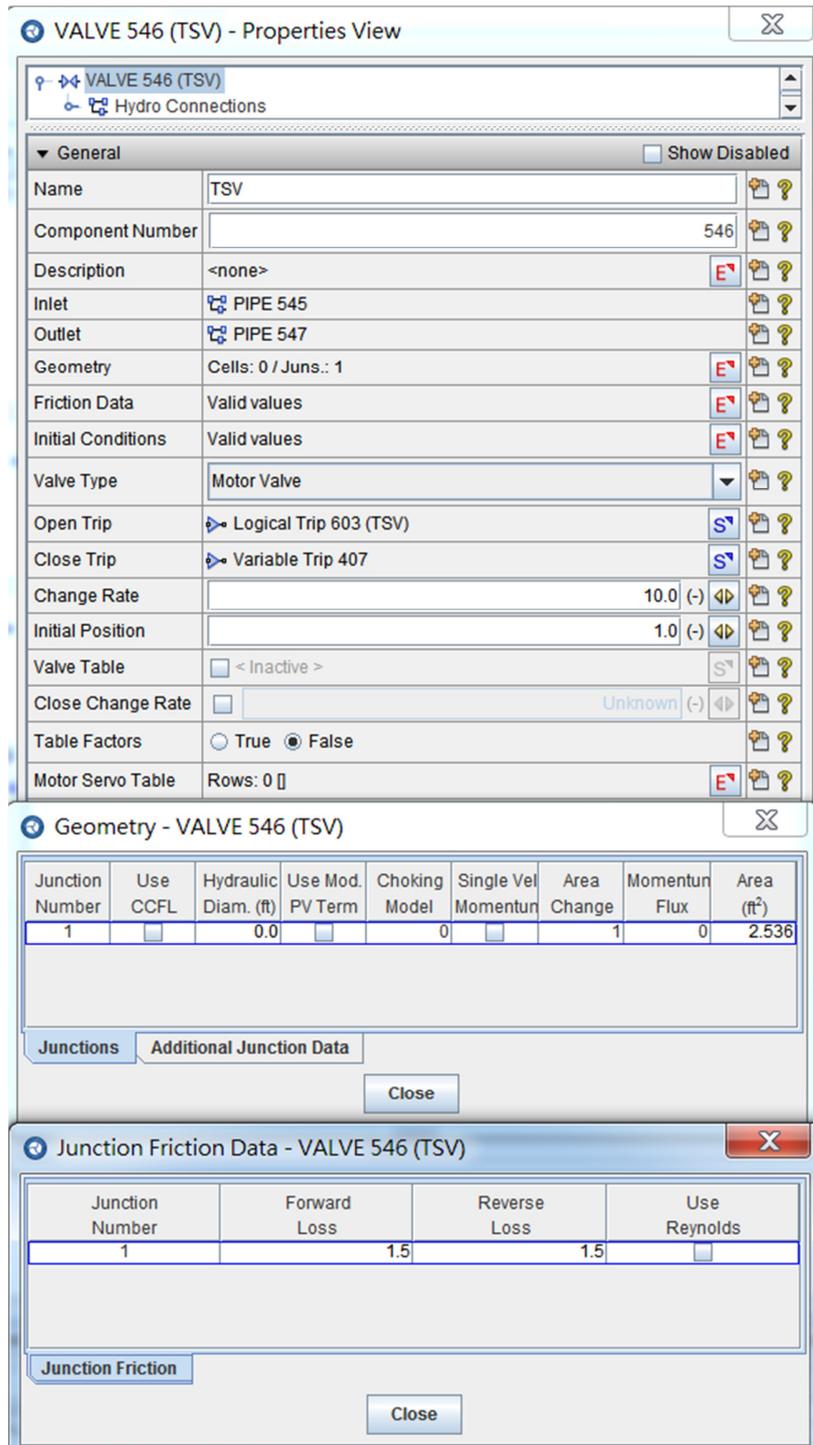
Figure 7   Properties of Turbine Stop Valves
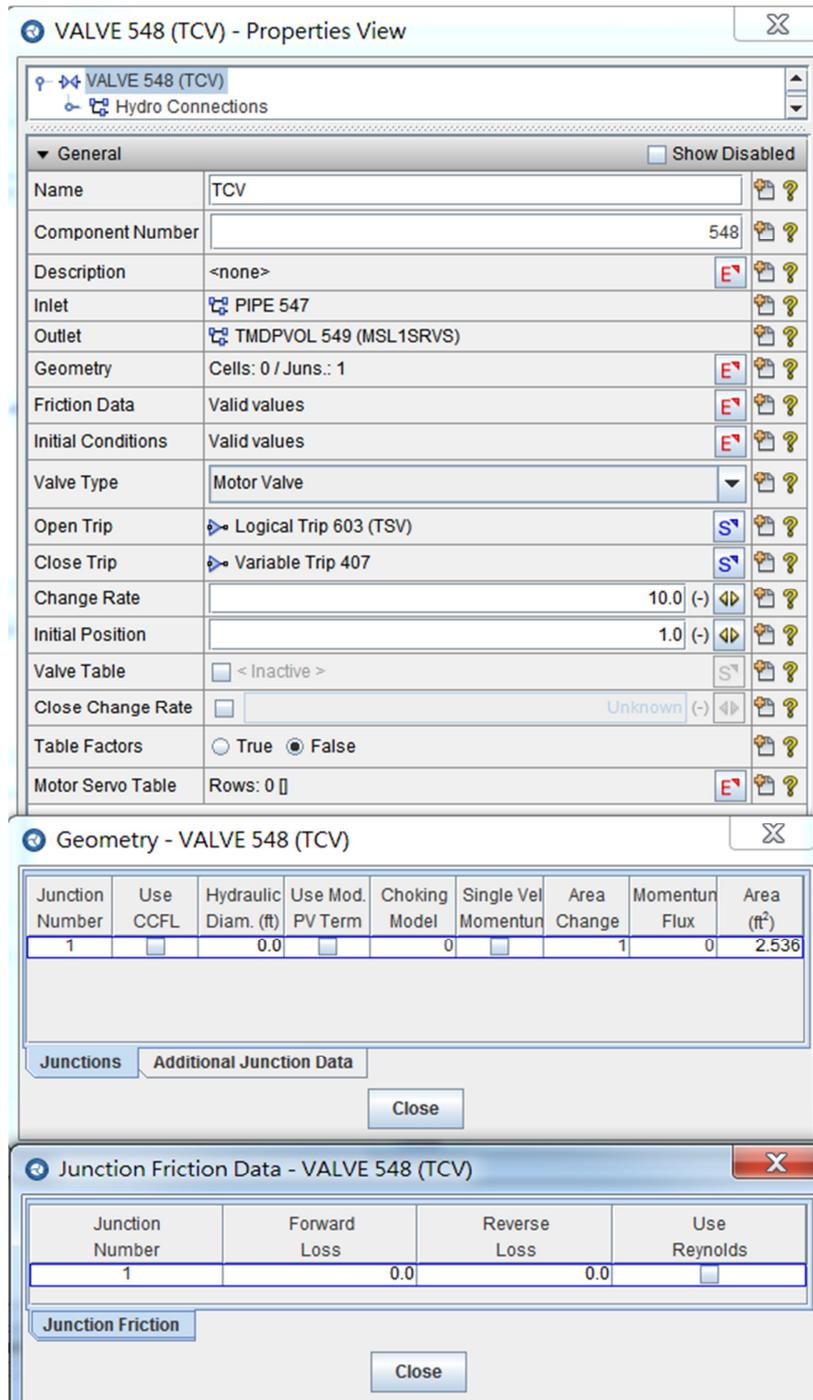
Figure 8   Properties of Turbine Control Valves
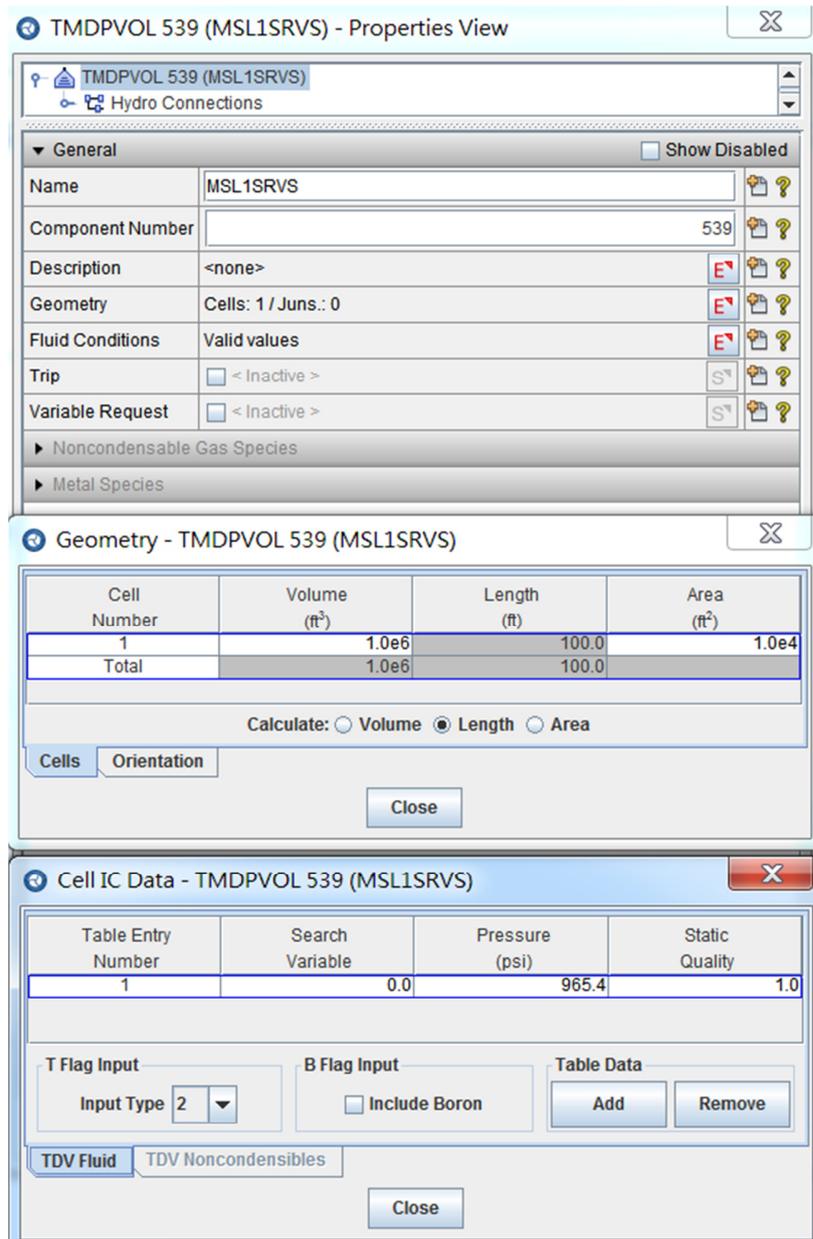
Figure 9   Time Dependent Volume Properties of the Main Steam Pipe Lines
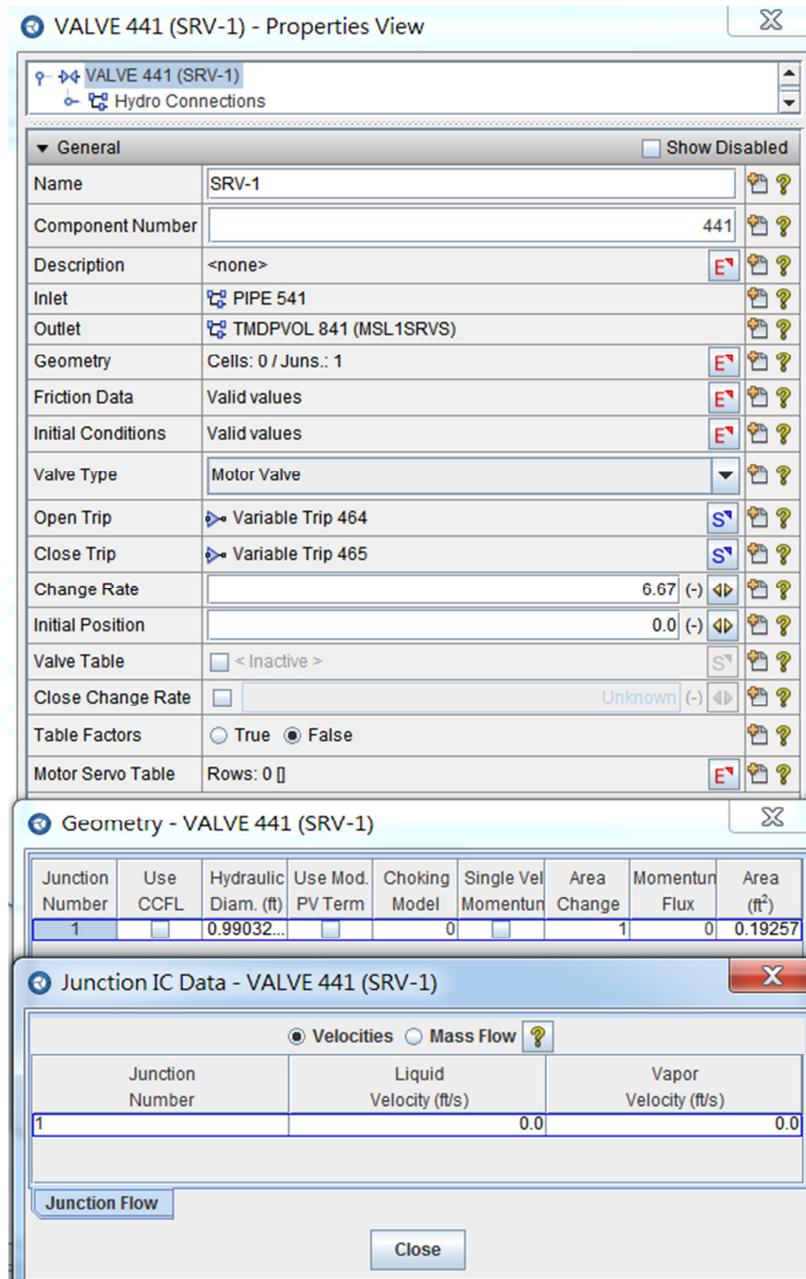
Figure 10   Properties of Safety/Relief Valves

<u>Pumps</u>

Typically, three types of pumps including feedwater pumps, recirculation pumps and jet pumps are built to supply core flow. Because the feedwater inlet assumed to be the boundary in this transient, three feedwater pumps in Kuosheng NPP were developed with the component Time Dependent Volume. After these components, there are three Time Dependent Junction components which are developed to control the feedwater flow rate. Properties of the Time Dependent Volume and Time Dependent Junction are shown in Figure 11 and Figure 12. There are two recirculation loops in Kuosheng NPP. As a result, two recirculation loops and flow control valves are developed in this RELAP5 model. Generally, rotation speed of the recirculation pumps would not be changed. The recirculation flow rates are varied by controlling the area of the recirculation control valves. Details of the recirculation loop setting are shown in Figure 13 and Figure 14. There are 20 jet pumps in the reactor vessel of Kuosheng NPP. However, to save the computational time, these jet pumps are merged into two jet pumps. Details of the jet pump properties are shown in Figure 15.
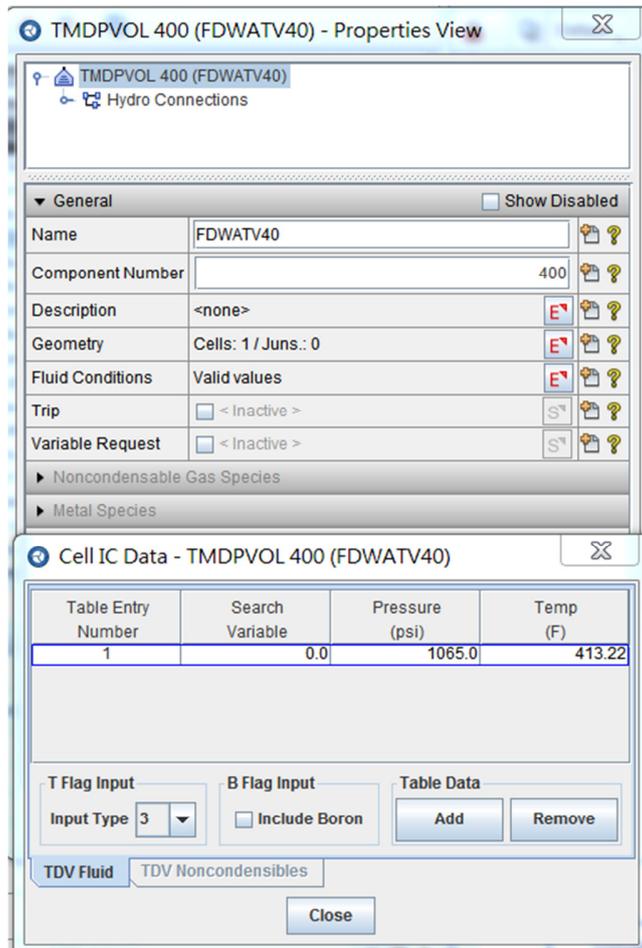
Figure 11   Details of Feedwater Time Dependent Volume
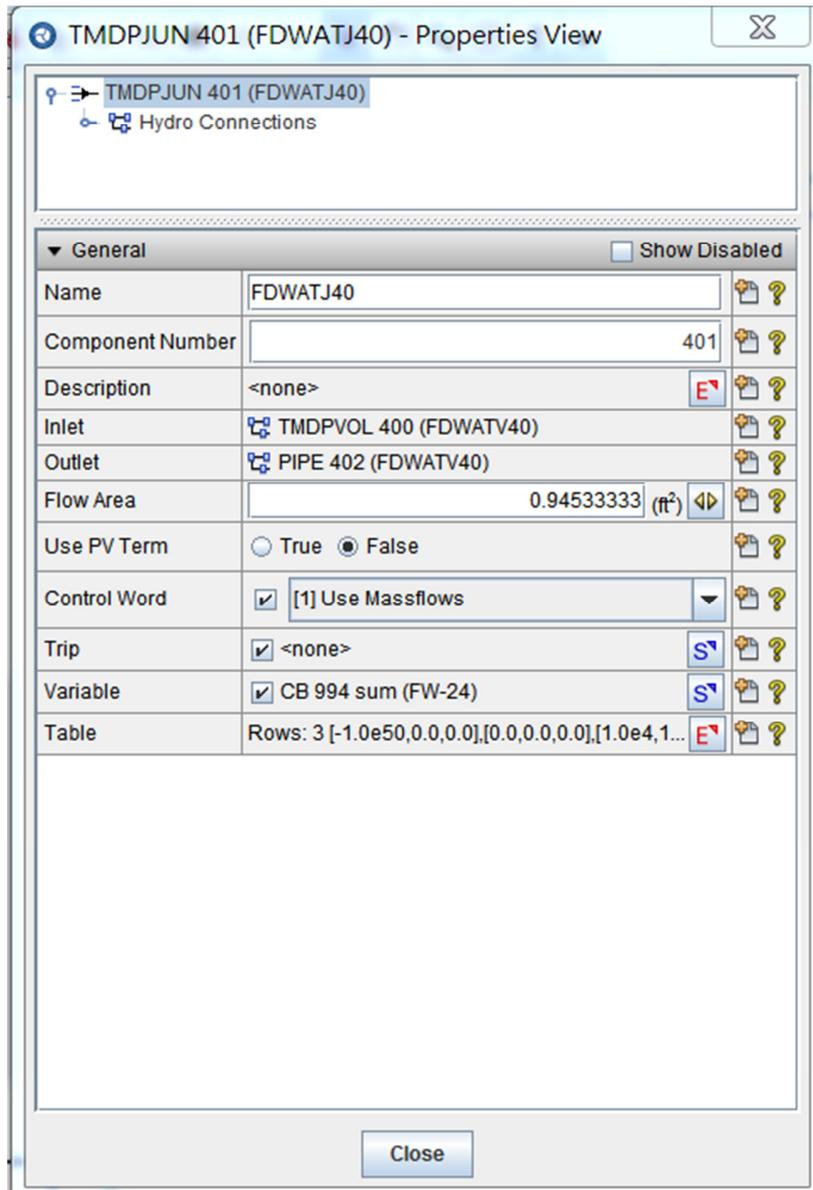
Figure 12   Details of Time Dependent Junction of Feedwater Pumps

PUMP 310 (CPUMPA31) - Properties View

- PUMP 310 (CPUMPA31)
  - Hydro Connections
  - Heatstructure Connections

▼ General   ☐ Show Disabled

Name	CPUMPA31
Component Number	310
Description	<none>
Geometry	Cells: 1 / Juns.: 2
Friction Data	Valid values
Initial Conditions	Valid values

▼ [CCC0301] Pump Options

Pump Indicator	Entered with this component
Multiplier Index	Active multiplier, entered in this component
Difference Table	Two-phase included in current pump
Motor Torque	No Torque Table Data
Pump Velocity	Use torque-inertia equation
Motor Trip	⊳ Variable Trip 405
Reverse Indicator	[1] Allow Reverse

▼ [CCC0302-4] Pump Description

Rated Velocity	1785.0	(rpm)
Ratio Velocity	1.0	(-)
Rated Flow	3.25e4	(gpm)
Rated Head	830.0	(ft)
Rated Torque	1.7185e4	(lbf*ft)
Rated Density	53.481666	(lbm/ft^3)
Moment of inertia	1.56e4	(lb*ft^2)
Rated Motor Torque	0.0	(lbf*ft)
TF2 Torque	0.0	(lbf*ft)
TF0 Torque	387.0	(lbf*ft)
TF1 Torque	0.0	(lbf*ft)
TF3 Torque	0.0	(lbf*ft)

- ▶ [CCC0308] Variable Inertia
- ▶ [CCC0309] Pump-Shaft Connection

[Close]

Cell/Junction IC Data - PUMP 310 (CPUMPA31)

Cell Number	Condition	Use Boron	Pressure (psi)	L.Energy (btu/lbm)	V.Energy (btu/lbm)	Void Frac.
1	[P,U$_f$,U$_g$,a$_g$]	☐	1177.607	521.309	1105.062	0.0

[Steam Tables]

Cell Fluid | Noncondensibles | Junction Flow

[Close]

Figure 13   Recirculation Pumps Properties

2-16
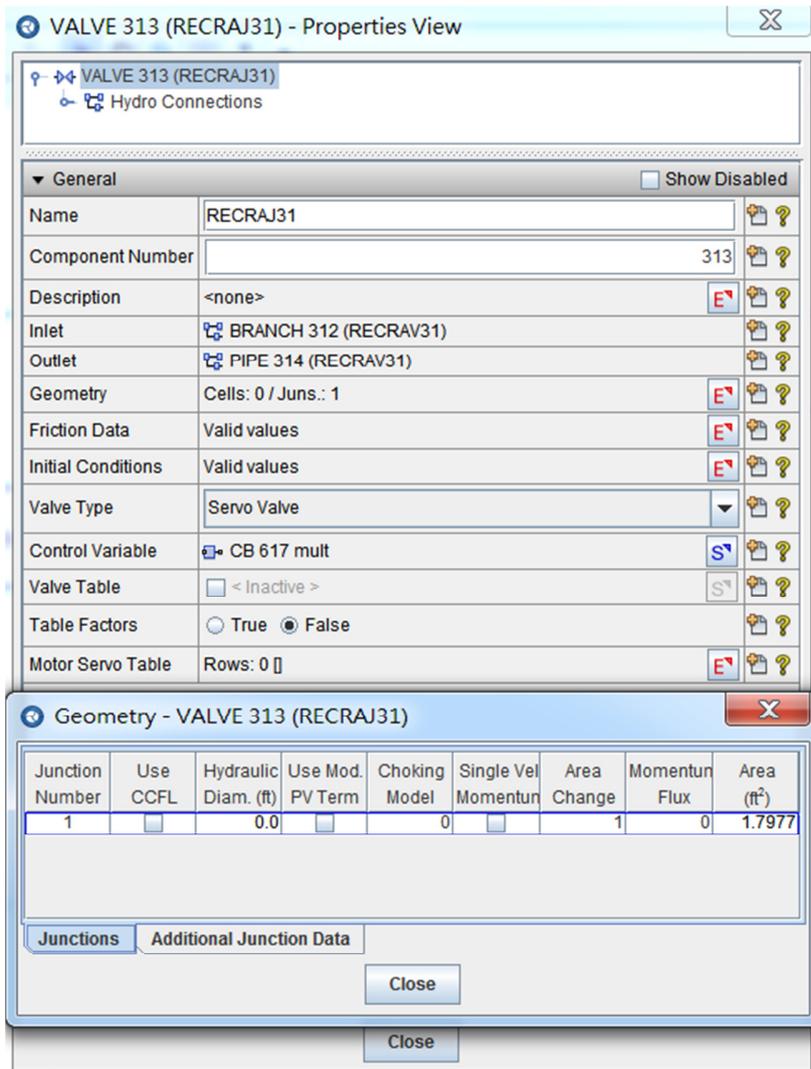
Figure 14   Properties of Recirculation Controlling Valves
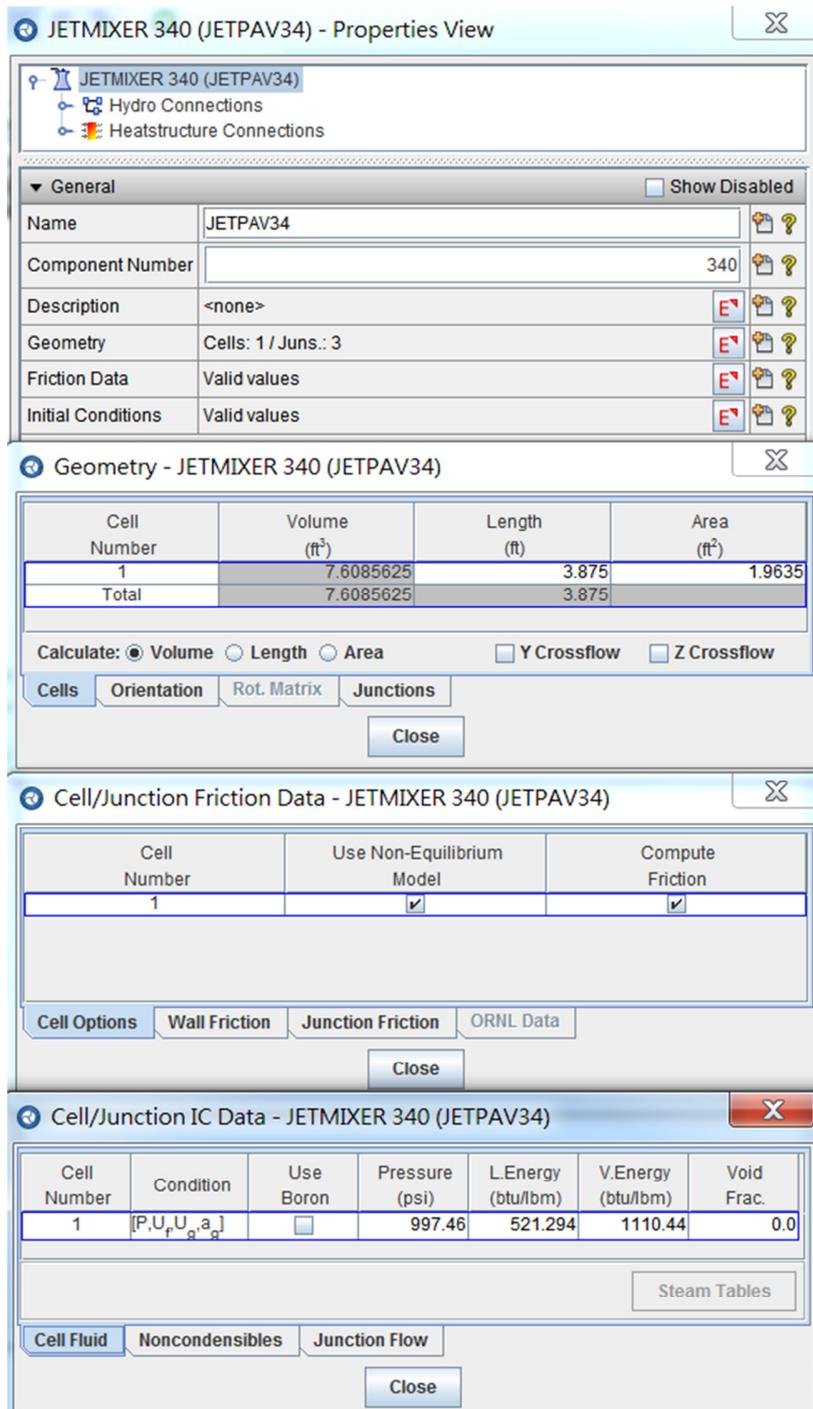
Figure 15   Jet Pumps Properties

Animation Model

Because of developing analysis model with SNAP interface, analysis data results could be transferred into animations which could illustrate the situation of NPP during transients more easily and clearly. Further, based on the TRACE animation model which had been developed by our group before, the RELAP5/MOD3.3 animation model just needs a few modifications in this research. As shown in Figure 16, the reactor vessel was developed with several "Plenum" animation components. Plenum at different height level would be connected to correspondent "Branch". Feedwater pumps were developed with "Fill" components because they were the boundary conditions of the RELAP5/MOD3.3 thermal hydraulic model. Two "Pipe" components which would change color depth with mass flow rate were connected to the feedwater pumps. Recirculation pumps were also developed with "Single Pump" components, which would be green if the pumps are working and would be red if the pumps are tripped. 4 main steam pipe lines were developed with "Pipe" animation components. These pipes would indicate the mass flow rate by changing color depth. Besides, all the valves including MSIVs, TSVs, TCVs and SRVs were also developed with animation components "Gate Valve". These valves would be green if they are open and would be red if they are closed. On the right side of the animation model, there are two bars which could immediately indicate the reactor power and dome pressure respectively during transients. With this animation model, the interactions of different components could be observed more directly and easily. It would also be more attractive and comprehensive on the presentation. Moreover, this RELAP5/MOD3.3 animation model was modified from the existing TRACE animation model. It would be convenient to do cross validation of TRACE and RELAP5 analysis results.
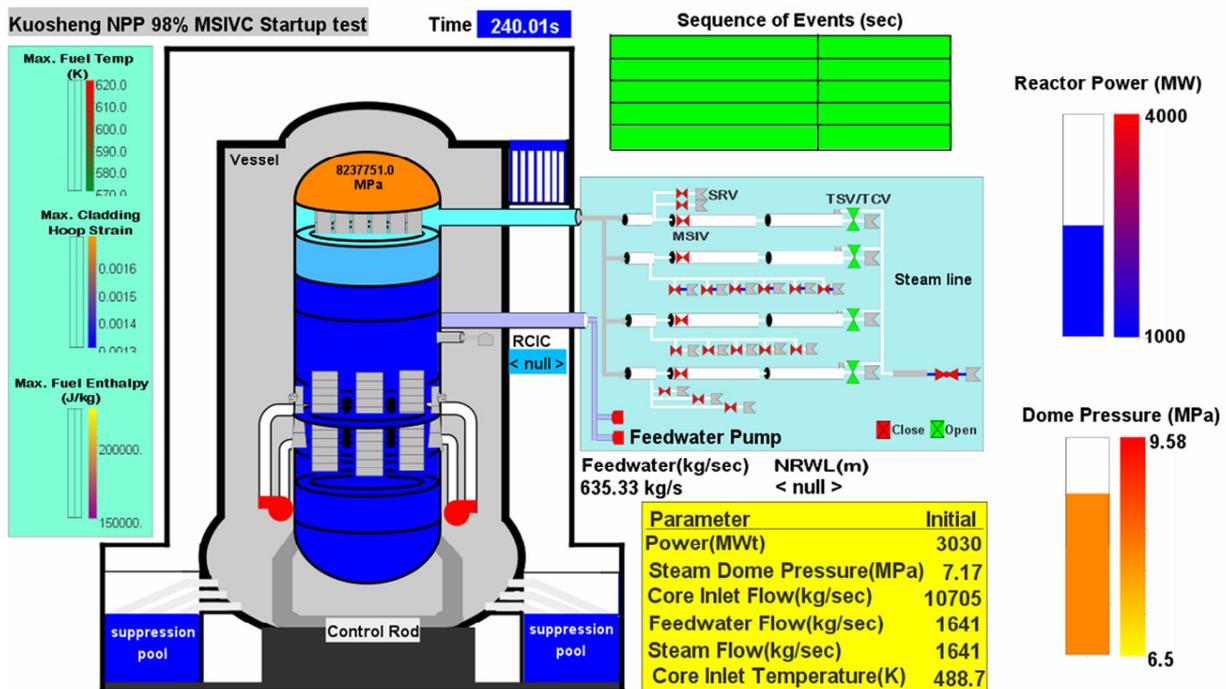
Figure 16   RELAP5/MOD3.3 Animation Model

3. START-UP TEST RESULTS

The startup tests data results collected in this research came from report prepared by INER in 1996. However, this report only recorded the data results before Measurement Uncertainty Recapture Power Uprate. To confirm the applicability of the RELAP5 model, the results of TRACE which had been performed for the startup tests transients were referenced. Hence, there are three curves (startup test, TRACE and RELAP5) compared in each case illustrated be low. Moreover, the steady state parameters of startup test will be a little lower than that of TRACE and RELAP5 computational results.

3.1 100% Load Rejection with Bypass Valve Open (LRWB)

Event Description

The LRWB test was held in November 11th, 1981. The operator triggered the Breaker of the main generator as the reactor was operated in 100% power. Then, the turbine control valves closed and turbine bypass valve opened which caused the transient start and reactor scram. The objective of this test is to ensure whether the turbine control valves could closed immediately or not. In addition, the responsibility of the turbine bypass valve, safety/relief valves and the reactor protection system should also be assessed. The acceptance criteria are shown as follows:
- Turbine bypass valve should open in 0.1 second after the turbine control valves closure. Further, the bypass valve should reach to 80% open of the design capacity in 0.3 second.
- The feedwater setting should avoid the main steam pipe lines being flooded.
- The feedwater control system should maintain the water level which would not trigger the MSIVs closure
- The dome pressure increasing should not be higher than 25 psi in 30 seconds and the heat flux ratio should not be over than 2%.

To verify the RELAP5 model, the turbine control valves and bypass valves were developed according to the FSAR of the power plant [1]. Table 1 lists the comparisons of the initial values from the RELAP5 analysis and startup test data. Due to the stability, a 210-second steady state was performed for ensuring that all the parameters matched the operating conditions as shown in Table 2. However, to compare the results with startup test and TRACE model data more easily, the steady state will be eliminated in the figures.

At 210.2 second, the TCVs started to close. Control system of the TCVs is shown in Figure 17. The control system of TBV opening is shown in Figure 18. After closure of TCVs for 0.011 second (variable trip 409), the TBV start to close. Further, when the dome pressure was lower than 940 psia (6.48 MPa, variable trip 410), the TBV would start to close as shown in Figure 18. The recirculation pumps were tripped as the dome pressure reached to 1134.7 psia (7.82 MPa). The control logic of recirculation pumps trip was shown in Figure 19. Three groups of the SRVs setpoints were developed as shown in Figure 20.

Table 1   Initial Conditions of RELAP5 and Startup Test Data in 100% LRWB Transient

Parameter	Test data	RELAP5
Power，MW	2894.0	2894.0
Steam Dome Pressure (MPa)	6.96	6.99
Feedwater Flow(kg/sec)	1551.66	1574.11
Steam Flow(kg/sec)	1510.91	1574.10
Narrow Range Water Level，cm	90	89.9
Test Date	'81/11/11	─

Table 2   Event Sequence of 100% LRWB Transient

Event(sec)	Test data	RELAP5
TCV Start to Close	0.2	210.2
Reactor Scram	0.236	210.29
BPV Fully Open	0.329	210.32
TCV Fully Close	0.394	210.35
Peak Dome Pressure Time (value)	3.9 (7.43MPa)	212.84 (7.55MPa)
End of Analysis	─	230
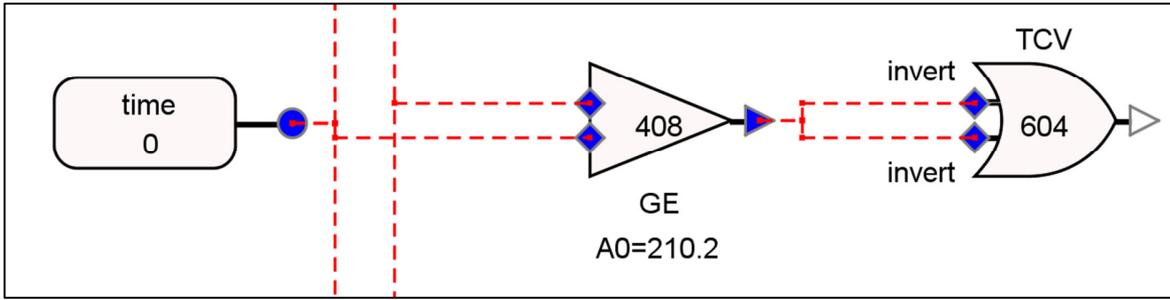
Figure 17   Controlling System of Turbine Control Valves in 100% LRWB Transient
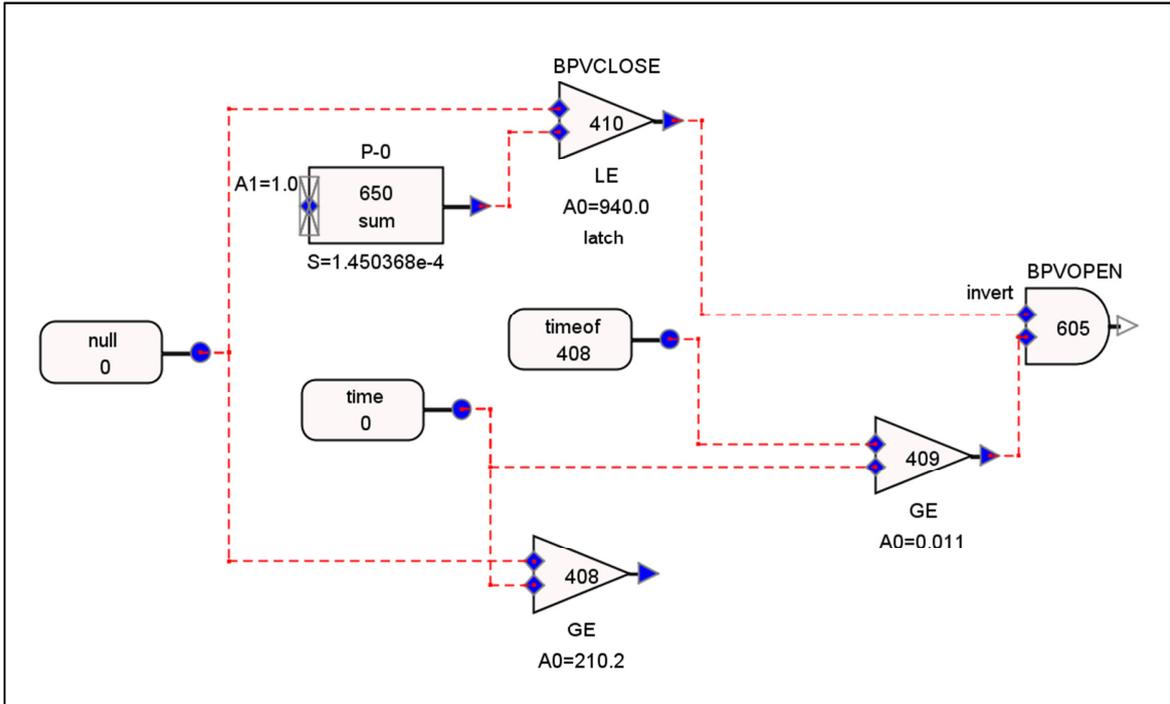
Figure 18   Controlling System of Turbine Bypass Valve in 100% LRWB Transient
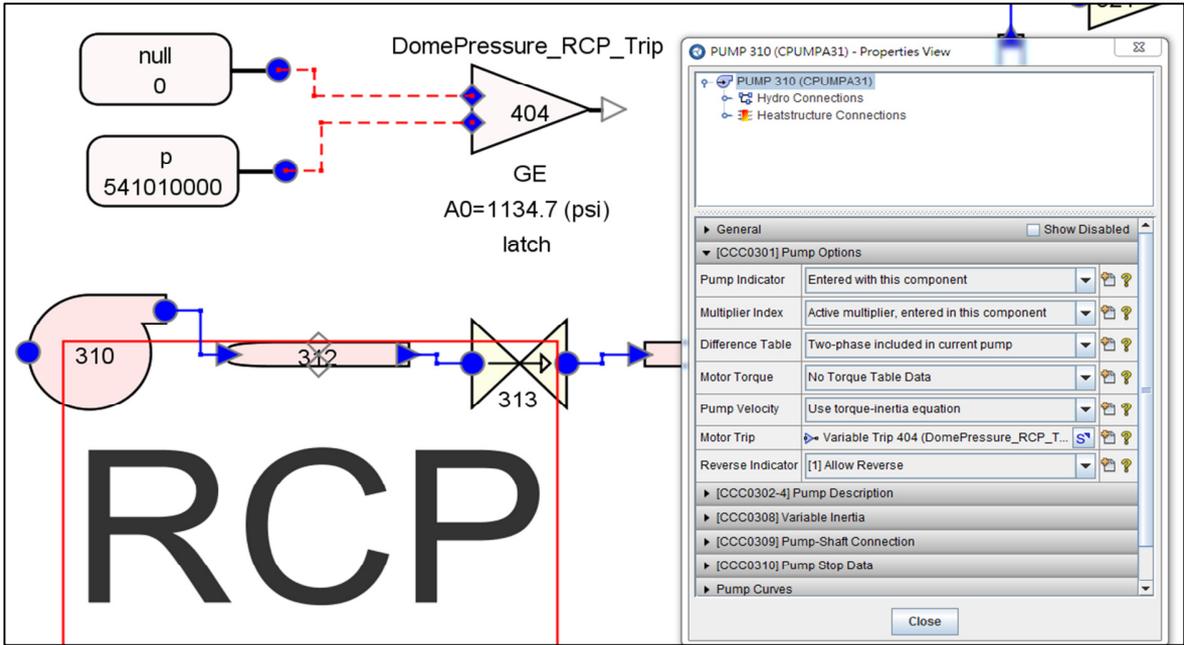
Figure 19   Controlling System of Recirculation Pumps Trip
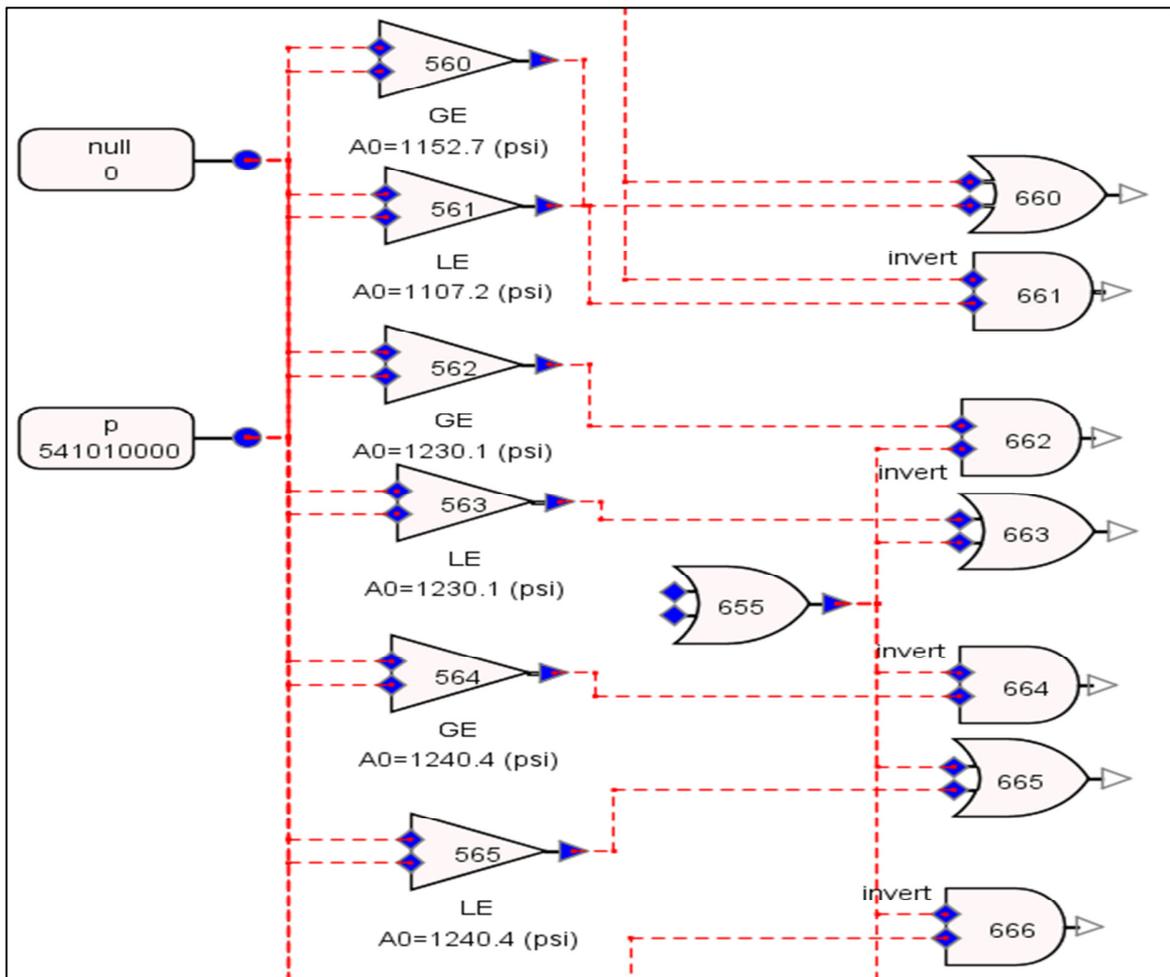
Figure 20   Controlling System of Safety Valves in 100% LRWB Transient

Analysis Data Results

Once the generator tripped, the turbine control valves started to close and fully close in 0.1 second. As the TCVs started to close, the steam flow decreased immediately as shown in Figure 21. Figure 22 shows great consistency between RELAP5 model and startup test result. As the TCVs started to close, the reactor scram signal was initiated. Due to the large negative reactivity feedback, the power decreased rapidly at first second as shown in Figure 22. Nonetheless, the reactor core still generated some decay heat by the fission product. Hence, the steam was still produced inside the reactor vessel. Even though the turbine bypass valve started to open at 0.1 second according to the acceptance criteria mentioned above, the dome pressure still increased as shown in Figure 23.
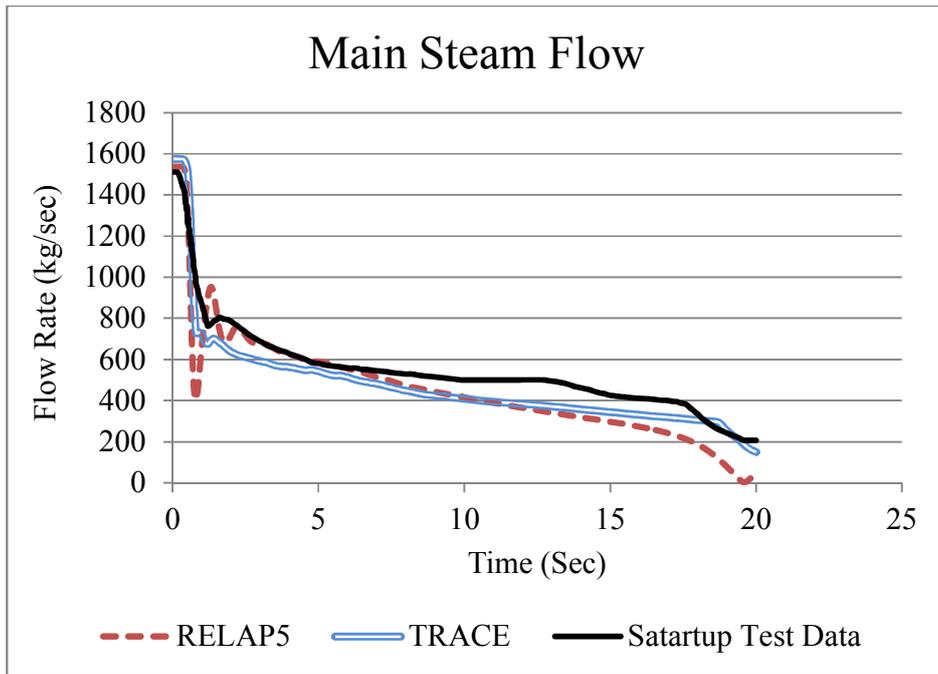
Figure 21   Steam Flow Variation during the 100% LRWB Transient
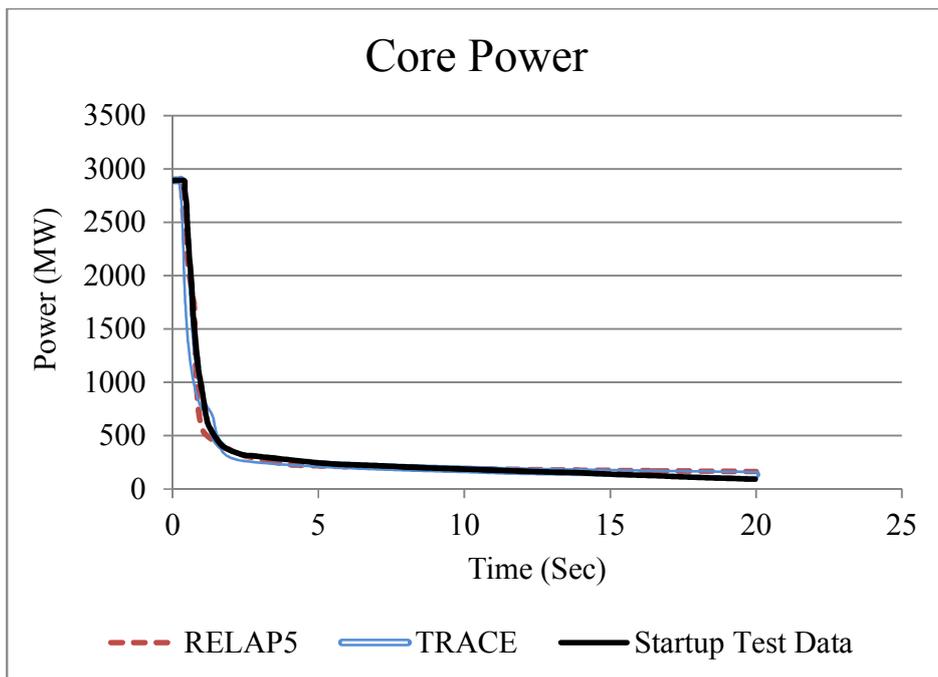
Figure 22   Core Power Variation during the 100% LRWB Transient
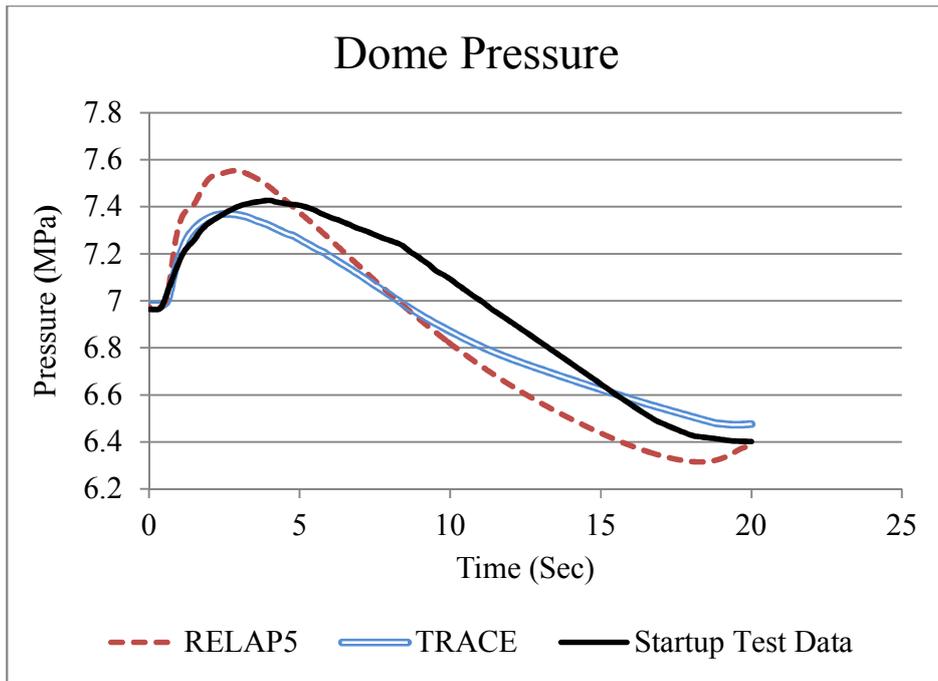
Figure 23   Dome Pressure Variation during the 100% LRWB Transient

3.2 96% Main Steam Line Isolation Valves Closure (MSIVC)

Event Description

The Main steam line Isolation Valves Closure test was held in November 6th in 1981. As the reactor operated at 96% power, the operator triggered the circuits F6A and F6B to close all the MSIVs. The objective of this test including:
(1) Whether the MSIVs work normally in such power and operating condition.
(2) Obtained the transient variation as the MSIVs were closing and fully closed.
(3) Ensure the fully closure time of the MSIVs.

The acceptance criteria are shown as follows:
(1) The closure time of each MSIV should not shorter than 3 seconds and should not longer than 5 seconds. Further, the electric signal delayed time should shorter than 0.5 second.
(2) After MSIVs closure, the dome pressure increment should not larger than 160 psi in 30 seconds. The heat flux increment should not exceed 2% of the initial value.
(3) The feedwater control system should avoid flooding of the main steam pipe lines.
(4) After the overpressurization transient, the safety/relief valves could be re-closed.

From the startup test results, the reactor scram as the MSIVs reached to 94% open. The MSIVs closure time was about 4.08 to 4.53 seconds. The dome pressure increment was about 118 psi during the transient. Because the dome pressure reached to the setpoints of the safety/relief valves, there were two groups of valves opened during the transient. Based on these data results, the RELAP5 model control system in the following description could be developed.

To ensure the analysis results were consistent to the operating conditions, a 210-second steady state was performed. Comparison of the startup test data and RELAP5 initial condition results was listed in Table 3 and the events sequence of this transient was shown in Table 4. At 210 second, the MSIVs started to close and fully closed in 4.9 second according to the startup test report. The reactor scram signal might come from three sources including neutron flux, reactor vessel pressure and main steam line isolation valves area as shown in Figure 24. In this transient, the reactor scram signal was initiated due to the MSIVs reached to 94% open with 0.94 second delayed time. Different from the startup test report, there are three groups of SRVs setpoints in this transient as shown in Figure 25 and Table 4, which lists the event sequence of the MSIVC transient.

Table 3 Initial Conditions of RELAP5 and Startup Test Data in 96% MSIVC Transient

Parameters	Test data	RELAP5
Power (MW)	2836.97	2836.72
Steam Dome Pressure (MPa)	6.92	6.96
Feedwater Flow (kg/sec)	1498.41	1519.29
Steam Flow (kg/sec)	1456.64	1519.29
Narrow Range Water Level，cm	90.0	89.9
Test Date	'81/11/06	--

Table 4 Event Sequence of 96% MSIVC Transient

Event (sec.)	Test data	RELAP5
MSIVFI Initiated	0.3	210.3
Reactor Scram	1	210.92
MSIV Full Isolation	4.83	214.825
Peak Dome Pressure Time (value)	18.4 (7.7MPa)	218.1 (7.44MPa)
End of Analysis	一	240
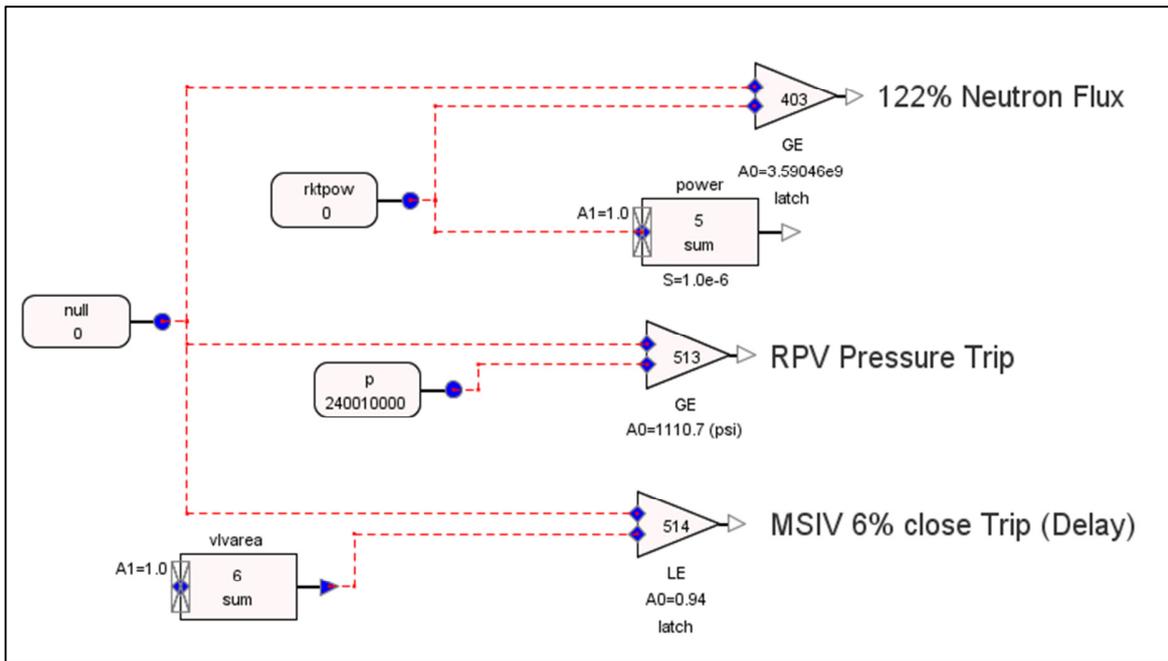
Figure 24　Controlling System of Reactor Scram in 96% MSIVC Transient
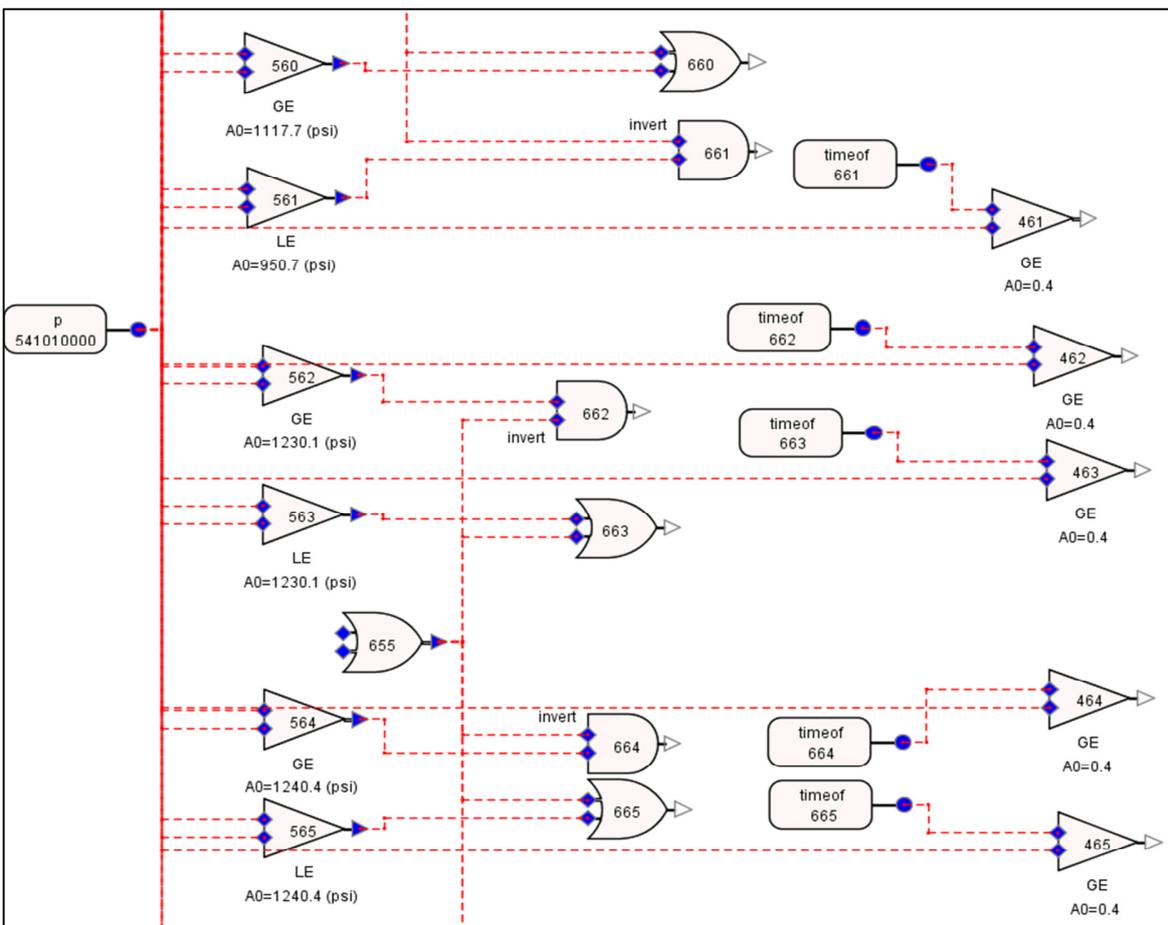
Figure 25　Controlling System of Safety/Relief Valves in 96% MSIVC Transient

Analysis Data Results

Figure 26 shows the steam flow comparison of startup test data, TRACE and RELAP5 results. As the MSIVs started to close, the steam flow decreased gradually. 5 seconds later, the MSIVs fully closed and the steam flow rate reached to 0 kg/sec. From Figure 27, it is obviously that after the reactor scram, the power variations of startup test and computational results were a little different. This difference might come from the decay heat model choice. In the RELAP5 model, only GAMMA-AC fission product decay model was chosen and the fission product type was not chosen. This model choice might be different from the real reactor core.

Because of the MSIV closure, the steam would be trapped inside the reactor vessel. Hence, the dome pressure might increase as shown in Figure 28. However, from this figure, it can be noticed that before the dome pressure increasing, it would decrease first because the reactor had scrammed at about 1 second, as shown in Figure 28. The steam produced inside the reactor core would be reduced. Hence, the dome pressure decreased until the MSIVs fully closed. In addition, from this figure, it can be found that the increasing of dome pressure in RELAP5 model was faster than that in startup test or in TRACE model. The SRVs would function earlier in RELAP5 model. Hence, the peak dome pressure of RELAP5 model would be lower than that of other two data results.

Figure 29 shows the comparison of feedwater flow among these three data results. For the computational results, they are consistent to each other; however, the feedwater flow of startup test data was much different. This difference might come from the setting of recirculation pumps. The curves of recirculation pumps of Kuosheng NPP were hard to be collected completely. Some of the pump curves in RELAP5 model were referred to the EPRI data. Although the pumps developed EPRI data would not be so different from the pumps installed in the NPP, however, the feedwater feedback system might be very sensitive to the core flow. Hence, the feedwater flow rate of computational result was much different from the startup test data.
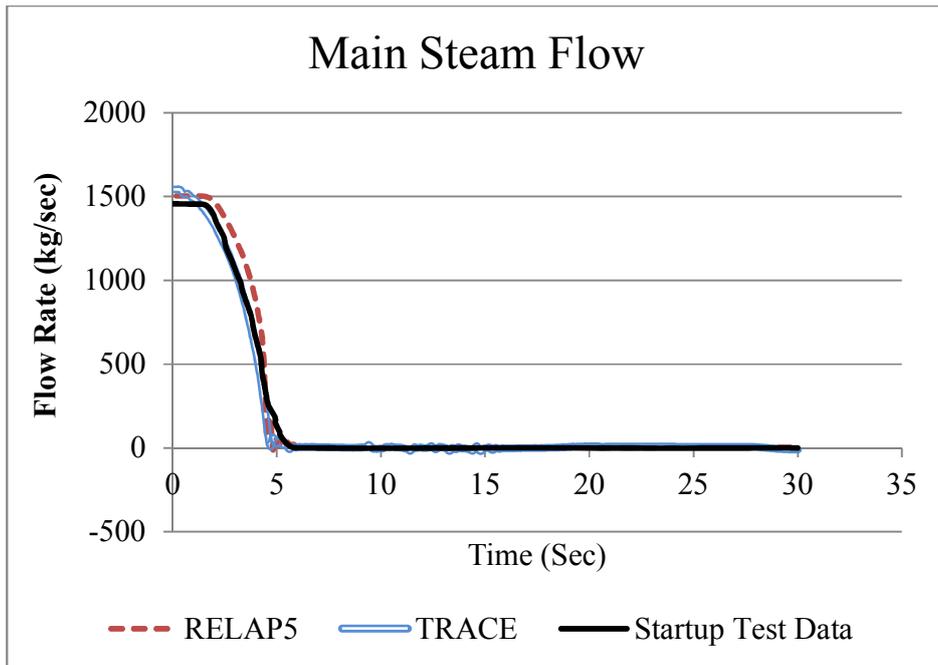
Figure 26   Steam Flow Variation during the 96% MSIVC Transient
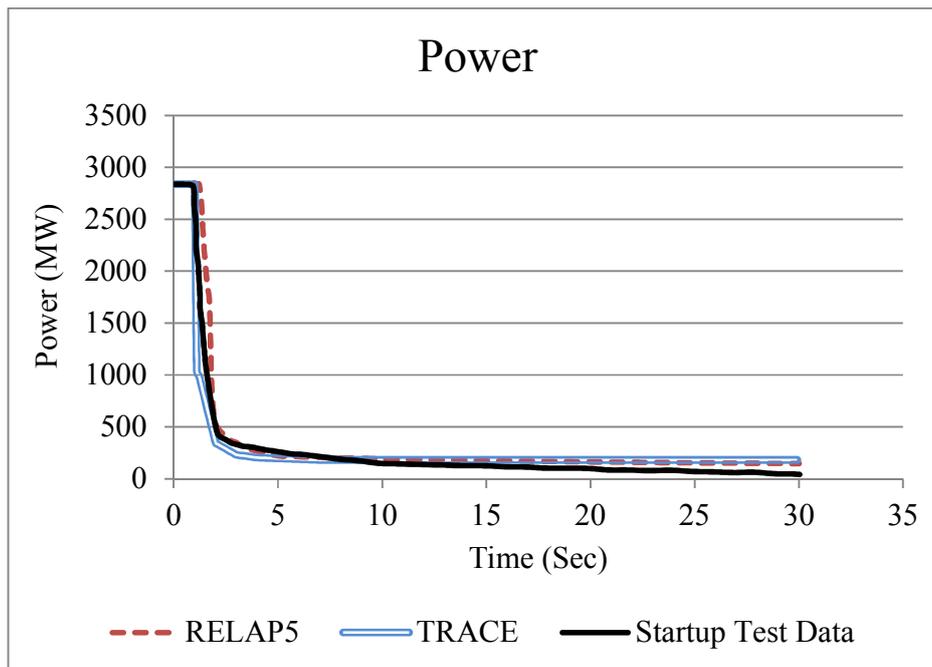
Figure 27   Core Power Variation during the 96% MSIVC Transient
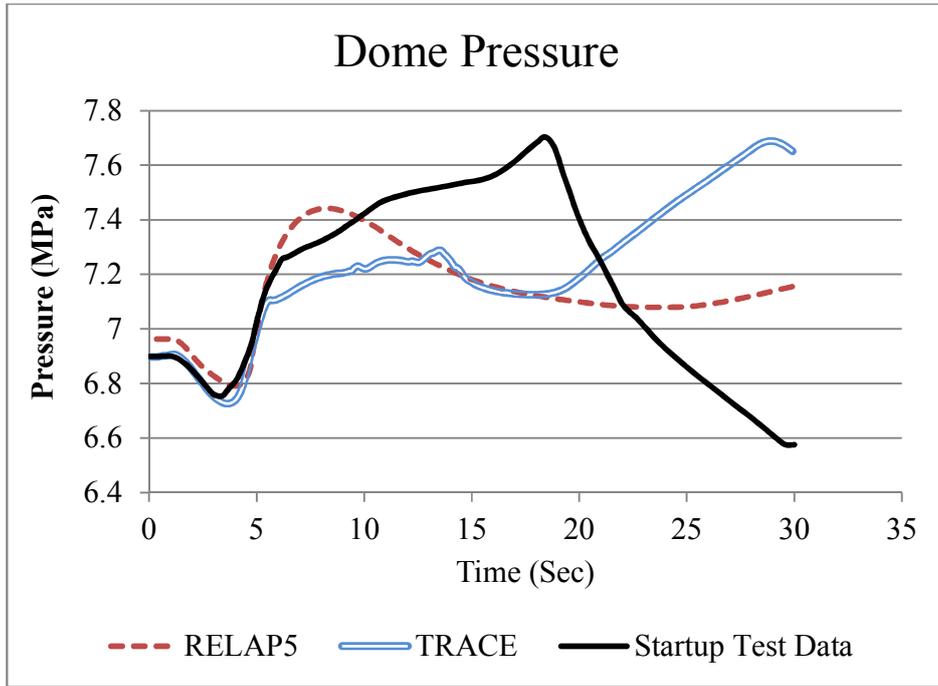
Figure 28   Dome Pressure Variation during the 96% MSIVC Transient
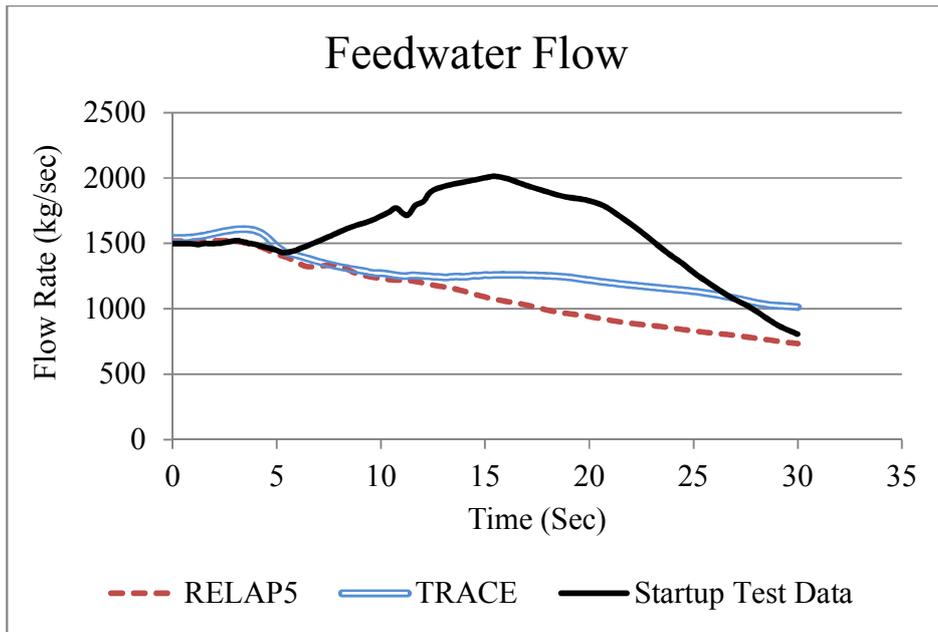
Figure 29   Feedwater Flow Variation during the 96% MSIVC Transient

3.3 94% Feedwater Pumps Trip (94% FWPT)

Event Description

The 94% Feedwater Pumps Trip was held in November 6th, 1981. The transient started as the operator tripped one of three operating feedwater pumps at 94% power. The objective of this test is to examine whether the flow control valves (FCV) run back in the recirculation loops could lower the core power or not when a feedwater pump failed. If the core power could be lowered by the reduction of the recirculation flow, the water level would not be below L3, which was the setpoint of the reactor scram.

Table 5 shows the initial conditions of RELAP5 analysis results and startup test data and Table 6 shows the event sequence of this transient. Same as previous two cases, a 210-second steady state analysis was performed to ensure all the parameters matched the operating conditions. At 214.9 second, one of the three feedwater pumps was tripped. Generally speaking, three feedwater inlet junctions were controlled by control block 994 (FW-24) as shown in Figure 30. To perform the FWPT transient, one of the feedwater junctions was controlled by control block 1 (FW-25), which was functioned by feedwater trip table 902. Further, the L4 water level signal would initiate FCV run back to lower the core power which would maintain the water level. According to the configuration of KS NPP, the FCV area of recirculation loop A would decline from 80% to 50% and the FCV area of recirculation loop B would decline from 69% to 46%. Control system of the FCV is shown in Figure 31.

Table 5   Initial Conditions of RELAP5 and Startup Test Data in 94% FWPT Transient

Parameters	Test data	RELAP5
Power (MW)	2790.40	2790.80
Steam Dome Pressure (MPa)	7.00	7.01
Feedwater Flow (kg/sec)	1470.35	1494.77
Steam Flow (kg/sec)	--	1494.76
Narrow Range Water Level，cm	90.0	89.9
Test Date	'81/11/06	--

Table 6   Event Sequence of 94% FWPT Transient

Event (sec.)	Test data	RELAP5
Feedwater pump Trip	4.9	214.9
L4 Singal	15.1	217.01
Minimum Power Value	18.5 (57.1%)	218.90 (54.2%)
Minimum Core Flow	19.4 (79.3%)	218.31 (78.76%)
END of Analysis	―	240
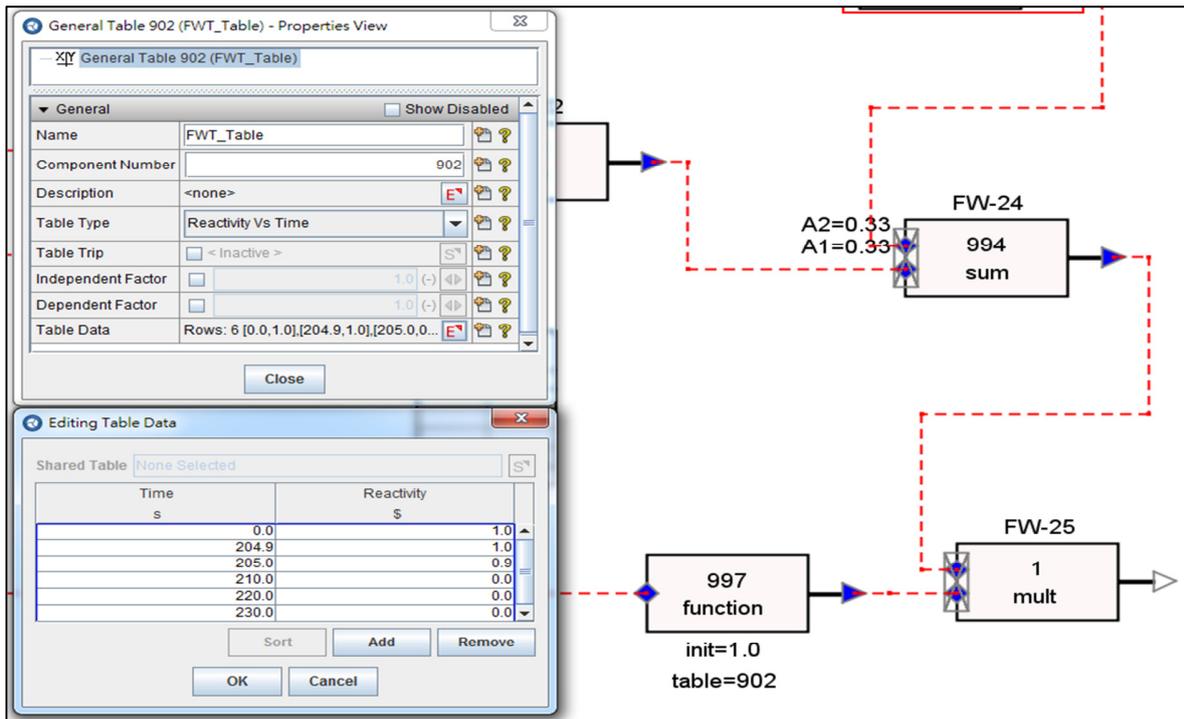
Figure 30   Controlling System of Feedwater Pump Trip in 96% FWPT Transient
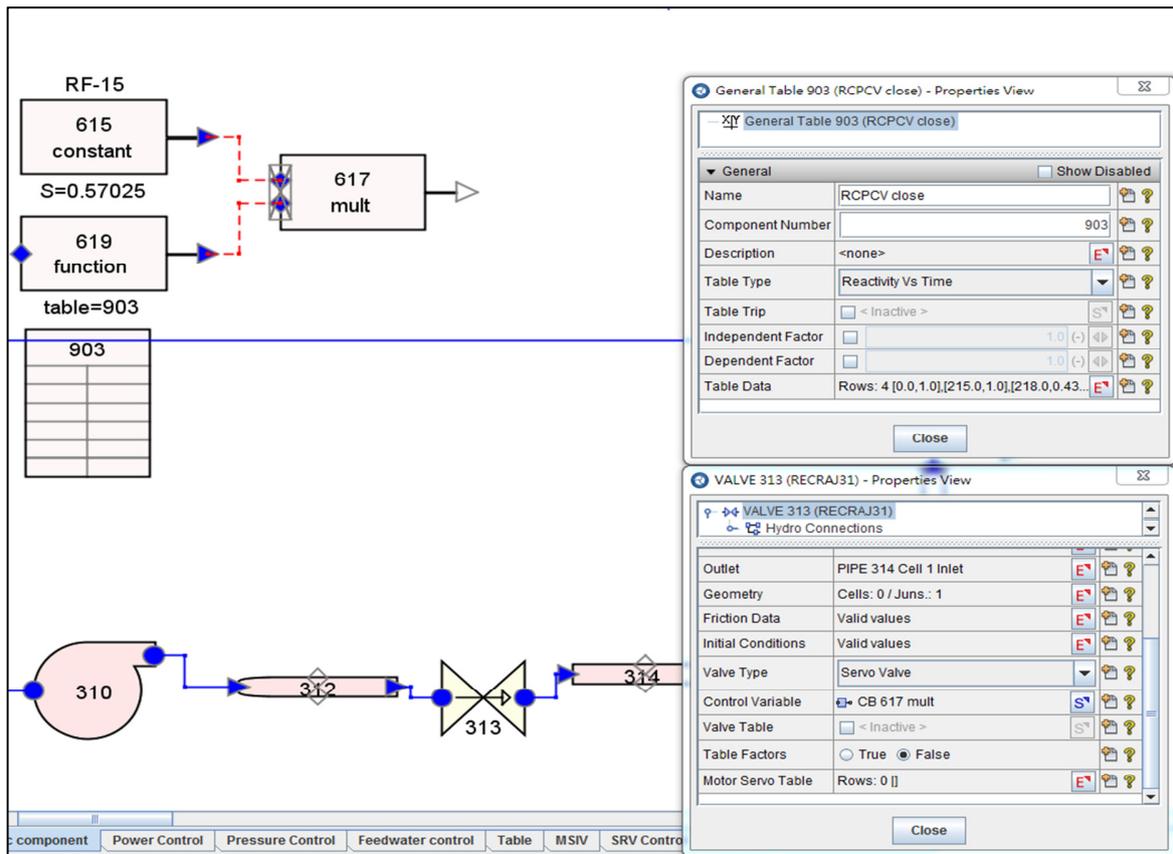
Figure 31   Controlling System of Flow Control Valves in 96% FWPT Transient

3-16

Analysis Data Results

Figure 32 shows the comparisons of feedwater flow among startup test, TRACE and RELAP models. At 4.9 seconds, one of three feedwater pumps was tripped. The feedwater flow decreased for about 5 seconds. About 17 seconds later, as shown in Figure 33, the NRWL reached to 0.848 m, which is L4 for Kuosheng NPP. The recirculation flow control valves ran back. Hence, core inlet flow rate decreased to 80 %, as shown in Figure 34. Decreasing of the core flow would increase the void fraction, which means that the density reactivity feedback would also decrease. The core power dropped down at about 17 seconds as shown in Figure 35. In addition, because the water level declination in TRACE model was a little slower than the other two cases, it can be noticed that the core power decreasing in TRACE model was later than that in the other two data results.

Due to the declination of feedwater flow rate and water level, the feedwater control system adjusted the other two available feedwater junctions and increased the flow rate as shown in Figure 32. However, at the end of the transient, core power of computational data results was different from that of startup test since the water level might be controlled in startup test as shown in Figure 33. The density reactivity feedback would be different from the computational models. Nonetheless, due to lack of some other transient scenario or information, the water level followed the computational results rather than matched the startup test data in both TRACE and RELAP5 models.
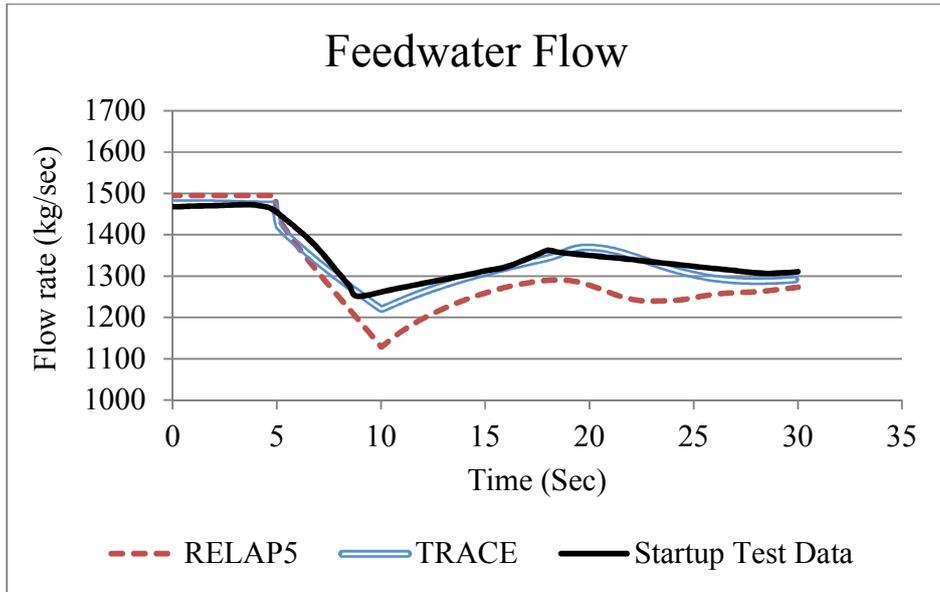
Figure 32   Feedwater Flow Variation during the 94% FWPT Transient
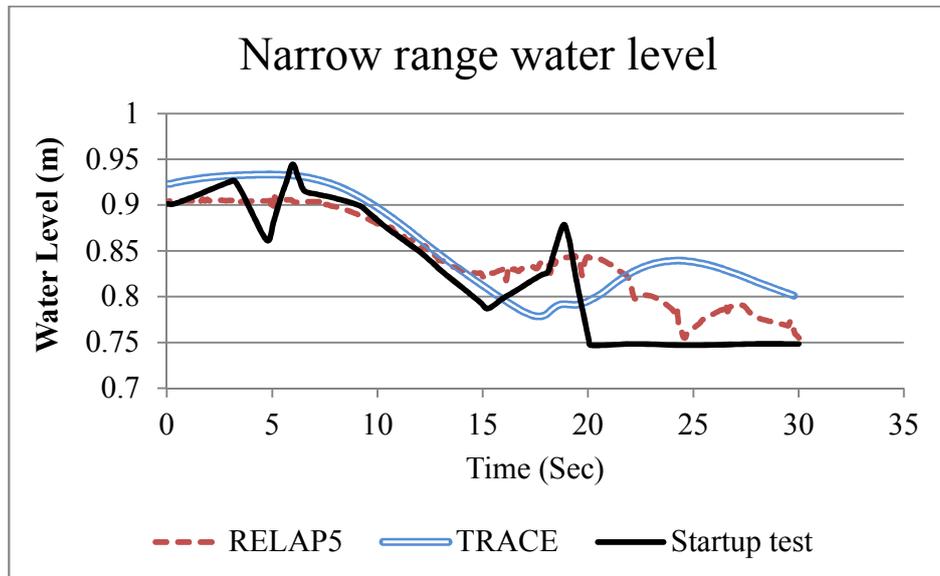
Figure 33   Narrow Range Water Level Variation during the 94% FWPT Transient
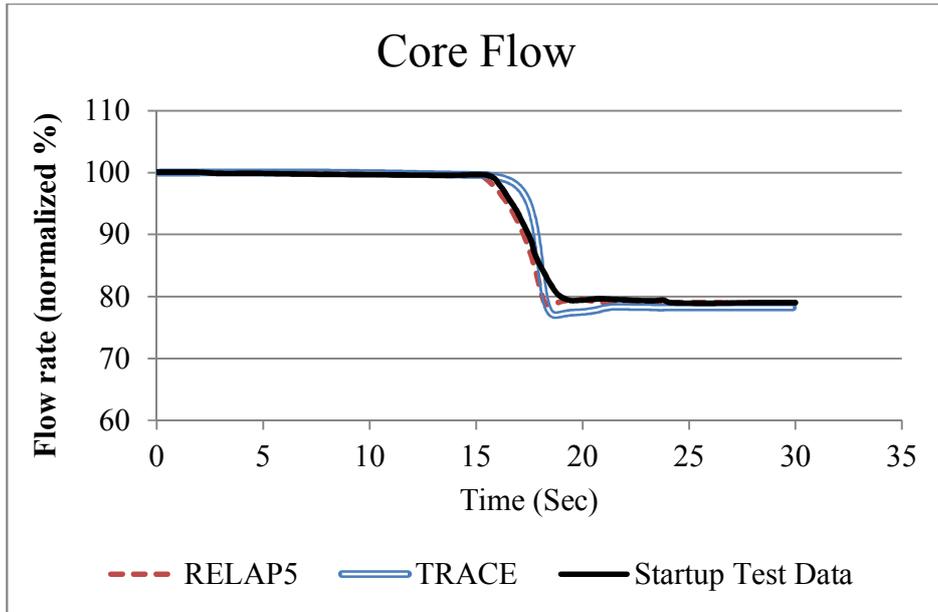
Figure 34　Core Flow Variation during the 94% FWPT Transient
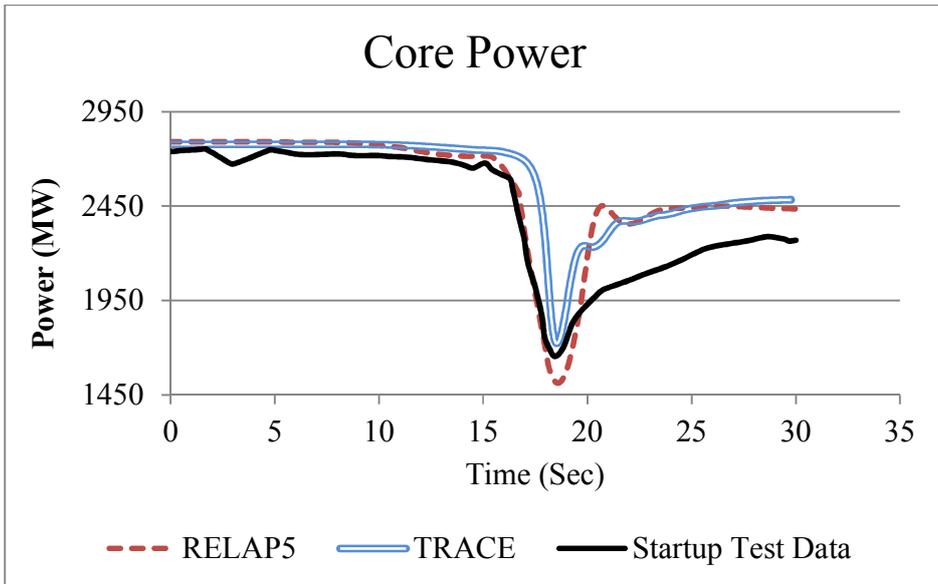
Figure 35　Core Power Variation during the 94% FWPT Transient

4. HYPOTHETICAL TRANSIENT RESULTS

To ensure the safety of SPU, a series of overpressurization transient test of TRACE code had been performed before by our group [5, 6]. In this research, same hypothetical accidents have been performed again by RELAP5 code to verify the accuracy of TRACE model. As a result, the initial conditions such as power, feedwater flow, steam flow and dome pressure would be same as the prediction of TRACE code as shown in Table 7.

Table 7   Initial Conditions of TRACE and RELAP5 Models

Parameters	Unit	FSAR	TRACE	RELAP5
Power	(MWt)	3030	3030	3030
Dome Pressure	(MPa)	7.17	7.17	7.17
Feedwater Flow	(kg/sec)	1645.3	1641	1640.6
Steam Flow	(kg/sec)	1645.3	1641	1640.6
Core inlet flow	(kg/sec)	10645	10704	10674
Recirculation flow (Single loop)	(kg/sec)	1549.5	1538.6	1548.4
Core exit pressure	(MPa)	7.23	7.3	7.23
NWRL	(m)	0.934	0.934	0.933

4.1 Main Steam Line Isolation Valves Closure with Bypass Failure

Hypothetical Event Sequence

In this hypothetical accident, a 210-second steady state was performed for ensuring that all the parameters matched the operating conditions. At time point 210 second, the main steam line isolation valves (MSIVs) closed in three seconds. In order to promote the difficulties of this transient for the plant, the closure time was assumed to be 3 seconds in this case, which was shorter than 5 seconds that described in FSAR. According to the FSAR description, reactor scram signal in this case might come from two ways including neutron flux exceeding 122% or dome pressure exceeding 7.66 MPa. The reactor scram control system is shown in Figure 36. Further, as the dome pressure reached to 7.82 MPa, recirculation pump trip signal would be sent out with delayed time 0.14 second. For the conservation reason, only 11 safety valves works even though that there are 16 safety/relief valves on the main steam pipelines. All these 11 safety valves were still divided into three groups, as shown in Figure 37, with different set-points including 8.38 MPa for two valves, 8.48 MPa for five valves and 8.55 MPa for four valves. Besides, after the closure signal sent out, the safety valves would fully close in 0.15 second with an electronic delayed time 0.4 second. Table 8 lists the MSIVC events sequence and important set-points. From the data results, the reactor scrammed due to neutron flux exceeding 122% in this transient. Further, the safety valves will fully open in 0.15 second with delayed time 0.4 second. Once the safety valves open, the dome pressure was under controlled and the NPP was back to the safe situation.
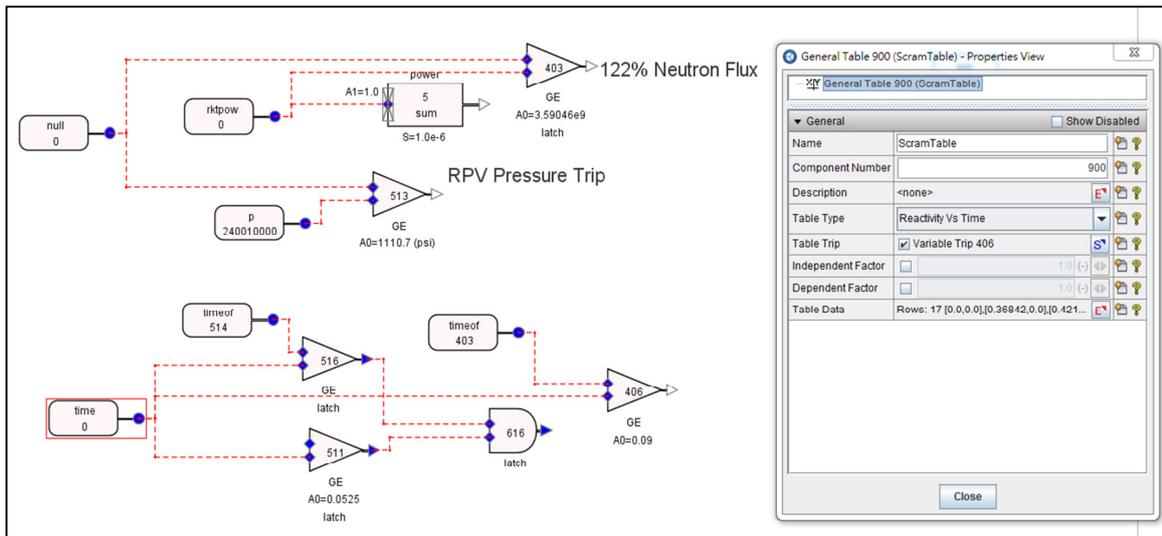
Figure 36   Controlling System of Reactor Scram in MSIVC Hypothetical Accident
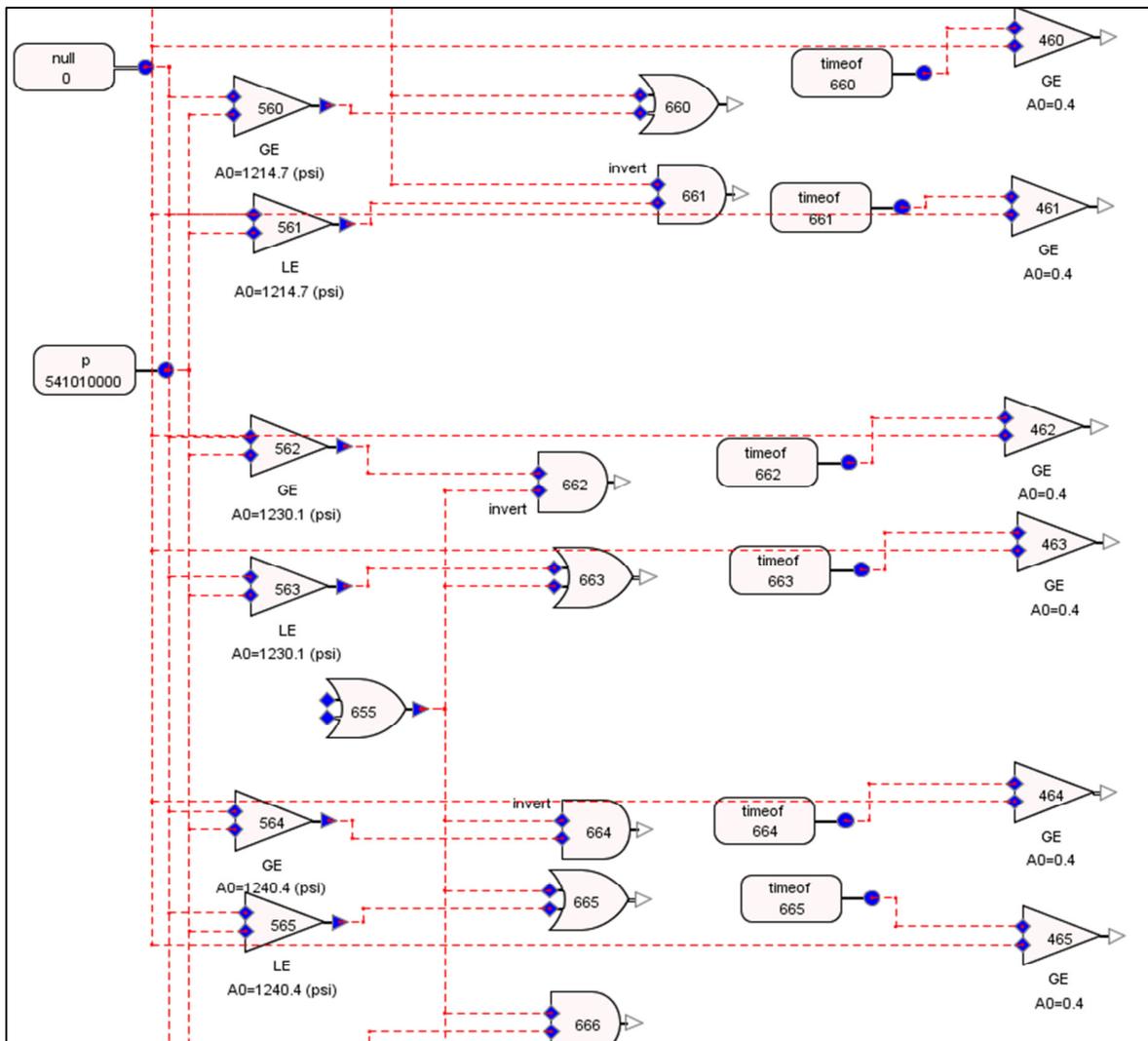
Figure 37   Controlling System of Safety Valves in MSIVC Hypothetical Accident

Table 8   Events Sequence and Comparison of MSIVC Hypothetical Accident

Events	RELAP5 (sec)	TRACE (sec)	Notes
Steady state	0~210	0~210	Power 3030 MW Feedwater flow rate 1641 kg/sec Feedwater temperature 488.7K Core inlet flow rate 11177.3 kg/sec Dome pressure 7.17MPa
MSIVs closure	210	210	Fully closure time 3 seconds
MSIVs fully closed	213.210	213.06	
Reactor scram signal initiated	213.211	213.14	Power reached to 122%
Reactor scram	213.310	213.23	Delayed time 0.09 second
Recirculation pumps tripped	213.315	213.46	Dome pressure 7.82MPa
Safety valves group 1 opened (8.38MPa，Group 1)	213.915	214.53	Delayed 0.4 second
Safety valves group 2 opened (8.48MPa，Group 2)	214.015	214.65	Delayed 0.4 second
Safety valves group 3 opened (8.55MPa，Group 3)	214.115	214.73	Delayed 0.4 second
End of the analysis	220	220	

Analysis Data Results

Figure 38 shows the comparison of steam flow rate between TRACE and RELAP5 model. From this figure, it is obviously that the steam flow rate dropped rapidly once the MSIVs closed at 210 second. At 213 second, the MSIVs fully closed and the dome pressure increased greatly as shown in Figure 39. Hence, the void fraction inside the reactor core decreased which increased the power greatly as shown in Figure 40. In this figure, the core power of RELAP5 model was much lower than that of TRACE model. This difference might came from the neutron kinetic feedback table setting. In the RELAP5 model, the neutron slowing effect would reference the table which includes density and reactivity. However, in the TRACE model, the reactor power iteration would reference the table containing void fraction and reactivity. As a result, the power calculation might have some differences during high-power transient.

Despite the fact that the reactor had scrammed at 213.23 second, the decay heat still heated the reactor vessel and generated steam in both the RELAP5 and TRACE model. As a result, the dome pressure kept increasing. At 214.53 second, the dome pressure was higher enough to meet the setpoint of group 1 safety valves. However, 3 opened safety valves discharged the steam insufficiently. The dome pressure kept increasing until the group 2 and group 3 safety valves opened at 214.65 and 214.73 second respectively in TRACE model. Further, because the dome pressure varied near the setpoints, the safety valves opened and closed for three times. On the other hand, in the RELAP5 model, safety valves of group 1 opened at 214.31 second, which was not far from that in TRACE model. Safety valves of group 2 and group 3 also opened at 214.40 second and 214.51 second. However, the dome pressure prediction of RELAP5 model was higher than that of TRACE model. As a result, the dome pressure variation did not triggered the setpoinhts of safety valves closure in the RELAP5 model. Hence, as shown in Figure 39, the reactor vessel in the RELAP5 model maintain a higher pressure than that in the TRACE model.

From both the RELAP5 TRACE data results, it is known that the core power could be controlled by the scram system. In addition, as long as the safety valves opened, the dome pressure would not exceed the ASME criteria 9.58 MPa. Kuosheng NPP was safe in the MSIVC hypothetical transient.
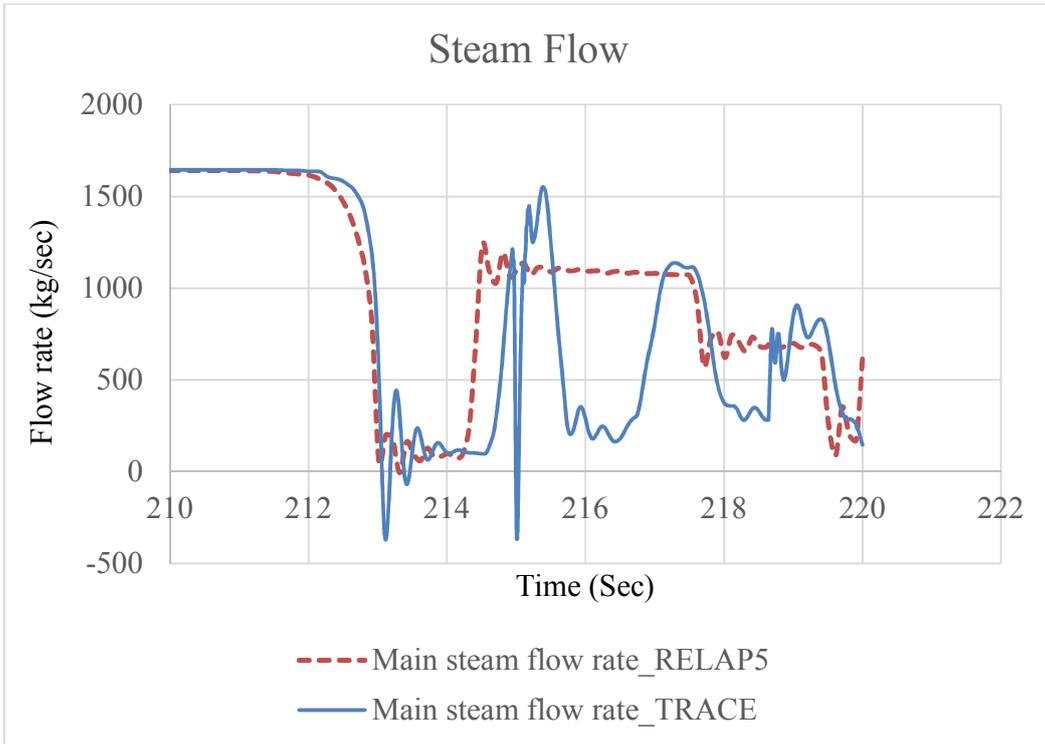
Figure 38   Steam Flow Variation during the MSIVC Hypothetical Accident
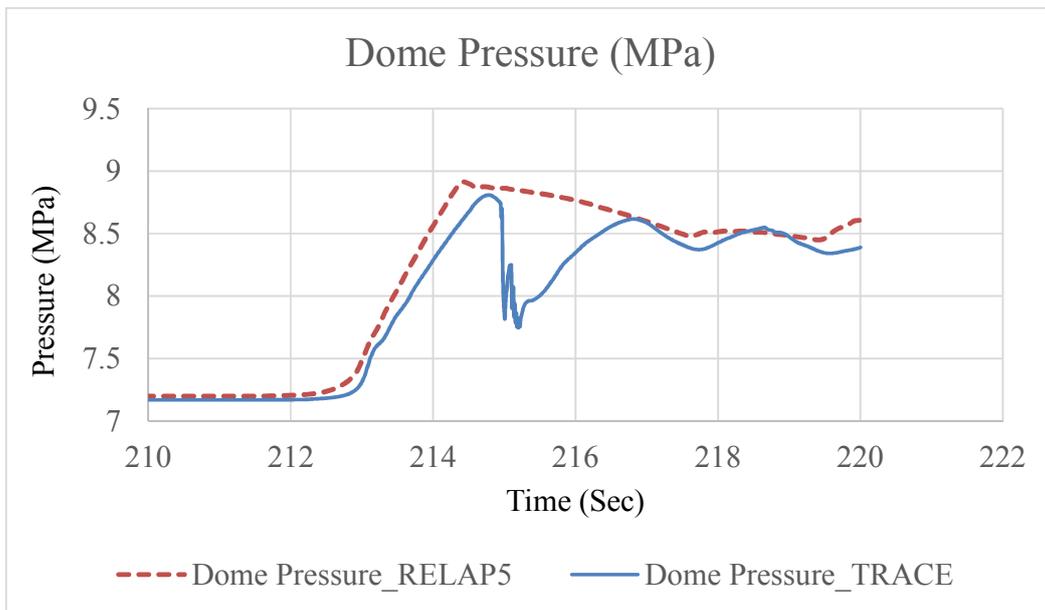
Figure 39   Dome Pressure Variation during the MSIVC Hypothetical Accident
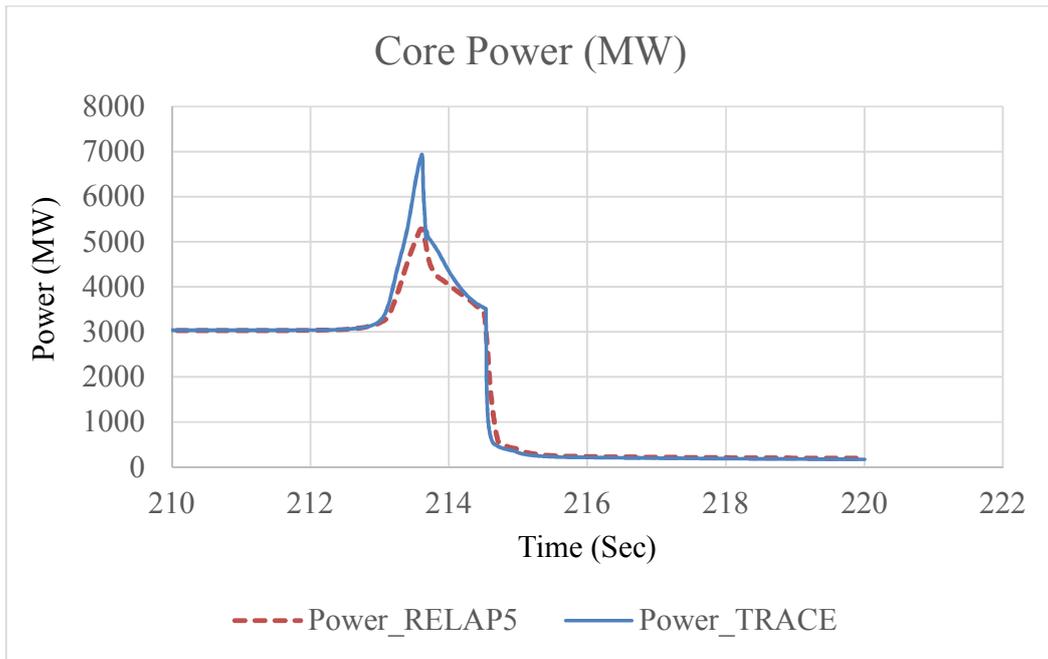
Figure 40   Core Power Variation during the MSIVC Hypothetical Accident

4.2 <u>Turbine Trip with Bypass Failure</u>

<u>Hypothetical Event Sequence</u>

In the hypothetical accident of TSVC, a 500 second steady state was performed to ensure all the parameters matched the operation conditions. Same as the MSIVC hypothetical accident, the reactor scram signal might come from neutron flux exceeding 122% or dome pressure exceeding 7.66 MPa. However, in the TSVC case, the scram signal might also come from the TSVs reaching 90% open. In the RELAP5 model, all the possibilities were concerned as shown in Figure 41.

At 500 second, the turbine tripped and the turbine stop valves started to close with closure time 0.1 second. As the TSVs reached to 90% open, the reactor scram signal was initiated. According to the FSAR, there would be 0.08 second delayed time. After the TSVs closure, the dome pressure increased and the void fraction inside the reactor core decreased. Hence, a positive reactivity feedback functioned and the power increased.

For conservative reason, on 6 safety relief valves opened with delayed time 0.4 second at the dome pressure 7.94 MPa and closed at dome pressure 7.62 MPa. The control system of SRVs was shown in Figure 42. Further, the safety relief valves would fully opened in 0.15 second. Once the safety relief valves opened, the dome pressure and the power plants would be under controlled. Table 9 listed the event sequences and important setpoints of the TRACE model during TSVC transient.
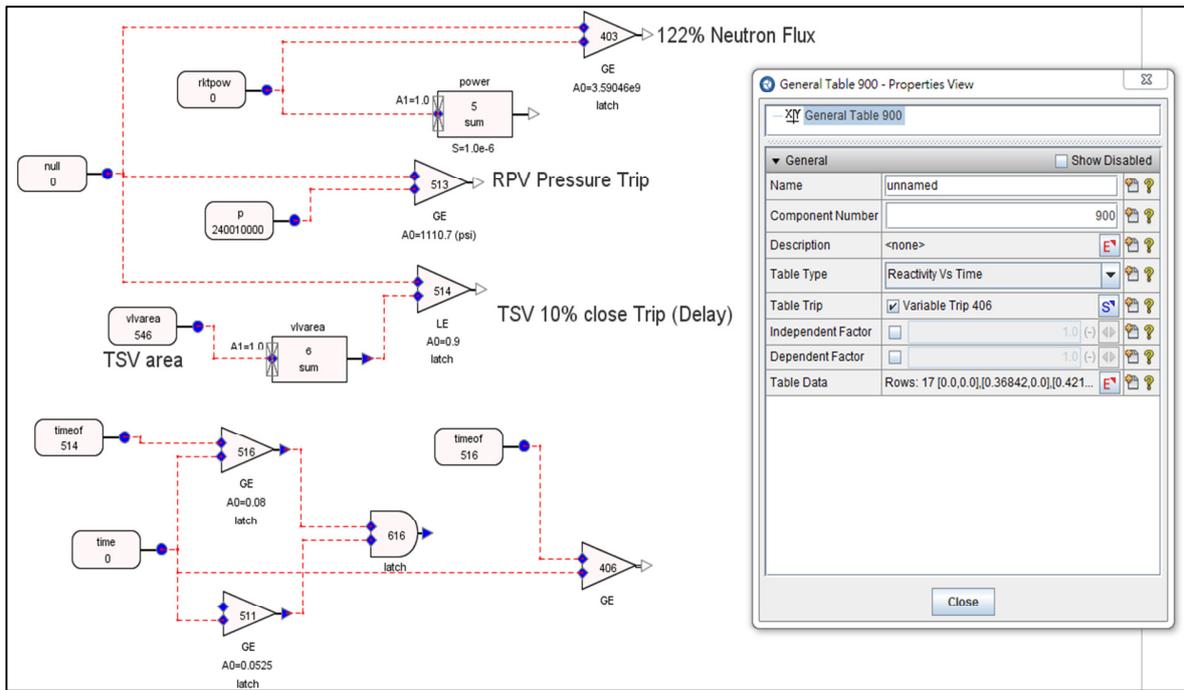
Figure 41　Controlling System of Reactor Scram in TTBF Hypothetical Accident
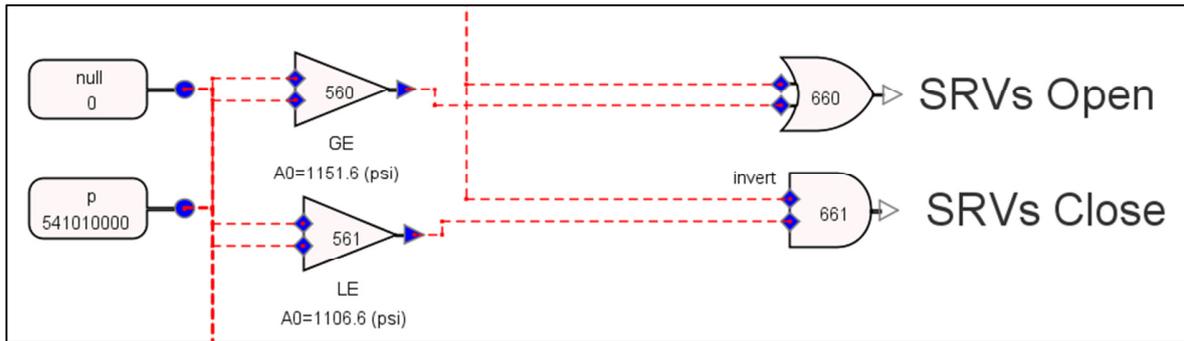
Figure 42　Controlling System of Relief Valves in TTBF Hypothetical Accident

Table 9   Events Sequence and Comparison of TTNB Hypothetical Accident

Events	RELAP5 (sec)	TRACE (sec)	Notes
Steady state	0~499	0~499	Power 3030 MW Feedwater flow rate 1640 kg/sec Feedwater temperature 488.7K Core inlet flow rate 11177.3 kg/sec Dome pressure 7.17MPa
TSVs start to close	500	500	Fully closure time 0.1 second
Reactor scram signal initiated	500.02	500.01	Signal initiated at TSVs 90% open
Reactor scram	500.11	500.09	Delayed time 0.08 second
TSVs fully closed	500.1	500.1	
Relief valves opened	501.34	501.77	Open at dome pressure 7.94MPa and closed at dome pressure 7.63MPa
Peak dome pressure (MPa)	503.81 (8.53MPa)	502.27 (8.2MPa)	
End of the analysis	505	505	

Analysis Data Results

The TSVs started to close at 500 second and fully closed at 500.1 second. Figure 43 shows the comparison of steam flow rate in TRACE and RELAP5 model. At the beginning of the transient, the steam flow dropped rapidly in both models. As mentioned in previous case, because the friction data is hard to be determined in the RELAP5 model, the steam flow rate might decreased (or increased) faster than that in TRACE model. The steam inside the reactor vessel was trapped; hence, the dome pressure increased as shown in Figure 44. Due to the increasing dome pressure, the void fraction of the reactor core would decrease, which caused a positive reactivity feedback. As a result, the power increased as shown in Figure 45. Once the TSVs reached to 90% open, reactor scram signal was sent out with a delayed time 0.08 second. That is, the negative scram feedback would function at 500.09 second. However, the scram reactivity feedback needed some time to dominate the core power. Hence, as shown in Figure 45, the core power kept increasing until the scram feedback dominated the power at 501.4 second. Despite that the reactor scram control system in RELAP5 model was same as that in TRACE model, the negative reactivity feedback dominated the core power with a timing difference. Although the reactor scrammed at 501.4 second, the dome pressure still increased as shown in Figure 44 because the decay heat still produced steam inside the reactor core both in TRACE and RELAP5 model.

As the dome pressure reached to 7.94 MPa, the relief valves open signal was sent out with delayed time 0.4 second. Hence, as shown in Figure 44, the relief valves really functioned until the dome pressure reached to 8.2 MPa. Once the relief valves opened, the dome pressure decreased in TRACE model. However, in the RELAP5 model, the dome pressure would maintain at 8.5 MPa, which was still under the ASME regulation 9.58MPa.

Due to the opening of the relief valves, the steam flow rate increased again as shown in Figure 43. However, only six relief valves functioned in this hypothetical accident, the steam flow was only 0.3 times of the normal operating flow rates. From the analysis results of RELAP5 and TRACE models, it is noticed that the reactor scram system can successfully inhibit positive reactivity which comes from the decreasing void fraction. In addition, even though the bypass valve is failed and only six relief valves are functioned, the dome pressure can still be controlled sufficiently.
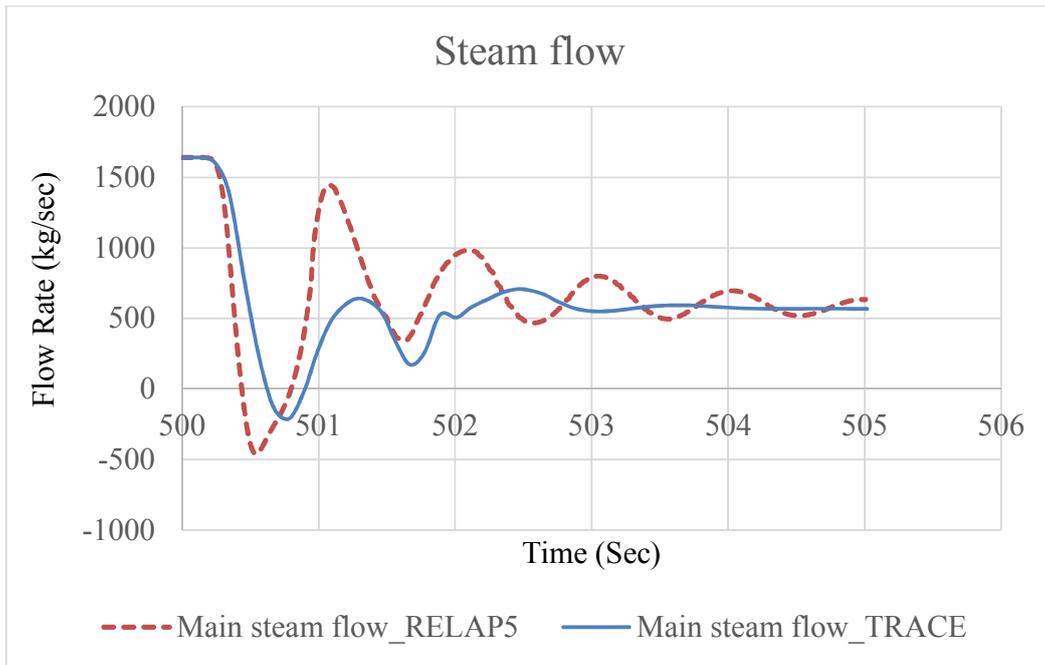
**Figure 43   Steam Flow Variation during the TTBF Hypothetical Accident**
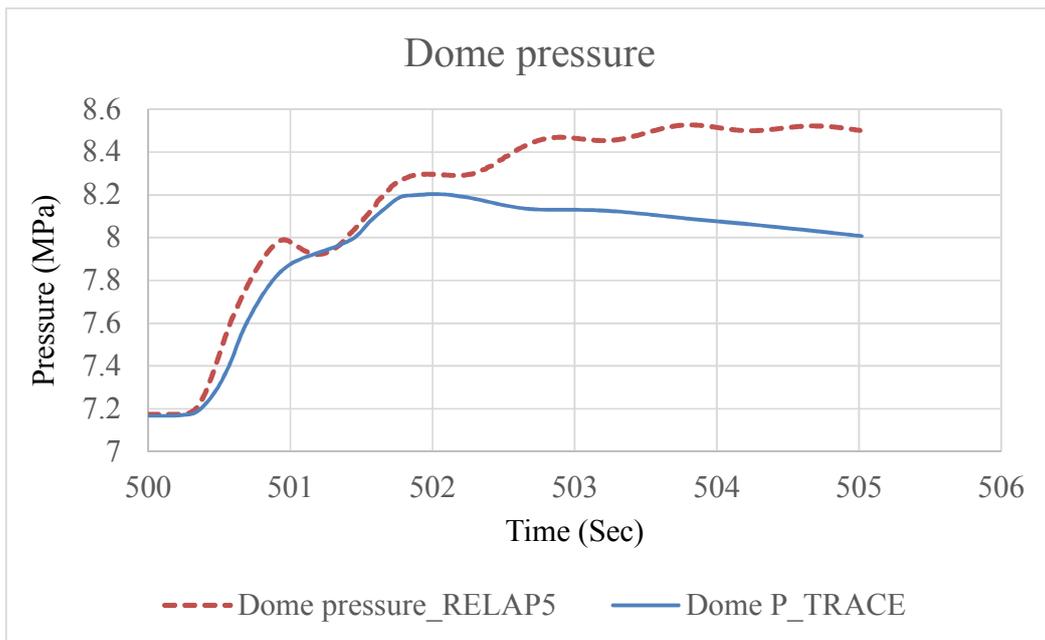
**Figure 44   Dome Pressure Variation during the TTBF Hypothetical Accident**
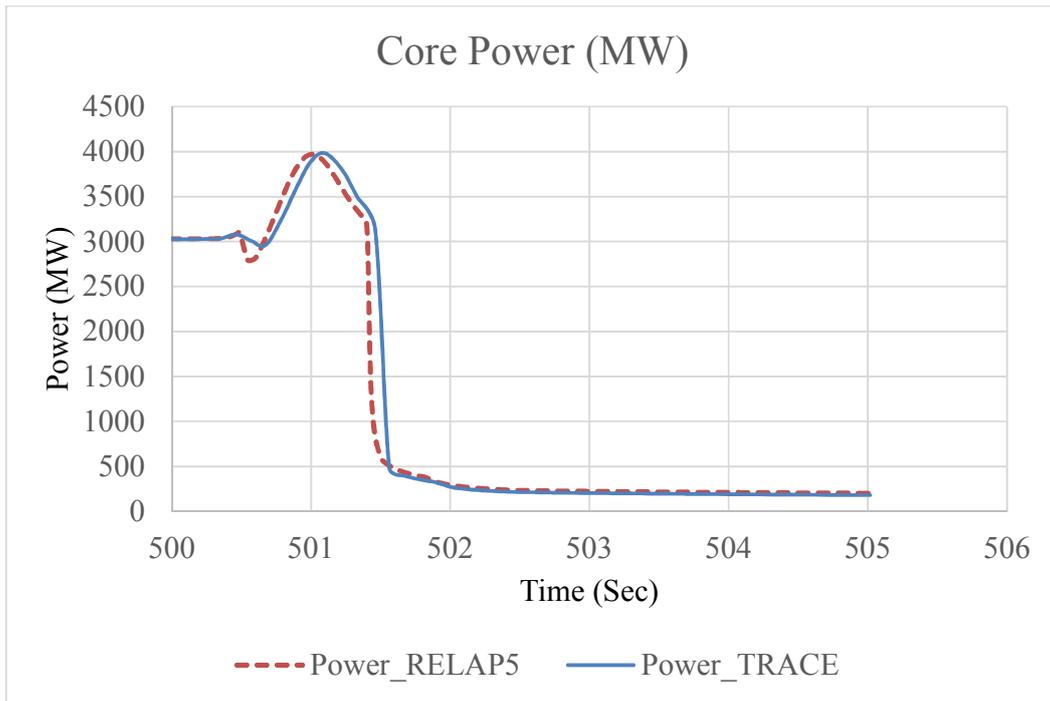
Figure 45   Core Power Variation during the TTBF Hypothetical Accident

4.3 Load Rejection with Bypass Failure

Hypothetical Event Sequence

In the TCVC hypothetical accident, a 210-second steady state simulation was performed to ensure all the parameters reached to the operating conditions. At 210 second, the load rejection happened and the TCVs closed immediately. Once the turbine control valves closed, the main steam line flow decreased immediately which increased the dome pressure. The positive void fraction reactivity feedback functioned and the power increased. Different from the previous two cases, the recirculation pumps trip signal and the reactor scram signal were sent out immediately as the TCVs closed. However, due to the signal delayed time, the reactor would scram 0.07 second later as shown in Figure 46. Further, the recirculation pumps would be out of service 0.14 second later. For conservative reason, there were only 7 safety valves and 6 relief valves functioned in this case with signal delayed time 0.4 second. The control system of the SRVs is shown in Figure 47. Same as previous two cases, once the safety/relief valves open, the dome pressure decreased and the NPP was under controlled. Table 10 lists the important setpoints and transient events of the TCVC hypothetical accident.
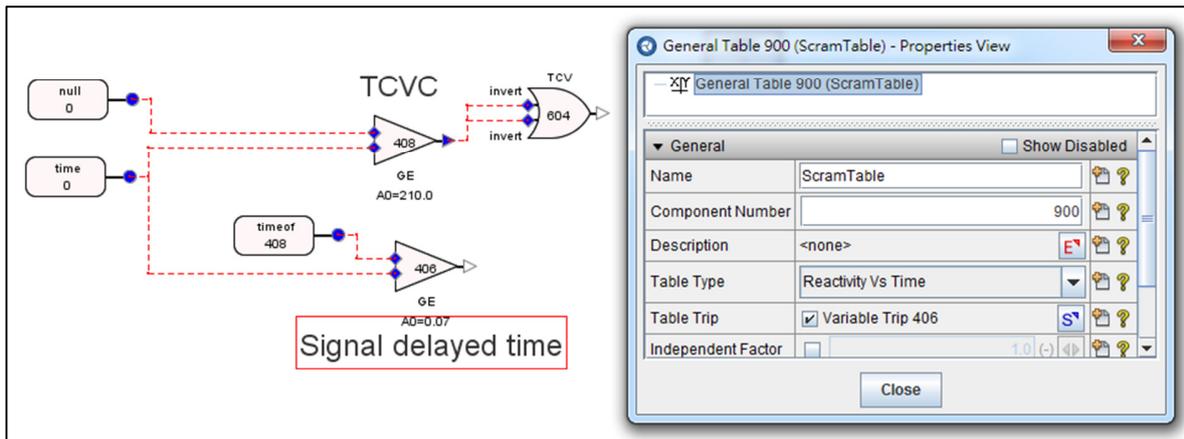
Figure 46   Controlling System of Reactor Scram in LRNB Hypothetical Accident
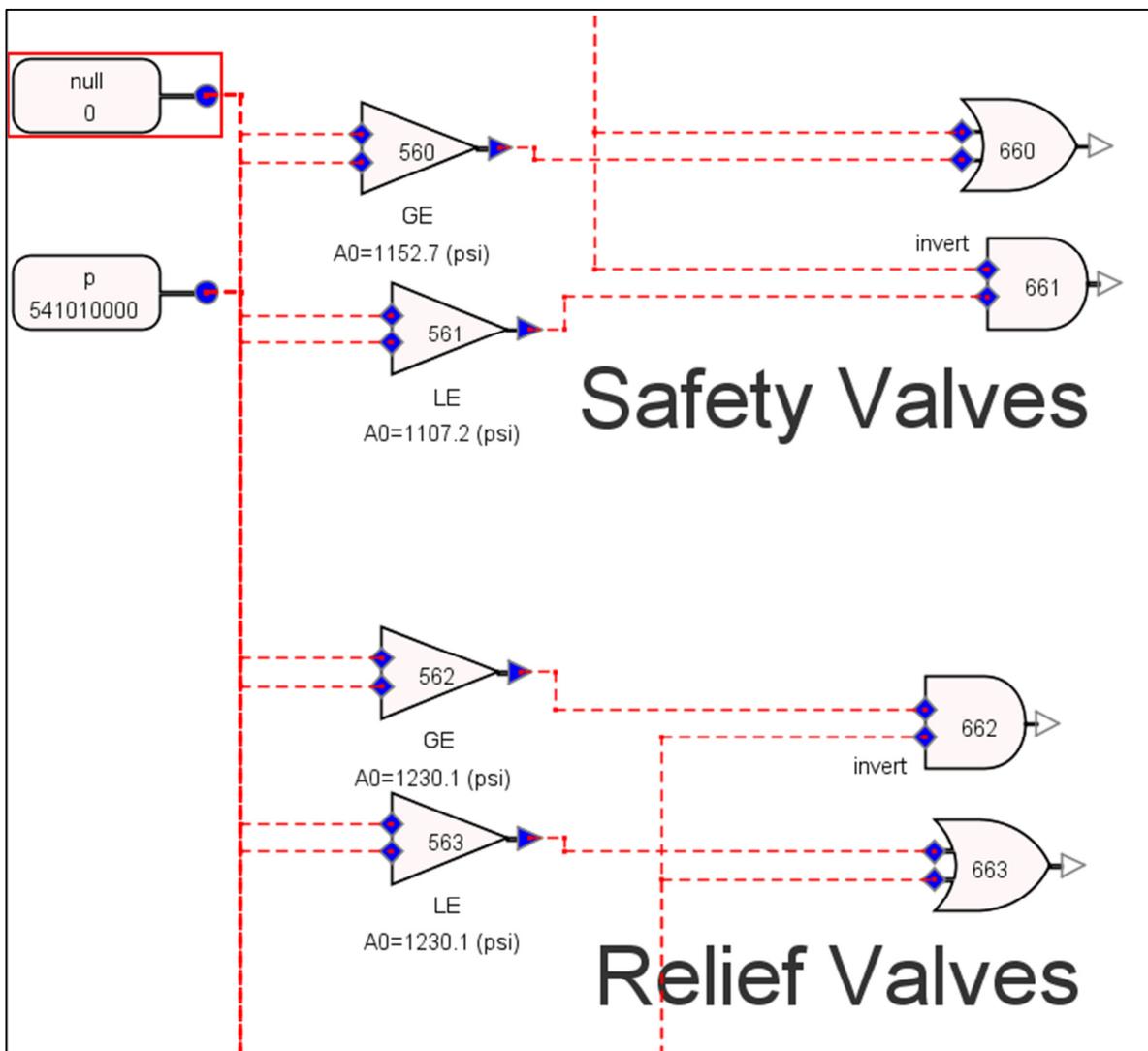
Figure 47   Controlling System of Safety/Relief Valves in LRNB Hypothetical Accident

Table 10   Events Sequence and Comparison of LRNB Hypothetical Accident

Events	RELAP5 (sec)	TRACE (sec)	Notes
Steady state	0~209	0~209	Power 3030 MW Feedwater flow rate 1641 kg/sec Feedwater temperature 488.7K Core inlet flow rate 11177.3 kg/sec Dome pressure 7.17MPa
Turbine control valves closed	210	210	Start of the transient; valve closed in 0.15 second
Reactor scram	210.07	210.07	Signal initiated at TCVs close with delayed 0.07 second
Recirculation pumps trip	211.4	211.4	Signal initiated at TCVs close with delayed for 0.14 second
Safety/relief valves open	211.36	211.76	7 safety valves and 6 relief valves functioned in this model with delayed time 0.4 second
Peak dome pressure (MPa)	213.75 (8.5 MPa)	211.83 (8.2 MPa)	
End of analysis	215	215	

Analysis Data Results

Figure 48 shows the comparison of steam flow rate in TRACE and RELAP5 models. Once the turbine control valves closed, the steam flow rate dropped in both models. However, the steam flow in RELAP5 model dropped more rapidly than that in TRACE model even with same TCVs closure time. Due to the declination of the steam flow, the dome pressure increased immediately as shown in Figure 49. As a result, the void fraction inside the reactor core decreased which caused a positive reactivity feedback. The power increased as shown in Figure 50. Different from previous two cases, the reactor scram signal was generated at once as the turbine control valves closed. The power would decrease at the beginning of the transient both in the TRACE and RELAP5 models. After 210.5 second, the power increased because the positive void fraction reactivity feedback dominated the power. However, at 211.1 second for RELAP5 model and 211.2 second for TRACE model, the negative scram reactivity feedback dominated the power again. The power dropped rapidly. After 211.76 second, the safety/relief valves opened and released the steam inside the reactor vessel. Same as previous two cases, the dome pressure of RELAP5 model would maintain at a higher pressure and the dome pressure of TRACE model would decrease a little after the SRVs opening. Nonetheless, both the dome pressure of TRACE and RELAP5 model were under controlled and the steam flow rate was back to a stable value. On the other hand, the recirculation pump and feedwater pump trip signal would be sent out to avoid the void fraction increased after reactor scram. As a result, Figure 51 shows that the core flow rate decreased.

From those data results mentioned above, the dome pressure did not exceed the criteria 9.58Mpa which was regulated by ASME. Further, all the inside and outside flow rates became stable. Kuosheng NPP was under control in the TCVC transient.
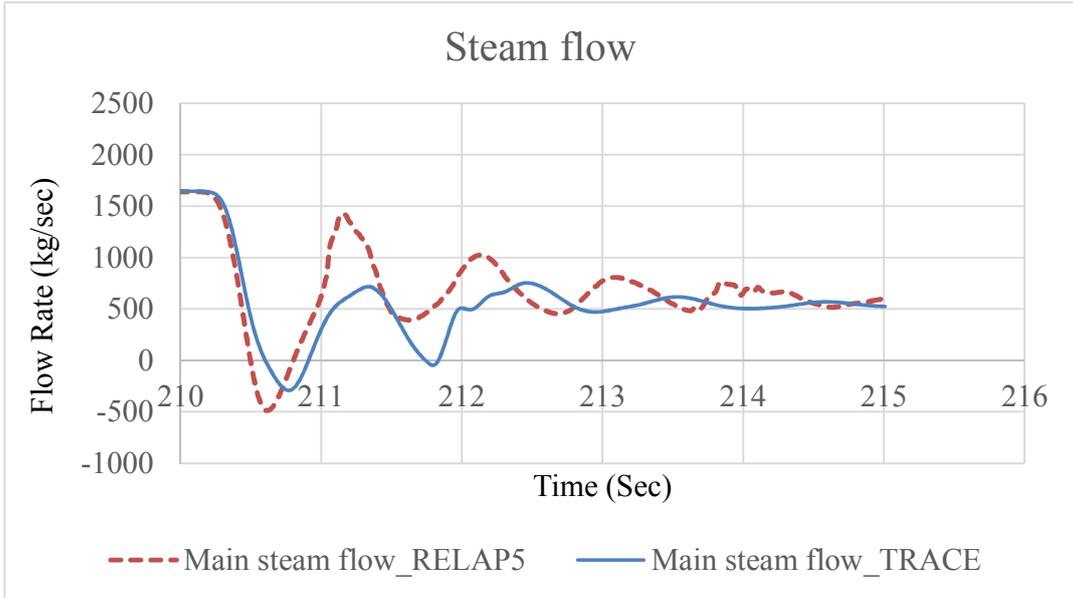
Figure 48   Steam Flow Variation during the LRNB Hypothetical Accident
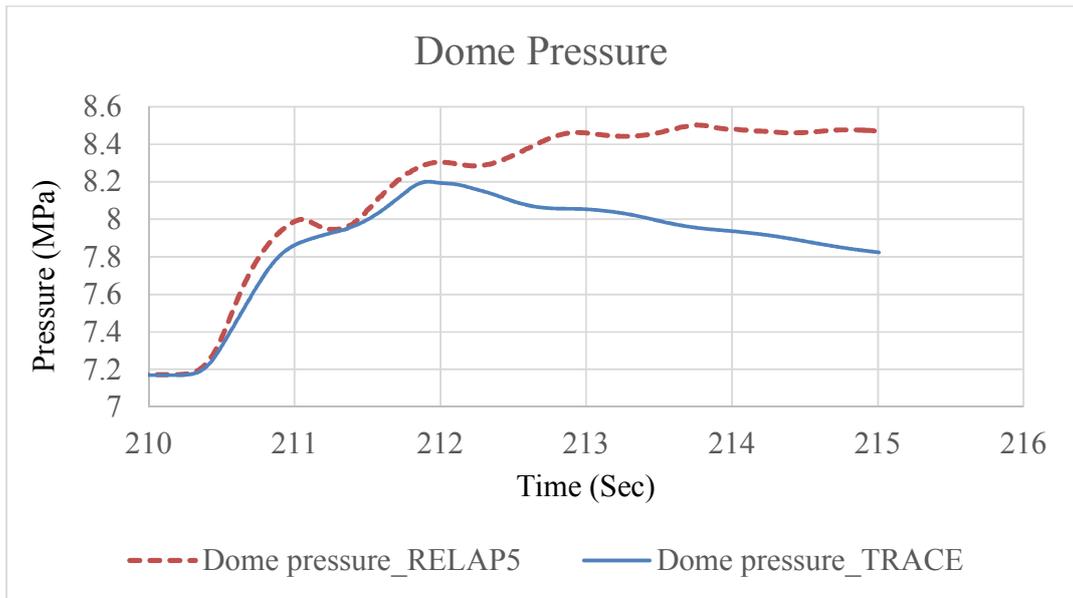
Figure 49   Dome Pressure Variation during the LRNB Hypothetical Accident
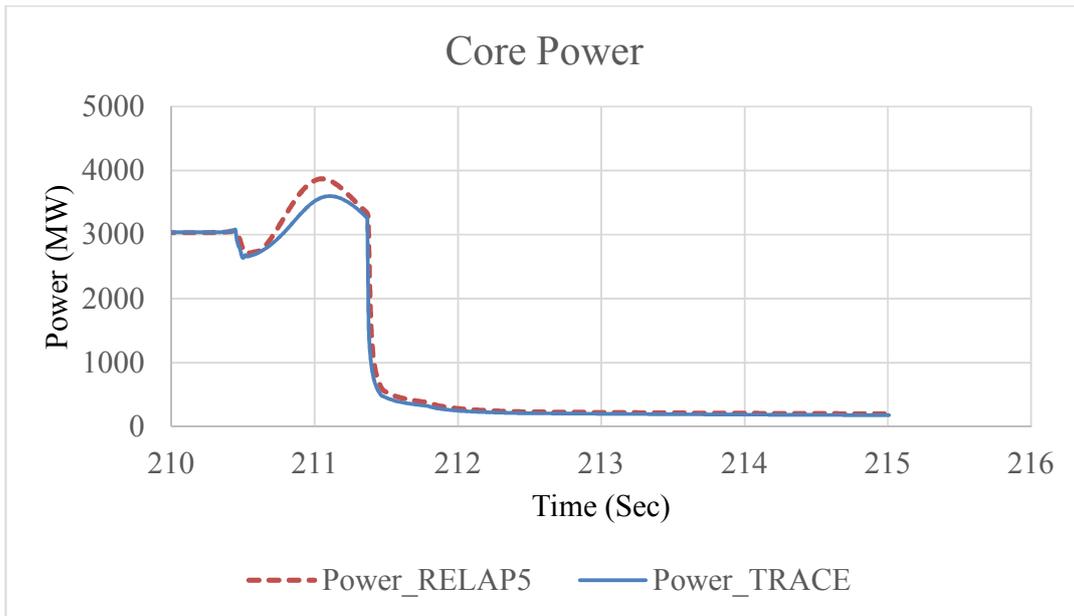
Figure 50   Core Power Variation during the LRNB Hypothetical Accident
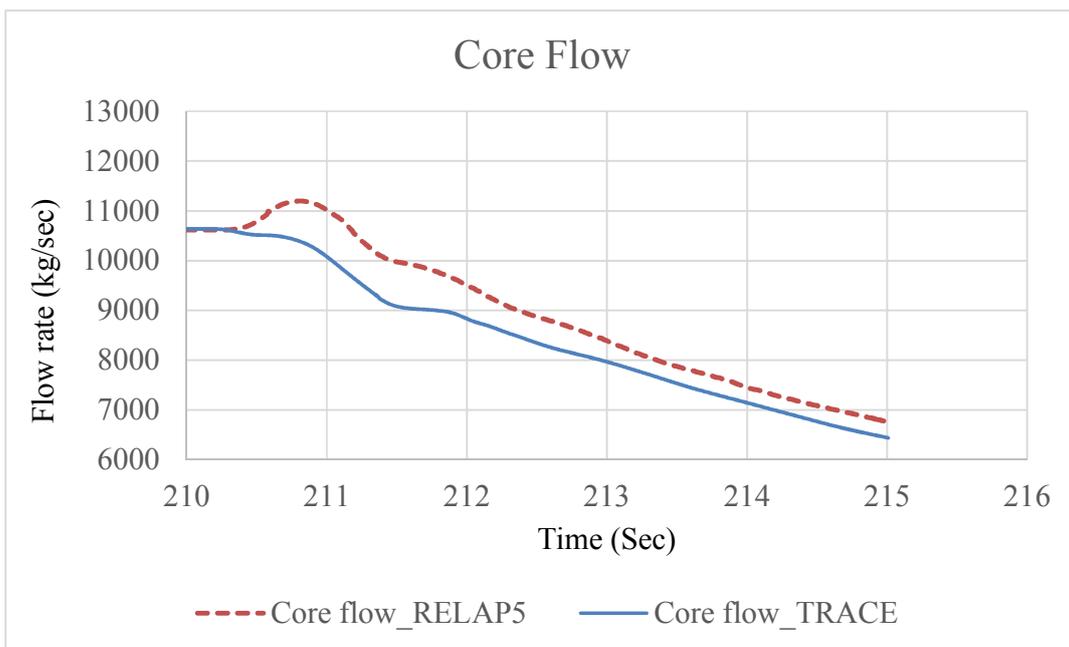
Figure 51   Core Flow Variation during the LRNB Hypothetical Accident

5. CONCLUSIONS

In this research, the RELAP5/MOD 3.3 model of Kuosheng NPP was successfully developed. The startup test data and TRACE model analysis results were used to compare with the results of RELAP5/MOD3.3 model for FWPT, LRWB and MSIVC transient. The predictions of RELAP5/MOD3.3 were consistent with the data of startup test and TRACE roughly. It indicates that there is a respectable accuracy in Kuosheng NPP RELAP5/MOD3.3 model. Additionally, the analysis of the above RELAP5/MOD3.3 model for the overpressurization hypothetical accidents was performed. The predictions of RELAP5/MOD3.3 were also consistent with the results of TRACE roughly. However, there is a difference between the prediction values of RELAP5/MOD3.3 and TRACE for the above transients. Therefore, we will study this difference and modify the models in the future.

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10. SUPPLEMENTARY NOTES

K.Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

After the measurement uncertainty recapture power uprates, Kuosheng nuclear power plant (NPP) has been uprated the power from 2894 MWt to 2943 MWt. For these power upgrade, several analysis codes were applied to assess the safety of Kuosheng Nuclear Power Plant. In our group, there were a lot of effort on thermal hydraulic code, TRACE, had been done before. However, to enhance the reliability and confidence of these transient analyses, thermal hydraulic code, RELAP5/MOD3.3 will be applied in the future. The main work of this research is to establish a RELAP5/MOD3.3 model of Kuosheng NPP with SNAP interface. Model establishment of RELAP5 code is referred to the Final Safety Analysis Report (FSAR), training documents, and TRACE model which has been developed and verified before. After completing the model establishment, three startup test scenarios would be applied to the RELAP5 model. With comparing the startup test data and TRACE model analysis results, the applicability of RELAP5 model would be assessed. Recently, Taiwan Power Company is concerned in stretch power uprated plan and uprates the power to 3030 MWt. Before the stretch power uprates, several transient analysis should be done for ensuring that the power plant could maintain stability in higher power operating conditions. In this research, three overpressurization transient scenario including main steam isolation valves closure, turbine trip with bypass failure and load rejection with bypass failure would be performed. These analysis results would be compared with the TRACE model results which had been performed before.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

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